ENCLOSURE

SAFETY EVALUATION REPORT

FOR

VIRGINIA ELECTRIC AND POWER COMPANY

REPORT VEP-NE-2

STATISTICAL DNBR EVALUATION METHODOLOGY

DOCKET NOS. 50-280, 50-281, 50-338, AND 50-339

## INTRODUCTION

Virginia Electric and Power Company (the licensee) submitted a topical report VEP-NE-2 "Statistical DNBR Evaluation Methodology" dated October 8, 1985. This report describes the licensee's methodology for statistically treating several of the important uncertainties in Departure from Nucleate Boiling Ratio (DNBR) analysis. Previously, these uncertainties were treated in a conservative deterministic fashion, with each parameter assumed to be simultaneously and continuously at the worst point in its uncertainty range with respect to the DNBR. Statistical combination of some of these uncertainties maintains the same uncertainty on each parameter, but permits a more realistic combination of the independent variable errors and thus provides a more realistic evaluation of DNBR margin. Similar statistical methodologies have been approved by the NRC. This methodology report uses typical data from the North Anna Power Station for the sample calculations. Before implementation of the method on a plant-specific basis, submittals will be made for the North Anna and Surry Power Stations based upon uncertainties determined from plant specific data.

DESCRIPTION OF METHODOLOGY

The licensee's DNBR evaluation methodology can be summarized as follows:

1. Statistically treated parameters and their uncertainties are defined.

2. At a specified set of nominal conditions, defined as a "nominal statepoint," 2000 sets of operating conditions are determined by randomly varying each statistically-treated parameter about its nominal value according to its distribution. Each perturbation from a nominal statepoint is defined as a "random statepoint."

3. The random statepoints are analyzed with the COBRA IIIC/MIT code, yielding a distribution of 2000 DNBR's at each nominal statepoint. Each random statepoint is corrected to account for correlation uncertainty.

4. The DNBR distribution is tested for normality with the D' normality test.

5. Steps 1-4 are performed at each of the different nominal statepoints which are selected for the analysis. The most conservative DNBR standard deviation which is obtained at any nominal statepoint is used in the further development of the Statistical DNBR Limit.

6. A total DNBR standard deviation is determined by combining the overall parameter/correlation standard deviation with factors which account for the code and model uncertainties.

7. A DNBR limit for the nominal statepoint is determined from the absolute DNBR limit of 1.0, using the distribution mean and standard deviation to provide protection from DNB with 95% probability at a 95% confidence level for the hot fuel rod.

8. Using the DNBR standard deviation, a core-wide DNB probability analysis is performed to determine the expected number of rods in DNB when the hot fuel rod in the core is at the Statistical DNBR Limit (SDL). This is done iteratively, if necessary, to find an SDL such that no more than 0.1% of the rods in the core are expected to be in DNB if the plant were to operate at the SDL.

   Thus, the SDL is the more restrictive of 1) single rod DNB probability, or 2) a minimum number of rods expected to be in DNB at the DNBR limit.

EVALUATION

The licensee selected vessel flow, pressure, inlet temperature, power, engineering enthalpy-rise hot channel factor (Fdhe), nuclear enthalpy-rise hot channel factor (measurement component) and effective flow fraction (EFF) as the parameters to be statistically treated. An analysis of plant hardware and procedures will be presented in the plant specific implementation submittals to justify the distribution, mean and standard deviation for each of the statistically treated parameters.

For each implementation, the nominal statepoints will span the full range of pressure/temperature power transients and low flow accidents. Justification that the chosen points represent the limiting conditions will be provided in the plant specific submittals.

Using on-line random number generators RANNOR and RANUNI, the 2000 random statepoints will be generated at each nominal statepoint. For each random statepoint a DNBR calculation is performed and its value is adjusted to account for the effect of the correlation uncertainty. The DNBR distribution at each nominal statepoint is tested with the D' normality test. For the limiting statepoint the 95% upper confidence limit on the standard deviation due to parameters and correlation, SIGMA (P/C) is calculated using standard statistical techniques.

The model uncertainty and code uncertainty are then applied in the following manner. The code uncertainty of 5% is used. This is consistent with the factors specified for other thermal/hydraulic codes (Ref. 1 & 2). The 5% penalty is conservative since Ref. 3 shows the COBRA IIIC/MIT/W-3 data are better than the THINCI/W-3 data. Thus, the 95% confidence level on SIGMA (Code) will be 5/1.645 = 3.04%. The model standard deviation will be evaluated at a 95% confidence level for each plant specific package.

The final DNBR standard deviation which accounts for all factors, each at its 95% confidence level will be obtained as the "Square Root of the Sum of the Squares."

SIGMA (Total) = [SIGMA (P/C, 95)$^2$ + SIGMA (Model)$^2$ SIGMA (Code)$^2$)]$^{1/2}$ The SDL is then determined using the total Standard Deviation.

$$SDL = 1.0 + 1.645\ (SIGMA\ (Total))$$

The licensee also used a criterion of finding an SDL which resulted in an expected number of rods in DNB which is less than 0.1% of the total in the core. To check this criterion the full core DNB probability must be evaluated. Given the DNBR standard deviation, the probability of DNB is calculated for each rod and summed over the entire core. The analysis is performed until the 0.1% criterion is met. This final SDL is the more restrictive of the peak rod DNB probability limit and the core-wide DNB probability.

CONCLUSION

We have reviewed the statistical methodology as presented in the report VEP-NE-2. Based on our review of the material presented we find the methodology acceptable with the following conditions.

1) Selection and complete justification of nominal statepoints to be used must be included in the plant specific implementation submittal.

2. For the statistically treated parameters - justification of the distribution, mean and standard deviation must be included in the plant specific implementation submittal.

3. Justification of the value of model uncertainty must be included in the plant specific implementation submittal.

4. Justification of the WRB-1 CHF correlation, with a 95/95 DNBR limit of 1.17 and that the M/P distribution is normal with a mean of one and a standard deviation of 0.838 must be provided, or VEP-NE-3 (Ref. 4) must be approved.

References

1. Letter from J. F. Stolz (NRC) to C. Eicheldinger (Westinghouse) "Staff Evaluation of WCAP-7956, WCAP-8054, WCAP-8567 and WCAP-8762, dated April 19, 1978.

2. Letter from R. A. Clark (NRC) to W. Cavanaugh, III (Arkansas Power & Light Co.), "Operation of ANO-2 During Cycle 2," dated July 21, 1981.

3. Letter from W. N. Thomas (Virginia Power) to H. R. Denton (NRC) "Topical Report VEP-FRD-33 Vepco Reactor Core Thermal-Hydraulic Analyses Using the COBRA IIIC/MIT Computer Code 2," dated June 12, 1981.

4. Letter from W. L. Stewart (Virginia Power) to Document Control Test SER No. 86-698 (NRC) Topical Report VEP-NE-3 Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code, January 29, 1987.