September 16, 1994

Mr. C. S. Hinnant, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant,
Unit No. 2
Post Office Box 790
Hartsville, South Carolina 29551-0790

Dear Mr. Hinnant:

SUBJECT: ISSUANCE OF AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-23 REGARDING CORE OPERATING LIMITS REPORT METHODOLOGY - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 (TAC NO. M85989)

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

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 151 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. This amendment changes the Technical Specifications (TS) in response to your request dated March 3, 1993, as supplemented October 22, 1993, and August 12, 1994.

The amendment will (1) add ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," to the approved methodologies list in TS Section 6.9.3.3.b, (2) clarify the wording in Sections 3.10.2.1 and 3.10.2.2.2 by describing more precisely how measurement uncertainty and engineering factors are considered, (3) correct the licensee's typographical error in Section 3.10.2.2, and (4) correct a reference in TS 3.10.8.3 on page 3.10-16a of the TS basis.

Sincerely,

Original Signed by:
Brenda L. Mozafari, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 151 to DPR-23
2. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	LA:PD21:DRPE	PM:PD21:DRPE	AD:PD21:DRPE	OGC No legal objection
NAME	PAnderson	BMozafari:jrm	DMatthews	
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AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

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PD II-1 Reading
S. Varga
D. Matthews
P. Anderson
B. Mozafari
OGC

G. Hill (2)	T5-C3
C. Grimes	11-E-22
R. Jones	8-E-23
T. Huang	8-E-23

ACRS (10)
OPA
OC/LFDCB
B. Boger, R-II

cc: Robinson Service List

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

September 16, 1994

Docket No. 50-261

Mr. C. S. Hinnant, Vice President
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant,
Unit No. 2
Post Office Box 790
Hartsville, South Carolina 29551-0790

Dear Mr. Hinnant:

SUBJECT: ISSUANCE OF AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-23 REGARDING CORE OPERATING LIMITS REPORT METHODOLOGY - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 (TAC NO. M85989)

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Sincerely,

Brenda Mozafari

Brenda L. Mozafari, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 151 to DPR-23
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

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

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
H. B. Robinson Steam Electric Plant
Route 5, Box 413
Hartsville, South Carolina 29551

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

Mr. Marc P. Pearson
Plant Manager
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant
Post Office Box 790
Hartsville, South Carolina 29551

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

Mr. R. M. Krich
Manager - Regulatory Affairs
Carolina Power & Light Company
H. B. Robinson Steam Electric Plant,
Unit No. 2
Post Office Box 790
Hartsville, South Carolina 29551-0790

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

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
Post Office Box 29520
Raleigh, North Carolina 27626-0520

Mr. Max Batavia, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

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

Hartsville Memorial Library
147 West College Avenue
Hartsville, South Carolina 29550

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 151
License No. DPR-23

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated March 3, 1993, as supplemented October 22, 1993, and August 12, 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 3.B. of Facility Operating License No. DPR-23 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 151 , are hereby incorporated in the license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David B. Matthews, 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: September 16, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 151

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

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
3.10-3	3.10-3
3.10-3a	3.10-3a
3.10-4	3.10-4
3.10-16a	3.10-16a
6.9-18	6.9-18

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the measurement uncertainty factor F_u^N = 1.05 and the engineering factor F_Q^E = 1.03. $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by a 1.04 measurement uncertainty factor. K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the CORE OPERATING LIMITS REPORT (COLR). F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER (RTP), $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER, $PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{RTP}$. F_Q^{RTP}, $F_{\Delta H}^{RTP}$, and $PF_{\Delta H}$ are specified in the COLR.

3.10.2.1.1 Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

* During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

3.10.2.2 $F_Q(Z)$ shall be determined to be within the limit given in 3.10.2.1 by satisfying the following relationship for the middle axial 80% of the core at the time of the target flux determination:

$$F_Q(Z) \leq (F_Q^{RTP}/P) \times [K(Z)/V(Z)] \text{ for } P > 0.5$$

$$F_Q(Z) < (F_Q^{RTP}/0.5) \times [K(Z)/V(Z)] \text{ for } P \leq 0.5$$

where V(Z) is specified in the COLR.

3.10.2.2.1 If the relationship specified in 3.10.2.2 cannot be satisfied, one of the following actions shall be taken:

a) Place the core in an equilibrium condition where the limit in 3.10.2.2 is satisfied and re-establish the target axial flux difference

b) Reduce the reactor power by the maximum percent calculated with the following expression for the middle axial 80% of the core:

$$\left[\left[\text{max. over Z of } \frac{F_Q(Z) \times V(Z)}{\left(F_Q^{RTP}/P\right) \times K(Z)}\right] - 1\right] \times 100\%$$

c) Comply with the requirements of Specification 3.10.2.2.2.

3.10.2.2.2 The Allowable Power Level above which initiation of the Axial Power Distribution Monitoring System (APDMS) is required is given by the relation:

$$\text{APL} = \text{minimum over Z of } \frac{F_Q^{RTP} \times K(Z)}{F_Q(Z) \times V(Z)} \times 100\%$$

where $F_Q(Z)$ is the measured $F_Q^N(Z)$, multiplied by the engineering factor F_Q^E = 1.03 and the measurement uncertainty factor F_u^N = 1.05 at the time of target flux determination from a power distribution map using the movable incore detectors. The V(Z) axial variation function and K(Z) functions are specified in the COLR.

The above limit is not applicable in the following core plane regions.

1) Lower core region 0% to 10% inclusive.
2) Upper core region 90% to 100% inclusive.

Specific numerical values for the number of twice burned non-blanketed assemblies allowed in the core and on the bounding bank D control rod reactivity worth are provided in Reference p) of Technical Specification 6.9.3.3.b (NRC-approved power distribution control methodology) which details the most recent application(s) of the power distribution control methodology to H. B. Robinson.

For transient events, the core is protected from exceeding 21.1 kw/ft locally, and from going below the DNBR safety limit by automatic protection on power, flux difference, pressure and temperature.

Measurements of the hot channel factors are required as part of startup physics tests and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors.

In the specified limit of F_Q^N there is a 5 percent allowance for uncertainties[5] which means that normal operation of the core within the defined conditions and procedures is expected to result in a measured F_Q^N 5 percent less than the limit, for example, at rated power even on a worst case basis. When a measurement is taken, experimental error must be allowed for, and 5 percent is the appropriate allowance for a full core representative map taken with the movable incore detector flux mapping system.

In the specified limit of $F_{\Delta H}^N$, there is an 8 percent allowance for design prediction uncertainties, which means that normal operation of the core is expected to result in $F_{\Delta H}^N$ at least 8 percent less than the limit at rated power. The uncertainty to be associated with a measurement of $F_{\Delta H}^N$ by the movable incore system, on the other hand, is 4 percent, which means that the normal operation of the core shall result in a measured $F_{\Delta H}^N$ at least 4 percent less than the value at rated power. The logic behind the larger design uncertainty in the case is that (a) abnormal perturbation in the radial power shape (e.g., rod misalignment) affects $F_{\Delta H}^N$ in most cases without necessarily

q) ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," Advanced Nuclear Fuels Corporation, Richland WA 99352, latest revisions and supplements.

(Methodology for Specifications 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.3.3.c 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.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated March 3, 1993, as supplemented October 22, 1993, and August 12, 1994, the Carolina Power & Light Company (licensee) submitted a request for changes to the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR2), Technical Specifications (TS). The amendment will (1) add ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," to the approved methodologies list of Section 6.9.3.3.b, (2) clarify the wording in Sections 3.10.2.1 and 3.10.2.2.2 by describing more precisely how measurement uncertainty and engineering factors are considered, (3) correct an error in Section 3.10.2.2, and (4) correct a reference in TS 3.10.8.3 on page 3.10-16a of the TS basis. The letters dated October 22, 1993, and August 12, 1994, provided clarification and did not effect the initial no significant hazards consideration determination published in the Federal Register.

2.0 EVALUATION

The licensee requested TS changes in accordance with the 10 CFR 50.90 and 2.101. The revised TS were proposed as follows:

(a) Technical Specifications 3.10.2.1 and 3.10.2.2.2

Clarify wording changes by replacing "$F_Q(z)$ and $F\Delta H$ including" with "$F^N_Q(z)$ and $F^N\Delta H$ multiplied by" to describe more precisely how measurement uncertainty and engineering factors are considered.

(b) Technical Specification 3.10.2.2

Correct the typographical error: ≤ should be changed to <.

(c) Technical Specification 3.10.8.3 on page 3.10-16a

Change Reference 2 to Reference p)

(d) Technical Specification 6.9.3.3.b

Add approved topical report ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," to Reference q) for supporting heat flux hot channel factor and nuclear enthalpy rise hot channel factor calculations.

As indicated, TS 3.10.2.1, 3.10.2.2.2, and 3.10.8.3 are clarifications or corrections that do not alter the meaning of the TS. The changes allow a more precise description on how the specified uncertainty and engineering factors are to be applied. In addition, changes to TS 3.10.2.2 and TS 3.10.8.3 correct errors.

The topical report which is to be added to TS 6.9.3.3.b was approved by the NRC on September 9, 1991, for referencing a design methodology. While the methodology provides an acceptable basis for the pressurized water reactor fuel rod designs to burnups of 62 Gwd/MTU, the licensee confirmed in their letter of October 22, 1993, that their current design basis limits the plant to a mechanical design peak bundle burnup of 52.5 Gwd/MTU. The corresponding peak rod average burnup is 57.5 Gwd/MTU. The NRC accepts the addition of the reference of the approved design methodology to the TS for documenting a design methodology. The August 12, 1994, letter provides a clarification concerning the revision of the methodology being added and commits to document the revision in the next COLR.

Based on our review, we conclude that the changes to these TS are acceptable since the changes are to incorporate an approved reference for a fuel design methodology that will ensure that values of core parameters are determined such that all applicable limits of the safety analysis are met. In addition, TS changes to correct typographical errors and make wording clarifications that are administrative in nature are also acceptable. No safety related equipment, safety functions, plant design basis or operational procedures are altered as a result of these changes.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South 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. 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 19472). 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 Contributors: T. Huang
B. Mozafari

Date: September 16, 1994