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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

In a submittal dated July 8, 1991, as supplemented April 15, 1992, the Virginia Electric and Power Company (VEPCO or licensee) proposed a change in the enthalpy rise hot channel factor (F-delta-h) for its Surry Units 1 and 2 plants from 1.55 to 1.62. The submittal described the application of the VEPCO statistical departure from nucleate boiling (DNB) methodology to the Surry Units 1 and 2 plants, discussed the impact of the 1.62 F-delta-h value on Surry non-LOCA event analyses, and provided a Surry small break loss-of-coolant accident (LOCA) analysis assuming the new F-delta-h value. The licensee also proposed Technical Specifications (TS) changes to reflect the methods and values discussed in the submittal. The April 15, 1992 letter provided supplemental information that did not change the initial proposed no significant hazards consideration determination.

The proposed F-delta-h increase would accommodate increased radial power factors resulting from installation of flux suppression inserts in Surry Unit 1. These inserts are designed to reduce peripheral core power and thereby reduce reactor vessel neutron radiation embrittlement.

2.0 EVALUATION

2.1 Methodologies

2.1.1 DNB Methodologies

In assessing the impact of the proposed 1.62 F-delta-h value, the licensee referenced the deterministic W-3 DNB methodology (and the deterministic application of the WRB-1 correlation for certain events within its range of applicability) currently applied to the Surry units, and the VEPCO statistical DNB methodology using the WRB-1 DNB correlation (the licensee's July 8, 1991 submittal contains a table identifying events for which this methodology will be used).

With either DNB methodology, the licensee determined a retained margin, the difference between the DNB ratio (DNBR) limit for the methodology and a design limit against which the plant has been explicitly analyzed. The licensee expresses this margin as a percent of the design limit for the methodology and assesses certain DNBR penalties (e.g., rod bow) against it when necessary.

The current W-3 deterministic methodology is applicable to both Westinghouse LOPAR fuel and Surry Improved Fuel (SIF), which are contained in the Surry cores. For application of this methodology to Surry, the licensee has determined a correlation DNBR limit of 1.24 as applicable, and has set a design limit of 1.46. The retained margin using the W-3 deterministic methodology is 18 percent. In cases where the WRB-1 correlation is used deterministically, the DNBR limit is 1.17 and the retained margin is 20 percent.

The statistical DNB methodology used for the Surry F-delta-h determination is described in the topical report VEP-NE-2-A. This methodology is applied only to SIF fuel and was previously approved for application to the VEPCO North Anna plants. The North Anna design is like the Surry design in all aspects pertinent to the applicability of the methodology. The staff therefore finds the statistical DNBR methodology described in VEP-NE-2-A is applicable to Surry Units 1 and 2.

In the application of the VEPCO statistical DNB methodology to Surry, the licensee determined the statistical DNBR limit (SDL) of the correlation using Surry-specific parameters (e.g., for vessel average temperature, pressurizer pressure, thermal power, vessel mass flow) uncertainties in the calculation of statepoint uncertainties. The licensee determined an SDL of 1.27 as applicable to the Surry units, and has set a design limit of 1.46 for consistency with the W-3 deterministic methodology design limit. The retained margin using the VEPCO statistical DNB methodology is 13 percent.

The above methodologies have previously been approved for existing Surry analyses and/or have been approved for application to the North Anna plants which are of similar design. The staff, therefore, finds them applicable to the Surry plants, as described in the licensee's July 8, 1991 submittal.

2.1.2 Small Break LOCA Methodology

The emergency core cooling system (ECCS) small break LOCA evaluation model (EM) with the Westinghouse NOTRUMP code used for the Surry small break reanalysis is described in WCAP-10079-P-A and WCAP-10054-P-A. This approved EM is applicable to the Surry plants.

2.2 F-Delta-H

In its July 8, 1991 submittal, the licensee proposed a design F-delta-h limit of 1.62. The proposed TS surveillance F-delta-h limit is 1.56, considering a 4 percent measurement uncertainty. Evaluation analyses, except those using the VEPCO statistical DNB methodology, assume a 1.62 value. Analyses using the statistical methodology assume the 1.56 value, because the measurement uncertainty is factored into the method.

Using the W-3 deterministic DNB methodology the licensee determined that the increase in F-delta-h to 1.62 would result in a 7.3 percent DNBR penalty. In an assessment of the reactor protection setpoints using the approved methodologies, the licensee determined that the existing TS core thermal limits (CTLs) were not bounding. The licensee constructed new CTLs reflecting the higher F-delta-h, which are presented in proposed TS Figure 2.1-1. Existing overpressure-delta-T (OPDT) and overtemperature-delta-T (OTDT) trip setpoints were found to be adequate. No change in these reactor protection setpoints are proposed.

Because the licensee used acceptable methodologies in making these assessments, the staff finds the resultant determinations regarding reactor protection setpoints acceptable.

2.3 Transient and Accident Analyses

In its July 8, 1991, the licensee provided an assessment of the impact of the proposed F-delta-h change on the Surry Final Safety Analysis Report (FSAR) Chapter 14 design basis event analyses.

2.3.1 Non-LOCA Events

The licensee addressed the impact of the proposed F-delta-h change on non-LOCA events covering both LOPAR fuel and SIF fuel.

For LOPAR fuel, analyzed using the W-3 deterministic methodology, the licensee indicated that existing analyses and protection setpoints bound or include an assumed 1.62 F-delta-h. The most limiting OTDT DNB event was identified to be a rod withdrawal at power with existing OTDT trip setpoints indicated to be adequate to bound the 1.62 F-delta-h assumption. The most limiting DNB event, which does not trip on OTDT, was identified to be a loss of flow event, whose existing analysis assumes a 1.62 F-delta-h.

For SIF fuel, the most limiting OTDT DNB event was identified to be a rod withdrawal at power, for which the licensee indicated that the current analysis is bounding for an assumed 1.62 F-delta-h.

The most limiting DNB event for SIF fuel which does not trip on OTDT was identified to be a loss of flow event. The licensee indicated that the existing analysis using the WRB-1 correlation deterministically is based on a 1.62 F-delta-h value. However, the licensee provided a reanalysis of this event using the VEPCO statistical methodology to enhance the analysis margin and to demonstrate application of the methodology. The calculated minimum DNBR for this event was about 1.5, which is higher than the 1.46 design limit and does not involve retained margin compensation.

The most limiting DNB event for SIF fuel analyzed using deterministic DNB methods was identified to be a locked rotor event. The licensee indicated that it had performed a thermal-hydraulic reanalysis of this event assuming 1.62 F-delta-h for both fuels and concluded that the existing 5 percent failed fuel assumption remains limiting.

The remainder of the non-LOCA Chapter 14 events are discussed in the licensee's July 8, 1991 submittal and indicate that DNBRs are not significantly reduced by the 1.62 F-delta-h, not affected by the change in F-delta-h, or not applicable to the present Surry core.

2.3.2 LOCA Analyses

The licensee indicates that the current large break LOCA of record assumes a 1.62 F-delta-h. The calculated peak cladding temperature (PCT) in that analysis is 1979°F.

The July 8, 1991 submittal provides the results of a small break LOCA reanalysis using the Westinghouse NOTRUMP code and assuming a 1.65 F-delta-h value. The calculated PCT was 1504°F. This is much lower than the large break PCT. Small break LOCAs continue to be less limiting than large break LOCAs with the 1.62 (or 1.65) F-delta-h assumptions.

2.3.3 Analysis Conclusions

Based on the assessments provided by the licensee, the staff concludes that Surry operation will continue to be bounded by Chapter 14 analyses with the F-delta-h raised to 1.62.

3.0 TS Changes

The licensee's submittal proposed the following TS changes to reflect the 1.62 F-delta-h value and the methodologies used to assess its impact.

a. TS 2.1-4, - change in discussion of TS Figure 2.1-1 to reflect 1.62 F-delta-h and statistical methodology implementation.

b. TS 2.1-5 - change in discussion of DNBR analyses to reflect differences in use of statistical DNB methodology versus deterministic DNBR methodology.

c. TS 2.1-6 - continuation of changes from previous page.

d. TS Figure 2.1-1 - change to reactor core thermal and hydraulic safety limits to reflect 1.62 F-delta-h.

e. TS 3.12-3 - change to equation for F(N)-delta-h to reflect F-delta-h surveillance limit; change in line referring to above equation; change to F-delta-h surveillance limit value in discussion of maintenance operation within hot channel factor limits.

f. TS 3.12-11 - adds surveillance requirements for DNB-related parameters: reactor coolant system (RCS) average temperature, pressurizer pressure, and RCS total flow rate, to reflect use of statistical DNBR methodology.

g. TS 3.12-11a - continuation of changes from previous page.

h. TS 3.12-14 - adds qualification to discussion of engineering heat flux hot channel factor (FQE) to clarify that the FQE penalty is applicable only in non-statistical analyses, to reflect use of statistical methodology.

i. TS 3.12-15 - Bases discussion of F-delta-h is updated to reflect 1.56 surveillance limit and use of statistical methodology.

j. TS 3.12-16 - continuation of changes from previous page.

k. TS 3.12-19 - Bases discussion is added for DNB parameters specified on TS pages 3.12-11 and 3.12-11a.

l. TS Figure 3.12-8 - change to hot channel factor normalized operating envelope, to reflect changed F-delta-h.

m. TS Table 4.1-2A - adds RCS flow to table of minimum frequency for equipment tests, to reflect use of statistical methodology.

These TS changes reflect use of the methodologies discussed in Section 2.1.1 and an increased F-delta-h value. The staff finds the TS changes acceptable because they are consistent with similar changes implemented at the North Anna plants, which are of like design.

4.0 SUMMARY

As discussed in Section 2.1.1, the staff finds that the W-3 and WRB-1 deterministic DNB methodologies are applicable to the Surry units, as limited in their present use, based on their currently approved usage. The staff finds that the VEPCO statistical DNB methodology is applicable to the Surry units based on its currently approved applicability to the North Anna plants of similar design.

Based on the justifications provided by the licensee, the staff finds the 1.62 enthalpy rise hot channel factor (F-delta-h) acceptable for operation of the Surry units with LOPAR and SIF fuels.

The staff also finds the proposed TS changes which accommodate the methodological and operational changes acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

7.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (56 FR 47246). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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