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December 3, 1976

Docket No. 50-261

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Senior Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The Commission has issued the enclosed Amendment No. 25 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. The amendment consists of a change to the Technical Specifications based upon review of your submittal of a reevaluation of ECCS performance in response to the Commission's Order of August 27, 1976. This change has been discussed with your staff. Effective upon issuance of this amendment, the Commission's Order for Modification of License dated August 27, 1976, relative to Facility Operating License No. DPR-23, is terminated.

The amendment revises the limit on the total nuclear peaking factor (F_Q).

In connection with refueling for Cycle 5, you stated in your letter of September 2, 1976, that on the basis of your evaluation there were no unreviewed safety questions associated with the refueling and therefore the refueling was a plant modification permitted under 10 CFR §50.59. At our request you also submitted Exxon Nuclear Company (ENC) Report XN-76-39 in support of your position. Upon reviewing XN-76-39, a question arose regarding the validity of your conclusions inasmuch as certain methods used to perform the physics calculations reported in XN-76-39 had been modified from previously approved methods. We have, therefore, reviewed these revised physics calculational methods. As a result of our review, we conclude that the revised methods are acceptable as they apply to your facility and XN-76-39. Accordingly, we conclude that their use in preparing XN-76-39 did not involve an unreviewed safety question.

cont'd

Copies of the Safety Evaluation performed for the ECCS reevaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 25
2. Safety Evaluation
3. Federal Register Notice

cc w/enclosures: See next page

12/4/76

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Carolina Power & Light Company

cc w/enclosure(s):
G. F. Trowbridge, Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Hartsville Memorial Library
Home and Fifth Avenue
Hartsville, South Carolina 29550

Mr. McCuen Morrell, Chairman
Darlington County Board of Supervisors
County Courthouse
Darlington, South Carolina 29532

John D. Whisenhunt, Esq.
Bridges and Whisenhunt
Bridges Building
P. O. Box 26
Florence, South Carolina 29501

cc w/enclosures & incoming
dated: 11/17/76 & 12/2/76
Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

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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. 25
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 November 17, 1976, as supplemented December 2, 1976, 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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 3, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 25

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A Technical Specifications as follows:

Remove Page	Insert Page
3.10-2	3.10-2

The changed area on the revised page is shown by a marginal line.

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:

$F_Q(Z) \leq (2.20/P) \times K(Z)$ for $P > .5$

$F_Q(Z) < (4.40) \times K(Z)$ for $P \leq .5$

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

where P is the fraction of licensed power at which the core is operating, K(Z) is the function given in Figure 3.10-3, and Z is the core height location of F_Q.

3.10.2.1.1 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:

$$[F_j(Z)S(Z)]_{max} \leq \frac{2.085/P}{\bar{R}_j(1+\sigma_j)}$$

where:

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

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

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

F^N_{qi} is the value obtained from a full core map without the measurement uncertainty factor F^N_u. The quantity $F(Z)_{ij}S(Z)$ is the measured value without inclusion of the instrument uncertainty factor F^a_q. Those uncertainty factors, $F^N_u = 1.05$ and $F^a_q = 1.02$, have been included in the limiting value of 2.085/P.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 25 TO LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

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

DOCKET NO. 50-261

Introduction

On August 27, 1976, the Commission ordered amendment of the H. B. Robinson Unit No. 2 (Robinson-2) Facility Operating License No. DPR-23 by adding the following new provisions:

1. As soon as possible, the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with an approved Westinghouse Evaluation Model, with appropriate correction for upper head water temperature.

2. Until further authorization by the Commission, the Technical Specification limit for total nuclear peaking factor (F_Q) shall be reduced to 2.05.

The reason for this order was the acquisition of experimental data indicating the temperature of the water in the upper head region of reactors designed by the Westinghouse Electric Corporation (Westinghouse) was higher than had been assumed in the Emergency Core Cooling System (ECCS) analyses previously performed pursuant to the requirements of 10 CFR 50.46 and the fact that this higher temperature could have the effect of increasing the calculated peak clad temperature in the event of a loss of coolant accident. The reduction in the limiting value of the total nuclear peaking factor (F_Q) was based on a conservative estimate of the magnitude of reduction required to assure that ECCS criteria would not be exceeded with the revised assumption of upper head temperature.

In conformance with the order, the Carolina Power and Light Company (the licensee) has submitted by letters dated November 17 and December 2, 1976, a reevaluation of ECCS cooling performance calculated in accordance with an approved Westinghouse Evaluation Model with appropriate correction for upper head water temperature. In addition, however, the licensee has also included in his submittals a similar reevaluation calculated in accordance with an Exxon Nuclear Company (ENC) Evaluation Model which was similar to the Robinson ENC Evaluation Model previously approved by the staff with the exception of a change to one portion of the model which was included for the first time in this submittal. This change to a portion of the model will be described and evaluated in the body of this safety evaluation.

The Commission's Order of August 27, 1976, did not require submission of a reevaulation using an approved ENC Evaluation Model because the ECCS evaluations based on the Robinson-2 core composition at that time demonstrated that the total nuclear peaking factor limits established for the Westinghouse fuel in the core were also conservative for the nuclear fuel in the core supplied by ENC (At the time of the order the core contained approximately 2/3 Westinghouse fuel and 1/3 ENC fuel.). Since issuance of the order, however, Robinson-2 has initiated refueling for Cycle 5. At the conclusion of refueling the core will contain approximately 1/3 Westinghouse fuel and 2/3 ENC fuel. Accordingly, it would not be possible to demonstrate that the operating limits determined for the new core composition using the Westinghouse model alone would still be conservative for the ENC fuel. It is for this reason that the licensee also submitted an evaluation of ECCS performance for Cycle 5 using an approved ENC evaluation model.

In the course of preparing the present ECCS analyses the licensee determined that the values assumed for the low pressure safety injection (LPSI) flow rates in the preceding ECCS evaluation (discussed in Amendment No. 16 to the Facility License, dated December 3, 1975) were in error and that the error was such as to cause the calculated peak clad temperature (PCT) following a postulated loss of coolant accident (LOCA) to be underestimated. While the present analyses correct this error, its presence in previous large break ECCS analyses invalidates those analyses and invalidates the bases which permitted continued operation of Robinson-2 under the Commission's Order of August 27, 1976 (the Order). However, since the present amendment is based upon updated and corrected ECCS evaluations performed using models wholly conforming to 10 CFR 50.46 and Appendix K, as described below, the requirements of the Order have been satisfied and effective upon issuance of this amendment, the Order terminates.

Discussion

As noted above the Robinson 2 core will contain during Cycle 5 fuel supplied by both Westinghouse and ENC. Since different evaluation models are employed by the suppliers of these fuels, the analyses will be discussed separately.

Westinghouse Fuel

The analysis submitted by the licensee for the Westinghouse fuel was performed using the previously approved October 1975 version of the Westinghouse Evaluation Model. Basic parameters used in the reanalysis were the same as used in the previous analyses submitted in 1975 except for the following:

1. The temperature of the fluid in the upper head region was assumed to be equal to the hot leg temperature.

2. Coolant flow rates during low pressure safety injection were changed to their correct (lower) rates.

3. Since all Westinghouse fuel contained in Cycle 5 is twice-burned, credit was taken for the effect of fuel burnup. The minimum fuel rod burnup used in the analysis (based on actual reactor operation) was 11,456 MWD/MTU.

4. Six percent of the tubes were assumed to be plugged in each steam generator.

5. Minor refinements were made in containment heat sink data based on operating experience at Robinson-2.

The reactor power level used in the reanalysis was unchanged from the proposed Robinson 2 uprated value of 2300 MWt. The total nuclear peaking factor (F_Q) value employed was 2.30.

A reanalysis of the small break LOCA was not required. The small break analysis submitted by the licensee on October 2, 1974 remains valid. The small breaks are essentially insensitive to the fluid temperature in the upper head region and to steam generator tube

plugging and are unaffected by small changes in low pressure safety injection flow rates. Consequently, these considerations would not affect the acceptability of the model for small breaks. The highest peak clad temperature (PCT) predicted for the worst small break in the October 2, 1974 submittal was 1926°F. We agree with the licensee that reanalysis of the small break LOCA is not required.

For the large break LOCA the licensee submitted analyses for the following breaks: 0.4 and 0.6 C_D Double Ended Cold Leg Guillotine (DECLG) and a 1.0C_D Double Ended Cold Leg Split (DECLS). The maximum calculated PCT of 1979°F occurred for the DECLG with C_D = 0.4. The maximum zirconium oxidation also occurred for this break, with 4.14% being the maximum local value and less than 0.3% being the core averaged value.

While the licensee has only analyzed three breaks he states that the guillotine break in the cold leg pump discharge piping is the most limiting break type and location on the basis of the Westinghouse sensitivity study for three loop plants[1] which considers the effect of upper head temperature on the type and location of the limiting break. We have reviewed this report (WCAP-8853) and found it acceptable. We therefore agree with its conclusion that for the type of plants considered, a change in upper head fluid temperature does not change the type or location of the worst break. Robinson-2, however, is a three loop plant utilizing 15 x 15 fuel elements and therefore WCAP-8853, which addresses three loop plants having 17 x 17 fuel elements may not be directly applicable to the Robinson-2 analyses. To support his use of this report to confirm his identification of the worst break type and location the licensee states that particular details of plant design do not alter the basic effect of upper head water temperature, i.e., the flashing of water at the hot leg saturation pressure versus flashing at cold leg saturation pressure does not change the limiting break type and location. In addition the licensee cites several previous sensitivity studies for three loop plants which have been consistent in identifying this break type and location as limiting.

We agree that the conclusions of WCAP-8853 are applicable to Robinson-2, and therefore agree that he has identified the worst break.

Regarding the assumption of 6% tube plugging in each steam generator, this assumption is in excess of the actual percentage of tubes presently plugged. The licensee has not demonstrated thatthis is a conservative assumption in his submittal for Westinghouse fuel. The assumption is addressed, however, for the ENC fuel which experiences significantly higher PCT's and is demonstrated to be conservative.

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1 Julian, H. V., Tabone, C. J. and Thompson, C. M., "Westinghouse ECCS - Three Loop Plant (17x17) Sensitivity Studies," WCAP-8853, September, 1976.

We therefore conclude that the analysis of ECCS performance submitted by the licensee for the Westinghouse fuel in Robinson 2 during Cycle 5 has been performed with an approved evaluation model and the results demonstrate conformance with the criteria contained in 10 CFR §50.46.

ENC Fuel

The analysis submitted by the licensee for the ENC fuel was performed using the ENC WREM-Based PWR Evaluation Model. The licensee states that the basic evaluation capabilities remain unchanged from the approved ENC WREM model used for the Robinson-2 analysis submitted in 1975, except that a phase separation model was used in the upper head region and certain minor revisions were made to correct errors, increase flexibility and improve running efficiency. Also basic parameters used in the reanalysis are the same as used in the previous analyses submitted in 1975 except for the following:

1. The temperature of the fluid in the upper head region was assumed to be equal to the core outlet temperature.

2. Coolant flow rates during low pressure safety injection were changed to their correct (lower) rates.

3. Input flows, resistance coefficients, flow areas, and junction elevations were refined based on detailed plant information.

4. Six percent of the tubes were assumed to be plugged in each steam generator.

The reactor power level used in the reanalysis was unchanged from the proposed Robinson 2 uprated value of 2300 MWt. The total nuclear peaking factor (F_Q) value employed was 2.20.

A reanalysis of the small break LOCA was not required. The small break analysis submitted by the licensee on November 13, 1975 remains valid. The small breaks are essentially insensitive to the fluid temperature in the upper head region and to steam generator tube plugging and are unaffected by small changes in low pressure safety injection flow rates. Consequently, these considerations would not offset the acceptability of the model for small breaks. The highest peak clad temperature (PCT) predicted for the worst small break in the November 13, 1975 submittal was 1457°F. We agree with the licensee that reanalysis of the small break cases is not required.

For the large break analysis the licensee submitted results for the following breaks: 0.6, 0.8 and 1.0 C_D Double Ended Cold Leg Guillotine (DECLG) and 0.8 and 1.0 C_D Double Ended Cold Leg Split (DECLS). The maximum PCT of 2152°F occurred for the DECLS with C_D = 0.8. The maximum zirconium oxidation also occurred for this break, with 7.89% being the maximum local value and less than 1% being the core averaged value.

From the spectrum of large breaks submitted we conclude that the licensee has identified the most limiting break, i.e. the DECLG with C_D = 0.8.

Regarding the licensee's assumption of 6% tube plugging in each steam generator, this assumption is in excess of the actual percentage of tubes presently plugged. To demonstrate that this is a conservative assumption the licensee repeated the analysis of the DECLG with C_D = 1.0, with 0% tube plugging. For the case of no tube plugging, the calculated maximum PCT was lower by 33°F. Based on these results and generic studies on the effect of steam generator tube plugging on PCT, we conclude that the assumption of 6% tube plugging is conservative and therefore acceptable.

Regarding the licensee's use of phase separation noding in the upper head region, the previous ENC model for the blowdown calculation assumes a homogeneous fluid model that was initially at the cold leg temperature. For the present analysis the licensee has proposed a phase separation model with the fluid temperature initially at the hot leg temperature. The phase separation model incorporated in RELAP4 is used with a bubble density gradient of 0.8 and a bubble rise velocity of 3 feet/second. The flow path junction at the top of the control rod guide tubes has also been modeled at the proper elevation. This flow path is the primary connection between the upper head and upper plenum.

The licensee has submitted sensitivity studies to show the effect on the calculated peak cladding temperature (PCT) for the proposed model changes. The postulated 1.0 DECLG break was evaluated at the hot leg temperature using both the phase separation and homogeneous fluid models. The PCT for the homogeneous model was lower by approximately 12°F. A separate evaluation made by Exxon and referenced by the licensee[2] indicated that it was conservative to assume the hot leg temperature rather than a temperature equal to the average for the hot and cold leg temperatures. Thus, both model changes are in the conservative direction.

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[2] G.F.Owsley (ENC) letter to D.L. Ziemann (NRC) of November 30, 1976.

We therefore conclude that use of a phase separation fluid model at the hot leg temperature provides a conservative model that is satisfactory for the H.B. Robinson upper head region.

Based on the foregoing we conclude that the analysis of ECCS performance submitted by the licensee for the ENC fuel in Robinson 2 has been performed with an approved evaluation model and the results demonstrate conformance with the criteria contained in 10 CFR §50.46. One of the constraints of the analysis of the ENC fuel, however, is the assumption of a total nuclear peaking factor (F_Q) value of 2.20. This is less than the value of 2.30 used in the analysis of the Westinghouse fuel and is also less than the value of 2.30 presently given in the facility Technical Specifications. Accordingly, to assure conformance with 10 CFR 50.46 criteria, this amendment will revise the technical specification limit on F_Q downward to a value of 2.20.

We have determined that this 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 environmental 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 the 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 accidents previously considered and does not involve a significant decrease in a safety margin, 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.

Dated: December 3, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 25 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, Hartsville, South Carolina. The amendment is effective as of its date of issuance.

The amendment revises the limit on the total nuclear peaking factor (F_Q) and terminates the Commission's Order for Modification of License dated August 27, 1976.

The application for the amendment complies 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 the 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 application for amendment dated November 17, 1976, as supplemented December 2, 1976, (2) Amendment No. 25 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 Avenues, Hartsville, South Carolina.

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 Operating Reactors.

Dated at Bethesda, Maryland, this 3rd day of December 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors