Enclosure

Evaluation of the Exxon Statistical Setpoint
Methodology for CE Reactors

1. Indroduction

Exxon Nuclear Company (ENC) has developed methods for statistically combining the parameter and measurement uncertainties that are applicable to thermal margin analysis for CE reactor reload cores. This ENC methodology as documented in References 1, 2 and 3 is used to replace or validate deterministic reactor local power density (LPD) and thermal margin/low pressure (TM/LP) trip setpoints and limiting conditions for operation (LCO) with 95% probability bounds (with 95% confidence).

We have reviewed the ENC statistical thermal margin setpoint analysis methodology for CE reactors. The review was based on the ENC reports pertaining to their setpoint methodology (Refs. 1 to 5), and on the ENC responses (Ref. 8) to the review questions. The results of the review are discussed in this report as follows.

2. The Evaluation of Methodology for the LPD Trip Setpoint and LCO

The LPD trip setpoint is established to scram the reactor before the peak fuel centerline temperature reaches the melting temperature, while the LPD LCO is an operational limit on reactor power to assure that the calculated peak cladding temperature will not exceed the safety limit of 2200°F during a large break loss-of-coolant-accident.

In the ENC LPD setpoint calculations, the radial power peaking factor with the worst axial power distribution which results in the highest peaking for LPD thermal margin is used for conservatism.

As indicated in the topical report, the power measurement and power peaking uncertainties are related to the nominal power trip setpoint (calculated based on the input parameters at best estimate values) to obtain an allowable power trip setpoint. The Monte Carlo Simulation (MCS) is used to statistically combine the power peaking and fuel rod engineering factor uncertainties. The convolution integral method is used to combine the uncertainties for the power measurement and the power trip signal process. We have reviewed the ENC use of MCS and the convolution integral method. We find that MCS is a valid technique for combining the probability densities of two random variables and the equations used in the convolution method are correct. Since the highest peaking, which results in a lower LPD thermal margin, is used and the methods to combine the uncertainties involved in the LPD setpoint are correctly used, we conclude that the methods for determination of LPD trip setpoint and LCO are acceptable. (The results of review for MCS are further discussed in Section 6).

3. The Evaluation of TM/LP Trip Setpoint Methodology

The TM/LP trip is established to scram the reactor to prevent the DNBR in the limiting coolant channel in the core from decreasing below that value corresponding to a 95% probability that DNB will not occur, at 95% confidence level.

An application technique common to both the TM/LP trip setpoint analysis and the DNB thermal margin allowable power LCO (discussed in Section 4) is the use of a "most sensitive" ASI point for the statistical analysis. The "most sensitive" point is the value of ASI, and its associated power distributions and operating conditions, where the largest difference between a deterministic setpoint (calculated with all key parameter uncertainties set to 95% probability or the tolerance bounds) and a nominal setpoint (calculated with all key parameters set at nominal values) is calculated. The resulting 95% probability bound at a 95% confidence (the 95/95 bound) for the trip setpoint or LCO calculated at this "most sensitive" ASI point is then applied to the calculated nominal values for the setpoint at all other ASI points.

We have reviewed the TM/LP trip setpoint methodology and found that the methodology for defining the 95/95 bound for the TM/LP setpoint is an adequate approach; the methodology for verification of the existing TM/LP setpoint algorithm is a conservative application of the statistical methodology. Therefore, we conclude that the TM/LP trip setpoint methodology is acceptable.

4. The Evaluation of DNB Allowable Power LCO Methodology

The DNB allowable power LCO (DNB LCO) is an operational limit on power as a function of ASI, designed to protect the fuel against DNB (i.e., a 95% probability with a 95% confidence that DNB will not occur on the hot fuel rod) during any limiting Anticipated Operational Occurrence (AOO). The DNB LCO is based on the results of thermal-hydraulic analyses for limiting AOOs. Potentially limiting AOOs were considered to define which transients are limiting for the DNB LCO. For CE reactors with an asymmetric steam generator protection trip, the identified limiting AOOs for the DNB LCO are the loss of flow (due to loss of pumping power) or the control element assembly drop transients. Both of these transients were analyzed to define which was most limiting for the DNB LCO.

Thermal-hydrualic calcualtions are done using the XCOBRA-IIIC computer code (Ref. 11) and the XNB DNB correlation (Ref. 10), which were previously approved by NRC (Ref. 6), to determine the reactor power for which the transient minimum DNBR reaches the limit for DNBR as a function of key statistical variables.

A response surface analysis performed at the "most sensitive" ASI point (defined in Section 3) is used to formulate, from XCOBRA-IIIC calcualtions, an equation (the response surface equation) for the power limit (for DNBR = DNBR safety limit), as a function of the statistical parameters. Monte Carlo Simulation (MCS) is used to propagate the uncertainties in the statistical parameters through the response surface equation. The MCS results are then used to define a 95/95 bound for the DNB allowable power LCO.

We have reviewed the statistical methodology used for the DNB LCO analysis and found that an approved computer code and DNB correlation are used and, the response surface analysis and Monte Carlo Simulation are adequate methods for calculating the uncertainties of the statistical parameters. We conclude that the DNB LCO methodology is acceptable.

5. The Evaluation of Application of the XNB DNB Correlation

The ENC analyses for establishing the 95/95 bound for the TM/LP trip setpoint and the DNB LCO involve calculating the margin from an approach to DNB by use of the XNB correlation (Ref. 10). The XNB correlation and its safety limit DNBR of 1.17 were previously approved by NRC (Ref. 6). However, the XNB correlation was based on DNB data for many different PWR fuel bundle geometries, and is thus a general PWR CHF correlation rather than a correlation defined for a specific fuel bundle design. As restricted by NRC in Reference 6, when the XNB correlation is applied for a specific type of fuel bundle which does not belong to the data base on which the XNB correlation was based, ENC should submit for NRC approval a compensating factor to account for the effect of the different fuel type on the DNBR calculation.

6. The Evaluation of Monte Carlo Simulation Methodology

In Monte Carlo Simulation, a random sampling routine is used to choose values for the input statistical variables according to their defined probability distributions, and then to exercise the relevant computer program. The MCS routine will continue this process for a specified number of trials, after which many output points are available for defining the probability distribution of the response variable by use of a histogram, and for defining the statistical characteristics of the response variable.

The MOCARS computer code (Ref. 12) was used to perform Monte Carlo Simulation for the ENC Statistical Setpoint analyses. In MOCARS, values for each input variable are randomly chosen from the uncertainty distribution of the respective variables.

We have reviewed the ENC use of MCS. We find that ENC sets the accuracy high enough (to include 99.5% of the population with a 99.5% confidence) to adequately envelope the 95% probability bound. Based on the review of Reference 13, we find that 95/95 bound is conservatively defined using distribution free order statistics, which bounds the true value for the desired probability level. Therefore, we conclude that the ENC use of MCS is adequate and acceptable.

7. The Evaluation of Responses Surface Analysis Methodology

In a response surface analysis (RSA), the computer models which are used to calculate the value of the response variable, such as DNBR, are exercised many times with different values for the key input variables in order to define a response surface; that is, the values of the response variables as function of the key input variables. The response surface is fitted by a response surface equation (RSE), a polynomial, by using a regression analysis. The RSE can be used as a substitute for the more complex algorithms or computer models for the many calculation trials needed to define the probability distribution of the response variable. ENC uses MCS approach to define the 95/95 probability bound in the RSA application.

The ENC use of RSA in their statistical setpoint methodology as described in their generic statistical uncertainty analysis methodology report was previously reviewed and approved by NRC (Ref. 7). We also find that all the limitations (Ref. 7) imposed by NRC on the ENC RSA are adequately met. Therefore, we conclude that the ENC use of RSA methods is acceptable.

8. Review of Input Parameter Uncertainties

8.1 The Staff Evaluation of Parameters Uncertainties

ENC generally assumes for non-data based parameter uncertainties a normal distribution. However, numerous examples of non-normal distributions for parameter uncertainties were found. Use of a normal distribution to represent a uniform distribution was also found to be not neccessarily conservative (Ref. 9).

We find that the general ENC approach for the non-data based uncertainties is not adequately supported, since the distribution of the uncertainties for plant system parameters may be a function of how the measurements are taken and of how the signals are processed. The magnitudes of the measurement uncertainties for system parameters as listed in References 2 and 8 appear to be conservative when compared to data for measurement uncertainties. However, ENC has not adequately supported the appropriateness (or conservativeness) of their assumption of normal distribution for all measurement uncertainties. Therefore, we require ENC to justify the measurement uncertainties and the assumed probability distributions for specific CE reactor core reload application (once for each CE reactor), or provide bounding values for all CE reactor reloads (once for all CE reactors). The affected system parameters are: reactor power, ASI, pressure, inlet coolant temperature, coolant flow, and scram delay time.

8.2 The Staff Evaluation of Power Distribution Uncertainty

ENC used the inferred power distribution and various PWR power distribution data sources to determine the power distribution uncertainty. However, as is discussed below, the PWR power distribution data used by ENC may not be applicable to CE reactor cores.

ENC indicates that the measurement data base for the inferred detector power uncertainty is derived from measurements of the detector signals taken during Cycles 4 and 5 at the St. Lucie Unit 1 plant. However, the measurement data base for axial power distribution were derived from the measurements performed at two Westinghouse reactors. Also, the measurement data base for the uncertainty in the assembly local peaking factor is composed of measurements performed by Babcock & Wilcox and Battelle Pacific Northwest Laboratories.

The above data bases for Westinghouse and Babcock & Wilcox fuel may not be applicable to CE reactors because of potential significant neutronic differences between CE fuel bundles and cores and the other PWR fuel bundles and cores. Therefore, we require that the data or analyses used to establish or justify the power distribution specifically address the CE reactor core and CE fuel bundle design.

8.3 The Staff Review of Operating Point Uncertainty

In the statistical setpoint analysis, ENC uses deterministic (worst case) assumptions to specify all operating state variables at the technical specification bounding values (or operational bounds) which minimize the DNB thermal margin. We find that this approach is conservative and is acceptable.

9. Conclusions

Based on the review described above we conclude that the topical reports XN-NF-507 (Supplements 1 and 2) are acceptable for referencing in licensing actions by ENC with respect to the Exxon statistical setpoint methodology for CE reactors. The following restrictions apply to the use of the methodlogy:

1. The statistical setpoint methodology, as defined by References 1, 2, 8, and in this report should not be altered or modified, except as approved herein, without additional NRC review.

2. ENC is required to justify the system parameter measurement uncertainties and their probability distributions for specific CE reactor core reload applications of the methodology. The affected system parameters are: reactor power, ASI, pressure, inlet coolant temperature, coolant flow and scram delay time. The justification should include consideration of the measurement system and signal processing when defining an appropriate probability density function for the measurement uncertainty.

3. ENC is required to justify the power distribution uncertainty assumed for specific applications of the methodology for CE reactor core reloads. The justification will need to demonstrate that the power distribution uncertainty is applicable and/or conservative based on analyses and/or data applicable to CE fuel and cores.

4. ENC should submit for NRC review and approval a CHF uncertainty compensating factor to include an adequate allowance for the uncertainty in the CHF when applied to fuel bundle designs different from test bundles used for obtaining the DNB test data.

10. References

1. N. F. Fausz, W. T. Nutt, "ENC Setpoint Methodology for CE Reactors-Statistical Setpoint Methodology," XN-NF-507(P) Supplement 1, Exxon Nuclear Co., Inc., September 1982.

2. W. T. Nutt, N. F. Fausz, "ENC Setpoint Methodology for CE Reactors-Sample Program," XN-NF-507(P) Supplement 2, Exxon Nuclear Co., Inc., November 1982.

3. "ENC Setpoint Methodology for CE Reactors," Attachment to Florida Power and Light Company letter L-82-384, R. E. Uhring to D. G. Eisenhut, USNRC, September 1, 1982.

4. N. F. Fausz, "Generic Statistical Uncertainty Analysis Methodology," XN-NF-81-22(P), Exxon Nuclear Co., April 1981.

5. R. B. Stout, "ENC Setpoint Methodology for CE Reactors," XN-NF-507, Rev. 1, July 18, 1980.

6. A letter dated April 12, 1983 from C. Thomas (NRC) to R. Stout (ENC), "Acceptance for Referencing of Licensing Topical Report XN-NF-621(P), Revision 1".

7. C. Thomas (NRC) letter to C. Chandler (NRC), dated October 28, 1983, "Transmitting the Staff Evaluation Report on ENC Statistical Setpoint Methodology Reports including XN-NF-81-82".

8. Attachment to J. C. Chandler (Exxon Nuclear Co.) letter to T. L. Huang (USNRC), JCC:044:83, "Responses to EG&G Questions on XN-NF-507 and Supplements", March 21, 1983.

9. R. Goldstein, "Statistical Methods for Establishing Safety-System Margins," EPRI NP-2468, July 1982.

10. R. B. MaDuff, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," XN-NF-621(P), Rev. 1, April 1982.

11. K. P. Galbraith and T. W. Pattern, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation," XN-75-21.

12. Scott D. Matthews and John P. Poloski, "MOCARS: A Monte Carlo Code for Determining Distribution and Simulation Limits and Ranking System Components by Importance," TREE-1138 (Revision 1), August 1978.

13. Paul N. Somerville, "Tables for Obtaining Non-Parametric Tolerance Limits," Annals of Mathematical Statistics, 2, Vol. 29, June 1958, pp. 599-601.