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

October 29, 2008

Mr. Dave Baxter
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 USE OF AREVA NP MARK-B-HTP FUEL (TAC NOS. MD7050, MD7051, AND MD7052)

Dear Mr. Baxter:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 362, 364, and 363 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 22, 2007, supplemented July 14, September 17, and October 27, 2008.

These amendments revise TSs to allow the accommodation of AREVA NP Mark-B-HTP fuel.

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

Sincerely,

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. 362 to DPR-38
2. Amendment No. 364 to DPR-47
3. Amendment No. 363 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

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. 362
Renewed License No. DPR-38

1. The U.S. 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 October 22, 2007, and supplemented July 14, September 17, and October 27, 2008, 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.362, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John Stang For

Melanie C. Wong, 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
and the Technical Specifications

Date of Issuance: October 29, 2008

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. 364
Renewed License No. DPR-47

1. The U.S. 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 October 22, 2007, and supplemented July 14, September 17, and October 27, 2008, 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. 364, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

For

Melanie C. Wong, 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 and the Technical Specifications

Date of Issuance: October 29, 2008

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 363
Renewed License No. DPR-55

1. The U.S. 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 October 22, 2007, and supplemented July 14, September 17, and October 27, 2008, 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-55 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 363, 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 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

For

Melanie C. Wong, 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
and the Technical Specifications

Date of Issuance: October 29, 2008

ATTACHMENT TO LICENSE AMENDMENT NO. 362

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 364

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 363

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the 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
2.0-1	2.0-1
5.0-26	5.0-26
5.0-27	5.0-27
B 2.1.1-1	B 2.1.1-1
B 2.1.1-2	B 2.1.1-2
B 2.1.1-4	B 2.1.1-4
B 3.4.1-1	B 3.4.1-1

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. 362 , 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. 364 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. 363 , 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.

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(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

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 4642 - (5.8 x 10^{-3} x (Burnup, MWD/MTU))° F. Operation within this limit is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18 for the BWC correlation, 1.19 for the BWU correlation, and 1.132 for the BHTP correlation. Operation within these limits is ensured by compliance with the Axial Power Imbalance Protective Limits and RCS Variable Low Pressure Protective Limits as specified in the Core Operating Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

5.6 Reporting Requirements

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; and

(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 Analyses

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.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance

5.6 Reporting Requirements

Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.10, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2), the BWU (Ref. 4), and the BHTP (Ref. 5) CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The BWU correlation applies to the Mark-B11 fuel. The BHTP correlation applies to the MARK-B-HTP fuel. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC), 1.19 (BWU) and 1.132 (BHTP).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the

BASES

BACKGROUND (continued)

resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) and main steam relief valves (MSRVs) prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and anticipated transients. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints (Ref. 3), in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for RCS temperature, flow and pressure, and THERMAL POWER level that would result in a DNB ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip;

e. Reactor Coolant Pump to Power trip;

f. Flux/Flow Imbalance trip;

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

REFERENCES

1. UFSAR, Section 3.1.
2. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1995.
3. UFSAR, Chapter 15.
4. BAW-10199P, "The BWU Critical Heat Flux Correlations," Addendum 1, April 2000
5. BAW-10241(P)(A), Revision 1, BHTP DNB Correlation Applied with LYNXT, Framatome ANP, July 2005.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated transients assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the unit to approach the DNB limit.

The LCO for maximum RCS coolant loop average temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial loop average temperature in the analyses. A loop average temperature lower than that specified will produce a higher minimum DNBR. A loop average temperature higher than that specified will cause the unit to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the unit to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of ≥ 1.18 for BWC correlation, ≥ 1.19 for BWU correlation, ≥ 1.132 FOR BHTP correlation, or an equally valid limit when the statistical DNBR limit is employed (SCD methodology). This is the acceptance limit for the RCS DNBR parameters.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 362 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 364 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AND

AMENDMENT NO. 363 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 October 22, 2007, to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 072990298), as supplemented by letters dated July 14, 2008 (ADAMS Accession No. ML082000134), September 17, 2008 (ADAMS Accession No. ML082700552), and October 27, 2008 (ADAMS Accession No. ML083020297), Duke Energy Carolinas, LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the Oconee Nuclear Station, Units 1, 2, and 3. DPC-NE-2015-P, "Oconee Nuclear Station, Mark-B-HTP Fuel Transition Methodology," was provided as Attachment 3 to the October 22, 2007, application; a non-proprietary version of DPC-NE-2015-P is available under ADAMS Accession No. ML082690091.

The supplements dated July 14, September 17, and October 27, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff original proposed no significant hazards consideration determination as published in the *Federal Register* on November 20, 2007 (72 FR 65365).

The proposed changes would revise the TSs to accommodate use of AREVA NP Mark-B-HTP fuel at Oconee.

The licensee plans to transition to the Mark-B-HTP fuel assemblies from the current Mark-B11 fuel assemblies, both AREVA NP fuel designs, for the core reloads beginning in 2008. The Mark-B-HTP fuel design is an evolution of the standard Mark-B fuel product line. The Mark-B-HTP fuel assembly is a 15x15 array design with M5 fuel rods, instrument tube, and guide tubes. The Mark-B-HTP fuel is more resistant to grid-to-rod fretting and uses the AREVA NP BHTP critical heat flux (CHF) correlation. The M5 material was approved in the topical report BAW-10227P-A, entitled "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel." Introduction of the Mark-B-HTP fuel design requires revision to seven of the approved analytical methodology reports in the reload design process. The licensee consolidated all revisions to these previously approved reports into one reload report, DPC-NE-2015-P, which provides supporting information for the license amendment request. This report describes revisions to the methodologies for performing the nuclear design, mechanical design, thermal-hydraulic design, and the Chapter 15 non-loss-of-coolant-accident (LOCA) transient and accident analyses that are needed to use Mark-B-HTP fuel at Oconee. Some of the revisions are not associated with the change in fuel design, but are included for improvements, error corrections, and editorial clarification. The proposed license amendments will revise the TSs and associated Bases, which is necessary for the methodology revisions.

2.0 REGULATORY EVALUATION

The licensee requested license amendments to revise the TSs to transition to the Mark-B-HTP fuel for Oconee.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit" states that the holder of a license that desires to amend the license may file the application for an amendment with the NRC. 10 CFR 50.92, "Issuance of amendment," specifies that the NRC staff will be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate in determining whether an amendment will be issued to the applicant.

The following criteria from Appendix A of Part 50, "General Design Criteria for Nuclear Power Plants," apply: Criterion 10 – Reactor design, Criterion 11- Reactor inherent protection, and Criterion 28 – Reactor limits.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed DPC-NE-2015 in its entirety. The following technical evaluation addresses the revisions in DPC-NE-2015-P that are essential to the NRC staff's findings necessary to grant the requested license amendments.

3.1 Mechanical Design

3.1.1 Burnup Limit

The Mark-B-HTP fuel assembly to be used at the Oconee is an AREVA NP 15x15 fuel design with M5 cladding, instrument, and guide tubes. The intermediate and top spacer grids are also made of M5. The bottom spacer grid and upper and lower end fittings are made of Inconel 718. The M5 material was approved in the topical report, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," BAW-10227P-A, Revision 1, June 2003. The Mark-B-HTP fuel is an evolution of the standard Mark-B fuel product. The M5 material and Mark-B fuel design were approved to a rod average burnup limit of 62 GWd/MTU. The licensee indicated that fuel rod mechanical analyses were performed with the TACO3 fuel performance code. The TACO3 code was approved for the licensee's licensing applications in the methodology report, "Duke Power

Company Fuel Mechanical Reload Analysis Methodology Using TACO3," DPC-NE-2008P-A, Revision 0, April 1995, to a rod average burnup limit of 62 GWd/MTU.

Based on the approved reports, the NRC staff concludes that the Mark-B-HTP fuel design is approved to the rod average burnup limit of 62 GWd/MTU for Oconee.

3.1.2 Cladding Corrosion

Previously, the NRC staff approved AREVA NP high burnup applications in the topical report, "Extended Burnup Evaluation," BAW-10186P-A, Revision 2, June 2003. BAW-10186P-A, Revision 2 includes a best-estimate cladding corrosion model, COROS02, for zircaloy-4 fuel rods with a design limit of 100 microns in the corrosion analysis. The licensee will perform cladding corrosion analysis based on the AREVA NP methodology approved by the NRC, including the COROS02 corrosion model. However, the licensee had indicated that the COROS02 calculated results are reduced by certain amount to determine the best-estimate oxide thickness. Since the NRC staff had no knowledge of this provision in determining the best-estimate results, the NRC staff informed the licensee that the NRC staff did not approve such a reduction, which amounted to certain credit, in previous safety evaluations. Since the Mark-B-HTP fuel assembly uses the M5 cladding, the NRC staff recognizes that the approval of M5 in BAW-10227P-A does not encompass such a reduction for the best-estimate corrosion calculation. In addition, the amount of corrosion in the M5 cladding is generally much less than the amount in the zircaloy-4 cladding; a reduction could render unrealistically low corrosion for the M5 cladding. In fact, the NRC staff considers that the corrosion model in BAW-10227P-A is a best-estimate model. The best-estimate results are directly calculated from the cladding corrosion model without any reduction. Therefore, the NRC staff informed the licensee that the use of the reduction in determining the best-estimate results in the cladding corrosion model was not acceptable. Therefore, by letter dated October 27, 2008, the licensee stated that it will take no reduction in the COROS02 corrosion model for the calculated oxide thickness.

The NRC staff concludes that the approved model and the design limit of 100 microns is acceptable for analyzing cladding corrosion for the Mark-B-HTP fuel design for Oconee.

3.1.3 LOCA and Seismic Loading

Earthquake and postulated pipe breaks in the reactor coolant system would result in extreme forces on the fuel assembly. Section 4.2 of Appendix A to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," states that the fuel system coolable geometry shall be maintained and damage should not be so severe as to prevent control rod insertion during seismic and LOCA events.

In its letter dated September 17, 2008, the licensee analyzed the worst case loading on fuel assemblies due to LOCA and seismic events for the beginning of life (BOL) and end of life (EOL). The loads include core flood and decay heat LOCA loads and safe-shutdown earthquake seismic loads. Mixed and all Mark-BHTP fuel assemblies were evaluated. The licensee uses the square-root-of-sum-of-squares (SRSS) method to combine the two loads. The results show that the maximum impact load on fuel assemblies remains below the grid crushing load for the worst case core configuration. Thus, the fuel rod fragmentation does not occur and fuel coolability is maintained for the LOCA and seismic events. The NRC staff reviewed the results and concludes that the LOCA and seismic analyses are acceptable.

The NRC staff concludes that, based on these analyses, the mechanical design of the Mark-B-HTP fuel for Oconee is acceptable.

3.2 Nuclear and Reload Design

Three approved methodology reports; "Oconee Nuclear Station Reload Design Methodology," NFS-1001A, Revision 5, January 2001; "Oconee Nuclear Station Reload Design Methodology II," DPC-NE-1002-A, Revision 2, October 1985; and "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," DPC-NE-1004-A, Revision 1, December 1997; form the basis of nuclear and reload design for Oconee.

The revisions in the nuclear and reload design include the effect of fuel assembly bow, nuclear uncertainty factors, and the fuel densification power spike factor. The effect of fuel assembly bow on the pin power distribution is accounted for by a penalty factor in the analyses of fuel melting, clad strain, departure from nucleate boiling ratio (DNBR) transient, and LOCA. In its letter dated September 17, 2008, the licensee demonstrated that the fuel assembly bow peaking factor was statistically combined with other factors to form a single uncertainty factor using the SRSS method. The NRC staff reviewed the response and concludes that the analysis of assembly bow on the pin power distribution is acceptable.

The nuclear uncertainty factors were revised using the CASMO-3/SIMULATE-3P code as described in the approved report DPC-NE-1004-A. The fuel densification power spike factor was revised with an axially-dependent factor based on the approved report NFS-1001A. Based on the approved methodology reports, the NRC staff reviewed the nuclear uncertainty and densification factors and concludes that the analyses using these factors are acceptable for Oconee.

The NRC staff concludes that the revisions to the approved methodology reports and the analyses performed by the licensee are acceptable for the nuclear and reload design of Mark-B-HTP fuel at Oconee.

3.3 Thermal-Hydraulic Design

Two approved methodology reports; "Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2003P-A, Revision 1, September 2000, and "Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005-PA, Revision 3, September 2002; form the basis of thermal-hydraulic design for Oconee.

The licensee used the VIPRE-01 code for steady-state core thermal-hydraulic analyses. The licensee modeled the Mark-B-HTP fuel using the methodology described in DPC-NE-2003P-A for the analysis. Two CHF correlations, BHTP and BWU-N, were used. The BHTP correlation, which was approved in "BHTP DNB Correlation Applied with LYNXT," BAW-10241(P)(A), Revision 1, September 2004, was used for the fuel above the first intermediate grid spacer. The BWU-N correlation, which was approved in "The BWU Critical Heat Flux Correlations," BAW-10199P-A, August 1996 (and including Addendum, December 2000), was used for the fuel below the first intermediate grid spacer. Based on the BHTP correlation, the licensee determined the DNBR safety limit of 1.132 for Modes 1 and 2. Based on the approved correlation, the NRC staff concludes that the DNBR limit is acceptable for Oconee.

A new Appendix F, "Application of BHTP CHF Correlation to the Mark-B-HTP Fuel Design," is added to DPC-NE-2005-PA. Appendix F describes a methodology of DNB statistical design limits, parameters and uncertainties, and transition cores using VIPRE-01 with the BHTP correlation for the Mark-B-HTP fuel design. Appendix F is essentially similar to the approved Appendix D in DPC-NE-2005-PA for the current Mark-B11 fuel design. The licensee will continue to conform to the Limitations and Conditions described in the safety evaluation of BAW-10241(P)(A), Revision 1 for the use of BHTP CHF correlation. The NRC staff concludes that Appendix F is acceptable to incorporate into the approved methodology report DPC-NE-2005-PA.

For mixed core of the Mark-B-HTP and Mark-B11 fuel assemblies, the licensee analyzed the transition core penalty using the approved VIPRE-01 code. The licensee also analyzed the statistical DNB results using the approved methodology in DPC-NE-2005-PA. Based on the licensee's analysis using approved methodologies with appropriate revisions, the NRC staff concludes that the mixed core analysis is acceptable for Oconee.

The NRC staff concludes that the revisions to the approved methodology reports are acceptable for analyzing the thermal-hydraulic design of Mark-B-HTP fuel at Oconee.

3.4 Non-LOCA Transient and Accident Analyses

Two approved methodology reports; "Thermal-Hydraulic Transient Analysis Methodology," DPC-NE-3000-PA, Revision 3, September 2004, and "UFSAR Chapter 15 Transient Analysis Methodology," DPC-NE-3005-PA, Revision 2, May 2005; form the basis for non-LOCA transient and accident analyses. DPC-NE-3000-PA includes two codes, RETRAN-3D and VIPRE-01. DPC-NE-3005-PA encompasses three different codes, RETRAN-3D, CASMO-3/SIMULATE-3 and SIMULATE-3K. DPC-NE-3005-PA was revised to provide initial conditions and boundary conditions for Mark-B-HTP fuel in the Updated Final Safety Analysis Chapter 15 analysis.

A new Appendix D, "Methodology Revisions for Mark-B-HTP Fuel," is added to DPC-NE-3000-PA. Appendix D provides a description of design parameters in developing the RETRAN-3D and the VIPRE-01 models for the Mark-B-HTP fuel design. The BHTP and BWU CHF correlations are used, as indicated in Section 3.3, for most of the DNBR analyses. For the main steam line break analysis, the approved Modified-Barnett correlation, as described in "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," EMF-2310(P)(A), Revision 1, May 2004, was used in the low pressure regime. The NRC staff concludes that Appendix D is acceptable to incorporate into approved methodology report DPC-NE-3000-PA.

The licensee included two new features in the Appendix E, "Expanded Oconee VIPRE-01 Methodology," to DPC-NE-3000-PA. The first feature is a larger nodalization for VIPRE that enables modeling of most of the hot assembly and parts of three adjacent fuel assemblies. The second feature is a revised core power distribution. The current core power distribution is very conservative based on a cosine power distribution provided by the fuel vendor. The licensee developed a revised core power distribution using the approved SIMULATE-3 model, which reflects the current reload core design of a flattened power distribution. The licensee reasons that a flattened power distribution will cause less cross flow in the sub-channels and increase the number of limiting sub-channels. Thus, a flattened power distribution is considered conservative for the DNB analysis. The revised core power distribution will be used in the same manner as the current vendor-supplied power distribution. The licensee will perform analyses to confirm that the revised core power distribution remains conservative for future reload cores, or a new revised core power distribution will be developed using the same process. Based on the adequate conservatism in the approved methodology report DPC-NE-3000-PA, the NRC staff concludes that the revised Appendix E is acceptable to incorporate into DPC-NE-3000-PA.

The licensee developed two approaches of the mixed core analysis to account for DNBR penalty. The first approach modeled the hot Mark-B-HTP fuel assembly surrounded by a lumped channel representing many co-resident Mark-B11 fuel assemblies. This approach maximizes the flow diversion out of the hot assembly, resulting in a very conservative mixed core penalty as described in the thermal-hydraulic design. The second approach explicitly modeled the actual mixed core loading of the Mark-B-HTP and Mark-B11 fuel assemblies. The second approach depicts realistically the flow diversion out of the hot assembly resulting in a conservative, though less conservative than the first approach, mixed core penalty. In its September 17, 2008, the licensee indicated that the second approach will be adopted for non-LOCA transient and accident analyses, because the first approach is too restrictive for the analysis. Based on the adequate conservatism in the approved methodology reports DPC-NE-3000-PA and DPC-NE-3005-PA, the NRC staff approves the second approach for the mixed core analysis.

Based on these approved methodology reports, the NRC staff concludes that the analysis methodology of non-LOCA transients and accidents is acceptable for Mark-B-HTP fuel at Oconee.

3.5 LOCA Analysis

The licensee will perform the LOCA analysis using the approved LOCA Evaluation Model (EM), as described in "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," BAW-10192P-A, June 1998. The approved "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," BAW-10164P-A, Revision 6, June 2007, describes the RELAP5 code that is used for simulating the LOCA conditions. The NRC staff has approved Revision 6 to BAW-10164P-A for analyzing the Mark-B-HTP fuel design.

The NRC staff concludes that the approved topical reports BAW-10192P-A and BAW-10164P-A are acceptable for LOCA analysis of Mark-B-HTP fuel at Oconee.

3.6 Technical Specification (TS) Revisions

3.6.1 Section 2.1.1.2, Reactor Core Safety Limits

The licensee will add the BHTP CHF correlation in TS Section 2.1.1.2 for the Mark-B-HTP fuel design. TS 2.1.1.2 will be revised as follows:

"In MODES 1 and 2, ...1.19 for the BWU correlation, and 1.132 for the BHTP correlation. Operation within these limits..."

Based on the preceding technical evaluation, the NRC staff finds this revision acceptable.

3.6.2 Section 5.6.5.b, Core Operating Limits Report (COLR)

The proposed revision to the COLR will add the BHTP CHF correlation, which is applicable to the Mark-B-HTP fuel design, to the RELAP5 code described in the approved topical report BAW-10164-PA. Based on the preceding technical evaluation, this revision is acceptable.

3.6.3 Bases, Section B 2.1.1, Reactor Core SLs

The licensee will revise TS Bases, Section B 2.1.1 to include the BHTP CHF correlation for the Mark-B-HTP fuel design and to add BAW-10241(P)(A). Based on the preceding technical evaluation, the revision is acceptable.

3.6.4 Bases, Section B 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

The licensee will add the BHTP CHF correlation for the Mark-B-HTP fuel design to the TS Bases, Section B 3.4.1. Based on the preceding technical evaluation, the revision is acceptable.

4.0 SUMMARY

In summary, the NRC staff has reviewed the licensee's license amendment request for TS revisions and concludes that the TS revisions are acceptable. The NRC staff has also reviewed DPC-NE-2015-P; which discusses changes to the following previously approved methodology reports: NFS-1001A, DPC-NE-1002-A, DPC-NE-2008P-A, DPC-NE-2003P-A, DPC-NE-2005P-A, DPC-NE-3000-PA, and DPC-NE-3005-PA. Based on the NRC staff's review of DPC-NE-2015-P, including the preceding technical evaluation, the NRC staff approves the methodology revisions in DPC-NE-2015-P for use of Mark-B-HTP fuel at Oconee.

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 (72 FR 65365, November 20, 2007). 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.

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.

Principal Contributor: Shih-Liang Wu, NRR/SNPB

Date: October 29, 2008

October 29, 2008

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

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, ISSUANCE OF AMENDMENTS REGARDING USE OF AREVA NP MARK-B-HTP FUEL (TAC NOS. MD7050, MD7051, AND MD7052)

Dear Mr. Baxter:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 362, 364, and 363 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 22, 2007, supplemented July 14, September 17, and October 27, 2008.

These amendments revise TSs to allow the accommodation of AREVA NP Mark-B-HTP fuel.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next 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. 362 to DPR-38
2. Amendment No. 364 to DPR-47
3. Amendment No. 363 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

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