July 23, 1982

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Docket No. 50-261

Mr. J. A. Jones, Vice Chairman
Carolina Power and Light Company
336 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Jones:

The Commission has issued the enclosed Amendment No. 71 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your applications transmitted by letters dated April 30, 1981, April 30, 1982, and July 13, 1982, as supplemented by letters dated April 20, 1982 and June 24, 1982.

The amendment:

1. Authorizes Cycle 9 operation at a reduced power level;

2. Revises the Appendix A Technical Specifications to:

   a. Incorporate changes resulting from the Cycle 9 reload core analysis, including administrative changes, and

   b. Incorporates changes to include specific surveillance of the emergency core cooling system motor operated valves; and

3. Revises the Operating License Condition (OLC) 3.I.a, b, c & d to include a steam generator inspection and steam generator tube leakage criterion. Some portions of your proposed OLC have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:
S. A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

RSB
B. MANN
BW
7/22/82

Enclosures:
1. Amendment No. 71 to DPR-23
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures: See next page

OFFICE	ORB#1:DL	ORB#1:DL	ORB#1:DL	AD/OR:DL	OELD
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Carolina Power and Light Company

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Home and Fifth Avenues
Hartsville, South Carolina 29550

Mr. McCuen Morrell, Chairman
Darlington County Board of Supervisors
County Courthouse
Darlington, South Carolina 29535

State Clearinghouse
Division of Policy Development
116 West Jones Street
Raleigh, North Carolina 27603

Attorney General
Department of Justice
Justice Building
Raleigh, North Carolina 27602

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
H. B. Robinson Steam Electric Plant
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CAROLINA POWER AND LIGHT COMPANY

DOCKET NO. 50-261

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

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. DPR-23

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

   A. The applications for amendment by Carolina Power and Light Company (the licensee) dated April 30, 1981, April 30, 1982, and July 13, 1982, as supplemented April 20, 1982 and June 24, 1982, comply 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 applications, and 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. 71, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Revise paragraph 3.I.a, b, c, & d of Facility Operating License No. DPR-23 to read as follows:

   a. Operation beyond three effective full-power months is contingent upon staff review and approval of information concerning steam generator integrity provided by the licensee to justify operation beyond three effective full-power months, but in no case shall operation continue beyond a total of six effective full-power months at which time a steam generator eddy current examination will be performed. The scope of the inspection to be performed will be submitted to the NRC for approval at least 45 calendar days prior to the inspection.

   b. During cycle 9 operations, the following steam generator tube leakage criteria shall be in effect. Specifically, the plant shall be shut down for appropriate corrective action if the verified primary to secondary leakage in one steam generator exceeds any of the following:

      1. A sudden increase of 0.1 gallon per minute (gpm) if the total leakage rate in that steam generator exceeds 0.2 gpm.

      2. If the leakage rate in that steam generator exceeds 0.2 gpm and an upward trend in leakage rate in excess of 0.02 gpm per day is verified. This trend will be established using at least five valid consecutive daily samples.

   c. Should the plant be required to shut down to repair a steam generator tube leak as indicated in item (b) above, an inspection shall be performed as mutually agreed upon by the NRC Staff and CP&L; except in the case of steam generator tube plug valves.

   d. The NRC staff will be provided with a summary of the results of the eddy current inspection described in item a.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 23, 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A as follows:

Remove Pages	Insert Pages
2.3-1	2.3-1
2.3-2	2.3-2
2.3-3	2.3-3
3.10-2	3.10-2
3.10-2a	3.10-2a
3.10-3	3.10-3
3.10-4	3.10-4
3.10-5	3.10-5
3.10-6	3.10-6
3.10-7	3.10-7
3.10-9	3.10-9
3.10-11	3.10-11
3.10-14	3.10-14
3.10-15	3.10-15
3.10-16	3.10-16
3.10-17	3.10-17
3.10-20	3.10-20
3.11-1	3.11-1
3.11-2	3.11-2
4.11-1	4.11-1
4.5-3 through 4.5-5	4.5-3 through 4.5-5

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, and flow and pressurizer level.

Objective

To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification

2.3.1 Protective instrumentation settings for reactor trip shall be as follows:

2.3.1.1 Start-up protection

a. High flux, power range (low setpoint)
$\leq$25% of rated power.

2.3.1.2 Core protection

a. High flux, power range (high setpoint)
$\leq$109% of rated power*

b. High pressurizer pressure $\leq$2385 psig.

c. Low pressurizer pressure $\geq$1835 psig.

*Rated power is defined here as 1955 Mwt under the reduced T_{avg} program and 2300 Mwt under the normal T_{avg} program.

d. Overtemperature ΔT

$$\leq \Delta T_o \;\{K_1 - K_2 (T - 575.4) + K_3 (P - 2235) - f(\Delta I)\} \quad *$$

where:
ΔT_o = Indicated ΔT at rated power**, °F;
T = Average temperature, °F;
P = Pressurizer pressure, psig;
K_1 = 1.1619;
K_2 = 0.01035;
K_3 = 0.0007978;

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant start-up tests such that:

(1) For $(q_t - q_b)$ within +12% and -17%, where q_t and q_b are percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is total core power in percent of rated power (2300 Mwt), $f(\Delta I) = 0$. For every 2.4% below rated power (2300 Mwt) level, the permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power (2300 Mwt) level, the permissible negative flux difference range is extended by -1 percent.

(2) For each percent that the magnitude of $(q_t - q_b)$ exceeds +12% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt).

(3) For each percent that the magnitude of $(q_t - q_b)$ exceeds -17%, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt).

*When operating under the reduced T_{avg} program, replace the number 575.4 with 537.9 in the overtemperature ΔT calculation.

**Rated power is defined here as 1955 Mwt under the reduced T_{avg} program and 2300 Mwt under the normal T_{avg} program.

e. Overpower ΔT
$\leq \Delta T_o [K_4 - K_5 \frac{dT}{dt} - K_6 (T - T') - f(\Delta I)]$,

where:

ΔT_o = Indicated ΔT at rated power**, °F;
T = Average temperature, °F;
T'* = Indicated average temperature at nominal conditions and rated power**, °F;
K_4 = 1.07;
K_5 = 0 for decreasing average temperature and 0.2 seconds per °F for increasing average temperature;
K_6 = 0.002235 for $T \geq T'$; K_6 = 0 for $T < T'$;
$f(\Delta I)$ = as defined in d. above.

f. Low reactor coolant loop flow $\geq$90% of normal indicated flow

g. Low reactor coolant pump frequency $\geq$57.5 Hz

h. Undervoltage $\geq$70% of normal voltage.

2.3.1.3 Other Reactor Trips

a. High pressurizer water level $\leq$92% of span

b. Low-low steam generator water level $\geq$14% of narrow range instrument span.

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

2.3.2.1 The low pressurizer pressure trip, high pressurizer level trip, and the low reactor coolant flow trip (for two or more loops) may be bypassed below 10% of rated power.

2.3.2.2 The single-loop-loss-of-flow trip may be bypassed below 45% of rated power.

*The value of T' for nominal conditions and rated power is 575.4°F. When operating under the reduced temperature conditions described in the November 11, 1981 license submittal, replace the number 575.4 with 537.9 in the overpower ΔT calculation.

**Rated power is defined here as 1955 Mwt under the reduced T_{avg} program and 2300 Mwt under the normal T_{avg} program

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* for three-loop operation.

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 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

Under the normal T_{avg} program	Under the reduced T_{avg} program
$F_Q (Z) \leq (2.20/P) \times K(Z)$ for $P > .5$,	$F_Q (Z) \leq (2.32/P1) \times K(Z)$ for $P1 > .5$,
$F_Q (Z) < (4.40) \times K(Z)$ for $P \leq .5$,	$F_Q (Z) < (4.64) \times K(Z)$ for $P1 \leq .5$

*Rated power is defined here as 2300 Mwt for the normal T_{avg} program and 1955 Mwt for the reduced T_{avg} program where P is the fraction of rated power (2300 Mwt) at which the core is operating under the normal T_{avg} program, P1 is the fraction of 1955 Mwt at which the core is operating under the reduced T_{avg} program, K(Z) is based on the function given in Figure 3.10-3, and Z is the core height location of F_Q.

Under both the normal T_{avg} program and the reduced T_{avg} program:

$$F^N_{\Delta H} < 1.55 \ (1 + 0.2(1-P))$$

where P is the fraction of rated power (2300 Mwt) at which the core is operating.

3.10.2.1.1 At power levels in excess of 94% rated power, or if the value of Fxy for the unrodded plane of the core exceeds 1.435 as determined from power distribution maps using the movable detector system, the Axial Power Distribution Monitoring System (APDMS) will be employed to monitor $F_Q(Z)$ above a predetermined power level, P_{APDMS}. The limiting value is expressed as:

Under the normal T_{avg} program	Under the reduced T_{avg} program
$[F_j(Z)S(Z)]_{max} \leq \dfrac{1.994/P}{\overline{R}_j(1+\sigma_j)}$	$[F_j(Z)\ S(Z)]_{max} \leq \dfrac{2.103/P1}{\overline{R}_j\ (1+\sigma_j)}$

where:

a. P is the fraction of rated power (2300 Mwt) at which the core is operating ($P \leq 1.0$) and P1 is the fraction of 1955 Mwt at which the core is operating ($P1 \leq 1.0$).

b. $\overline{R}_j$, for thimble j, is determined from core power maps i and is by definition:

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

F_{qi}^{N} is the value obtained from a full core map without the measurement uncertainty factor F_u^N. The quantity $F(Z)_{ij}S(Z)$ is the measured value without inclusion of the instrument uncertainty factor F_q^a. Those uncertainty factors, $F_u^N = 1.05$ and $F_q^a = 1.02$, have been included in the limiting value of 1.994/P for the normal T_{avg} program and 2.103/P1 for the reduced T_{avg} program.

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

d. S(Z) is the inverse of the K(Z) function given in Figure 3.10-3.

This limit is not applicable during physics test and excore calibrations.

3.10.2.1.2 The predetermined power level at which APDMS initiation is required is given by the relation

For the normal T_{avg} program	For the reduced T_{avg} program
$P_{APDMS} \leq \frac{1.435}{Fxy} \times 0.94$	$P1_{APDMS} \leq \frac{1.435}{Fxy}$

where P_{APDMS} is the fraction of rated power and $P1_{APDMS}$ is the fraction of 1955 Mwt.

3.10.2.1.3 Fxy shall be determined for the unrodded core plane regions away from fuel support grids, located between a core plane elevation 2.0 feet from the top of the core and a core plane elêvation 2.0 feet from the bottom of the core with no control rod inserted more than 2.0 feet into the core. This determination shall be made from the movable incore detector maps specified in 3.10.2.3.

3.10.2.2 If either measured hot channel factor exceeds these values, the reactor power shall be reduced so as not to exceed a fraction of the design value* equal to the ratio of the F_Q^N or $F_{\Delta H}^N$ 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.

3.10.2.3 Following initial loading and at regular monthly intervals thereafter, 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. For the purpose of this confirmation:

a. The measurement of total peaking faactor, F_Q^{Meas}, shall be increased by three percent to account for manufacturing tolerances and further increased by five percent to account for measurement error.

*Design value is defined here as the maximum power at which the F_Q or $F_{\Delta H}$ limit is defined in specification 3.10.2.1.

b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by four percent to account for measurement error.

3.10.2.4 The reference equilibrium-indicated axial flux difference for each encore channel as a function of power level (called the target flux difference) shall be measured at least once per effective full power quarter*. If the axial flux difference has not been measured in the last effective full power month, the target flux difference must be updated monthly by linear interpolation, using the most recent measured value and the value predicted for the end of the cycle life.

3.10.2.5 The indicated axial flux difference shall be considered outside of the limits of Sections 3.10.2.6 through 3.10.2.9 when more than one of the operable excore channels are indicating the axial flux difference to be outside a limit.

3.10.2.6 Except during physics tests, during excore detector calibration, and except as modified by 3.10.2.7 through 3.10.2.9 below, the indicated axial flux difference shall be maintained within a ± 5 percent band about the target flux difference (defines the target band on axial flux difference).

3.10.2.7 At a power level greater than 90 percent of rated power*, if the indicated axial flux difference deviates from its target band, the flux difference shall be returned to the target band immediately or reactor power shall be reduced to a level no greater than 90 percent of rated power*.

3.10.2.8 At a power level no greater than 90 percent of rated power*,

a. The indicated axial flux difference may deviate from its ± 5 percent target band for a maximum of one hour (cumulative) in any 24-hour period provided the flux difference

*Full power and rated power are defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.

does not exceed an envelope bounded by and ±11 percent at 90 percent of rated power* and increasing by ±1 percent for each 2 percent of rated power below 90 percent of rated power*. 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* is contingent upon the indicated axial flux difference being within its target band.

3.10.2.9 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*.

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.7 or the flux difference-time requirement of 3.10.2.8.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. The requirement for alarms becomes effective December 1, 1975.

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*Rated power is defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.

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

b. If the tilt condition is not eliminated after 24 hours, the power range high flux setpoint shall be reset to 55 percent of rated power*. Subsequent reactor operation would be permitted up to 50 percent of rated power* for the purpose of measurement and testing to identify the cause of the tilt condition.

3.10.3.2 Except for low power physics tests, if the indicated quadrant tilt exceeds 1.09 and there is simultaneous indication of a misaligned rod:

a. The core power level shall be reduced by 2 percent of rated values* for every 1 percent of indicated power tilt exceeding 1.0, and

b. If the tilt condition is not eliminated within two hours, the reactor shall be brought to a hot shutdown condition.

c. After correction of the misaligned rod, reactor operation will be permitted to 50 percent of rated power* until the indicated quadrant tilt falls below 1.09.

3.10.3.3 If the indicated quadrant tilt exceeds 1.09 and there is not a simultaneous indication of rod misalignment, except as stated in Specification 3.10.3.2.c, the reactor shall immediately be brought to a hot shutdown condition.

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*Rated power and rated values are defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.

3.10.6.2 No more than one inoperable control rod shall be permitted during power operation.

3.10.6.3 If a full length control rod cannot be moved by its mechanism, boron concentration shall be changed to compensate for the withdrawn worth of the inoperable rod such that a shutdown margin equal to or greater than shown on Figure 3.10-2 results.

3.10.7 <u>Power Ramp Rate Limits</u>

3.10.7.1 During the return to power following a shutdown where fuel assemblies have been handled (e.g., refueling, inspection), the rate of reactor power increase shall be limited to 3 percent of rated power in an hour between 20 percent and 100 percent of rated power. This ramp rate requirement applies during the initial startup and may apply during subsequent power increases, depending on the maximum power level achieved and length of operation at that power level. Specifically, this requirement can be moved for reactor power levels below a power level P (20 percent $<$ P $\leq$ 100 percent), provided that the plant has operated at or above power level P for at least 72 cumulative hours out of any seven-day operating period following the shutdown.

3.10.7.2 The rate of reactor power increases above the highest power level sustained for at least 72 cumulative hours during the preceding 30 cumulative days of reactor power operation shall be limited to 3 percent of rated power in an hour. Alternatively, reactor power increase can be accomplished by a single step increase less than or equal to 10 percent of rated power followed by a maximum ramp rate of 3 percent of rated power in an hour beginning three hours after the step increase.

3.10.8 <u>Required Shutdown Margins</u>

3.10.8.1 When the reactor is in the hot shutdown condition, the shutdown margin shall be at least that shown in Figure 3.10-2.

shutdown margin. The specified control rod insertion limits meet the design basis criteria on (1) potential ejected control rod worth and peaking factor,[4] (2) radial power peaking factors, $F_{\Delta H}$, and (3) required margin shutdown.

The various control rod banks (shutdown banks, control banks) are each to be moved as a bank; that is, with all rods in the bank within one step (5/8 inch) of the bank position. Position indication is provided by two methods: a digital count of actuation pulses which shows the demand position of the banks, and a linear position indicator (LVDT) which indicates the actual rod position.[2] At rod positions $\geq$200 steps, full power reactivity worths of the control rods are sufficiently small such that a 15-inch indicated misalignment from the rod bank has no significant effect on the incore power distribution and is therefore allowable. For rod positions <200 steps, maintaining indicated rod position within 7.5 inches of the average of the indicated bank position provides an enforceable limit which assures design distribution is not exceeded. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. The determination of the hot channel factors will be performed by means of the movable in-core detectors.

The two hours in 3.10.1.5 are acceptable because complete rod misalignment (control rod 12 feet out of alignment with its bank) does not result in exceeding core safety limits in steady state operation at rated power and is short with respect to probability of an independent accident. If the condition cannot be readily corrected, the specified reduction in power will ensure that design margins to core limits will be maintained under both steady state and anticipated transient conditions.

e. Axial power distribution control procedures, which are given in terms of flux difference control, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in power between the top and bottom halves of the core.

For operation at a fraction P of full power, the design limits are met, provided the limits of Specification 3.10.2.1 are not exceeded.

The permitted relaxation in $F_{\Delta H}^{N}$ with reduced power allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed, these hot channel factors limits are met.

The procedures for axial power distribution control referred to above include operator control of flux difference to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of Flux Difference (ΔI) and a reference value which corresponds to the full power equilibrium value of Axial Offset (Axial Offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup but expressed as axial offset, it varies primarily with burnup.

The target (or reference) value of flux difference is determined as follows: At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with control Bank D more than 190 steps withdrawn. This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and the specified deviation of ΔI is permitted from the indicated reference value. During periods where extensive load following is required, it may be impossible to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power, and allowance has been made in predicting the heat flux peaking factors for less strict control at part power.

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 specifications on power distribution are not applicable during physics tests, control rod exercises, or excore calibrations; this 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 difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent for each 2 percent decrease in rated power*. Therefore, while the deviation exists, the power level is limited to 90 percent of rated power* or lower depending on the indicated flux difference.

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*Rated power is defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.

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.

For transient events, the core is protected from exceeding 21.1 kw/ft locally, and from going below a minimum of DNBR of 1.30 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[1] 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

*Rated power is defined here as 2300 Mwt for the normal T_{avg} program and 1955 Mwt for the reduced T_{avg} program.

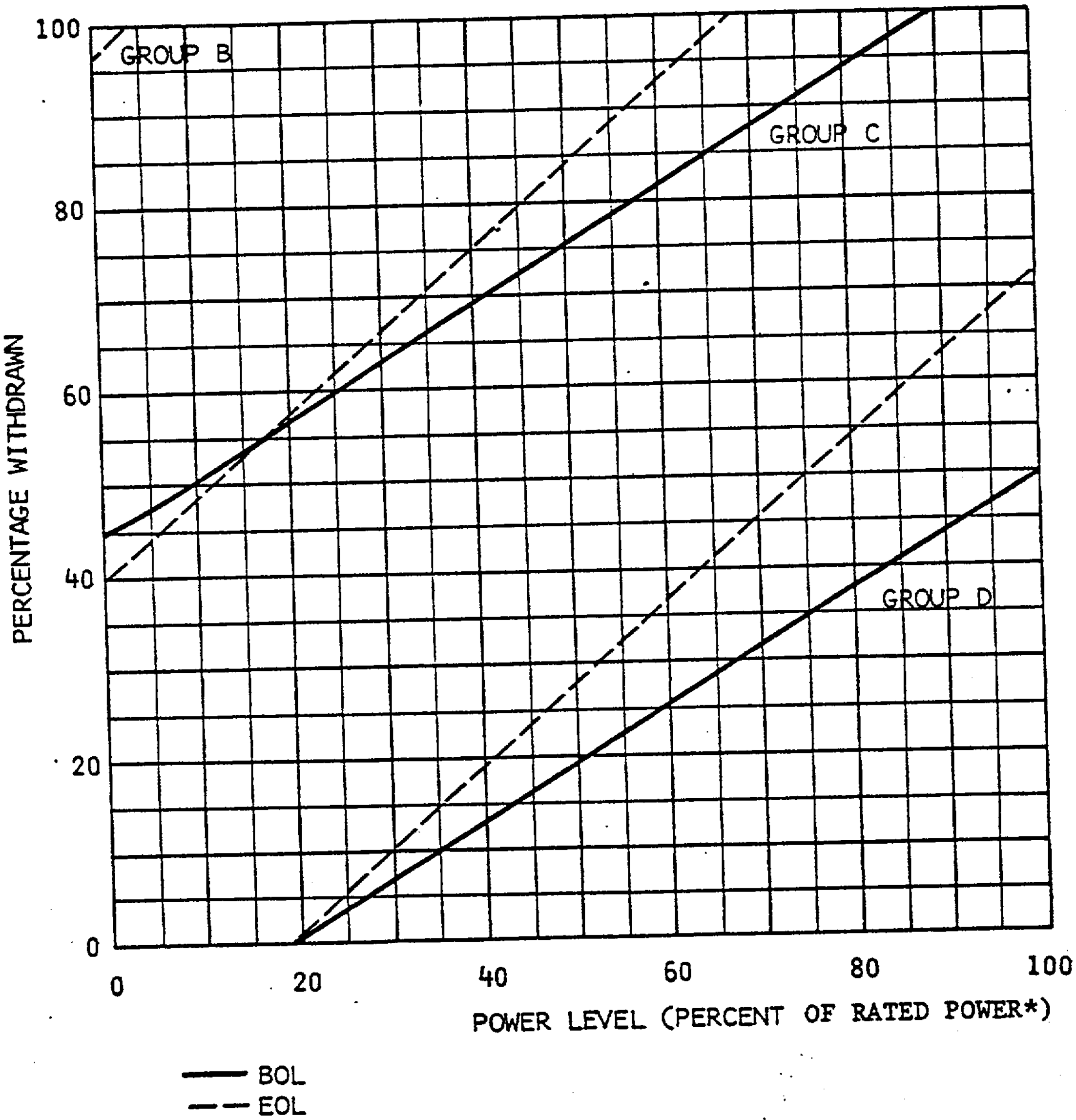
less than the limit, for example, at rated power* even on a worst case basis. When a measurement is taken, experimental error must be allowed for, and 5 percent is the appropriate allowance for a full core representative map taken with the movable incore detector flux mapping system.

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 this 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 affecting F_{Q}^{N} through movement of part length rods, and can limit it to the desired value; (b) while the operator has some control over F_{Q}^{N} through F_{z}^{N} by motion of control rods, he has no direct control over $F_{\Delta H}^{N}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests, can be compensated for in F_{Q}^{N} by tighter axial control, but compensation for $F_{\Delta H}^{N}$ is less readily available.

Quadrant power tilts are based upon the following considerations. The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions, measured as part of the startup physics testing, are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions is consistent with the assumptions used in power capability analyses. It is not intended that extended reactor operation would continue with a power tilt condition which exceeds the radial power asymmetry considered in the power capability analysis.

During normal plant startup, quadrant power tilt ratio may exceed 1.02 due to instrumentation instabilities as a result of rodded configurations

*Rated power is defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.



CONTROL GROUP INSERTION LIMITS FOR
THREE LOOP OPERATION

FIGURE 3.10-1

*Rated Power is defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.

3.11 MOVABLE IN-CORE INSTRUMENTATION

Applicability

Applies to the operability of the movable detector instrumentation system.

Objective

To specify functional requirements on the use of the in-core instrumentation systems, for the calibration of the excore symmetrical offset detection system.

Specification

3.11.1 A minimum of 16 total accessible thimbles and at least 2 per quadrant and sufficient movable in-core detectors shall be operable during recalibration of the excore symmetrical offset detection system.

3.11.2 Power shall be limited to 90% of rated power* for 3-loop or 40% of rated power* for 2-loop operation if recalibration requirements for the excore symmetrical offset detection system identified in Table 4.1-1 are not met.

Basis

The Movable In-Core Instrumentation System[1] has five drives, five detectors, and 46 thimbles in the core. Each detector can be routed to twenty or more thimbles. Consequently, the full system has a great deal more capability than would be needed for the calibration of the excore detectors.

To calibrate the excore detector system, it is only necessary that the Movable In-Core System be used to determine the gross power distribution in the core as indicated by the power balance between the top and bottom halves of the core.

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*Rated power is defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.

After the excore system is calibrated initially, recalibration is needed only infrequently to compensate for changes in the core, due, for example, to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor since it will componsate for an error of 10% in the excore protection system. Experience at the Beznau No. 1 and R. E. Ginna plants has shown that drift due to the core on instrument channels is very slight. Thus, limiting the operating levels to 90% of the rated* two and three-loop powers is very conservative for both operational modes.

Reference

(1) FSAR Section 7.4

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*Rated power is defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.

4.11 REACTOR CORE

Applicability

Applies to surveillance of the reactor core.

Objective

To ensure the integrity of the fuel cladding.

Specification

4.11.1 APDMS Operation

4.11.1.1 Prior to establishing normal operation with APDMS, at least six maps will be taken to determine applicable values of $\overline{R}$ and σ for surveillance thimbles.

4.11.1.2 Plant operation up to rated power* shall be permitted for the purposes of obtaining the initial maps of Specification 4.11.1.1, provided the APDMS is operational and hot channel factors are shown to be below the limiting values set forth in Specification 3.10.2. Suitably conservative values of $\overline{R}$ and σ shall be derived from maps previously run during the current fuel cycle for use in the APDMS system during this initial period.

4.11.1.3 Subsequent updates of $\overline{R}$ and σ shall employ the last six maps in accordance with Specification 4.11.1.1.

4.11.1.4 Each power distribution map will be based on flux traverses obtained from 36 or more of the 46 monitoring channels.

4.11.2 Except during physics tests and EXCORE calibrations, axial surveillance of F(Z)S(Z) shall consist of traverses with the movable incore detectors in appropriate pairs of detector paths, taken every eight hours, or a frequency of approximately 0, 10, 30, 60, 120, 180, 240, 360, and 480 minutes following accumulated control rod motion in any one direction of five.

*Rated power is defined here as 2300 Mwt under the normal T_{avg} program and 1955 Mwt under the reduced T_{avg} program.

4.5.2.2 Acceptable levels of performance shall be that the pumps start, reach their required developed head on recirculation flow and the control board indications and visual observation indicate that the pumps are operating properly. The pumps should be run for at least 15 minutes.

Valves

4.5.2.3 The boron injection tank isolation valves shall be opened and closed one at a time by operator action at intervals not to exceed one month when the reactor coolant system is above the cold shutdown condition. When periods of reactor cold shutdown extend this interval beyond one month, the test shall be performed prior to reactor heatup.

4.5.2.4 The spray additive valves shall be tested with the pumps shut down and the containment spray pump suction valves closed. Each spray additive valve will be opened and closed by operator action at each cold shutdown which extends more than 48 hours but not more often than once each quarter.

4.5.2.5 The accumulator check valves will be checked for operability during each refueling shutdown.

4.5.2.6 The refueling water storage tank outlet valves shall be tested at each cold shutdown which extends more than 48 hours but not more often than once each quarter.

4.5.2.7 When the reactor coolant pressure is in excess of 1,000 psi, it shall be verified at least once per 12 hours (from the RTGB indicators/controls) that the following valves are in their proper position with control power to the valve operators removed.

Valve Number	Valve Position
1- MOV 862 A&B	Open
2- MOV 863 A&B	Closed
3-MOV 864 A&B	Open
4- MOV 866 A&B	Closed

4.5.28 At monthly intervals during power operations each valve (manual, power operated, or automatic) in the safety injection (low and high pressure) and containment spray system flow paths that is not locked, sealed, or otherwise secured in position shall be verified as correctly positioned.

Basis

The Safety Injection System and the Containment Spray System are principal plant safeguards that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes reactor trip, main feedwater isolation and containment isolation, and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be peroframed during annual plant shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. A test signal is applied to initiate automatic action and verification made that the components receive the safety injection in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.(1) (2) (4)

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked each shift and the initiating circuits are tested monthly (in accordance with Specification 4.1). The testing of the analog channel inputs is accomplsihed in the same manner as for the reactor protection system. The engineered safety features logic system is tested by means of test switches to simulate inputs from the analog channels. The test switches interrupt the logic matrix output to the master relay to prevent actuation. Verification that the logic is accomplished is indicated by the matrix test light. Upon completion of the logic checks, verification that the circuit from the logic matrices to the master relay is complete is accomplished by use of an ohmmeter to check continuity. In addition, the active components (pumps and signal valves) are to be tested monthly to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of one month is based on the judgement that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), and that more frequent testing would result in increased wear over a long period of time.

Monthly testing of the safety injection pumps, residual heat removal pumps, containment spray pumps and the boron injection tank isolation valves is not required when in the cold shutdown condition. These components are not required for plant safety when the reactor is in cold shutdown and testing during this condition will result in unnecessary wear on the equipment.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 4.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

The surveillance requirements are provided to ensure that the ECCS valves are in their proper position during operation in which ESF equipment could be required.

References

(1) FSAR Section 6.2
(2) FSAR Section 6.4
(3) FSAR Section 6.1
(4) CP&L report and supplemental letters of September 29, November 5, December 8, 1971, and March 29, 1972.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

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

DOCKET NO. 50-261

1.0 Introduction

By applications dated April 30, 1981, April 30, 1982 and July 13, 1982, and supplemental information dated April 20, 1982 and June 24, 1982, Carolina Power and Light Company (the licensee) requested amendment to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2 (the facility). The amendment requests consist of:

a. Appendix A Technical Specifications (TSs) changes resulting from the analysis of the Cycle 9 reload.

b. Contined approval to operate through Cycle 9 at reduced power.

c. Appendix A Technical Specification (TS) changes resulting from surveillance requirements for ECCS Motor Operated valves.

d. Approval of an Operating License change for steam generator inspection and surveillance.

Carolina Power and Light Company (CP&L), proposes operation of HBR-2 at reduced power, primary temperature flow. Table 1 presents a comparison of rated power and reduced power major plant parameters. The licensee's new analysis was performed by Exxon Nuclear Company

(ENC). The program of reduced temperature, flow and power is proposed to improve the operating conditions of the steam generators, and to allow up to 20% tube plugging. This program is expected to result in a maximum power output of 85% of rated power.

TABLE 1

	Rated Conditions	Cycle #9
Power	2300 Mwt	1955 Mwt
Primary Flow	89965 gpm/loop	82700 gpm/loop
T_{ave}	$575^{o}F$	$537^{o}F$
Primary Pressure	2250 psia	2250 psia
Steam Generator Pressure	800 psig	580 psig

Operation at reduced power and temperature was started during Cycle #8. HBR-2 licensing Amendment No. 61, issued by NRC on November 13, 1981, consisted of changes to the Operating License and Technical Specifications to allow HBR-2 operation at reduced power, primary temperature and flow for the remainder of Cycle #8. This amendment stipulated that if the licensee wished to continue operation at reduced power, primary temperature, and flow after refueling, a detailed transient and accident analysis would have to be submitted for NRC review and approval. The licensee submitted this analysis in Reference (1). Reference (1) includes evaluation of the following anticipated operational occurrences (AOOs) and accidents:

AOO's - Uncontrolled rod withdrawal

- Three reactor coolant pump coastdown
- Loss of external load
- Excess load

Accidents - Loss of Coolant Accident (LOCA)
- locked rotor
- steam line break (SLB)

The following transients and accidents were not initially reanalyzed: startup of an inactive loop, loss of feedwater, loss of A.C. power, chemical and volume control system (CVCS) malfunction, steam generator tube rupture (SGTR) and reduction in feedwater enthalpy accident. Of the above, startup of an inactive loop and reduction in feedwater enthalpy were analyzed in Reference (2) and in the FSAR under full power conditions and showed acceptable consequences. Based on our request, the licensee provided information which discussed the consequences of the following transients at reduced power, temperature and flow: SGTR, CVCS malfunction, loss of offsite A.C. power, and loss of normal feedwater. The SGTR and loss of normal feedwater transients are evaluated in their respective sections. ENC has further indicated that the CVCS malfunction transient consequences are bounded by the rod withdrawal event and that the consequences of the loss of offsite A.C. power event are bounded by the 3 RCP coastdown transient with regard to minimum DNBR and loss of load event with regard to peak pressure. We conclude that operation at reduced power will not adversely affect the consequences of these transients.

2.0 Discussion and Evaluation

2.1 Fuel Design

The reload core design for Cycle 9 utilizes gadolinia as a burnable poison. The reload analysis makes use of gadolinea fuel properties described in Exxon topical report, XN-NF-79-56, which has been reviewed and approved by the NRC staff. Carolina Power and Light has stated that the gadolinia concentration in the fuel will be within those limits specified in our review of XN-NF-79-56. We find this to be acceptable.

2.1.1 Fuel ECCS Analysis

The staff has been generically evaluating three fuel material models that are used in ECCS analyses. Those models predict cladding rupture temperature, cladding burst strain (ballooning), and fuel assembly flow blockage. The staff has (a) discussed its evaluation with vendors and other industry representatives (Ref. 3), (b) published NUREG-0630 (Ref. 4), and (c) required licensees to confirm that their operating reactors would continue to be in conformance with the ECCS Acceptance Criteria of 10 CFR Part 50.46 if the NUREG-0630 correlations were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (Refs. 5 and 6) to offset penalties incurred due to the use of the NUREG-0630 correlations.

Although Exxon has submitted a new ECCS evaluation model (EXEM/PWR, see Ref. 7) that incorporates revised materials models (Ref. 8), the NRC review of the new ECCS evaluation model has not been completed and this model has not been used for the HBR LOCA analysis. Hence, in accordance with the requirements discussed in the preceeding paragraph, the HBR analysis has been augmented by a supplemental ECCS assessment that addresses the predicted effect of NUREG-0630 correlations on the HBR analysis.

In Reference 9, CP&L has provided this supplemental ECCS assessment. For operation at reduced temperature and power, the ECCS analysis of the HBR limiting double-ended cold-leg quillotine break at beginning-of-life conditions predicts reflood rates greater than 1 inch per second and peak cladding temperature (PCT) occurring on the burst node. Hence, reflood heat transfer calculations are performed with the FLECHT correlation and cladding rupture and burst strain models impact PCT analyses only at the burst node.

Exxon has performed sensitivity calculations using the ENC WREM-II PWR and EXEM/PWR ECCS evaluation models. The latter EM is the most recent and is currently under NRC review. It contains (a) cladding models that are slightly modified versions of the NUREG-0630 correlations and (b) various other model revisions such as cladding radiation heat transfer. Exxon has found that, with the new EM, an analysis of a burst-node-limited plant that uses FLECHT heat transfer correlations (such as HBR) will exhibit reduced LOCA PCTs compared with the old EM primarily because of the beneficial effect of the new radiation heat transfer model, which delays fuel rod rupture thus resulting in less cladding inner surface oxidation and the concurrent reduction in heat production associated with the metal-water reaction.

We thus conclude that the inclusion of the NUREG-0630 correlations into the HBR ECCS analysis would not result in predictions that exceed the ECCS Acceptance Criteria. Therefore, the issue of cladding swelling and rupture is resolved for HBR.

2.2 Nuclear Design

Physics parameters remain essentially unchanged from those for previous cycle (Cycle 8) operation at reduced primary coolant temperature and, therefore, are acceptable. However, more detailed information regarding transient and accident analyses was reviewed.

Transient analyses for the uncontrolled control rod withdrawal events from hot zero power and from 1955 MWt were presented in XN-NF-82-18.(Ref. 1). These were reviewed and found to be acceptable. The basis for acceptance in the staff review is that the applicant's analyses of the maximum transients for single error control rod withdrawal from low power and full power conditions have been confirmed, that the analytical methods and input data are reasonably conservative, and that fuel damage limits are not exceeded. The staff concludes that the calculations contain sufficient conservatism, with respect to both assumptions and models, to assure that fuel damage will not result from such control rod assembly accidents.

The staff also requested additional information on the control rod ejection accident which was supplied (Ref. 9). The assumptions and calculational techniques used are the same as those which have previously been evaluated by the staff and found to be acceptable. Since the calculations resulted in peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The radial peak power value at BOC is less than that calculated in the reference analyses and is, therefore, acceptable. However, at EOC conditions, a peak radial power about 8 percent above the reference calculation peaking factor prior to ejection is calculated. This 8 percent increase, however, is more than offset by the 15 percent reduction in reactor operating power for Cycle 9. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained during a control rod ejection transient.

2.3 Thermal-Hydraulics

To support the reduced temperature program, the licensee has performed a review of anticipated operational transients at the proposed operating conditions and reactor protection system setpoints. The thermal-hydraulic calculations for the steady-state conditions at the reduced power and coolant temperature have shown about a 65 percent increase in MDNBR as compared to the rated full load operating conditions. Based on this substantial increase in thermal margin, the licensee concludes that the anticipated operational transients will satisfy the Specified Acceptable Fuel Design Limits (SAFDLs) since the changes in MDNBR during these transients will not be greater than those previously evaluated for rated full power. The staff agrees with this conclusion although additional information for certain reactivity initiated transients (discussed below) were requested.

For large steam line break analysis, the modified Barnett critical heat flux (CHF) correlation (Ref.10) is employed for DNBR calculation. However, no DNBR limit, which will ensure avoidance of a fuel rod experiencing DNB with 95 percent probability at 95 percent confidence level, was described in XN-NF-82-18 (Ref. 1). In a telecommunication (Y. Hsii of NRC and J. C. Chandler of ENC on June 9, 1982), Exxon indicated the DNBR limit for the modified Barnett correlation was 1.135. This 95/95 DNBR limit was developed from the CHF data presented in the Appendix A of Reference 10using the Non-Parametric Tolerance Limit Method (Ref. 11). Our evaluation has found that the modified Barnett correlation with a DNBR limit of 1.135 is acceptable for the steam line break analysis based on the following observations: (1) The non-parametric method is a distribution-free tolerance limit determination method with no assumption of normal distribution regarding the measured-to-predicted CHF ratio data. Therefore, it is a proper method for determining the DNBR limit. (2) The modified Barnett correlation has been approved in 10 CFR Part 50, Appendix K as an acceptable CHF correlation for LOCA analysis. We conclude that it is also acceptable for the steam line break transient analysis where the primary system pressure falls within the pressure range of 150 to 725 psia of the modified Barnett correlation. (3) The DNBR limit of 1.135 is determined with 95/95 probability/confidence level from the existing CHF data described in Reference 10.

3.0 Anticipated Operational Occurrences

3.1 Three Rector Coolant Pumps (RCP) Coastdown

This analysis assumed loss of power to all three RCPs at 1955 Mwt power level, beginning of cycle reactor kinetics coefficient, and reactor trip on low flow signal (more conservative than the more realistic assumption of reactor trip due to bus undervoltage or underfrequency). A multiplier of 0.8 was applied to the Doppler coefficient for conservatism. The pressurizer was assumed to be in automatic control with pressurizer spray available. While this takes credit for non-safety grade equipment, it is more conservative with regard to DNBR prediction, since actuation of the pressurizer spray results in a lower DNBR. The minimum DNBR was 2.58 at 3.5 seconds. The peak primary pressure is bounded by the loss of external load event (see item 3 below). We conclude that this analysis is acceptable.

3.2 Excess Load

This analysis assumed increase in turbine load causing a power mismatch between reactor power and steam generator demand. A 10% step increase in rated turbine load was analyzed at an initial power of 1955 Mwt, end of core life, with no automatic control rod or pressurizer control assumed. Core power reached 2115 Mwt after 42 seconds. Minimum DNBR was 2.79 at 51 seconds. Both primary and secondary pressure decreased. We conclude that this analysis is acceptable.

3.3 Loss of Load

This analysis assumed a tubine trip without a direct reactor trip, an initial power level of 1955 Mwt at beginning of core life, thus providing a positive moderator coefficient. For conservatism, a multiplier of 0.8 was applied to the Doppler coefficient. No credit was taken for automatic reactor control, steam dumps and tubine bypass. However, the initial reanalysis assumed that pressurizer spray and the power relief valves (PORVs) were operational. This assumption was conservative for DNBR prediction because of lower pressures as a result of pressurizer spray and PORV actuation, but not for predicting peak pressure. Reactor trip on high pressure occurred in 12.5 seconds, and primary pressure peaked at 2460 psia in 14 seconds. By comparison PORV actuation is at 2335 psig and primary safety valve actuation at 2485 psig. The minimum DNBR was 2.91.

Based on our request this transient was reanalyzed for peak primary system pressurization (Ref. 9). In this reanalysis, the PORVs and pressurizer spray were assumed inoperable. The predicted peak primary pressure was 2585 psia. The primary safety valves would be actuated. There was no decrease in DNBR from its original value. Therefore, we conclude that this analysis is acceptable..

3.4 Loss of Normal Feedwater

This event as analyzed in the original FSAR and was not reanalyzed in references (1) and (2). The FSAR analysis indicated that for rated power conditions T(average) peaked at 605°F approximately 1500 sec after initiation of the transient, and that there was no water relief from the pressurizer relief or safety valves. Based on our request for additional information, the licensee provided an estimate of the results of this transient during reduced power and primary temperature operation, which predicts a maximum T(average) of 608°F, and pressurizer safety valve actuation, resulting in expulsion of 140 cubic feet of primary fluid. The time after transient initiation for occurrence of these events was not given.

These analyses were based on the assumptions of a reactor trip on steam flow/feedwater flow mismatch coincident with steam generator low water level or on low-low steam generator level, natural circulation in the primary loops, one auxiliary feedwater pump starting at one minute and delivering 300 gpm to two steam generators, no credit for steam dump valves, and steam generator safety valve actuation. These assumptions are conservative. The licensee stated that there would be no fuel damage since about 850 cubic feet of liquid remains above the core, and that sufficent auxiliary feedwater capacity exists to remove decay heat. Results of loss of mainfeedwater analyses for other Westinghouse plants at full power conditions also indicate that there is no DNBR problem. The licensee has further indicated that the DNBR for this event is bounded by the DNBR for the 3 reactor coolant pump coastdown transient (See Section 3.0). We conclude based on our review of other plants as well as the H. B. Robinson 2 submittal, that DNBR will remain acceptable.

However, since the licensee's analysis is unrealistically conservative and may mask other effects in the transient, we require that the licensee perform a more detailed analysis for this transient. The results of this analysis should include plots of T(average), primary and secondary pressure versus time for the full extent of the transient, and the value for the minimum DNBR attained. These results should be submitted to NRC by October 31, 1982.

4.0 Accidents

4.1 LOCA

A new LOCA ECCS analysis for only the limiting break was performed for the HBR-2 reduced power and primary temperature operation. The licensee states that the analysis was performed in accordance with 10 CFR Part 50, Appendix K, for the limiting double ended cold leg guillotine break at beginning of life fuel conditions. Previous analyses showed this to be the limiting break with regard to peak cladding temperature (PCT) (see References 12 and 13). The EMC WREM-IIA model was utilized. A discharge coefficient (C_D) of 0.8 was assumed, as previous analysis had shown this to be conservative. (see Reference 14). Loss of offsite power was assumed.

The ENC analysis identifies a number of detrimental effects for the reduced power, temperature and flow operation as compared to rated conditions for the LOCA consequences. These included: reduced heat transfer during blowdown because of decreased core flow; a slower core power decay due to reduced voiding; reduced reflood rates due to lower containment pressure; longer blowdown times because of reduced saturation pressures with lower pressures earlier in the blowdown, which in turn result in earlier accumulator injection and flow for a longer time during blowdown, with consequent greater loss of accumulator inventory, since 10 CFR Part 50 Appendix K requires all ECCS coolant injected during blowdown to be assumed lost. Nevertheless, the reduction in linear heat generation rate associated with the 15% reduction in power more than offsets these detrimental effects and results in a PCT of 2077^{O}F compared with a PCT of 2185^{O}F for a LOCA at full power and at rated temperature and flow. The maximum local metal-water reaction is 6.05% and total core-wide metal-water reaction is less than 1%, thus meeting the requirements of 10 CFR Part 50.46.

Based on our request, the licensee provided information (Ref. 9) which indicates that consideration of the cladding swelling and rupture model in NUREG-0630 would not adversely affect prediction of PCT (discussed in Section 2.1.1). We conclude that the LOCA analysis at reduced power and temperature is acceptable.

4.2 Locked Rotor

This analysis assumes three loop operation at 1955 Mwt, with instantaneous seizure of one RCP. The reactor is tripped by the resulting low flow signal. The feedwater pumps were assumed to trip with the reactor, but offsite power is retained and continued operation of the intact RCPs is assumed. Beginning-of-cycle reactor kinetics coefficients are assumed. A 0.8 multiplier is applied to the Doppler coefficient for conservatism. A 0.95 multiplier was applied to the DNBR to account for assymetric core flow because of loop flow differences due to steam generator tube plugging. Based on these assumptions, the minimum predicted DNBR is 2.19 and peak primary pressure is 2321 psig. We conclude that this analysis is acceptable.

4.3 Steam Line Break (SLB)

The SLB was reanalyzed for the most severe case i.e., an SLB inside containment at end of core life and at hot zero power conditions, corresponding to a core average temperature of 530°F. At this time the steam generator secondary side inventory is at a maximum, prolonging the duration and increasing the magnitude of the primary loop cooldown. For additional conservatism, offsite power is assumed available, the most reactive control rod is assumed to be stuck out of the core, the break is assumed to occur at the steam generator with the fewest plugged tubes and blowdown occurs also from the other two steam generators until closure of the main steam isolation valves.

The analysis shows very rapid loss of both primary and secondary pressure when compared to other SLB analyses on similar PWRs. The faulted steam generator is almost completely depressurized in 1-2 seconds and primary pressure decreases to about 250 psia in 50 seconds. In addition the licensee's analysis shows that the core returns to power at 7.5 seconds. These results appear to be inconsistent with analyses for other Westinghouse plants which show a much slower depressurization of the faulted steam generator and considerably higher minimum primary pressure. The peak power reached is approximately 940 Mwt at 43 seconds, after which boron addition terminates the power increase. The minimum critical heat flux (CHF) is calculated to be 1.19 at the time of peak core heat flux, utilizing the modified Barnett CHF correlation (discussed in Section 2.3). This value appears adequate based on a minimum acceptable CHF of 1.135. Discussions with the licensee indicated that the SLB model utilized does not consider asymmetric core temperatures, nor the mass input and RCS cooldown due to accumulator actuation or SIS input. The analysis does assume the boron addition from high pressure SIS to shutdown the reactor after its return to criticality due to the cooldown. The model utilized appears to provide conservative values and the resulting CHF appears acceptable. Therefore, we conclude, based on our review of MSLBs at other W plants and our review of the H. B. Robinson information, that the consequences of a MSLB at reduced power and temperature will not result in unacceptable fuel performance. However, since the licensee's analysis is excessively conservative and does not assume the mass input from the SIS, the analyses may mask important system effects. Therefore, we require that the licensee provide additional information that justifies the adequacy and conservatism of the model utilized in the SLB analysis, prior to the next refueling.

4.4 Steam Generator Tube Rupture (SGTR)

This event was analyzed in the original FSAR and was not reanalyzed in Reference (1) and (2). Based on our request, the licensee provided information which indicates that, despite the larger initial primary to secondary pressure differential, total primary to secondary leakage is estimated to be 4000 lbs. less for reduced power operation than for full power operation, and thus the consequences of this accident would be less severe. The consequences of this accident at rated conditions was previously reviewed and found acceptable. We conclude that the consequences of this event at reduced power conditions are acceptable .

5.0 Technical Specifications

5.1 Reduced Temperature Program

For the reduced temperature program, the licensee proposes changes to the technical specifications (Ref. 15). These changes include:

5.1.1 The peak F_Q (including uncertainties) assumed for Cycle 9 operation is revised to 2.32 at 85% of rated power. The revised F_Q limit of 2.32, corresponding to a linear heat generation rate of 11.8 KW/ft, is used in the LOCA ECCS analysis for reduced temperature operation and results in acceptable consequences. For additional analyses of the more limiting transients for reduced temperature operation, a more conservative value of 2.55 is used, also with acceptable consequences. The revised F_Q limit is bounded by the value used in the LOCA and other limiting transient analyses and is, therefore, acceptable.

5.1.2 The terms "rated power", "full power", "rated values", and "design values" are redefined under the reduced temperature program with power operation at 1955 MWt. The identification of the power level that various Limiting Conditions of Operation (LCO) are related to during the reduced temperature operation is primarily for clarification and is acceptable.

5.2 Additional Technical Specification Change

By application dated April 30, 1981, the licensee requested a change in the Technical Specifications to require specified surveillance of the Emergency Core Cooling System (ECCS) Motor Operated Valves which is required as a result of modifications to the ECCS electrical control circuits.

These changes were requested by our letter dated March 9, 1981 which suggested acceptable surveillance. The licensee responded to our request and used our suggested surveillance. Therefore, this change is acceptable.

6.0 Licensing Condition

By letter dated July 13, 1982, the licensee requested a modification to the Operating License Condition 3.I.a, b, c & d.

6.1 Steam Generator

As a result of a high level of stress corrosion cracking activity above the tubesheet area observed during August 1981, license conditions were imposed for the balance of Cycle 8 operation which included periodic primary to secondary hydrostatic tests, and more stringent limits on allowable primary to secondary leakage. The eddy current inspection results performed during the current outage indicates that reduced temperature operation since November 1981 has been successful in sharply reducing stress corrosion cracking activity above the tubesheet. The licensee plans to continue reduced temperature operation (T_{av} = 537°F) during the next cycle. For this reason, the staff has concluded that there is reasonable justification for not reimposing the license condition for periodic hydrostatic tests during the next operating cycle. Stress corrosion cracking and intergranular attack continues to be active within the tubesheet crevice region. However, the narrow tube to tubesheet crevices or gaps severely limit the potential for any high leakage such as could occur as a result of a rupture in free span portions of tubing (i.e., above the tubesheet). The licensee has proposed to continue the license condition for reduced limits on primary to secondary leakage which were imposed for the balance of Cycle 8 operation following August 1981.

Eddy current inspections have indicated an accelleration of phosphate wastage corrosion during the past operating cycle. By letter dated July 13, 1982, the licensee has proposed a licensing change which would require shutdown of H. B. Robinson within 6 EFPM of restart from the current outage for additional steam generator inspections to ensure that further progression of wastage does not become excessive. The licensee provided the staff with the eddy current inspection results, eddy current error estimates, and projected corrosion rates for the next cycle of operation to justify six months operation. This information is still being reviewed by the staff. However, based upon our preliminary findings, we have concluded that H. B. Robinson can be operated safely for at least three EFPM in a manner reasonably consistent with the criteria (per Regulatory Guide 1.121) which the staff generally employs for this type of evaluation. We plan to complete our evaluation of the licensee's proposed six EFPM operating interval by September 3, 1982. Operation beyond three EFPM to six EFPM as proposed by the licensee will be subject to approval by the staff.

7.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of enviornmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: July 23, 1982

References

1. Exxon Nuclear Company (ENC) Report XN-NF-82-18 "ECCS and Plant Transient Analyses for H. B. Robinson Unit 2 Reactor Operating at Reduced Primary Temperature" March 1982.

2. ENC Report XN-75-14 "Plant Transient Analysis of the H. B. Robinson Unit 2 for 2300 Mwt" July 1975.

3. R. P. Denise (NRC) memorandum for R. J. Mattson, "Summary Minutes of Meeting on Cladding Rupture Temperature, Cladding Strain, and Assembly Flow Blockage," November 20, 1979.

4. D. A. Powers and R. O Meyer, "Cladding Swelling and Rupture Models for LOCA Analysis," NRC Report NUREG-0630, April 1980.

5. D. G. Eisenhut (NRC) letter to All Operating Light Water Reactors, November 9, 1979.

6. H. R. Denton (NRC) memorandum for Commissioners, "Potential Deficiencies in ECCS Evaluation Models," November 26, 1979.

7. "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon report XN-NF-82-20, March 1982.

8. "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon report XN-NF-82-07, March 1982.

9. S. Zimmerman (CP&L) letter to S. Varga (NRC), June 24, 1982.

10. E. D. Hughes, "A Correlation of Rod Bundle Critical Heat Flux For Water in the Pressure Range 150 to 725 PSIA," Idaho Nuclear report IN-1412, July 1970.

11. P. N. Sommerville, "Tables for Obtaining Non-Parametric Tolerance Limits," Annals of Mathmetical Statistics, Vol. 29, No. 2, pp. 599-601, 1958.

12. ENC Report XN-75-57 "HBR-2 LOCA Analyses Using the ENC WREM Based PWR ECCS Evaluation Model" October 1975.

13. ENC Report XN-75-57 "HBR-2 LOCA Analyses Using the ENC WREM Based PWR ECCS Evaluation Model" Rev. 1, November 1975.

14. ENC Report XN-76-54 "LOCA Analyses for HBR-2 Using WREM Based PWR ECCS Evaluation With Reduced LPSI Flow, Steam Generator Plugging and Increased Upper Head Temperature" December 1976.

15. B. J. Furr (CP&L) letter to S. A. Varga (NRC), April 30, 1982.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 71 to Facility Operating License No. DPR-23 issued to Carolina Power and Light Company (the licensee), which revised Technical Specifications for operation of the H. B. Robinson Steam Electric Plant, Unit No. 2 , (the facility) located in Darlington County, South Carolina. The amendment is effective as of the date of issuance.

The amendment authorizes Cycle 9 operation at a reduced power level; revises the Appendix A Technical Specifications to: (a) incorporate changes resulting from the Cycle 9 reload core analysis, including administrative changes, and (b) incorporates changes to include specific surveillance of the emergency core cooling system motor operated valves; and revises the Operating License Condition 3.I.a, b, c & d to include a steam generator inspection and steam generator tube leakage criterion.

The applications for amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since this amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated April 30, 1981, April 30, 1982, and July 13, 1982 (as supplemented April 20, 1982 and June 24, 1982), (2) Amendment No. 71 to License No. DPR-23, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Hartsville Memorial Library, Home and Fifth Avenuves, Hartsville, South Carolina 29550. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 23rd of July 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing