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