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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION UNIT 2

DOCKET NO. 50-281

Introduction

By letter dated May 31, 1979 (Reference 1), as supplemented October 16, 1979 (Reference 2), October 25, 1979 (Reference 3), January 11, 1980 (Reference 4) and February 20, 1980 (Reference 5), Virginia Electric and Power Company (the licensee) requested an amendment to Operating License No. DPR-37 for Surry Power Station, Unit 2. References 1 and 2 contain a LOCA analysis and proposed Technical Specification changes in connection with the operation of Unit 2 with 3 percent of steam generator tubes plugged, with modifications made during the steam generator replacement operation and with a peaking factor F_Q of 2.19.

The changes to the Technical Specifications for Unit 2 requested by the licensee are the following:

(a) Change of the heat flux hot channel factor, F_Q to 2.19, the limiting enthalpy rise hot channel factor to 1.55 and the corresponding limiting assembly enthalpy rise factor to 1.476 for plant operation with 3 percent of steam generator tubes plugged.

(b) Change of the Hot Channel Factor Normalized Operating Envelope for Unit 2 (TS Fig. 3.12-8b).

(c) Removal of the specifications for the augmented surveillance of core power distribution and change of the axial flux difference limits (TS Fig. 3.12-10).

Since the limiting value of F_Q is below the level at which the excore detectors could provide reliable readings, the licensee has analytically predicted the maximum values for the total peaking factor, $F_Q(Z)$ using the "3 case analysis" methodology (Reference 6). These predicted values lie below the LOCA predicted limits (Reference 5) and augmented power distribution surveillance is therefore not required during Cycle 5 operation.

Evaluation

The licensee has provided an evaluation of the performance of Emergency Core Cooling System (ECCS) for Unit 2 corresponding to the hot channel peaking factor value of F_Q=2.19 and assuming a steam generator plugging level of 3 percent. In addition, the following changes in plant operational parameters were introduced into the LOCA analysis:

(a) Change of the Low Head Safety Injection flow due to NPSH consideration.

(b) Modification of the Containment Spray System.

(c) Change of the containment initial temperature from 90°F to 80°F.

(d) Changes of the steam generator model parameters reflecting the modification caused by the replacement of the steam generators.

The change in the low safety injection flow was needed in order to meet the NPSH requirement of the LPSI pumps. The flow was limited by means of a venturi flow restrictor. This change resulted in a slightly lower safety injection flow. However, the licensee has demonstrated (Reference 3) that this flow is still higher than the value assumed in the LOCA analysis.

The modification of the containment spray system consisted of adding additional spray header capacity. This additional capacity increased the fill time and resulted in a greater time to actuation of the spray system. The time assumed in the analysis was therefore conservatively changed from 52 to 59 seconds. The additional modification consisted of removing the flow reducing device, which had been installed as a part of the interim NPSH solution. This required increasing the assumed pump runout flow rates from 2250 to 3500 gpm. Although this modification caused the actuation time to change from 410 to 365 seconds, the results of the analysis were not affected because peak cladding temperature occurred at a much earlier time.

The reduction of the containment initial temperature from 90 to 80°F is in itself a conservative change in LOCA analysis. It is consistent with the minimum value of the allowable containment temperature range restrictions required by the NPSH considerations.

The changes introduced to the steam generator model parameters were mostly in thermal-hydraulic area and were caused by the modified reactor coolant side pressure drop and by flow area and tube length changes (References 7 and 8). The magnitude of these changes was small and it did not significantly influence the results of the LOCA analysis.

The LOCA analysis was performed using the February 1978 version of the Westinghouse Evaluation Model (Reference 9) which was reviewed and approved by us. It was performed for a spectrum of three double ended cold leg guillotine breaks (DECLG) with discharge coefficients of C_D=0.4, 0.6 and 1.0. The input parameters assumed in the analysis are listed below:

Core Power: 102 percent of 2441 MWt (rated power).
Peak Linear Power: 102 percent of 13.59 Kw/ft.
Peaking Factor: 2.19
Accumulator Water Volume: 975 cu ft/each

The results of the analysis indicate a peak cladding temperature of 2190°F, a maximum local Zr-water reaction of 7.99 percent and a total Zr-water reaction of less than 0.3 percent, all these values occurring at the critical break size of C_D=0.4.

The licensee did not include a small break analysis since neither the change in the steam generator tube plugging level, nor the other modifications introduced to the plant affected significantly the results of the original analysis.

The licensee has addressed (Reference 4) the potential impact of the recent concerns related to fuel performance model changes included in draft report NUREG-0630 (Reference 10). The licensee has shown that although these model changes by themselves could cause quite significant peak cladding temperature increases, there are at least two compensating effects which could provide credits required to offset the penalties causing these increases. These effects are due to the changes involving the slip and break flow models which have been approved by us for UHI plants after an extensive review. As a result the effects produced by the fuel performance model change could be excluded from the present LOCA analysis without reducing its degree of conservatism.

The licensee has predicted the maximum values of the total peaking factor reached by the Surry Unit 2 plant during its Cycle 5 operation. The prediction was made using the NRC approved, "3 case FAC" methodology (Reference 11) and the axial flux difference limits specified in the licensee's submittal (Reference 2). The predicted peaking factor, $F_Q(Z)$ is below the LOCA determined limit and no augmented core power distribution surveillance is required during the Cycle 5 operation. The licensee is therefore justified in removing the augmented surveillance requirement from the plant's Technical Specifications.

Based on our review of the submittal documents, we conclude that the results of the LOCA analysis performed with F_Q=2.19 are conservative relative to the 10 CFR 50.46 criteria. We consider the resultant changes to the Technical Specifications acceptable for operating Unit 2 with up to a maximum 3 percent of steam generator tubes plugged.

The submitted LOCA analysis was reviewed for Unit 2, however, the evaluation could be extended to Unit 1 after the unit is suitably modified to comply with the assumptions made in the analysis.

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 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 this amendment.

Conclusion

We have concluded, based on the consideration 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 16, 1980

References

1. Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Serial No. 388, dated May 31, 1979.
2. Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Serial No. 820, dated October 16, 1979.
3. Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Serial No. 388 A, dated October 25, 1979.
4. Letter from C. M. Stallings (VEPCO) to D. G. Eisenhut (NRC), Serial No. 039, dated January 11, 1980.
5. Letter from C. M. Stallings (VEPCO) to H. R. Denton (NRC), Serial No. 388 B, dated February 20, 1980.
6. Letter from C. Eicheldinger (Westinghouse) to J. F. Stolz (NRC), Serial No. NS-CE-1749, dated April 6, 1978.
7. Letter from C. M. Stallings (VEPCO) to E. G. Case (NRC), Serial No. 351, dated August 17, 1977.
8. Letter from A. Schwencer (NRC) to W. L. Proffitt (VEPCO), dated December 15, 1978.
9. WCAP-9220, Westinghouse ECCS Evaluation Model, February 1978 Version, February 1978.
10. NUREG-0630 (Draft), Cladding Swelling and Rupture Models for LOCA Analysis, November 1979.
11. Letter from R. A. Wiesemann (Westinghouse) to J. F. Stolz (NRC), Serial No. AW-78-34, dated April 6, 1978.