UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO TOPICAL REPORT EMF-96-029(P), "REACTOR ANALYSIS SYSTEM FOR PWR'S" SIEMENS POWER CORPORATION

1. INTRODUCTION

In a letter of May 8, 1996 (Ref. 1), Siemens Power Corporation (SPC) submitted Volumes 1 and 2 of the topical report EMF-96-029(P), "Reactor Analysis System for PWR's" (Ref. 2) for U.S. Nuclear Regulatory Commission (NRC) review. This topical report presents a core physics computer code system for pressurized water reactors (PWRs) which will be used by SPC to perform neutronics design analyses. The code system, named SAV95, will replace the XTGPWR/PDQ/CASMO-2E codes currently used by SPC.

SAV95 consists of a neutron cross section generator computer code system (MICBURN-3/CASMO-3G) and a reactor core simulator computer code system (PRISM). The cross section generator is used to calculate basic nuclear parameters that are required by the reactor core simulator. Input to the cross section generator includes a nuclear data library as well as user input describing the assembly lattice. The reactor core simulator is used to model the reactor core and perform the basic core calculations required for fuel cycle design and safety analyses, which include the required input for an incore monitoring code.

2. SUMMARY OF TOPICAL REPORT

Section 1 of Volume 1 of the topical report presents an introduction describing the type of neutronic calculations to be performed by the SAV95 code system and the improvements of SAV95 over SPC's previous neutronics design codes. Section 2 presents the development goals and basis, the new or improved models, and the major characteristics of the SAV95 code system. The neutron cross section generation scheme, including the nuclide chain used in MICBURN-3 and the representation of nodal cross sections for PRISM, are described in Section 3. Section 4 presents the structure and main components of the PRISM code, and references are given in Section 5.

Volume 2 of the report presents comparisons of data calculated with the SAV95 code system to measured data obtained from critical experiments, startup physics tests, and core follow data obtained from commercial power reactors. Additionally, comparisons to data calculated with previously approved analytical models are presented.

3. TECHNICAL EVALUATION OF REPORT

The MICBURN-3 (Ref. 3) and CASMO-3G (Ref. 4) codes are used to generate the microscopic and macroscopic lattice cross sections versus burnup (exposure) and fuel rod by rod power distributions versus burnup. The NRC has approved

these codes for use by SPC for boiling water reactor (BWR) neutronics calculations (Ref. 5). MICBURN-3 is a multigroup, one-dimensional transmission probability code that calculates neutron cross sections as a function of burnup in an absorber rod containing an initially homogeneous distribution of burnable absorber. These cross sections as functions of absorber number density are input to CASMO-3G. CASMO-3G is a multigroup, two-dimensional transport theory code used for burnup calculations on PWR and BWR fuel rods or assemblies. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with a burnable absorber, burnable absorber rods, cluster control rods, incore instrument channels, water gaps, boron steel curtains, and cruciform control rods in the regions separating fuel assemblies. The assembly lattice cross section data and reactor core description (e.g., assembly loading pattern, control rod position, core power, core inlet temperature) are input to the reactor core simulator code system, PRISM, which calculates k_{eff}, boron concentration, power distributions, peaking factors, control rod worths, and input for safety analyses and incore monitoring. PRISM uses the nodal expansion method (NEM) to solve the two-group diffusion theory representation of the reactor core.

A number of new models have been incorporated into SAV95. These include an extension of the available nuclide chain for isotopic depletion, a continuous neutron cross section representation covering all possible combinations of thermal-hydraulic parameters, a faster and more efficient flux solution module for stationary reactor states, the introduction of discontinuity factors at the boundary between assemblies or quadrants and for reflector regions, and a full three-dimensional fuel rod interpolation scheme in the determination of rodwise flux, power, and burnup values. These changes are acceptable because they offer improvements which result in both increased accuracy and efficiency and which enhance ease of use.

In order to qualify the SAV95 code system, SPC has compared data calculated with SAV95 to measured data from critical experiments as well as to startup physics test data and core follow data obtained from commercial power reactors. The validation criteria used for the SAV95 model are based on those suggested in ANSI/ANS-19.6.1-1985, "American National Standard Reload Startup Tests for Pressurized Water Reactors" (Ref. 6).

3.1 Critical Experiment Reactivity Measurements

Calculated results from MICBURN-3/CASMO-3 were compared to measured data from the Strawbridge-Barry (Ref. 7), KRITZ (Ref. 8), and the Babcock and Wilcox (Ref. 9) critical experiments and the calculated mean value of k_{eff} was 1.00039 ± 0.00107. The calculations were performed by Studsvik, not by SPC. However, the very good agreement between the calculations and the measurements validates the ability of the cross section generation portion of the SAV95 code system to accurately predict reactivity.

3.2 Power Reactor Measurements

Two Westinghouse (W) plants containing 157 fuel assemblies (one 15x15 fuel rod array and one 17x17 fuel rod array) and one Combustion Engineering (CE)

reactor containing 217 fuel assemblies with a 14x14 fuel rod array were used in the validation against commercial power plant measurements. A total of 14 cycles of operation were evaluated with at least three cycles evaluated for each plant. SPC compared the SAV95 model predictions to startup physics test data as well as to core follow measurements from each plant.

Startup physics test measurements are typically performed at hot zero power (HZP) conditions, and include critical boron concentrations, control bank worths, and isothermal temperature coefficients. The results of SAV95 comparisons with startup physics test measurements show a maximum absolute difference of 46 ppm in the all rods out (ARO) HZP critical boron concentration compared to the recommended validation test criterion of ±50 ppm (Ref. 6). The maximum absolute difference between calculated and measured individual HZP control bank worths was 14.3% and 93 pcm (where 1 pcm = $1x10^{-5}$ $\Delta k/k$) compared to the Reference 6 criteria of ±15% or ±100 pcm, whichever is larger. The maximum absolute difference between predicted and measured total HZP control bank worths was 7.3% compared to the recommended validation criterion of ±10%. The maximum absolute difference between the predicted and measured ARO HZP isothermal temperature coefficient was 1.01 pcm/°F compared to the validation criterion of ±2 pcm/°F. Therefore, SAV95 predictions of startup physics measurements met all applicable criteria.

Core follow measurements obtained during operation typically are measured at hot full power (HFP), and include critical boron concentrations and power distributions as a function of burnup. The results of SAV95 comparisons with core follow measurements show a maximum absolute difference of less than 50 ppm between the measured and calculated HFP critical boron concentration as a function of core burnup, thus meeting the validation criterion of ≤50 ppm recommended in Reference 6. The root mean square (RMS) difference of 0.018 between predicted and measured assembly average power distributions at beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) is well within the recommended criterion (Ref. 6) of <0.05. Comparisons of measured and predicted core average axial power distributions at BOC, MOC, and EOC show a maximum RMS difference of 0.049, which is within the recommended criterion of <0.05. Therefore, SAV95 predictions of core follow measurements met all applicable criteria.

3.3 Previously Approved Methodology Calculations

In addition to comparisons involving measured data, selected safety analysis parameters (Doppler coefficient, differential boron worth, and delayed neutron fraction) were compared to values calculated with the previously approved neutronics design methodology, CASMO-2/XTGPWR (Ref. 10). The good agreement in these comparisons demonstrate that the proposed new SAV95 methodology is compatible with the safety and licensing analyses.

3.4 Verification of Power Distribution Measurement Uncertainty

Since a replacement is proposed for the previously approved neutronics methodology, verification of the continued applicability of previously approved power distribution measurement uncertainties was performed. The measurement uncertainty for the power distribution peaking factors was

verified for two specific incore monitoring code systems, the Westinghouse design using movable incore detectors (Ref. 11), and the CE design using fixed incore detectors (Ref. 12).

The standard deviations of the relative uncertainties associated with the assembly, planar, or nodal power distribution were determined for the movable detector incore monitoring system (Westinghouse), using measured data from 11 cycles of operation of two different reactors, and for the fixed detector system (CE), using measured data from three cycles of a single reactor. The standard deviations of the relative uncertainty in the local peaking factor were determined by comparisons of calculated rod-by-rod fission rate distributions with critical experiment measurements. The standard deviations were statistically combined and expressed in terms of relative standard deviations. The data reduction and statistical techniques described in References 11 and 12 were used to verify that the one-sided 95/95 relative uncertainties are less than the measurement uncertainties previously approved by the NRC for the specified incore monitoring system. Therefore, the continued use of the previously approved power distribution measurement uncertainties for the Westinghouse and the CE detector systems with the proposed new methodology is acceptable. An extension of the methodology to other incore monitoring systems will require additional validation and verification of the acceptable uncertainties.

4. SUMMARY AND CONCLUSIONS

The NRC staff has reviewed the proposed SAV95 methodology as well as comparisons of the SAV95 code system with measured data from critical experiments, operating reactors, and previously approved methodology calculations. On the basis of this review, the staff finds the use of SAV95 acceptable for use by SPC in PWR reload core design, safety analysis parameter calculations, and startup and operations calculations. As stated in a letter of October 11, 1996, from SPC to the NRC (Ref. 13), SPC will impose the following restrictions on their use of the SAV95 neutronics methodology described in EMF-96-029(P), Volumes 1 and 2. The specific restrictions are:

1) SAV95 application will be supported by additional code validation to insure that the methodology and uncertainties are applicable:

   a) For designs differing from the Westinghouse reactors with 157 fuel assemblies with either 15x15 or 17x17 fuel rod arrays, and CE reactors with 217 fuel assemblies with 14x14 fuel rod arrays.

   b) When using incore monitoring systems differing from the INPAX-W and INPAX-2 systems contained in this safety evaluation when SPC provides input from SAV95.

2) Modifications to the code and methodology will be validated using the criteria approved in EMF-96-029(P).

3) The validation will be maintained by SPC and be available for NRC audit.

5. REFERENCES

(1) Letter from R. A. Copeland (SPC) to Document Control Desk (NRC), transmittal of EMF-96-029(P), Volumes 1 and 2, and EMF-96-029(NP), Volumes 1 and 2, "Reactor Analysis System for PWR's", RAC:96:043, May 8, 1996.

(2) S. K. Merk, et al, "Reactor Analysis System for PWR's," Volume 1, "Methodology Description," and Volume 2, "Benchmarking Results," EMF-96-029(P), May 1996.

(3) M. Edenius, et al, "MICBURN-3, Microscopic Burnup in Burnable Absorber Rods: Methodology," STUDSVIK/NFA-86/28, 1986.

(4) M. Edenius, et al, "CASMO-3, A Fuel Assembly Burnup Program: Methodology," STUDSVIK/NFA-86/8, 1986.

(5) Letter from A. C. Thadani (NRC) to R. A. Copeland (ANF), "Acceptance for Referencing of Topical Report XN-NF-80-19(P), Volume 1, Supplement 3, Advanced Nuclear Fuels Methodology for Boiling Water Reactors; Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology,'" August 13, 1990.

(6) American National Standard Reload Startup Tests for Pressurized Water Reactors, ANSI/ANS-19.6.1-1985, American Nuclear Society, 1985.

(7) L. E. Strawbridge and R. F. Barry, "Criticality Calculations for Uniform Water-Moderated Lattices," Nuclear Science and Engineering, Vol. 23, pp. 58-73, 1965.

(8) R. Persson, et al, "High-Temperature Critical Experiments with H_2O-Moderated Fuel Assemblies in KRITZ," Technical Meeting No. 2/11, NUCLEX 72, 1972.

(9) L. W. Newman, "Urania-Gadolinia: Nuclear Model Development and Critical Experiment Benchmark," BAW-1810, Babcock and Wilcox Company, April 1984.

(10) "Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors," XN-75-27(A), and Supplements 1, 2, 3, 4, and 5, Exxon Nuclear Company.

(11) "Power Distribution Measurement Uncertainty for INPAX-W in Westinghouse Plants," EMF-93-164(P)(A), Siemens Power Corporation, February 1995.

(12) "Exxon Nuclear Analysis of Power Distribution Measurement Uncertainty for St. Lucie Unit 1," XN-NF-83-01(P), Exxon Nuclear Company, January 1983.

(13) Letter from R. A. Copeland (SPC) to Document Control Desk (NRC), SPC Restrictions on SAV95, RAC:96:066, October 11, 1996.