Docket No. 50-261 July 15, 1992

Mr. R. A. Watson
Senior Vice President
Nuclear Generation
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

See correction letter of 7/29/92

Dear Mr. Watson:

SUBJECT: ISSUANCE OF AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DPR-23 REGARDING REMOVAL OF CYCLE-SPECIFIC PARAMETER LIMITS TO CORE OPERATING LIMITS REPORT - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 (TAC NO. M81333)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 141 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 January 7, 1991, as supplemented April 16, 1992, and June 4, 1992.

The amendment (1) allows the use of a Core Operating Limits Report (COLR), (2) inserts a definition of the COLR into the TS, (3) amends the affected TS to reflect the fact that numerical values for the cycle-specific limits and restrictions are being relocated to the COLR, and (4) adds the reference to the COLR to the Administrative Control Section of the TS to specify COLR contents, approved methodologies to be used for updating the COLR, and the reporting requirements for revision of the COLR.

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

Sincerely,
Orignal 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. 141 to DPR-23
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PD21:DRPE	PM:PD21:DRPE	OGC w/ noted revision	D:PD21:DRPE
PAnderson	BMozafari:dt	MYoung	EAdensam
7/7/92	7/7/92	7/16/92	7/15/92

Document Name: ROB81333.AMD

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1/1

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Carolina Power & Light Company

H. B. Robinson Steam Electric
Plant, Unit No. 2

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AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

Docket File
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Local PDR
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210002

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

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

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
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 January 7, 1991, as supplemented April 16, 1992, and June 4, 1992, 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. 141, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Brenda Mozafari, for

Elinor G. Adensam, 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: July 15, 1992

ATTACHMENT TO LICENSE AMENDMENT NO.141

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
v	v
1-7	1-7
3.1-11	3.1-11
3.1-12	3.1-12
3.10-1	3.10-1
3.10-2	3.10-2
3.10-3	3.10-3
-	3.10-3a
3.10-4	3.10-4
3.10-5	3.10-5
-	3.10-5a
3.10-7	3.10-7
3.10-7a	3.10-7a
3.10-10	3.10-10
3.10-13	3.10-13
3.10-15	3.10-15
3.10-16	3.10-16
--	3.10-16a
3.10-20	3.10-20
3.10-22	3.10-22
3.10-23	--
3.10-24	--
6.5-7	6.5-7
--	6.9-12
--	6.9-13
--	6.9-14
--	6.9-15
--	6.9-16
--	6.9-17
--	6.9-18

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3.1.4-1	Percent of Rated Thermal Power	3.1-15a
3.1-1	Reactor Coolant System Heat Up Limitations - Applicable for Records Up to 20 Effective Full Power Years	3.1-21
3.1-2	Reactor Coolant System Cool Down Limitations - Applicability for Periods Up to 20 Effective Full Power Years	3.1-22
3.10-1	(DELETED)	3.10-20
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6.2-2	Conduct of Operations Chart	6.2-4

1.19 SITE BOUNDARY

The site boundary shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee, as defined by Figure 1.1-1.

1.20 MEMBER(S) OF THE PUBLIC

Member(s) of the public shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen, or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for the purposes of protection of individuals from exposure to radiation and radioactive materials.

1.21 UNRESTRICTED AREA

Unrestricted area shall be any area at or beyond the Site Boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the Site Boundary used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

1.22 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT (COLR) is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.3.3. Unit operation within these operating limits is addressed in individual specifications.

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, at which the moderator temperature coefficient is outside the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limits shall be less than or equal to:

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1a or 3.1-1b (as appropriate per 3.1.2.1).

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is outside the limits specified in the COLR, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator temperature coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator temperature coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator temperature coefficient outside the limits specified in the COLR has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase in moderator temperature or decrease in coolant pressure. This requirement is

waived during low power physics tests to permit measurement of reactor moderator temperature coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[1] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The heatup curve of Figure 3.1-1 includes criticality limits which are required by 10 CFR Part 50, Appendix G, Paragraph IV.A.2.c. Whenever the core is critical, additional safety margins above those specified by the ASME Code Appendix G methods are imposed. The core may be critical at temperatures equal to or above the minimum temperature for the inservice hydrostatic pressure tests as calculated by ASME Code Appendix G methods, and an additional safety margin of 40°F must be maintained above the applicable heatup curve at all times.

If the specified shutdown margin is maintained (Section 3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[1]

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of one percent subcriticality will assure that the Reactor Coolant System will not be solid when criticality is achieved.

References

(1) FSAR Section 4.3

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

3.10.1.2 When the reactor is critical, except for physics tests and full length control rod exercises, the shutdown control rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

3.10.1.3 When the reactor is critical, except for physics tests and full length control rod exercises, the control rods shall be limited in physical insertion as specified in the COLR.

3.10.1.4 At 50 percent of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core values as specified in the COLR.

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions $\geq$ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

3.10.2 <u>Power Distribution Limits</u>

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

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

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

$$F_{\Delta H} < F_{\Delta H}^{RTP} \ (1 + PF_{\Delta H} \ (1-P) \ )$$

Amendment No. 87, 115, 119, 141

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q(Z)$ including the measurement uncertainty factor $F_u^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}$ including 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.

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* 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:

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

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

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:

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

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 $F_Q(Z)$, including 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.

At power levels in excess of APL of rated power, the APDMS will be employed to monitor $F_Q(Z)$. The limiting value is expressed as:

$$[\ F_j(Z)\ \ S(Z)\ ]_{max} \leq \frac{[F_Q^{RTP} / (F_u^N \times F_Q^E \times F_Q^a)]/P}{\bar{R}_j\ (1 + \sigma_j)}$$

where:

a. P is the fraction of rated power (2300 Mwt) at which the core is operating ($P \leq 1.0$).

b. $F_u^N = 1.05$ is the measurement uncertainty factor.

c. $F_Q^E = 1.03$ is the engineering uncertainty factor.

d. $F_Q^a = 1.02$ is the instrument uncertainty factor.

e. $\bar{R}_j$ for thimble j, is determined from the core power maps and is by definition:

$$\bar{R}_j = \frac{1}{6} \sum_{i=1}^{6} \frac{F_{Qi}}{[F(Z)_{ij} S(Z)]_{max}}$$

i) F_{Qi} is the value obtained from a full core map including S(Z), but without the uncertainty factors F_u^N and F_Q^E.

ii) $F(Z)_{ij}$ S(Z) is the measured value without inclusion of the instrument uncertainty factor F_Q^a.

f. σ_j is the standard deviation associated with the determination of $\bar{R}_j$.

g. S(Z) is the inverse of the K(Z) function specified in the COLR.

This limit is not applicable during physics tests and excore detector calibrations.

3.10.2.2.3 With successive measurements indicating the enthalpy rise hot channel factor, $F^N_{\Delta H}$, to be increasing with exposure, the total peaking factor, $F_Q(Z)$, shall be further increased by two percent over that specified in Specifications 3.10.2.2, 3.10.2.2.1, and

a. The indicated axial flux difference may deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference does not exceed the limits specified in the COLR. If the cumulative time exceeds one hour, then the reactor power shall be reduced immediately to no greater than 50 percent of rated power and the high neutron flux setpoint reduced to no greater than 55 percent of rated power.

b. A power increase to a level greater than 90 percent of rated power or 0.9 x APL (whichever is less) is contingent upon the indicated axial flux difference being within its target band.

3.10.2.8 At a power level no greater than 50 percent of rated power

a. The indicated axial flux difference may deviate from its target band.

b. A power increase to a level greater than 50 percent of rated power is contingent upon the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half of the time the indicated axial flux difference is out of its target band up to 50 percent of rated power is to be counted as contributing to the one-hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90 percent of rated power or 0.9 x APL (whichever is less).

3.10.2.9 Calibration of excore detectors will be performed under the following conditions:

a. at power levels greater than 90 percent of rated power or 0.9 x APL (whichever is less) provided the axial flux difference does not exceed the specified target bands, or

b. at power levels less than 90 percent of rated power or 0.9 x APL (whichever is less) provided the indicated axial flux difference does not exceed the limits specified in the COLR.

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.6 or the flux difference-time requirement of 3.10.2.7.a. If the alarms are temporarily out of service, the axial flux difference shall be logged, and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter.

3.10.2.11 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ as specified in 3.10.2.1.1. The allowable values of the target band are specified in the COLR. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target axial flux difference.

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests and during power increases below 50 percent of rated power, whenever the indicated quadrant power tilt ratio exceeds 1.02, the tilt condition shall be eliminated within two hours or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint to be less two percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and

3.10.8.2 When the reactor is in the cold shutdown condition, the shutdown margin shall be at least 1 percent $\Delta k/k$.

3.10.8.3 When the reactor is in the refueling operation mode, the shutdown margin shall be at least 6 percent $\Delta k/k$.

Basis

The reactivity control concept is that reactivity changes accompanying changes in reactor power are compensated by control rod motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of reactor power is by the control groups. A reactor trip occurring during power operation will put the reactor into the hot shutdown condition.

The control rod insertion limits provide for achieving hot shutdown by reactor trip at any time assuming the highest worth control rod remains fully withdrawn with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted rod worth in the unlikely event of hypothetical rod ejection and provide for acceptable nuclear peaking factors. The control rod insertion limits are specified in the CORE OPERATING LIMITS REPORT (COLR) and are appropriately chosen to meet the shutdown requirements shown in Figure 3.10-2. The maximum shutdown margin requirement occurs at end of core life and is based on the value used in analysis of the hypothetical steam break accident. Early in core life, less shutdown margin is required, and Figure 3.10-2 shows the shutdown margin required at end of life with respect to an uncontrolled cooldown. All other accident analyses are based on 1 percent reactivity

are of the fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

d. $F^N_{\Delta H}$, 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.

It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes through the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

It has been determined by extensive analysis of possible operating power shapes that the design limits on peak local power density and on minimum DNBR at full power are met, provided the values of F_q and $F_{\Delta H}$ in Specification 3.10.2.1 are not exceeded.

For normal operation, it is not necessary to measure these quantities. Instead, it has been determined that, provided certain conditions are observed, the above hot channel factor limits will be met; these conditions are as follows:

a. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.

b. The specific control rod sequence and overlap requirements are based on the rod insertion limits of specification 3.10.1.

c. The control bank insertion limits are not violated.

d. Deleted

Strict control of the flux difference is not possible during certain physics tests, control rod exercises, or during the required periodic excore calibration which require larger flux differences than permitted. Therefore, the specification on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this is acceptable due to the extremely low probability of a significant accident occurring during these operations. Excore calibration includes that period of time necessary to return to equilibrium operating conditions. In some instances of rapid plant power reduction, automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux differences in the allowable range specified in the COLR for 90 percent of rated power or 0.9 x APL (whichever is less). Therefore, while the deviation exists, the power level is limited to 90 percent of rated power or 0.9 x APL (whichever is less) or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled with the target band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent of rated power is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

An upper bound envelope of peaking factors has been determined from extensive analysis considering all operating maneuvers consistent with the technical specifications on power distribution control as given in Section 3.10.2 The specifications on power distribution control ensure that xenon distributions are not developed, which at a later time, could cause greater local power peaking even though the flux difference is then within limits. The results of a Loss-of-Coolant Accident analysis based on this upper bound envelope indicate that a peak clad temperature would not exceed the 2200°F limit. The nuclear analyses of credible power shapes consistent with the power distribution control procedures have shown that the F_Q^T limit is not exceeded.

Current power distribution control methodology, as applied on a H. B. Robinson Unit 2 plant specific basis, places certain restrictions on core characteristics which affect the validity of the power distribution control curves as provided each cycle in the COLR. The restricted core characteristics are:

a) Restrictions are placed on the maximum number of twice burned non-blanketed fuel assemblies which may be placed in the core and,

b) The bank D control rod reactivity worth is restricted such that its value must be bounded by those values assumed in the most recent application of the power distribution control methodology to H. B. Robinson.

The purpose of these restrictions is to make the power distribution curves plant specific but not core or reload specific, that is, if current core characteristics meet the restrictions on a) and b) above, the most recently developed power distribution control curves remain valid for the current reload. If at any time, the noted restrictions cannot be met for a proposed core reload, the current power distribution control curves are not valid and re-analysis using the NRC-approved methodology is necessary to provide new curves.

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 2 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

FIGURE 3.10-1 DELETED

FIGURES 3.10-3 THROUGH 3.10-5 DELETED

(next page is 3.11-1)

6.5.1.6.5 A quorum of the PNSC shall consist of the Chairman, and four members, of which two may be alternates.

6.5.1.6.6 The PNSC activities shall include the following:

a) Perform an overview of Specifications 6.5.1.1 and 6.5.1.2 to assure that processes are effectively maintained.

b) Performance of special reviews, investigations, and reports thereon requested by the Manager - Nuclear Assessment Department.

c) Annual review of the Security Plan and Emergency Plan.

d) Perform reviews of Specifications 6.5.1.1.6, 6.5.1.2.4, 6.5.1.3.1, and 6.5.1.4.1.

e) Perform review of all reportable events.

f) Review of facility operations to detect potential nuclear safety hazards.

g) Review of every unplanned on site release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrences to the Vice President - Robinson Nuclear Project, Manager - Nuclear Assessment Department.

h) Review of changes to the Process Control Program and the Offsite Dose Calculation Manual.

i) Review of major changes to radioactive liquid, gaseous, and solid waste treatment systems.

j) Review of changes to the CORE OPERATING LIMITS REPORT.

6.9.3.3 Core Operating Limits Report

6.9.3.3.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

1. Moderator Temperature Coefficient limits for Specification 3.1.3.1.

2. Shutdown Bank Insertion Limits for Specification 3.10.1.2.

3. Control Bank Insertion Limits for Specification 3.10.1.3 and 3.10.1.4.

4. Heat Flux Hot Channel Factor limit (F_Q^{RTP}), Nuclear Enthalpy Rise Hot Channel Factor limit ($F_{\Delta H}^{RTP}$), K(Z), and Power Factor Multiplier ($PF_{\Delta H}$) for Specification 3.10.2.

5. Axial Flux Difference limits and V(Z) for Specification 3.10.2

6.9.3.3.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

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 Specifications 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

b) XN-NF-84-73(P), latest Revision and Supplements, "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Exxon Nuclear Corporation, Richland WA 99352 (Accepted by the NRC for H. B. Robinson Unit 2 in the Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 87 to Facility License No. DPR-23, 7 Nov. 84).

(Methodology for Specifications 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

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 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor)

d) XN-NF-84-93(A), latest Revision and Supplements, "Steamline Break Methodology for PWR's," Exxon Nuclear Corporation, Richland, WA 99352.

(Methodology for Specifications 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor.)

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

(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)

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

(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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

g) EXEM PWR Large Break LOCA Evaluation Model as accepted in Letter, D. M. Crutchfield (NRC) to G. N. Ward (ENC), "Safety Evaluation of Exxon Nuclear Company's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Reports," July 8, 1986.

EXEM PWR LBLOCA Model includes the following references:

XN-NF-82-20(P), 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, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Richland WA 99352.

XN-NF-85-16(P), 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(P), and Supplements, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Company, Richland WA 99352.

(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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

h) XN-NF-78-44(A), latest Revision, "Generic Control Rod Ejection Analysis," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specifications 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor)

i) XN-NF-621(A), latest Revision, "XNB Critical Heat Flux Correlation," Exxon Nuclear Company, Richland WA 99352.

(Methodology for Specification 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor)

j) ANF-1224(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Advanced Nuclear Fuels Corporation, Richland WA 99352.

(Methodology for Specification 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor)

k) XN-NF-82-06(A), latest Revisions and Supplements, "Qualification of Exxon Nuclear Fuel for Extended Burnup", Exxon Nuclear Corporation, Richland, WA 99352.

(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)

l) Meyer, P. E. and Kornfilt, J., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10080-A, August 1985.

(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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

m) Lee, N., Tauche, W. D., Schwartz, W. R., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP code," WCAP-10081-A, August 1985.

(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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

n) Borderon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301, (Proprietary) and WCAP-8305, (Nonproprietary), June 1974 (accepted by the NRC in the SER related to WCAP-8472-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information", April, 1975).

(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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

o) "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 87 to Facility Operating License No. DPR-23, Carolina Power and Light Co., H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261," USNRC, Washington D. C. 20555, 7 Nov. 84.

(Methodology for Specifications 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

p) ANF-88-054(P), "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, latest revisions and supplements. (Accepted by the NRC for H. B. Robinson Steam Electric Plant, Unit 2, in the Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 128 to Facility License No. DPR-23, Docket No. 50-261, USNRC, Washington D.C., 20555, August 22, 1990).

(Methodology for Specifications 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141 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 January 7, 1991, (Ref. 1), as supplemented by letters dated April 16, 1992 (Ref. 2), and June 4, 1992, Carolina Power & Light Company (CP&L or the licensee), submitted a request for changes to the H. B. Robinson Steam Electric Plant, Unit No. 2 (HBR2) Technical Specifications (TS). The proposed changes would modify TS having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) containing the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls section of the TS. Guidance on the proposed changes was developed by NRC and provided to all power reactor licensees and applicants by Generic Letter (GL) 88-16, dated October 4, 1988 (Ref. 3). The April 16, 1992, and June 4, 1992, letters provided clarifying information and updated TS pages reflecting pages changed by recent amendments and did not change the initial determination of significant hazards consideration as published in the Federal Register.

2.0 EVALUATION

The proposed changes to the TS are in accordance with the guidance provided by GL 88-16 and are addressed below:

(1) The Definition section of the TS was modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC-approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following TS were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) TS 3.1.3.1 and 3.1.3.3

The moderator temperature coefficient (MTC) limits for these TS are specified in the COLR.

(b) TS 3.10.1.2

The shutdown rod insertion limit for this specification is specified in the COLR.

(c) TS 3.10.1.3 and 3.10.1.4

The control rod insertion limits for these specifications are specified in the COLR.

(e) TS 3.10.2.1, 3.10.2.2, 3.10.2.2.1, and 3.10.2.2.2

The heat flux hot channel factor (F_Q) limit at rated thermal power (F_Q^{RTP}), and the normalized F_Q limit as a function of core height K(z) for these specifications are specified in the COLR.

(f) TS 3.10.2.1

The nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) limit at rated thermal power ($F_{\Delta H}^{RTP}$) and the power factor multiplier ($PF_{\Delta H}$) for this specification is specified in the COLR.

(g) TS 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 2.10.2.11

The axial flux difference limits, the target band, and the axial variation function corresponding to the target band V(z) for these TS are specified in the COLR.

The bases of affected TS have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(3) TS 6.9.3.3 is revised to include the COLR under the reporting requirements of the Administrative Control section of the TS. This TS requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this TS requires that the NRC-approved methodologies be used in establishing the values of these limits for the relevant TS and that the values be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(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 TS 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

(b) XN-NF-84-73(P), latest revision and supplements, "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Exxon Nuclear Corporation, Richland, WA 99352 (Accepted by the NRC for HBR2 in the SE related to Amendment No. 87 to Facility License No. DPR-23, November 7, 1984).

(Methodology for TS 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

(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 TS 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor).

(d) XN-NF-84-93(A), latest revision and supplements, "Steamline Break Methodology for PWR'S," Exxon Nuclear Corporation, Richland, WA 99352.

(Methodology for TS 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor).

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

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

(Methodology for TS 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

(g) EXEM PWR Large Break LOCA Evaluation Model as accepted in Letter, D. M. Crutchfield (NRC) to G. N. Ward (ENC), "Safety Evaluation of Exxon Nuclear Company's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Reports," July 8, 1986.

EXEM PWR LBLOCA Model includes the following references:

XN-NF-82-20(P), 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, "RODEXZ Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Richland, WA 99352.

XN-NF-85-16(P), 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(P), and supplements, "Scaling of FCTF Based Reflood Heat Transfer Correlation for other Bundle Designs," Exxon Nuclear Company, Richland, WA 99352.

(Methodology for TS 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

(h) XN-NF-78-44(A), latest revision, "Generic Control Rod Ejection Analysis," Exxon Nuclear Company, Richland, WA 99352.

(Methodology for TS 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 3.10.2.1, 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2 - Heat Flux Hot Channel Factor).

(i) XN-NF-621(A), latest revision, "XNB Critical Heat Flux Correlation," Exxon Nuclear Company, Richland, WA 99352.

(Methodology for TS 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor).

(j) ANF-1224(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Advanced Nuclear Fuels Corporation, Richland, WA 99352 .

(Methodology for TS 3.10.2.1 - Nuclear Enthalpy Rise Hot Channel Factor).

(k) XN-NF-82-06(A), latest revisions and supplements, "Qualification of Exxon Nuclear Fuel for Extended Burn-up," Exxon Nuclear Corporation, Richland, WA 99352.

(Methodology for TS 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).

(l) Meyer, P. E. and Kornfilt, J., "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10080-A, August 1985.

(Methodology for TS 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 2.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

(m) Lee, N., Tauche, W. D., Schwartz, W. R., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP code," WCAP-10081-A, August 1985.

(Methodology for TS 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

(n) Bordelon, F. M., et al., "LOCA-IV Program: Loss of Coolant Transient Analysis," WCAP-8301 (Proprietary), and WCAP-8305 (Nonproprietary), June 1974 (accepted by the NRC in the SE related to WCAP-8472-A, "The Westinghouse ECCS Evaluation Model: Supplementary Information," April 1975).

(Methodology for TS 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

(o) "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 87 to Facility Operating License No. DPR-23, Carolina Power & Light Company, H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261," November 7, 1984.

(Methodology for TS 3.1.3.1 - Moderator Temperature Coefficient, 3.10.1.2 - Shutdown Bank Insertion Limits, 3.10.1.3 and 3.10.1.4 - Control Bank Insertion Limits, 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, and 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

(p) ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit No. 2," Advanced Nuclear Fuels Corporation, Richland, WA 99352, latest revisions and supplements. (Accepted by the NRC for H. B. Robinson Steam Electric Plant, Unit No. 2, in the NRC SE related to Amendment No. 128 to Facility License No. DPR-23, Docket No. 50-261, August 22, 1990).

(Methodology for TS 3.10.2.2, 3.10.2.2.1, 3.10.2.2.2, 3.10.2.7, 3.10.2.9, and 3.10.2.11 - Axial Flux Difference).

Finally, the TS requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review, the NRC staff concludes that the licensee provided an acceptable response to the items in GL 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC-approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the NRC finds that the proposed changes are acceptable.

As part of the implementation of GL 88-16, the staff has also reviewed a sample COLR provided by the licensee and concludes that the format and content of the sample COLR are acceptable.

3.0 SUMMARY

We have reviewed the request by CP&L to revise the HBR2 TS by removing the specific values of some cycle-dependent parameters from the TS and placing the values in a COLR referenced by the TS. Based on this review, we conclude that these revisions are acceptable.

4.0 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.

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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 6868). The amendment also changes recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 (NLS-90-248) from G. E. Vaughn (CP&L) to NRC, dated January 7, 1991.

2. Letter (NLS-92-101) from R. B. Starkey, Jr. (CP&L), dated April 16, 1992.

3. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

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Date: July 17, 1992