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

December 23, 2010

Christopher L. Burton, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – ISSUANCE OF AMENDMENT TO ALLOW THE USE OF THERMAL HYDRAULIC ANALYSIS CODE S-RELAP5 FOR NON-LOSS-OF-COOLANT ACCIDENT TRANSIENTS (TAC NO. ME1735)

Dear Mr. Burton:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 135 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant (HNP), Unit 1, in response to your application dated July 21, 2009, as supplemented by letters dated March 3, and July 28, 2010.

The amendment revises Technical Specification Section 6.9.1.6 to add NRC-approved Topical Report (TR) EMF-2310(P)(A), "SRP [Standard Review Plan] Chapter 15 Non-LOCA Methodology for Pressurized-Water Reactors," to the Core Operating Limits Report methodologies list.

This change will allow the use of thermal-hydraulic analysis code S-RELAP5 for Final Safety Analysis Report (FSAR) Chapter 15 non-loss-of-coolant accident (non-LOCA) transients in the HNP safety analyses. TR EMF-2310(P)(A), Revision 0, was approved by the NRC on May 11, 2001, for the application of the S-RELAP5 thermal-hydraulic analysis computer code to FSAR Chapter 15 non-LOCA transients. EMF-2310(P)(A), Revision 1, approved by the NRC on May 19, 2004, updated Section 5.6 of the TR.

A copy of the related NRC staff safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's regular biweekly *Federal Register* notice.

Sincerely,

Brenda Mozafari

Brenda Mozafari, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No.135 to NPF-63
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 135
Renewed License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated July 21, 2009, as supplemented by letters dated March 3, 2010, and July 28, 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 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;

   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, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 135, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-63
and the Technical Specifications

Date of Issuance: December 23, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 135

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace Page 4 of Renewed Operating License No. NPF-63 with the attached Page 4.

Replace the following page of Appendix A, "Technical Specifications," to Renewed Facility Operating License No. NPF-63 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove Page	Insert Page
6-24b	6-24b
6-24c	6-24c
6-24d	6-24d

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and 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.

(1) Maximum Power Level

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2900 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 135, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at §15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

[1]The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

h. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.

   (Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).

i. EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

j. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

   (Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

k. XN-NF-82-49(P)(A), "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," approved version as specified in the COLR.

   (Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

l. EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," approved version as specified in the COLR.

   (Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

m. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.

   (Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

n. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", approved version as specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

o. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.1. 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,

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT (Continued)

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, and

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10CFR50.4 within the time period specified for each report.

6.10 DELETED

(PAGE 6-25 DELETED By Amendment No.92)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 135 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By application dated July 21, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML092150228), as supplemented by letters dated March 3, 2010 (ADAMS Accession No. ML100700076), and July 28, 2010 (ADAMS Accession No. ML102160371), Carolina Power & Light Company (the licensee), now doing business as Progress Energy Carolinas, Inc., submitted a proposed amendment for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP).

The proposed amendment would revise Technical Specification (TS) Section 6.9.1.6 to add U.S. Nuclear Regulatory Commission (NRC) approved Topical Report (TR) EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA [Loss-of-Coolant Accident] Methodology for Pressurized-Water Reactors," to the Core Operating Limits Report (COLR) methodologies list. HNP TS 6.9.1.6.2 requires that the analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and that the approved revision number shall be identified in the COLR.

Currently, HNP TS 6.9.1.6.2.b authorizes the use of Advanced Nuclear Fuels (ANF) Corporation guidance ANF-89-151(P) (A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," to determine the core operating limits. The change will allow the use of thermal-hydraulic analysis code S-RELAP5 for Final Safety Analysis Report (FSAR) Chapter 15 non-loss-of-coolant accident (non-LOCA) transients in the HNP safety analyses. TR EMF-2310(P)(A), Revision 0, was approved by the NRC on May 11, 2001, for the application of the S-RELAP5 thermal-hydraulic analysis computer code to FSAR Chapter 15 non-LOCA transients. EMF-2310(P)(A), Revision 1, approved by the NRC on May 19, 2004, updated Section 5.6 of the TR.

The supplements dated March 3, and July 28, 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 November 10, 2009 (74 FR 58060).

2.0 REGULATORY EVALUATION

Topical Report EMF-2310(P)(A) pertains to non-LOCA accident and transient analyses that are part of the HNP licensing basis. The regulatory bases for these analyses are found in the General Design Criteria (GDC) of Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. The GDCs that pertain to each of the analyses are listed in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition," which is also referred to as simply the Standard Review Plan (SRP).

Evaluation models of LOCA events are defined in 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." This definition, which can also be applied to non-LOCA analyses, states that:

> An evaluation model is the calculational framework for evaluating the behavior of the reactor system during a postulated loss-of-coolant accident (LOCA). It includes one or more computer programs and all other information necessary for application of the calculational framework to a specific LOCA, such as mathematical models used, assumptions included in the programs, procedure for treating the program input and output information, specification of those portions of analysis not included in computer programs, values of parameters, and all other information necessary to specify the calculational procedure.

Section II of Appendix K, "Emergency Core Cooling System (ECCS) Evaluation Models," to 10 CFR Part 50, which is also written for LOCA analyses, but considered to be applicable to non-LOCA analyses, contains the documentation requirements for evaluation models. It states:

1. a. A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

   b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

5. General Standards for Acceptability — Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by § 50.46(a)(1)(ii), compliance with required features of section I of this appendix K; and, for models covered by § 50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of § 50.46(b) would not be exceeded.

Section III of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50 governs references to design control measures in the COLR. It states that "design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests."

The NRC staff's technical review consisted of comparing the licensee's proposed changes against the applicable regulatory criteria cited above, as well as reviewing the information provided by the licensee as it relates to the methodology for concluding that existing regulatory criteria will continue to be satisfied after the proposed changes are implemented.

3.0 <u>TECHNICAL EVALUATION</u>

The proposed change will revise HNP TS 6.9.1.6, "Core Operating Limits Report," to add the NRC-approved TR EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," as a COLR reference in HNP TS 6.9.1.6.2, which contains a listing of the analytical methods used to determine the core operating limits.

TR EMF-2310(P)(A), Revision 1, will be added to the HNP TS as 6.9.1.6.2.n. The current TS 6.9.1.6.2.n, "Mechanical Design Methodologies," will be re-designated as TS 6.9.1.6.2.o in order to retain the current grouping of the reference reports.

The TR EMF-2310(P)(A) methodology applies the NRC-accepted S-RELAP5 thermal-hydraulic analysis computer code to the analysis of Updated FSAR Chapter 15 non-LOCA transients. S-RELAP5 is an updated version of the NRC-accepted ANF-RELAP thermal-hydraulic code.

By letters dated May 11, 2001 [Reference 1], November 13, 2003 [Reference 2], and May 19, 2004 [Reference 3], the NRC staff accepted the use of this methodology for conducting analyses of certain non-LOCA events in pressurized-water reactors of the Combustion Engineering (2x4 plants) and Westinghouse (3- and 4-loop plants) designs.

The NRC staff's acceptance also noted that a generic description of S-RELAP5 cannot provide a detailed justification for all plant applications. Therefore, each applicant must also provide justification for its specific application of the S-RELAP5 code, which is expected to include, at a minimum: (1) the nodalization, (2) defense of the chosen parameters, (3) any needed sensitivity studies, (4) justification of the conservative nature of the input parameters, and (5) the calculated results. Accordingly, the licensee has provided this information as it relates to a Loss of Forced Reactor Coolant Flow (LOCF) analysis at HNP [Reference 4].

The LOCF event is among the non-LOCA events that may be analyzed using the methodology contained in TR EMF-2310(P)(A). The specific events for which this methodology has been accepted are:

<u>Increase in Heat Removal by the Secondary System</u>

Increase in Feedwater Flow

Increase in Steam Flow

Inadvertent Opening of Steam Generator Relief/Safety Valve

Steam System Piping Failures Inside and Outside Containment

<u>Decrease in Heat Removal by Secondary System</u>

Loss of Outside External Load

Turbine Trip

Loss of Condenser Vacuum

Closure of Main Steam Isolation Valve

Steam Pressure Regulator Failure

Loss of Non-Emergency AC Power to the Station Auxiliaries

Loss of Normal Feedwater Flow

Feedwater System Piping Breaks Inside and Outside Containment

<u>Decrease in Reactor Coolant Flow Rate</u>

Loss of Forced Reactor Coolant Flow

Flow Controller Malfunctions

Reactor Coolant Pump Rotor Seizure

Reactor Coolant Pump Shaft Break

<u>Reactivity and Power Distribution Anomalies</u>

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low Power Startup Condition

Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

Rod Cluster Control Assembly Misoperation

Dropped Rod/Bank

Single Rod Withdrawal

Statically Misaligned Rod Cluster Control Assembly

Startup of an Inactive Loop at an Incorrect Temperature

Chemical and Volume Control System (CVCS) Malfunction that Results in a Decrease of Boron Concentration

Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

Spectrum of Rod Ejection Accidents

<u>Increase in Reactor Coolant Inventory</u>

Inadvertent Operation of the Emergency Core Cooling System that Increases Reactor Coolant Inventory

CVCS Malfunction that Increases Reactor Coolant Inventory

<u>Decreases in Reactor Coolant Inventory</u>

Inadvertent Opening of a Pressurizer Pressure Relief Valve

Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment (for primary and secondary mass and energy release calculations)

Radiological Consequences of Steam Generator Tube Rupture (for primary and secondary mass and energy release calculations)

HNP is a Westinghouse 3-loop plant, and the core is composed solely of AREVA NP High Temperature Performance fuel assemblies. Accordingly, mixed core considerations do not apply. The licensee's analysis of a LOCF event for HNP includes a discussion of input assumptions, which notes, in part, the conservative direction of errors or uncertainties within the input values. There are also nodalization diagrams of the S-RELAP5 models used for the analysis. The nodalization is comparable in detail to nodalizations that are used for other codes that have been approved by the NRC staff for analysis of licensing basis non-LOCA events.

The reactor coolant system (RCS) is modeled by multi-node representations for the reactor vessel, which is comprised of an active core region, inlet and outlet plena, a downcomer and baffel region, and an upper head. Three reactor coolant loops are modeled, connected to three steam generators and a pressurizer. The steam generator models contain inlet and outlet plena and multi-node U-tubes for the primary side, and there are multinode downcomers, U-tube boiling regions, separators, and steam domes in the secondary side. Steam lines, steam safety valves, and steam line isolation valves are also represented.

The results of the LOCF analyses for HNP are typical of those for a 3-loop plant of Westinghouse design. The analysis was performed with assumptions and models that are designed to minimize the departure from nucleate boiling ratio (DNBR). For example, operation of the pressurizer power-operated relief valves (PORVs) was modeled, since that would tend to keep the core pressure low, which would reduce the core thermal margin, as represented by the DNBR. The results indicate that the PORVs open and relieve steam (i.e., the pressurizer does not become water-solid), and the transient DNBR does not fall below the safety limit DNBR. Thus, it can be concluded that there would be no fuel clad damage, and the event would not develop into a more severe event.

Because the core power does not increase appreciably during the LOCF event, the challenge to the fuel centerline melt Specified Acceptable Fuel Design Limit (SAFDL) is not limiting and margin exists to the DNB SAFDL. Additionally, because system temperatures and pressures increase less significantly for a LOCF event when compared to complete loss of load type events, the pressurization transient does not present a severe challenge to the maximum pressure criterion, and hence is bounded by a more challenging event. Based on these results, the acceptance criteria for this event are satisfied by the use of the S-RELAP5 model.

The results of the licensee's LOCF analysis using S-RELAP5 indicate that (1) the applicable acceptance criteria used in the licensing basis for HNP are satisfied, and (2) the transient trends and values of key parameters (e.g., RCS flow rate, core power level, and RCS temperatures) are consistent with those produced by other NRC-approved codes that are used to analyze licensing basis Updated FSAR Chapter 15 events.

Based on the NRC staff's review of the licensee's LOCF analysis, which was submitted in accordance with the requirement expressed in the acceptance correspondence for this non-LOCA analysis methodology, the staff accepts that the methodology contained in TR EMF-2310(P)(A) may be added to the list of methodologies in the HNP COLR.

Furthermore, by letter dated July 28, 2010, the licensee made the following commitment, as it applies to this LAR:

Commitment	Completion Date
After implementation of the HNP TS Amendment approving use of the thermal-hydraulic analysis methodology EMF-2310 (S-RELAP5) for Chapter 15 non-loss-of-coolant accident (LOCA) transients, EMF-2310 will be used for new replacement safety analysis currently performed using ANF-89-151 methodology. If it becomes necessary to correct an error in the current ANF-89-151 analyses, ANF-89-151 methodology will be used for the error correction.	Upon implementation of the approved TS Amendment (60-day implementation period requested). This commitment will remain in effect until all ANF-89-151 analyses are replaced with EMF-2310 analyses.

The NRC staff considers the above licensee commitment acceptable because it ensures that the new (EMF-2310) methodology will be utilized during all subsequent thermal-hydraulic analyses involving non-LOCA scenarios, while errors in the current analyses will be addressed in a "like for like" manner using the old (ANF-89-151) methodology until those analyses can be superseded. In approving the proposed change, the NRC staff relied upon this commitment, and it is expected that the commitment will be completed in the manner stated by the licensee.

As a result of the NRC staff's review of the licensee's plant-specific analysis, which was submitted in accordance with the limitation and condition contained in the acceptance of the non-LOCA analysis methodology, the staff concludes that a reference to TR EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," may be added to the COLR methodologies list of HNP TS 6.9.1.6.2 as TS 6.9.1.6.2.n. This addition, as well as the subsequent renumbering of the remainder of TS 6.9.1.6.2, is acceptable as this is an administrative TS change necessary to reflect the current non-LOCA analysis methodology.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment on August 19, 2010. The North Carolina State official had no comments regarding the proposed amendment.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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, which was published in the *Federal Register* on November 10, 2009 (74 FR 58060). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Letter from the U.S. Nuclear Regulatory Commission to Framatome ANP, dated May 11, 2001, "Acceptance for Referencing of Licensing Topical Report EMF-2310(P), Revision 0, 'SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,' (TAC NO. MA7192)" (ADAMS Accession No. ML011310533).

2. Letter from the U.S. Nuclear Regulatory Commission to Framatome ANP, dated November 13, 2003, "Clarification of Safety Evaluation for EMF-2310(P)(A), 'SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,' (TAC NO. MB7338)" (ADAMS Accession No. ML033140185).

3. Letter from the U.S. Nuclear Regulatory Commission to Framatome ANP, dated May 19, 2004, "Final Safety Evaluation for Topical Report EMF-2310(P), Revision 1, 'SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors,' (TAC NO. MC0329)" (ADAMS Accession No. ML041400499).

4. ANP-2693(P), Revision 0 (Proprietary), "Loss of Forced Reactor Coolant Flow Analysis for Harris Nuclear Plant, Unit 1," AREVA Report, May 2009.

Principal Contributor: Muhammad Razzaque

Date: December 23, 2010

December 23, 2010

Christopher L. Burton, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – ISSUANCE OF AMENDMENT TO ALLOW THE USE OF THERMAL HYDRAULIC ANALYSIS CODE S-RELAP5 FOR NON LOSS OF COOLANT ACCIDENT TRANSIENTS (TAC NO. ME1735)

Dear Mr. Burton:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 135 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant (HNP), Unit 1, in response to your application dated July 21, 2009, as supplemented by letters dated March 3, and July 28, 2010.

The amendment revises Technical Specification Section 6.9.1.6 to add NRC-approved Topical Report (TR) EMF-2310(P)(A), "SRP [Standard Review Plan] Chapter 15 Non-LOCA Methodology for Pressurized-Water Reactors," to the Core Operating Limits Report methodologies list.

This change will allow the use of thermal-hydraulic analysis code S-RELAP5 for Final Safety Analysis Report (FSAR) Chapter 15 non-loss-of-coolant accident (non-LOCA) transients in the HNP safety analyses. TR EMF-2310(P)(A), Revision 0, was approved by the NRC on May 11, 2001, for the application of the S-RELAP5 thermal-hydraulic analysis computer code to FSAR Chapter 15 non-LOCA transients. EMF-2310(P)(A), Revision 1, approved by the NRC on May 19, 2004, updated Section 5.6 of the TR.

A copy of the related NRC staff safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's regular biweekly *Federal Register* notice.

Sincerely,
/RA/
Brenda Mozafari, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No.135 to NPF-63
2. Safety Evaluation

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