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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO REVISION 2 OF TOPICAL REPORT NSPNAD-8609

"QUALIFICATION OF REACTOR PHYSICS METHODS FOR APPLICATION TO MONTICELLO"

FOR

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated August 23, 1994 (Ref. 1), the Northern States Power Company (NSP) submitted Revision 2 of the Topical Report NSPNAD-8609, "Qualification of Reactor Physics Methods for Application to Monticello," (Ref. 2) for NRC review. NSPNAD-8609, Rev. 1-A describes the currently approved methodology for the Monticello Nuclear Generating Plant. This revision documents the capability of NSP to implement and apply new methods, based on CASMO-3/SIMULATE-3 methodology, to boiling water reactor (BWR) core reload physics design activities for the Monticello unit. Both the CASMO-3 and SIMULATE-3 computer program packages have been reviewed and accepted for referencing (with certain restrictions) by separate NRC safety evaluations (Refs. 3 and 4) regarding the Yankee Atomic Electric Company (YAEC) Topical Reports YAEC-1363 (Ref. 5) and YAEC-1659 (Ref. 6). Specific limitations imposed on the use of these models at that time were:

1) that CASMO-3 is to be used for the core parameter ranges and configurations that were verified; i.e., new fuel designs will require additional validation, and

2) that SIMULATE-3 is to be used for steady-state physics analyses only with the approved versions of the CASMO-3 and TABLES-3 codes.

NSP intends to use the CASMO-3/SIMULATE-3 programs in licensing applications, including BWR reload physics design, calculations for startup predictions, generation of physics input for reload safety evaluation (RSE) analyses, core physics data books and setpoint updates for both the reactor protection and monitoring systems.

2.0 SUMMARY OF THE TOPICAL REPORT

Topical Report NSPNAD-8609, Revision 2, describes the NSP qualification of new reactor physics methods (CASMO-3/SIMULATE-3) for application to the Monticello BWR and addresses the reactor model description, qualification and

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quantification of reliability factors and applications to operations and reload safety evaluations of Monticello. The qualification benchmarking compares the CASMO-3/SIMULATE-3 model results with measurements obtained from benchmarking data covering operating cycles 11 through 15 of the Monticello unit. The plant analyses were performed over a wide range of conditions from cold (ambient) temperature to hot full power operation. The good agreement between the measured and calculated values presented in the topical report is used to validate the NSP application of these computer programs for analysis of the Monticello BWR unit.

NSP intends to use these methods for steady-state BWR core physics reload design and licensing applications, including fuel bundle and loading pattern analysis; for the generation of core physics control rod worth and startup predictions, reactivity coefficients for transient and safety analyses input; and for the potential support of the process computer core monitoring system.

2.1 Overview

Section 1 of the topical report provides introductory and background information and an overview of the scope of the report. The philosophy for determining the calculational uncertainties (and bias) and reliability factors is presented in Appendix A of the topical.

2.2 Methodology

Section 2 of the topical report describes the NSP-specific CASMO-3/SIMULATE-3 computer program package methodology, provides references for each of the individual components, and gives a flowchart for the model application.

CASMO-3 is the Studsvik Energiteknik lattice physics code (Ref. 7) used by NSP in determining the neutronics input to SIMULATE-3 for BWR core performance analyses. CASMO-3 uses a binary-format cross section library based on the standard ENDF/B-IV cross-section set with some ENDF/B-V fission spectrum updates.

SIMULATE-3 was also acquired from Studsvik of America (Ref. 8). The code is based on a modified coarse mesh (nodal) diffusion theory calculational technique, with coupled thermal hydraulic and Doppler feedback. The code includes the following modeling capabilities: solution of the two group neutron diffusion equation, fuel assembly homogenization, baffle/reflector modeling, cross-section depletion and pin power reconstruction. In order to ensure the flux continuity at nodal interfaces and perform an accurate determination of the pin-wise power distribution, SIMULATE-3 uses assembly discontinuity factors that are pre-calculated by CASMO-3. These factors are related to the ratio of the nodal surface flux in the actual heterogeneous geometry to the cell averaged flux in an equivalent homogeneous model and are determined for each energy group as a function of exposure, moderator density and control-rod-state.

The two-group model solves the neutron diffusion equation in three dimensions, and the assembly homogenization employs the flux discontinuity correction factors from CASMO-3 to combine the global (nodal) flux shape and the assembly heterogeneous flux distribution. The flux discontinuity concept is also applied to the baffle/reflector region in both radial and axial directions to eliminate the need for user-supplied albedoes, normalization, or other adjustment at the core/reflector interface.

The SIMULATE-3 fuel depletion model uses tabular and functionalized macroscopic and/or microscopic cross sections to account for fuel exposure without tracking the individual nuclide concentrations. Depletion history effects are calculated by CASMO-3 and then processed by the TABLES-3 code (Ref. 9) for generation of the cross-section library used by SIMULATE-3.

SIMULATE-3 can be used to calculate the three-dimensional pin-by-pin power distribution in a manner that accounts for individual pin burnup and spectral effects. SIMULATE-3 also calculates control rod worth and moderator, Doppler and xenon feedback effects.

ESCORE is an Electric Power Research Institute (EPRI) developed computer code (Ref. 10) for predicting best-estimate, steady-state fuel rod performance parameters for light-water reactor (LWR) fuel rods. This program has been previously reviewed and approved (Ref. 11) for use in calculating fuel rod temperatures for input to reload and safety analyses as a function of burnup and power history.

2.3 Benchmarking and Model Verification

Section 3 of the topical report discusses benchmarking of the NSP models to the five operating cycles which provided measured plant data from a range of plant startup and normal operation conditions.

2.4 Model Applications for Reactor Operating Support

Section 4 of the topical report discusses the application of the NSP models to both predictive and plant monitoring modes.

2.5 Model Applications to Safety Evaluation Analyses

Section 5 of the topical report describes the methods used to apply the reliability factors and biases to calculational physics results for safety applications.

3.0 TECHNICAL EVALUATION

Background

The previously approved YAEC topical report (YAEC-1363) for CASMO-3 applications included a detailed description of the neutronics modeling methodology together with the YAEC validation of the code system. The basic nuclear cross-section data, unit cell calculation, two-dimensional transport theory and diffusion theory calculations, and the determination of flux discontinuity factors for use in SIMULATE-3 were described.

The original CASMO-3 validation was carried out by the code developer - Studsvik Energiteknik. This benchmarking included the calculation of a set of pin-cell critical experiments, with varying pin radius and pitch, and fuel enrichments. The YAEC validation was based on comparisons with measured critical experiments, measured fuel isotopics, and measured pin-wise La-140 distributions. These comparisons were intended to exercise and validate the depletion calculation, the spatial transport calculation and the nuclear data library. The fuel depletion calculation was validated by comparisons with the Yankee Core-1 and Zion measured uranium and plutonium isotopics which are industry-standard benchmark sources. These comparisons were performed for a range of pin-cell spectra and indicated good agreement for the fuel isotopics versus burnup. As further validation, a set of uniform critical measurements were also calculated. CASMO-3 reproduced 74 criticals to within 1 percent delta-k/k. The comparisons were analyzed as a function of rod pitch, fuel enrichment, H_2O/U-ratio, soluble boron, buckling and moderator temperature, and no significant dependence of the calculation/measurement differences was observed.

In addition to the measurement benchmarks, the YAEC CASMO-3 calculation of the Brookhaven National Laboratory (BNL) Fuel Assembly Standard Problem was compared to the BNL reference solution. Comparisons of reactivity defects, control rod worth, boron worth, fuel isotopics, and pin-wise power distributions were made. The agreement was found to be very good, with the observed differences within the stated uncertainty of the BNL reference solution.

The previously approved YAEC topical report (YAEC-1659) for SIMULATE-3 applications focused upon three major areas. The first was application to operating pressurized water reactors (PWRs) and included comparisons of SIMULATE-3 generated parameters to measured data, as well as to the BNL PWR Core Standard Problem. The second application was to operating BWRs and included comparisons to measured data. The third area focused on the pin-by-pin power distribution capabilities of SIMULATE-3. This application compared multi-assembly SIMULATE-3 pin-by-pin power distributions to higher order transport theory solutions. In addition, pin-by-pin power distributions were compared between SIMULATE-3 and previously accepted PDQ-07 methods of pin power distribution calculations.

The statistics from the cold (85°F to 209°F) zero-power comparisons quantify the model accuracy for predicting reactivity for beginning-of-cycle (BOC), xenon-free and in-cycle restart conditions. Thirty-three measurements from the five operating cycles are included. Sixty-eight at-power statepoints with TIP [traversing incore probe] traces are used for reactivity comparison and power distribution reliability factors.

The statistical analysis described in Appendix A was performed on the measured versus the SIMULATE-3 calculated reactivities and TIP reaction rates.

The sample distributions were tested for normality using standard methods. The normality test is used since the standard 95 percent probability at the 95 percent confidence level [95/95] tolerance limit method assumes that the population has a normal distribution. If the distributions are not normal, but are known, a special treatment (Appendix A) allows equivalent 95/95 statistics to be generated. Parameters not covered by the above are conservatively bounded.

Control rod worths

The SIMULATE-3 prediction of control rod worths was compared by NSP with the BOC zero-power startup measurements for the five operating cycles. A statistical analysis of the control rod worth differences determined the bias, standard deviation and the normality of the difference distribution. The SIMULATE-3 capability to predict the shutdown margin with the worst stuck rod was qualified by comparison to local critical measurements as well as in-sequence rod withdrawal criticals

Assembly power distribution

The SIMULATE-3 calculated incore detector reaction rates and assembly power distributions were verified at NSP by comparison with direct incore signal measurements. A total of 68 incore detector (TIP) statepoints were taken at close to Hot-Full-Power (HFP) conditions from Cycles 11 through 15. The predicted reaction rates were compared with the measured signals by individual detector, assembly location, and radial level to determine the mean and standard deviation for the observed differences. The 95/95 tolerance limits for assembly peaking factors were calculated from multiplying the standard deviations by the k-value corresponding to the sample size for all statepoint conditions.

4.0 SUMMARY AND CONCLUSIONS

Northern States Power Company (NSP) has performed extensive benchmarking using the CASMO-3/SIMULATE-3 methodology. This effort consisted of detailed comparisons of calculated key physics parameters with the measurements obtained from five operating cycles of the Monticello BWR plant. These results were used to determine the set of 95/95 (probability/confidence) tolerance limits for application to the calculation of the stated BWR physics parameters. This effort also demonstrated the ability of NSP to use the CASMO-3/SIMULATE-3 computer program package for application to the Monticello BWR unit.

Based on the analyses and results presented in the topical report, the staff concludes that the CASMO-3/SIMULATE-3 methodology as validated by NSP can be applied to steady-state BWR reactor physics calculations for reload applications as discussed in the above technical evaluation. The accuracy of this methodology has been demonstrated to be sufficient for use in licensing applications, including BWR reload core physics analysis, generation of safety and transient analysis inputs, startup and control rod worth predictions, and core monitoring system support.

As in the earlier approvals, application of the approved package is limited to the range of fuel configurations and core design parameters verified and referenced by this topical report; introduction of significantly different fuel designs may require further validation by the licensee.

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5.0 REFERENCES

1. Letter from R. O. Anderson (NSP) to Document Control Desk (USNRC), regarding "Revision 2 of Topical Report NSP-8609, 'Qualification of Reactor Physics Methods for Application to Monticello'," dated August 23, 1994.

2. NSPNAD-8609, Rev 2, "Qualification of Reactor Physics Methods for Application to Monticello," Northern States Power Company, August 1994.

3. Letter from A. C. Thadani (USNRC) to G. Papanic, Jr. (YAEC), regarding "Acceptance for Referencing of Topical Report YAEC-1363, 'CASMO-3G Validation'," March 21, 1990.

4. Letter from A. C. Thadani (USNRC) to G. Papanic, Jr. (YAEC), regarding "Acceptance for Referencing of Topical Report YAEC-1659, 'SIMULATE-3, Validation and Verification'," February 20, 1990.

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7. M. Edenius, A. Ahlin, B. H. Forssen, "CASMO-3, A Fuel Assembly Burnup Program, User's Manual, Version 4.4," STUDSVIK/NFA-89/3, Studsvik Energiteknik AB, Sweden, November 1989. [Proprietary - Not publicly available]

8. K. S. Smith, J. A. Umbarger, D. M. Ver Planck, "SIMULATE-3, Advanced Three-Dimensional Two-Group Reactor Analysis Code, User's Manual, Version 3.0," STUDSVIK/SOA-89/03, Studsvik of America, Inc., November 1989. [Proprietary - Not publicly available]

9. K. S. Smith, J. A. Umbarger, D. M. Ver Planck, "TABLES-3, Library Preparation Code for SIMULATE-3, User's Manual, Version 3.0," STUDSVIK/SOA-89/05, Studsvik of America, Inc., November 1989. [Proprietary - Not publicly available]

10. "ESCORE - The EPRI Steady-State Core Reload Evaluator Code: General Description," EPRI-NP-5100, February 1987.

11. Letter from A. C. Thadani (USNRC) to C. R. Lehmann (Pennsylvania Power and Light Co.), regarding "Acceptance for Referencing of Licensing Topical Report EPRI-NP-5100, 'ESCORE - The EPRI Steady-State Core Reload Evaluator Code: General Description'," May 23, 1990.