March 1, 1994

Docket No. 50-400

Mr. W. R. Robinson
Vice President - Harris Plant
Nuclear Generation
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

Dear Mr. Robinson:

SUBJECT: ISSUANCE OF AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING CYCLE 6 FUEL TRANSITION FOR SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 (TAC NO. M86223)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 44 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP). This amendment changes the Technical Specifications (TS) in response to your request dated July 16, 1993, as supplemented February 17, 1994.

The amendment revises the SHNPP TS 3/4.2.1, Axial Flux Difference, and 3/4.2.2, Heat Flux Hot Channel Factor (F_Q), deletes surveillance requirements 4.2.2.3 and 4.2.2.4, and changes 6.9.1.6, Core Operating Limit Report, and associated Bases related to the transition from nuclear fuel supplied by Westinghouse to nuclear fuel supplied by Siemens Power Corporation (SPC) beginning with Cycle No. 6.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's regular bi-weekly Federal Register notice.

Sincerely,

Original Signed by:
Ngoc B. Le, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 44 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

*See Previous Concurrence

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

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Office of Nuclear Reactor Regulation

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1. Amendment No. 44 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

Mr. H. Ray Starling
Manager - Legal Department
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Resident Inspector/Harris NPS
c/o U.S. Nuclear Regulatory Commission
Route 1, Box 315B
New Hill, North Carolina 27562

Karen E. Long
Assistant Attorney General
State of North Carolina
Post Office Box 629
Raleigh, North Carolina 27602

Public Service Commission
State of South Carolina
Post Office Drawer 11649
Columbia, South Carolina 29211

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., N.W. Suite 2900
Atlanta, Georgia 30323

Mr. W. R. Robinson
Vice President - Harris Plant
Carolina Power & Light Company
Shearon Harris Nuclear Power Plant
Post Office Box 165, MC: Zone 1
New Hill, North Carolina 27562-0165

Mr. Dayne H. Brown, Director
Division of Radiation Protection
N.C. Department of Environmental
Commerce & Natural Resources
Post Office Box 27687
Raleigh, North Carolina 27611-7687

Mr. H. W. Habermeyer, Jr.
Vice President
Nuclear Services Department
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Admiral Kinnaird R. McKee
214 South Morris Street
Oxford, Maryland 21654

Mr. Robert D. Martin
3382 Sean Way
Lawrenceville, Georgia 30244

AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

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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 FACILITY OPERATING LICENSE

Amendment No. 44
License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated July 16, 1993, as supplemented February 17, 1994, 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 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; and paragraph 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

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(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. 44 , 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

S. Singh Bajwa, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 1, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 44

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-4	3/4 2-4
3/4 2-6	3/4 2-6
3/4 2-7a	3/4 2-7a
3/4 2-7b	3/4 2-7b
3/4 2-7c	DELETED
3/4 2-7d	DELETED
3/4 10-2	3/4 10-2
B3/4 2-1	B3/4 2-1
B3/4 2-2	B3/4 2-2
B3/4 2-4	B3/4 2-4
6-24	6-24
6-24a	6-24a
6-24b	6-24b
----	6-24c

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a band about the target AFD as specified in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the limits specified in the COLR, either:

   1. Restore the indicated AFD to within the limits specified in the COLR within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

* See Special Test Exception 3.10.2

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 The target AFD of each OPERABLE excore channel shall be determined by excore measurement at least once per 31 Effective Full Power Days in conjunction with the requirements of Specification 4.2.2.2. The target AFD may be updated between measurements by adding the most recently measured value and the change in the predicted value since the measurement. The provisions of Specification 4.0.4 are not applicable.

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT, plant procedure PLP-106.

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 $F_Q(Z)$ shall be evaluated to determine if it is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

b. Increasing the measured $F_Q(Z)$ component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

c. Satisfying the following relationship:

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times V(Z)} \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{V(Z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and V(Z) is the function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(Z), and V(Z) are specified in the COLR.

d. Measuring $F_Q^M(Z)$ according to the following schedule:

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(Z)$ was last determined,* or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

e. With measurements indicating $F_{\Delta H}^{N}$ has increased since the previous determination of $F_Q^M(Z)$ either of the following actions shall be taken:

1) $F_Q^M(Z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c, or

2) $F_Q^M(Z)$ shall be measured and a target AFD reestablished at least once per 7 Effective Full Power Days until two successive maps indicate that $F_{\Delta H}^{N}$ is not increasing.

f. With the relationships specified in Specification 4.2.2.2c above not being satisfied:

1) Calculate the percent $F_Q(Z)$ exceeds its limit by the following expression:

$$\left\{ \left[\text{maximum} \left[\frac{F_Q^M(Z) \times V(Z)}{\frac{F_Q^{RTP}}{P} \times K(Z)} \right] \right] -1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \left[\text{maximum} \left[\frac{F_Q^M(Z) \times V(Z)}{\frac{F_Q^{RTP}}{0.5} \times K(Z)} \right] \right] -1 \right\} \times 100 \text{ for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the COLR by 1% AFD for each percent $F_Q(Z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_Q(Z)$ exceeding its limit by the percent calculated above.

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e, and 4.2.2.2f above are not applicable in the following core plane regions:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100%, inclusive.

4.2.2.3 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

(Pages 3/4 2-7c and 3/4 2-7d have been deleted)

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specification 4.2.2.2 and

b. Specification 4.2.3.2.

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to the design DNBR value during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power;

$F_{\Delta H}$ Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power, with an allowance to account for measurement uncertainty.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference (target AFD) is determined at equilibrium xenon conditions. The rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target AFD at RATED THERMAL POWER for the associated core burnup conditions. Target AFD for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic measurement of the target flux difference value is necessary to reflect core burnup considerations. The target AFD may be updated between measurements based on the change in the predicted value with burnup.

AXIAL FLUX DIFFERENCE (Continued)

The target band about the target AFD is specified in the COLR. The target band limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits.

The computer determines the one-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the acceptable AFD target band. These alarms are active when power is greater than 50% of RATED THERMAL POWER.

HEAT FLUX HOT CHANNEL FACTOR, AND RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The combination of the RCS flow requirement and the measurement of $F_{\Delta H}$ ensures that the calculated DNBR will not be below the design DNBR value. The relaxation of $F_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

When an $F_{\Delta H}$ measurement is taken, an allowance for measurement error must be applied prior to comparing to the $F_{\Delta H}^{RTP}$ limit(s) specified in the CORE OPERATING LIMITS REPORT (COLR). An allowance of 4% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System.

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. The margin is more than sufficient to offset any rod bow penalty and transition core penalty.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The hot channel factor $F_Q^M(Z)$ is measured periodically and increased by a cycle and height dependent power factor V(Z) to provide assurance that the limit on the hot channel factor, $F_Q(Z)$, is met. V(Z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. The V(Z) function is specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106, prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

b. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5,

c. Control Bank Insertion Limits for Specification 3/4.1.3.6,

d. Axial Flux Difference Limits for Specification 3/4.2.1,

e. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), and V(Z) for Specification 3/4.2.2,

f. Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

g. Boron Concentration for Specification 3/4.9.1.

6.9.1.6.2 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 the approved revision number shall be identified in the COLR.

a. XN-75-27(A), latest Revision and Supplements, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Richland WA 99352.

   (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, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

b. ANF-89-151(A), latest Revision, "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, Richland WA 99352.

   (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).

c. XN-NF-82-21(A), latest Revision, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Richland WA 99352.

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

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

d. XN-75-32(A), Supplements 1, 2, 3, and 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, Richland WA 99352.

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

e. XN-NF-84-93(A), latest Revision and Supplements, "Steamline Break Methodology for PWRs," Exxon Nuclear Company, Richland WA 99352.

(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, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

f. EXEM PWR Large Break LOCA Evaluation Model as defined by:

XN-NF-82-20(A), latest Revision and Supplements, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Richland WA 99352.

XN-NF-82-07(A), latest Revision, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Richland WA 99352.

XN-NF-81-58(A), latest Revision and Supplements, "RODEX2 Fuel Rod Thermal Response Evaluation Model," Exxon Nuclear Company, Richland WA 99352.

XN-NF-85-16(A), Volume 1 and Supplements, Volume 2, latest Revision and Supplements, "PWR 17x17 Fuel Cooling Test Program," Exxon Nuclear Company, Richland WA 99352.

XN-NF-85-105(A), and Supplements, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company, Richland WA 99352.

(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).

g. XN-NF-78-44(A), latest Revision, "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

h. ANF-88-054(A), latest Revision, "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland WA 99352.

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

i. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).

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

j. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the WESTINGHOUSE ECCS EVALUATION MODEL USING THE BASH CODE", March 1987 (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

k. WCAP-11837-P-A, "EXTENSION OF METHODOLOGY FOR CALCULATING TRANSITION CORE DNBR PENALTIES", January 1990 (W Proprietary).

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

l. EMF-92-081(A), latest Revision and Supplements, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Siemens Power Corporation, Richland WA 99352.

(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).

m. EMF-92-153(A), latest Revision and Supplements, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Nuclear Power Corporation, Richland WA 99352.

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

n. XN-NF-82-49(A), latest Revision and Supplements, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Exxon Nuclear Company, Richland WA 99352.

(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).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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.

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 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 44 TO 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 letter dated July 16, 1993, as supplemented February 17, 1994, the Carolina Power & Light Company (CP&L or the licensee) proposed changes to Technical Specifications (TS) TS 3/4.2.1, Axial Flux Difference, and 3/4.2.2, Heat Flux Hot Channel Factor (F_Q); deletes surveillance requirements 4.2.2.3 and 4.2.2.4; and changes to 6.9.1.6, Core Operating Limit Report, and the associated Bases. This amendment is related to the transition from nuclear fuel supplied by Westinghouse to nuclear fuel supplied by Siemens Power Corporation (SPC) beginning with Cycle No. 6. for the Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP). The proposed changes will incorporate references to the SPC methodologies that will support the transition. In the February 17, 1994 submittal, the licensee provided clarifying information within the scope of the original application and did not change the initial proposed no significant hazards consideration determination. This submittal was made at the request of the NRC staff.

2.0 EVALUATION

The SHNPP will begin transition from nuclear fuel supplied by Westinghouse Electric Corporation to nuclear fuel designed and fabricated by SPC. The SPC-designed fuel will be supported by SPC-designed methodology, while the Westinghouse fuel will continue to be supported by the original Westinghouse calculation.

The proposed changes to the current TS to support the transition from Westinghouse-supplied fuel to SPC-supplied fuel are addressed below.

2.1 Specifications 3/4.2.1, Axial Flux Difference, and 3/4.2.2, Heat Flux Hot Channel Factor (F_Q)

The specific proposed TS changes reflect the manner in which target axial flux difference (AFD) will be controlled and how the associated penalties to the measured F_Q are applied. The plant systems, including excore and incore hardware and general methods of operation (i.e., axial power distribution control) are not affected by the proposed changes; plant procedures (except that base load is no longer an option) and allowed action times for limiting conditions for operation are also unchanged.

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The proposed changes would replace one approved methodology (WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specification," June 1983) with another approved methodology (ANF-88-054(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." These methodologies are designed to provide the same function and control of axial flux difference.

The staff reviewed information provided in the licensee's submittal and related reference documents. The approved methodologies assure that the reactor control system will function consistent with safe plant operation. The relaxed axial offset control (RAOC) and the base load operation are not applicable under these approved methodologies and, consequently, are not required for SHNPP Cycle 6 operation. Therefore, the staff finds the proposed TS changes and associated Bases related to the transition to be acceptable.

2.2 Specification 6.9.1.6, Core Operating Limits Report (COLR)

This proposed change reflects the application of appropriate SPC generic methodology used to establish the limits provided in the COLR. This proposed change allows the incorporation of the three (staff-approved versions) SPC methodologies which are EMF-92-081, EMF-92-153, and XN-NF-82-49.

The staff reviewed information in the submittal and finds the proposed change conforms to the staff guidance contained in Generic Letter 88-16 in that plant operation will continue to be limited in accordance with the values of cycle specific parameters that are established using NRC-approved methodology. Therefore, the proposed change is acceptable. However, the XN-NF-82-49, Revision 1, Supplement 1, shall be reviewed and approved by the staff prior to the application to the SHNPP reload analysis.

Based on the above review, the staff concludes that the proposed TS changes are acceptable.

3.0 STATE CONSULTATION

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

4.0 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 and changes the Surveillance Requirements. 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 41500). Accordingly, the amendment meets 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 amendment.

5.0 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.

Principal Contributor: T. Huang

Date: March 1, 1994