Enclosure

SAFETY EVALUATION REPORT ON SUPPLEMENTS 1, 2 AND 3
TO XN-NF-79-71 (REV. 2)
FOR EXXON NUCLEAR PLANT TRANSIENT METHODOLOGY
FOR BOILING WATER REACTORS

1. INTRODUCTION

The Exxon Nuclear Company (ENC) methodology and computer codes for BWR plant transient analyses are described in detail in Reference 2. The statistical treatment of the uncertainties for calculation of the Critical Power Ratio (CPR) is performed according to the ENC Generic Statistical Uncertainty Analysis Methodology (GSUAM) (Ref. 3). These documents were reviewed by NRC as part of the review for the ENC reload application for the Dresden 3, Cycle 8 reload in 1982 (Ref. 4). The review found that ENC had not included computer code uncertainty in the calculation of transient CPR, and determined that the ENC approach of using conservative code input did not directly address code uncertainty and could not be shown to sufficiently include or bound the effect of code uncertainty. As a result of this review conclusion, a compensation factor was assigned to the operating limit minimum CPR to account for potential code uncertainty. In response to the NRC staff review, ENC submitted reports (Refs. 1, 9 and 11) to address the code uncertainties.

We have reviewed the ENC methodology for determination of code uncertainty during transient analyses for a BWR. The review was based on the ENC report pertaining to their code uncertainty methodology (Ref. 1), and on ENC responses (Refs. 9 and 11) to the review questions. The results of the review are discussed as follows. The review was conducted with the assistance of Idaho National Engineering Laboratory (INEL) under Contract FIN-A6499.

2. CALCULATION OF A CONTRANSA CODE UNCERTAINTY

2.1 Summary of the Methodology

Computer code uncertainty is best defined in terms of the ability of the code to accurately predict the value of the output parameter of interest when the input parameter values are known or best estimate values are used. This is the approach that was used by ENC for the determination of the code uncertainty for their COTRANSA code for the calculation of decreases of CPR (ΔCPR) for BWR transients (Ref. 1). COTRANSA code uncertainty was defined in terms of the transient integral power for limiting BWR turbine trip transients or other rapid pressurrization transients. The COTRANSA transient integral power uncertainty was determined by a comparison of COTRANSA calculated transient power to transient power data for the Peach Bottom Unit 2 (PB2) turbine trip tests (Reference 5).

During a rapid power transient, such as the turbine trip transient, neutron power will rapidly peak and decrease before bundle thermal power significantly increases. Peak bundle thermal power and minimum CPR will occur when the neutron power transient is nearly over. The time delay for the fuel rod heat flux and bundle thermal power is due to high thermal capacitance and low thermal conductivity of the UO_2 fuel and the thermal transport resistance of the fuel to cladding gap. Because of this thermal transport delay and its influence on heat flux and CPR, ENC indicates that the data for neutron power would not be an appropriate parameter for defining COTRANSA code uncertainty for ΔCPR calculations. The ENC proposed parameter is the transient integral power, defined in Reference 1, as it is a measure of the total energy deposited in the fuel which will, after some delay, result in an increase in fuel bundle heat flux, temperatures, and coolant enthalpy. There were three turbine trip tests. During the tests, the transient neutron power data were obtained for four axial levels in the PB2 core average power. The COTRANSA code uncertainty was defined in Reference 1 as the standard deviation of the ratio of measured transient integral power (from the PB2 turbine trip tests core average power data) to the COTRANSA predicted transient integral power. The probability

distribution for the Transient Integral Power Ratio (TIPR) uncertainty was assumed to be a normal distribution.

2.2 Review of the Methodology

The review of the code uncertainty consists of three parts: (1) a review of the appropriateness of defining the code uncertainty by use of the transient integral power parameter, (2) code uncertainty applicabilty, and (3) the adequacy of the transient integral power ratio (TIPR) uncertainty as calculated by ENC. The results of the review are discussed below.

2.2.1 Code Uncertainty Parameter Appropriateness

The ENC transient integral power parameter was reviewed for its appropriateness for the calculations of uncertainties for ΔCPR in a BWR transient analysis.

We have reviewed the limiting BWR transient. It is usually a turbine trip or load rejection transient. The limiting transient results in a short neutron burst of less than one second. Because of the thermal transport delay in the fuel rods, the fuel rod heat flux and coolant enthalpy do not significantly increase and peak until the neutron power burst is nearly over. CPR and ΔCPR are direct functions of the fuel bundle heat flux and coolant enthalpy which are functions of the integrated power added to the fuel rods during the transient because of the thermal transport delay of the BWR fuel rods. Therefore, we conlude that the ENC use of the transient integral power in defining the code uncertainty is appropriate and acceptable.

2.2.2 Code Uncertainty Applicability

A review was conducted for the applicability of the ENC code uncertainty (calculated from PB2 benchmark calculations) to other BWR cores. It was concluded that the ENC code uncertainty as defined and utilized by ENC for the calculation of ΔCPR is applicable to ENC BWR transient analyses for other reactors and cores. This conclusion is limited, however, to those BWR's for

which the limiting anticipated transient is a turbine trip transient or a similar rapid pressurization transient.

The PB2 benchmark calculations indicate the ability of the COTRANSA code to predict important transient phenomena and reactor consequences during a turbine trip test. These phenomena and the resulting consequences of a turbine trip are generic to BWR's. The code uncertainty (transient integral power uncertainty) calculated from the PB2 benchmark analysis is applied as a multiplier on COTRANSA calculated transient integral power in subsequent analyses for ΔCPR. This application of the core uncertainty as a multiplier makes it independent of the reactor and core for which the uncertainty was originally calculated since the uncertainty factor proportionally modifies the specific COTRANSA calculated results for transient integrated power.

The neutron cross section tables used in COTRANSA will change with time in the fuel burnup cycle. The PB2 turbine tests for which the COTRANSA code uncertainty was calculated were for a specific cycle time, end-of-cycle (Ref. 5). But, end-of-cycle is the most limiting time for the turbine trip transient when the control rods are fully withdrawn and scram time is a maximum. Therefore, the data upon which the code uncertainty was based is appropriate. The calculated ΔCPR is very sensitive to the potential uncertainty in the void reactivity calculated from the neutron cross section tables. Even though the effect of uncertainty in the neutron cross section tables is included in the code uncertainty, the cross section tables used in specific applications to licensing analyses change for each plant for which analyses are done or even for each fuel cycle. The uncertainty in the values in the cross section tables should not exceed those used for the PB2 benchmark comparisons in Reference 1. ENC is required to validate the accuracy of the cross section values for specific plant reload analyses by checking COTRANSA reactivity calculations against startup data for either or both rod withdrawal tests and load rejection tests.

ENC indicates that additional measured data would be needed to derive a value for ΔCPR for the PB2 tests since ΔCPRs are not measured quantities but are dependent on many other quantities. Some of the needed data, such as core mass flow, were not measured for the PB2 tests. Calculation of some needed quantities, such as core flow for the purpose of deriving test values for ΔCPR would introduce additional uncertainties. ENC concludes that it would not be desirable to determine the code uncertainty by comparing with ΔCPR derived from the PB2 test data. We agree with the ENC position and find the position to be acceptable. Based on the discussion above, we conclude that the ENC use of transient integral power to assess the code uncertainty is appropriate and acceptable.

2.2.3 Adequately Bounded Value for STIPR.

The ENC calculated code uncertainty, expressed as a standard deviation for the transient integral power ratio (STIPR), was reviewed for adequacy. The ENC value for STIPR was reviewed for conservatism and to determine if the STIPR was a sufficiently bounding value. The conclusion of the review was that the ENC calculated code uncertainty as expressed by a value for STIPR in Reference 1, is based on only three data points, is not necessarily conservative and was not shown to be a sufficiently bounding value for use in licensing calculations for ΔCPR and for the OLMCPR. The value for STIPR proposed by ENC for a code uncertainty is not acceptable.

In response, ENC submitted a revised approach for the COTRANSA code uncertainty (defined as the uncertainty in the transient integral power parameter) which was defined and discussed in Reference 11. This revised code uncertainty, ±10% for the integral of transient power, is treated as a bounding, deterministic factor instead of a statistical parameter (STIPR). Based on our independent study, we find that ENC proposed value (±10% for the transient integral power) envelopes not only the COTRANSA calculated total transient integral power for the PB2 benchmark tests, but it also is equal to or enveloping for all of the PB2 benchmark calculations for TIPR by the COTRANSA (Ref. 14), BNL-TWIGL (Ref. 15) and RAMONA-III (Refs. 6, 7) codes.

Therefore, we conclude that the ENC deterministic code uncertainty factor (± 10% for the transient integral power ratio) defined in Reference 11 is sufficiently bounding and acceptable.

3. EVALUATION OF THE OLMCPR CALCULATION METHODOLOGY

The operating limit minimum power ratio (OLMCPR) is defined as the sum of the safety limit MCPR (SLMCPR) and the 95 percent probability bound ΔCPR, which is the maximum decrease in CPR calculated for a limiting BWR transient. The SLMCPR is the minimum allowed value for CPR for which 99.9 percent of the fuel rods in the core will not be expected to experience boiling transition under steady-state conditions. The ENC methodology for the determination of the SLMCPR is described in Reference 16, which has been previously approved by NRC. ΔCPR is determined by use of a response surface analysis (RSA) to define a functional relationship for ΔCPR as a function of the key variables, and a statistical analysis of the functional relationship to determine ΔCPR (at the 95 percent probability bound). The response surface and statistical analysis methodology used by ENC is their Generic Statistical Uncertainty Analysis Methodology (GSUAM) described in detail in Reference 3, which was previously approved by NRC (Ref. 17). Based on our review, we find that all the limitations imposed by NRC on the ENC response surface analysis are adequately met. We, therefore, conclude that the ENC use of the response surface analysis methods is acceptable. ENC states that all non-statistical input parameters are given deterministic (conservative) and/or bounding values in the COTRANSA/XCOBRA/HUXY calculations for ΔCPR (References 10, 12, 13). This is an appropriate and acceptable approach for non-statistical input parameter values. Since deterministic and/or bounding values are used for all non-statistical input parameters and the NRC approved methods are used for determination of SLMCPR and ΔCPR, we conclude that methodology to determine the operating limit MCPR is acceptable.

4. CONCLUSIONS

Based on the review described above we conclude that the topical reports (Supplements 1, 2 and 3) are acceptable for referencing in licensing actions by ENC with respect to the determination of code uncertainties for the ΔCPR calculation for a BWR. The following restrictions apply to the use of this methodology:

1. Since different neutron cross section tables are used in COTRANSA from the set used for the Peach Bottom 2 benchmark analyses in Reference 1 for each specific core reload analysis, the uncertainty of the values in the cross section tables, or reactivity values calculated using the tables, will need to be shown to not exceed the cross section table uncertainty for the Peach Bottom 2 benchmark calculations for a plant specific application.

2. The deterministic or bounding input values for the important input parameters which are not handled as statistical variables should be used for a plant specific application. Of particular importance are those system modeling parameters and initial conditions which influence the differential pressure and pressure wave propagation calculations between the steam line and the core. The important system modeling parameters are: gap conductance of the fuel rods, steam piping and system volume element lengths and inertias, piping and upper vessel volumes, flow resistance coefficients, initial steam flow, and turbine stop value closure time.

3. If applicable measured data are not available for the determination of the statistical characteristics of the statistical variable (involved in the code uncertainty determination), then appropriate and conservatively bounding estimates for the standard deviations and probability (or uncertainty) distributions should be used.

5. REFERENCES

1. S. E. Jensen, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, Methodology for Including Code Uncertainties in Determining Operating Limits for Rapid Pressurization Transients in BWR's," Supplement 1, XN-NF-79-71(P), Revision 2, October 24, 1983.

2. R. H. Kelley and G. L. Cooke, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," XN-NF-79-71(P), Revision 2, November 16, 1981.

3. N. F. Fausz, "Generic Statistical Uncertainty Analysis methodology," XN-NF-81-22(P), April 27, 1981.

4. J. Hegner (NRC) letter to L. DelGeorge (Commonwealth Edison Co.), "Dresden 3 Cycle 8 Reload," April 29, 1982.

5. L. A. Carmichael and R. O. Niemi, "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2," EPRI-NP-564, June 1978.

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8. R. B. Macduff and T. W. Patten, "XN-3 Critical Power Correlation," XN-NF-512, Rev. 1, October 1982.

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10. Reference 1, P. 22.

11. S. E. Jensen, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, Revised Methodology for Including Code Uncertainties In Determining Operating Limits for Rapid Pressurization Transients in BWR's," Supplement 3 to XN-NF-79-71(P), Revision 2, March 4, 1985.

12. Reference 6, pg. 28.

13. Reference 2, Part IV, pg. 111.

14. Reference 2, Part V.

15. M. S. Lu, et. al., "Analysis of Licensing Basis Transients for a BWR/4," BNL-NUREG-26684, September 1979.

16. "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," XN-NF-524(P), Rev. 1, May 1983.

17. C. Thomas (NRC) letter to C. Chandler (ENC), October 28, 1983, transmitting the Staff Evaluation Report on ENC Statistical Setpoint Methodology reports including XN-NF-81-22.