UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 2, 2011

Mr. Preston Gillespie
Site Vice President
Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, ISSUANCE OF AMENDMENTS REGARDING THE USE OF CASMO-4/SIMULATE-3 METHODOLOGY FOR REACTOR CORES CONTAINING GADOLINIA BEARING FUEL (TAC NOS. ME4646, ME4647, AND ME4648)

Dear Mr. Gillespie:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 377, 379, and 378 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively.

These amendments authorize changes to the Technical Specifications and authorize changes to the "Updated Final Safety Analysis Report" for Oconee Nuclear Station, Unit 1, 2, and 3, to allow the use of CASMO-4/SIMULATE-3 methodology for application to reactor core designs containing low enrichment uranium fuel bearing lumped burnable and/or gadolinia integral absorbers in response to your application dated June 10, 2009, as supplemented by letters dated December 18, 2009, and August 25, 2010.

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. If you have any questions, please call me at 301-415-1345.

Sincerely,

John Stang, Senior 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. 377 to DPR-38
2. Amendment No. 379 to DPR-47
3. Amendment No. 378 to DPR-55
4. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 377
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 Energy Carolinas, LLC (the licensee), dated June 10, 2009, as supplemented by letters dated December 18, 2009, and August 25, 2010, 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 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. 377 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, Renewed Facility Operating License No. DPR-38 is amended to authorize a change to the "Updated Final Safety Analysis Report" (UFSAR) to allow the use of CASMO-4/SIMULATE-3 Methodology for application to reactor core designs containing low enrichment uranium fuel bearing lumped burnable and/or gadolinia integral absorbers, as set forth in the application dated June 10, 2009, as supplemented by letters dated December 18, 2009, and August 25, 2010. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

4. 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

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-38

Date of Issuance: August 2, 2011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 379
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 Energy Carolinas, LLC (the licensee), dated June 10, 2009, as supplemented by letters dated December 18, 2009, and August 25, 2010, 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 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. 379 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, Renewed Facility Operating License No. DPR-47 is amended to authorize a change to the "Updated Final Safety Analysis Report" (UFSAR) to allow the use of CASMO-4/SIMULATE-3 Methodology for application to reactor core designs containing low enrichment uranium fuel bearing lumped burnable and/or gadolinia integral absorbers, as set forth in the application dated June 10, 2009, as supplemented by letters dated December 18, 2009, and August 25, 2010. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

4. 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

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-47

Date of Issuance: August 2, 2011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 378
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 Energy Carolinas, LLC (the licensee), dated June 10, 2009, as supplemented by letters dated December 18, 2009, and August 25, 2010, 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 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. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 378 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Further, Renewed Facility Operating License No. DPR-55 is amended to authorize a change to the "Updated Final Safety Analysis Report" (UFSAR) to allow the use of CASMO-4/SIMULATE-3 Methodology for application to reactor core designs containing low enrichment uranium fuel bearing lumped burnable and/or gadolinia integral absorbers, as set forth in the application dated June 10, 2009, as supplemented by letters dated December 18, 2009, and August 25, 2010. The licensee shall update the UFSAR by adding a description of this change, as authorized by this amendment, and in accordance with 10 CFR 50.71(e).

4. 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

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-55

Date of Issuance: August 2, 2011

ATTACHMENT TO LICENSE AMENDMENT NO. 377

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 379

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 378

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Renewed Facility 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 Pages	Insert Pages
Licenses	Licenses
License No. DPR-38, page 3	License No. DPR-38, page 3
License No. DPR-47, page 3	License No. DPR-47, page 3
License No. DPR-55, page 3	License No. DPR-55, page 3
TSs	TSs
5.0.26	5.0.26

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 377 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or

Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 379 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

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Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 378 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000-P-A, Thermal Hydraulic Transient Analysis Methodology;

(8) DPC-NE-2005-P-A, Thermal Hydraulic Statistical Core Design Methodology;

(9) DPC-NE-3005-P-A, UFSAR Chapter 15 Transient Analysis Methodology;

(10) BAW-10227-P-A, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel;

(11) BAW-10164P-A, RELAP 5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and non-LOCA Transient Analysis; and

(12) DPC-NE-1006-P-A, Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3 (Revision 0, May 2009).

The COLR will contain the complete identification for each of the Technical Specifications referenced topical reports used to prepare the COLR (i.e., report number, title, revision number, report date or NRC SER date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 377 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 379 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AND

AMENDMENT NO. 378 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By application dated June 10, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML091630712), as supplemented by letters dated December 18, 2009 (ADAMS Accession No. ML100060881), and August 25, 2010 (ADAMS Accession No. ML102430157), Duke Energy Carolinas, LLC (Duke, the licensee), requested approval of changes to the Technical Specifications (TSs) and the "Updated Final Safety Analysis Report" (UFSAR) for Oconee Nuclear Station, Units 1, 2, and 3 (Oconee 1/2/3), to allow the use of CASMO-4/SIMULATE-3 methodology for application to reactor core designs containing low enrichment uranium (LEU) fuel bearing lumped burnable and/or gadolinia integral absorbers. Specifically, the licensee requests the U.S. Nuclear Regulatory Commission (NRC) to review and approve the methodology report, DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3 Revision 0" (Proprietary).

Methodology report DPC-NE-1006-P Revision 0, "Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3," describes the methodology for application to core designs containing LEU fuel bearing lumped burnable and/or gadolinia integral absorbers and its associated technical justification. This methodology is consistent with that used for the Catawba Nuclear Station and McGuire Nuclear Station reload core designs (References 1 and 2).

These amendments are only authorizing the methodology associated with CASMO-4/SIMULATE-3. The licensee by letter dated October 19, 2009 (ADAMS Accession No. ML092960626), requested approval of the use of gadolinia as an integral burnable absorber in a uranium oxide fuel matrix. This action was approved by the NRC staff, and the separate amendments dated July 21, 2011 (ADAMS Accession No. ML11137A150).

The supplements dated December 18, 2009, and August 25, 2010, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 19, 2010 (75 FR 13314).

2.0 REGULATORY EVALUATION

The principal design criteria for Oconee Nuclear Station Units 1, 2, and 3 were developed in consideration of the seventy General Design Criteria for Nuclear Power Plant Construction Permits proposed by the Atomic Energy Commission (AEC) in a proposed rulemaking published in the *Federal Register* on July 11, 1967 (32 FR 10213). The applicable regulatory requirements for reactor core design are defined in the Oconee 1/2/3 TSs Section 5.6.5 and the UFSAR, Chapter 3, Criterion 6, and referenced in Chapters 4 and 15 of the UFSAR.

The licensee performed an evaluation on the proposed changes to the UFSAR pursuant to Title 10 of the *Code of Federal Regulations* (10CFR) Section 50.59, The licensee determined that the proposed changes require prior NRC approval as a departure from methodology in accordance with 10 CFR 50.59(c)(2)(viii). Therefore, the licensee requested approval of the UFSAR changes by submitting a license amendment request (LAR) to allow the use of CASMO-4/SIMULATE-3 methodology pursuant to 10 CFR 50.90 in addition to the changes to the TSs. The nuclear design review of fuel assemblies, control systems, and reactor core is carried out to aid in confirming that fuel design limits will not be exceeded during normal operation or anticipated operational transients. The NRC staff's acceptance criteria are based on Chapter 4.3, "Nuclear Design," of the Standard Review Plan for the "Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," May 1980 (NUREG 75/087).

The CASMO–4/SIMULATE–3 code system has been previously approved by the NRC for analyzing reactor cores with LEU fuel bearing lumped burnable absorbers in analysis for:

- Catawba and McGuire Nuclear Stations (References 1 and 2)
- Palo Verde Nuclear Station
- North Anna and Surry Nuclear Stations

3.0 TECHNICAL EVALUATION

Methodology report DPC-NE-1006-P, "Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3," describes the methodology for application to core designs containing LEU fuel bearing lumped burnable and/or gadolinia integral absorbers and its associated technical evaluation. The CASMO-4/SIMULATE-3 code system, as it pertains to this evaluation, is applied to reactor cores containing LEU fuel with gadolinia. Accordingly, the code system is hereinafter referred to as CASMO-4/SIMULATE-3.

3.1 Computer Codes and Models

The codes that the licensee used for benchmarking comparisons with reactor cores containing gadolinia are: CASMO-4, SIMULATE-3, and CMS-LINK. This code is a multi-group, two-dimensional transport theory model with a microscopic depletion model for burnable absorbers. CASMO-4 is used to calculate lattice physics parameters, including cross sections,

pin power distributions and other nuclear data, which are used as inputs to SIMULATE-3. The SIMULATE-3 code is a two-group, three-dimensional coarse-mesh nodal diffusion theory simulator. SIMULATE-3 combines the nodal solution with the heterogeneous lattice solution from CASMO-4 to calculate pin power distributions. To model reactor cores containing gadolinia, no modifications to the nodal solution or pin power reconstruction routines are necessary. The other code that is used by the licensee with this package is CMS-LINK, which processes data from CASMO-4 to produce multi-dimensional tables for input to SIMULATE-3.

3.2 Oconee Nuclear Station Benchmark Analyses

The licensee used the CASMO-4/SIMULATE-3 code system to calculate predicted reactivity parameters and fuel assembly power distributions for comparisons to measured data from the Oconee 1/2/3 reactor cores. Measured data for critical boron concentration, isothermal temperature coefficient, and control rod bank worth was compiled from startup physics testing and routine monitoring during the operation of Oconee Unit 1 fuel cycles 16 through 22; Oconee Unit 2 fuel cycles 15 through 21; and Oconee Unit 3 fuel cycles 16 through 22. In addition, core power distribution measurements for these cycles were taken at regular intervals with the Oconee incore system. The licensee compared these measured data to the predicted values in order to assess the uncertainty of the CASMO-4/SIMULATE-3 models of reactor cores containing gadolinia fuel.

3.2.1 Critical Boron Concentration Comparisons

Critical boron concentration measurements for the Oconee Units were taken during start-up physics testing at the beginning of each fuel cycle and throughout full power operation by sampling the reactor coolant. The measurements made during start-up physics testing were taken at beginning of cycle (BOC), hot zero power (HZP) conditions, with all rods out of the core (ARO), peak samarium, and no xenon. The full power measurements were taken at or near hot full power (HFP) nominal conditions at several burnups throughout the fuel cycle. Full power measurements were corrected for Boron (B)-10 depletion, and from the measured rodded condition to the ARO condition.

Using the CASMO-4/SIMULATE-3 models, the licensee calculated a predicted value of critical boron concentration for each measurement from the Oconee Units. The predicted critical boron concentrations agree well with the measurements. The NRC has previously reviewed critical boron concentration comparisons of CASMO-4/SIMULATE-3 predictions to previously approved Duke Energy Carolinas, LLC Methodology (References 1 and 2). The results of these previous comparisons are similar to the results of the comparisons with Oconee cores.

3.2.2 Isothermal Temperature Coefficient Comparisons

The isothermal temperature coefficient (ITC) was measured at BOC, HZP, and ARO conditions during startup physics testing for all three Oconee units. The licensee's predictions of ITC using CASMO-4/SIMULATE-3 were found, by the NRC staff to be comparable to the accuracy of the previously approved Duke Energy Carolinas, LLC methodology.

3.2.3 Control Rod-worth Comparisons

During the Oconee startup testing, control rod bank worth measurements were made at BOC, HZP, peak samarium, and no xenon conditions. The Oconee 1/2/3 control bank worth measurements were performed using the Boron Rod Swap technique, as described in the June 10, 2009, application. Comparisons to CASMO-4/SIMULATE-3 predictions were made for each of the individual bank worth's and for the sum of bank worth's for each cycle. All predicted values are in good agreement with the measured control rod bank worth. The results of these comparisons are also similar to previous control rod bank worth comparisons reviewed and approved by the NRC staff at other Duke Energy Carolinas, LLC, power plants such as Catawba Nuclear Station and McGuire Nuclear Station reactor cores (References 1 and 2).

3.2.4 Fuel Assembly Power Distribution Analysis and Uncertainty Factors

Core power distributions were measured at regular intervals during operation using the Oconee fixed self-powered neutron detectors (SPNDs) placed in instrument tube of selected fuel assemblies during reactor operation. During the cycles power distribution measurements were made to determine the following: the measured assembly peaking factors assembly/pin radial power – average relative power in each fuel assemble pin ($F\Delta H$); assemble/pin maximum power – largest relative power in each assemble/pin (Fq); and assembly/pin axial power ($Fz = Fq/F\Delta H$) (Fz). The licensee used SIMULATE-3 to model the reactor conditions for all power distribution measurements and produce predicted values for the assembly peaking factors. The uncertainties between the predicted and measured values are characterized by assembly uncertainty factors. The licensee defines assembly uncertainty factors, referred to as observed nuclear reliability factors (ONRFs), according to the following expression:

$$ONRF = 1 - bias + K_a\sigma_a$$

The bias is the mean of the predicted value, minus measured values, and $K_a\sigma_a$ is the statistical deviation of the bias. The "a" subscript in $K_a\sigma_a$ is used to represent assembly-averaged values. The σ_a is the standard deviation of the bias distribution, and the K_a factor is determined from a 95% one-sided upper tolerance interval with a 95% confidence level, as described in References 3 and 4. The determination of $K_a\sigma_a$ required that the data passes a normality test (Reference 5), at the 1 percent significance level. If the data fails the normality test, then a conservatively large value is assigned to $K_a\sigma_a$ using a non-parametric evaluation. Note that all data used in these calculations were deemed to be normally distributed. The NRC staff reviewed the statistical methods used to calculate the ONRFs and have found them to be acceptable.

The calculated ONRF values for assembly $F\Delta H$, Fq, and Fz indicate good agreement between measured power distributions and those predicted with SIMULATE-3. The ONRFs from this comparison to Oconee data have similar values to the ONRFs that the licensee has previously calculated for other Duke Energy Carolinas, LLC, plants.

The complete power distribution uncertainties incorporate the bias and uncertainty of the assembly average power distributions along with the uncertainty from predictions of fuel pin power distributions. The licensee's determination of uncertainty in the pin power distribution is discussed in Section 3.3 below. The licensee's December 18, 2009 supplement provided additional information and clarifications in determining the power distribution uncertainties. The

licensee addressed such issues as the applicability of the licensee's uncertainty methodology specific to the Oconee Nuclear Station. The licensee also addressed the applicability of the CASMO-4/SIMULATE-3 benchmarking simulations to the Oconee Nuclear Station. The NRC staff reviewed of the licensee's supporting technical basis and found them to be acceptable.

3.2.5 TMI Benchmarking Results

The licensee also compared measured core physics parameters from Three Mile Island (TMI) nuclear plant against predicted values of CASMO-4 and SIMULATE-3 predictions. Since TMI fuel contains gadolinia, it serves a purpose to demonstrate the ability of this methodology predictions to reactor core containing fuel assemblies bearing gadolinia burnable absorber. The average deviation between measured and calculated values and the associated standard deviation for each of the four reactor physics parameters evaluated (BOC HZP ARO critical boron concentration, HFP critical boron concentration, BOC HZP control rod worth and BOC HZP isothermal temperature coefficient) were determined from the TMI benchmark calculations. These variations are summarized in Table 3-13 of the June 10, 2009, application.

Results were also obtained from the power distribution benchmark analyses. The ONRFs for the F_h, Fq, and Fz peaking factors that were developed from comparisons of the TMI measured power distribution data and CASMO-4/SIMULATE-3 predicted values are summarized in Table 3-12 of June 10, 2009, application.

The results presented demonstrate that the performance of the CASMO-4/SIMULATE-3 core model is acceptable for modeling gadolinia-bearing fuel based on the extensive benchmarking against measured reactivity and power distribution data obtained from TMI Cycles 13-16. Based on the NRC staff's review of the licensee's benchmarking, the results and conclusions drawn from the benchmarks are considered applicable to future Oconee core designs containing gadolinia-bearing fuel. This determination is based on diverse set of benchmark calculations performed encompassing transition and full gadolinia core designs, and the large range of gadolinia concentrations and absorber patterns evaluated.

3.3 Pin Power Uncertainty Factor

To determine the accuracy of the CASMO-4 and SIMULATE-3 models for predicting pin power distributions in reactor cores containing gadolinia, the licensee compared these models to results from the Babcock and Wilcox (B&W) Urania Gadolinia critical experiments (Reference 6). These benchmark comparisons were used to develop uncertainty factors for both LEU fuel pin powers and gadolinia fuel pin powers. The LEU pin power uncertainty was determined by direct comparison of SIMULATE-3 predictions with measurements from the critical experiments. For the uncertainty in gadolinia fuel pin power, the licensee employed an alternate approach. The B&W critical experiment measurements were taken at, or near, BOC conditions, where the gadolinia pin power density is non-limiting and significantly lower than that for the LEU pins. The licensee based the gadolinia pin power uncertainty on a combination of comparisons to B&W critical experiment data and an evaluation of a series of theoretical infinite lattice 2x2 colorset calculations at different burnups.

3.3.1 LEU Pin Power Uncertainty

The licensee determined the LEU pin power uncertainty by modeling the power distributions from the B&W critical experiments for core configurations 5, 14, and 20, containing gadolinia fuel. LEU pin power distributions were calculated separately using CASMO-4 and SIMULATE-3, and uncertainties were developed for each code. The uncertainty is based on the predicted minus measured percent error and on the $K\sigma$ value, which is described in Section 3.2.4 of this SE and derived from References 4 and 5. The CASMO-4 and SIMULATE-3 predictions produce similar pin power uncertainty values, with the SIMULATE-3 uncertainty being slightly larger than CASMO-4. Both uncertainties agree very well with the CASMO-4 and SIMULATE-3 LEU pin power uncertainties from Revision 0 of the topical report (Reference 7), which compared predictions to B&W critical experiments with non-gadolinia cores.

The licensee uses the SIMULATE-3 LEU pin power uncertainty value in the final calculation of combined power distribution uncertainty factors (discussed in Section 3.4 below). The licensee does not provide a justification for using the SIMULATE-3 uncertainty rather than the CASMO-4 uncertainty, but the NRC staff finds this decision is acceptable since the SIMULATE-3 uncertainty is larger than the CASMO-4 uncertainty.

3.3.2 Gadolinia Pin Power Uncertainty

The first component that the licensee used in the determination of the gadolinia pin power uncertainty was the benchmark comparison of the B&W critical experiment data to predicted values. The licensee used the CASMO-4 code to calculate the pin power distribution for the gadolinia fuel rods from the B&W critical experiment core configurations 5, 14, and 20 (Reference 6). The predicted powers were compared to measured data to find the bias (i.e. the mean difference of predicted values minus measured values) and the standard deviation.

The bias and standard deviation were divided by the average gadolinia pin power to find the percent uncertainty. The average gadolinia pin power, as measured in the B&W critical experiments, was quite low. Using such a low value in these calculations is not meaningful. The low gadolinia pin power, measured at BOC conditions in the critical experiments, is not representative of the higher pin powers that are reached after the gadolinia is depleted. The licensee chose a conservative value to use for the average gadolinia pin power in calculating the percent uncertainty. The value that the licensee chose is acceptable and recognized as conservative in the uncertainty calculation.

The gadolinia pin power data from the B&W critical experiments were demonstrated to be normally distributed by the "W" test (Reference 6). A K-factor for a 95/95 upper tolerance was applied to the statistical uncertainty (as described in Reference 5). The NRC staff reviewed the data sets utilized for all the comparisons, as well as the methodology used in the uncertainty analysis, and determined that the results for the CASMO-4 gadolinia pin power uncertainty were conservative and therefore acceptable.

The licensee determined that a comparison of the CASMO-4 predicted values to the B&W critical experiment data was, by itself, not sufficient to establish a gadolinia pin power uncertainty. The B&W critical experiment data taken at BOC conditions are non-limiting for gadolinia fuel pins. The gadolinia pin power is of most concern after the gadolinia is depleted and the gadolinia pin power approaches, or exceeds, the assembly average power. To resolve the uncertainty for this burnup

range, the license performed a series of theoretical infinite lattice 2x2 colorset calculations with CASMO-4 and SIMULATE-3. The results of the two codes were compared to characterize the SIMULATE-3 to CASMO-4 pin power reconstruction uncertainty.

The licensee selected a diverse set of 2x2 fuel assembly loading patterns for the infinite lattice calculations with SIMULATE-3 and CASMO-4. The licensee evaluated 11 cases with different combinations of burned fuel and feed fuel assemblies with varying numbers of gadolinia fuel pins with concentrations up to 8.0 w/o Gd_2O_3. Each colorset was modeled for a number of different burnups up to approximately 20 GWD/MTU, and gadolinia depletion was considered.

The licensee based the SIMULATE-3 gadolinia pin power reconstruction uncertainty on comparisons between the CASMO-4 and SIMULATE-3 power distributions for the 2x2 colorset cases described above. Both code calculations were normalized to an average assembly power of 1.0; therefore, the mean difference between the two predictions was 0. So, the gadolinia pin power uncertainty was based on the broadness of the distribution. The data set was tested for normality with the "D" tests (Reference 6) and found to be not normal. Consequently, the uncertainty was based on a non-parametric evaluation of the data set. References 3 and 4 were consulted to determine the 95/95 one-sided tolerance for the data. This value was taken to represent the SIMULATE-3 to CASMO-4 gadolinia pin power reconstruction uncertainty.

The licensee calculated the total gadolinia pin power uncertainty by combining the SIMULATE-3 to CASMO-4 uncertainty with the $K\sigma$ value from the B&W critical experiment comparison with CASMO-4. These values were combined by taking the square-root of the sum of the squares. The bias term from the CASMO-4 comparison was also added to the uncertainty. The combination of these terms yields the total gadolinia pin power reconstruction uncertainty for SIMULATE-3. Based on the NRC staff review, this methodology is deemed acceptable for producing a suitable value of gadolinia pin power uncertainty.

3.4 Statistically Combined Power Distribution Uncertainty Factors

The licensee has defined power distribution uncertainty factors to be applied to peaking factors for design of reload cores and for surveillance of operating cycles. These uncertainty factors, referred to as statistically combined uncertainty factors (SCUFs), combine the inter-assembly power uncertainty and the intra-assembly pin power uncertainty.

The SCUF is calculated for each of the power distribution peaking factors $F\Delta H$, Fq, and Fz. The SCUFs are determined for LEU fuel and gadolinia fuel separately. These factors are applied to core reload designs and to surveillance tests to assure a conservative evaluation of fuel pin performance. Based on the NRC staff's review, the values of the uncertainty factors are reasonable, and the licensee's methodology to determine the SCUF values is acceptable.

3.5 Summary of Assessment of DCP-NE- Revision 0 of 1006-P

The licensee intends to use the CASMO-4/SIMULATE-3 code system for reload design analyses for reactor cores containing gadolinia for the Oconee Nuclear Station. To qualify this code system, the licensee has performed a series of benchmark comparisons. Reactivity and assembly power distribution predictions were compared to data from four TMI fuel cycles. Fuel pin power distributions were compared to measurements from the B&W Urania Gadolinia critical experiments (Reference 6). The comparisons demonstrate the capability of the

CASMO-4/SIMULATE-3 code system to adequately reproduce reactivity and power distribution calculations for reactor cores containing gadolinia.

The CASMO-4 based SIMULATE-3 predictions of reactivity parameters were compared to measurements from TMI, and the NRC staff found acceptable accuracy. Comparisons were made for critical boron concentrations, both at BOC HZP and HFP conditions. Isothermal temperature coefficient and control rod worth comparisons were made for BOC HZP conditions. All deviations between measurement and prediction produced similar results to those in the licensee's prior submittal (Reference 3) for non-gadolinia cores.

CASMO-4 based SIMULATE-3 calculations were also used to predict the assembly average power distributions from the Sequoyah Unit 2 cycles containing gadolinia fuel. The calculated uncertainty factors for assembly FΔH, Fq, and Fz indicate good agreement between measured power distributions and the SIMULATE-3 predictions.

The pin power distribution uncertainty for LEU fuel rods was resolved by comparing CASMO-4 and SIMULATE-3 predictions to data from the B&W Gadolinia critical experiments. For the gadolinia fuel pins, the determination of pin power uncertainty was based on two inputs: 1) the comparison of CASMO-4 predictions to measurements from the B&W critical experiments, and 2) comparison of SIMULATE-3 to CASMO-4 calculations for a set of theoretical 2x2 assembly configurations at a number of different burnups. The gadolinia fuel comparison of CASMO-4 to the B&W critical experiments relies on a small number of data points. Despite this limitation, the comparison, along with the SIMULATE-3 to CASMO-4 comparisons, demonstrates the licensee's ability to satisfactorily reconstruct pin power distributions with the CASMO-4 and SIMULATE-3 codes. The NRC staff reviewed the data and finds the licensee can predict the pin power distribution using CASMO-4/SIMULATE-3.

The licensee uses the combined assembly average power uncertainties and pin power uncertainties to calculate FΔH, Fq, and Fz uncertainty factors for LEU fuel and gadolinia fuel. The peaking factor statistically combined uncertainties are used for analysis of reload designs for reactor cores containing gadolinia fuel. The calculation of these uncertainty factors are found to be acceptable for both LEU and gadolinia fuel.

Based on the evaluation of topical report DPC-NE-1006-P Revision 0, as delineated above, the NRC staff finds CASMO-4/SIMULATE-3 methodology is acceptable for calculating steady-state physics parameters for use in reload design analyses for Oconee 1/2/3 reactor cores containing gadolinia fuel.

4.0 SUMMARY

The proposed changes to the UFSARs based on Revision 1 to DPC-NE-1005-P, Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX, have been reviewed by the NRC staff. Revision 0 of DPC-NE-1006-P presents results of benchmarking studies comparing CASMO-4/SIMULATE-3 reactivity and power distribution predictions to measurements from operating reactors and critical experiments. The report details the methodology used to calculate uncertainties for reload core designs with LEU fuel containing gadolinia.

Based on the technical evaluation above, the NRC staff finds the proposed changes to the TSs and UFSAR acceptable. The NRC staff's approval is based on the range of fuel configurations and core design parameters as stated and referenced in the licensee's June 10, 2009, application, as supplemented by letters dated December 18, 2009 and August 25, 2010.

Introduction of significantly different or new fuel designs will require further validation of the above stated physics methods for application to Oconee 1/2/3 by the licensee, and will require approval by the NRC staff

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components 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 published in the *Federal Register* on March 19, 2010, (75 FR 13314). The amendments also relate to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusions set forth in 10 CFR 51.22(c)(9) and (c)(10). 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.

7.0 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.

8.0 REFERENCES

1. Catawba Nuclear Station, Units 1 and 2, Issuance of License Amendments Regarding Revision 1 to DPC-NE-1005-P, "Nuclear Design Methodology Using CASMO-4/SIMULATE-3," MOX, dated November 12, 2008 (ADAMS Accession No. ML082820047).

2. McGuire Nuclear Station, Units 1 and 2, Issuance of License Amendments Regarding Revision 1 to DPC-NE-1005-P, "Nuclear Design Methodology Using CASMO-4/SIMULATE-3," MOX, dated November 12, 2008 (ADAMS Accession No. ML082820015).

3. M. G. Natrella, "Experimental Statistics," National Bureau of Standards Handbook 91, October 1966, http://www.itl.nist.gov/div898/handbook/.

4. "Assessment of Assumption of Normality (Employing Individual Observed Values)," ANSI-N15.15-1974, October 1973.

5. "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification", U.S. Nuclear Regulatory Commission, Regulatory Guide 1.126, Revision 1, March 1978 (ADAMS Accession No. ML003739385).

6.. "Urania Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," DOE/ET/34212-41, BAW-1810, April 1984.

7. DPC-NE-1005-P-A "Nuclear Design Methodology Using CASMO-4/SIMULATE-3," MOX, Revision 0, SER, dated August 20, 2004 (ADAMS Accession No. ML042370178).

Principal Contributor: A. Attard

Date: August 2, 2011

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. If you have any questions, please call me at 301-415-1345.

Sincerely,

/RA/

John Stang, Senior 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. 377 to DPR-38
2. Amendment No. 379 to DPR-47
3. Amendment No. 378 to DPR-55
4. Safety Evaluation

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