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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT ENEAD-01-P

"QUALIFICATION OF REACTOR PHYSICS METHODS FOR APPLICATION TO THE PWRs OF THE ENTERGY SYSTEM"

ENTERGY OPERATIONS, INC.

1. INTRODUCTION

In a letter of January 7, 1994 (Ref. 1), from F. W. Titus, Entergy Operations, Inc. (EOI) to the U.S. Nuclear Regulatory Commission (NRC), EOI submitted Topical Report "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," ENEAD-01-P, Revision 0 (Ref. 2), for NRC review. The report describes the CASMO/SIMULATE reactor models used at EOI and the methodology used to determine uncertainties and the resultant 'reliability factors'. CASMO/SIMULATE can be used to provide operations support in the form of reactivity balance data and predictions of physics parameters, and for reload physics, core operating limits and reload safety analyses for Pressurized Water Reactor (PWRs). Both the CASMO-3 and SIMULATE-3 computer program packages have been reviewed and accepted for referencing (with certain restrictions) by separate NRC Safety Evaluation Reports (SERs) (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 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.

EOI intends to use the CASMO/SIMULATE programs in steady-state PWR reload core physics design and licensing applications, calculations for startup predictions, generation of physics input for safety analyses, and input for the reactor core monitoring systems (CMS). EOI provides reload cycle specific support to Arkansas Nuclear One, Unit 1 (ANO-1), Arkansas Nuclear One, Unit 2 (ANO-2) and the Waterford Steam Electric Station, Unit 3 (WSES-3). The existing models are based on using the PDQ and EPRI-NODE-P computer codes, described in the NRC approved Entergy PWR Physics Topical Report, MSS-NA1-P (Ref. 7).

2. TOPICAL REPORT SUMMARY

Topical Report ENEAD-01-P compares the CASMO/SIMULATE model results with measurements obtained from benchmarking data covering 22 operating cycles of

Enclosure

the Arkansas Nuclear One - Unit 1 (ANO-1), Arkansas Nuclear One - Unit 2 (ANO-2), and the Waterford Steam Electric Station - Unit 3 (WSES-3). Comparisons were also made to results from ten critical experiments measured by Babcock and Wilcox (B&W) and ABB Combustion Engineering (ABB/CE), and to spent fuel isotopic data from five fuel rods from the Zion Unit 1 high burnup irradiation program (Ref. 8). The plant analyses were performed over a wide range of conditions from hot zero power (HZP) temperature to full power (HFP) operation. The good agreement between the measured and calculated values presented in the topical report is used to validate the application of the computer programs for analysis of the EOI PWR units.

Section 1 of the topical report provides introductory background information and an overview of the scope of the report. It also references the key PWR physics parameters for which the bias and reliability factor are determined. These are:

- Core reactivity (hot zero power, critical ppm boron)
- Assembly and pin power peaking (F_q, F_r, and F_{xy})
- Control rod worth (%ρ)
- Inverse boron worth (ppm boron per %ρ)
- Isothermal temperature coefficient (delta ρ per degree F)
- Doppler coefficient
- Isotopics (U-235, Pu-239)
- Delayed neutron parameters (core effective delayed neutron fraction, effective neutron lifetime)

Section 2 of the topical report describes the three EOI reactors (ANO-1, ANO-2 and WSES-3) which provided measured plant data from a range of plant startup and normal operation conditions.

ANO-1 is a Babcock and Wilcox PWR lowered-loop plant with a 15x15 fuel rod array, 177 fuel assembly core, generating 2568 megawatts-thermal (MWt) at rated power, which began commercial operation in 1974. There are currently 60 full-length and eight part-length control rod element assemblies (CEAs). Burnable poison rod assemblies (BPRs) may be included in the reload fuel. The in-core flux instrumentation consists of 52 strings of seven 4.75 inch long fixed self-powered rhodium (Rh) in-core neutron detectors.

ANO-2 and WSES-3 are both two-loop Combustion Engineering (CE) PWR plants with a 16x16 fuel rod array. The control element assemblies typically consist of five neutron absorber rods, with the part-length (PL) CEAs containing non-absorber segments. In addition, burnable poison absorber rods, or integral fuel-burnable poison shim rods are selectively used to replace standard fuel rods for reactivity depletion control.

ANO-2 is a 177 fuel assembly core, generating 2815 MWt at rated power, which began commercial operation in 1980. ANO-2 has 73 full-length and eight (PL) CEAs. The ANO-2 Cycle 1 core also had part-length axial and asymmetric

radial BPR distributions. The in-core instrumentation consists of 44 fixed strings each containing five 15.75 inch long fixed axial self-powered Rh detectors.

WSES-3 is a 217 fuel assembly core, generating 3390 Mwt, which began commercial operation in 1985. WSES-3 has 83 full-length and eight PL CEAs. There are 56 strings of five 15.75 inch long fixed Rh detectors

For ANO-2 and WSES-3, the neutron flux detector signals are processed with the CE CECOR program (Ref. 9) to infer the measured 3D core power distribution. The algorithms employed by CECOR synthesize the three-dimensional power distribution from the incore detector readings using the coefficient library generated from the CASMO/SIMULATE model.

Section 3 of the topical report describes the EOI-specific CASMO/SIMULATE computer program package methodology, provides references for the individual components, and outlines the procedures used for the model applications.

CASMO is the lattice physics code (Ref. 10) originally developed by Studsvik Energiteknik and obtained by EOI from Studsvik of America (SOA). It is used by EOI to determine the neutronics input to SIMULATE for PWR core performance analyses. CASMO uses a binary-format cross section library based on standard ENDF/B cross-section sets, that is created by an auxiliary code CASLIB.

SIMULATE (Ref. 11) was also acquired from SOA. The code is based on a modified coarse mesh (nodal) diffusion theory calculational technique, with coupled thermal hydraulic and Doppler feedback. SIMULATE also calculates control rod worth and moderator, and xenon feedback effects. 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 flux continuity at nodal interfaces and perform an accurate determination of the pin-wise power distribution, SIMULATE uses assembly discontinuity factors that are pre-calculated by CASMO. 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 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 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 and then processed by the TABLES code (Ref. 12) for generation of the cross section library used by SIMULATE.

SIMULATE 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, and also calculates neutron flux at the in-core detector locations. An auxiliary code, RHOBURN, calculates fixed RH in-core detector signals from the CASMO/SIMULATE flux and power distributions. RHOBURN also generates data for the CECOR data library generation program.

INTERPIN (Ref. 13) is an SOA computer code developed for predicting steady-state fuel rod performance parameters for zircaloy-clad light-water reactor (LWR) fuel rods. This program is used in calculating best-estimate fuel rod temperatures for input to CASMO and SIMULATE codes, as a function of burnup and power history.

The previously approved YAEC topical report (YAEC-1363) for CASMO 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 were described. The original CASMO 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 reproduced seventy-four criticals to within 1% 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 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 applications focused upon three major areas. The first was application to operating PWRs and included comparisons of SIMULATE generated parameters to measured data, as well as to the BNL PWR Core Standard Problem. The second application was to operating Boiling Water Reactors (BWRs) and again included comparisons to measured data. The final application focused on the pin-by-pin power distribution capabilities of SIMULATE. This application compared multi-

assembly SIMULATE pin-by-pin power distributions to higher order transport theory solutions. In addition, pin-by-pin power distributions for an operating PWR were compared between SIMULATE and the previously accepted PDQ-7 method of pin power distribution calculations.

Section 4 of the topical report presents the model verification by comparisons of the calculated key parameters (listed in Table 4.1-1) with the measured or inferred plant data and develops both the calculational uncertainty and tolerance factors to define the 'reliability factor'. The measured data cover the range from hot zero power startup testing to normal full power operations at the three EOI units. Six cycles from ANO-1, ten cycles from ANO-2, and six cycles from WSES-3 were included for a total of 22 operating cycles, covering initial and reload cores.

In a related submittal (Ref. 14), Topical Report ENEAD-02-NP, Revision 0, EOI proposes to use the reload design computer codes CASMO and SIMULATE to support the CECOR CMS at ANO-2 and WSES-3. These codes will be used to generate input (library coefficients) to the Entergy CECOR core power distribution monitoring computer program. CECOR synthesizes detailed three-dimensional fuel assembly and peak fuel pin power distributions from fixed incore detector signals.

3. TOPICAL REPORT EVALUATION

The NRC staff has reviewed the information presented in topical report ENEAD-01-P, Revision 0, with regard to the analytical methods and the statistical methods. The data base used to establish the comparison between measurement and calculation in determining the basic measurement uncertainty was examined. For each of the parameters compared, the sample mean (x-bar) and standard deviation (S) of the observed relative or absolute differences were calculated using standard accepted statistical techniques, including the ANSI standard normality tests (Ref. 15), and the Bartlett test for poolability (Ref. 16). Based on the sample means, the standard deviations, and the sample size, a set of conservative 95/95 tolerance limits (bias ± reliability factor) was calculated using the methods of Reference 17.

Critical boron concentration and target eigenvalue

The SIMULATE model predictions of critical boron concentration (CBC) and reactivity were compared to zero-power startup test measurements and to power operating data for the six cycles of ANO-1, ten cycles of ANO-2 and six cycles of WSESS-3 operation. Differences between calculated and measured boron ppm data are stated in absolute terms, measured minus predicted (m-p).

The statistics from the zero-power comparisons quantify the model accuracy for predicting the CBC for beginning-of-cycle (BOC), xenon-free conditions. Nineteen measurements from 20 cycles of startup tests are included. Two of the measurements were taken with the reactor critical during initial cycle startups. A statistical analysis was performed on the measured versus the SIMULATE calculated CBC differences. First, the sample mean and standard

deviations were calculated for CBC differences. These differences are due to SIMULATE model uncertainties, variations in B-10 isotopic concentrations and measurement (titration) uncertainties. For example, boron concentration measurement errors (estimated at ± one percent) can be as high as ±15 ppm. For conservatism, all observed differences are assumed due only to SIMULATE calculational uncertainties. Second, the two sample distributions were tested for normality. The normality test is used since the 95/95 tolerance limit method assumes that the population has a normal distribution. The test concluded that the CBC difference distribution was normal. Finally, the bias, 95/95 reliability factor and tolerance limit were calculated. The resulting 95/95 tolerance limits for hot zero power CBC predictions are +7.89 (bias) + 37.29 (RF) equals 45.18 ppm.

The at-power reactivity comparison results, accounting for control rod insertion, power (60-100%) with thermal feedback and xenon conditions, are used as estimates of the model uncertainty for all power conditions. There are a total of 279 measurements from 22 operating cycles. The SIMULATE reactivity eigenvalue (lambda) is calculated for each case. The mean eigenvalue and standard deviation for all at-power reactivity calculations are 0.9991 and 0.0017 delta-k, respectively. All data statepoints were within ±0.00548 delta-k.

Inverse boron worth

The SIMULATE calculated inverse boron worths (IBW) were compared with a total of 17 measurements from 18 cycles of operation for the three units. The IBW is calculated using:

$$IBW = -(CBC1 - CBC2) / (\text{delta-Reactivity})$$

where CBC1 is the critical boron concentration for Statepoint 1), CBC2 is for Statepoint 2, and delta-Reactivity is the reactivity change (% delta-rho) required to go from the SP-1 to SP-2 conditions. This reactivity change is accomplished by control rod insertion or withdrawal. The m-p differences at BOC, zero-power conditions show a sample mean and standard deviation of -0.81 and 3.60 ppm/% delta-rho, respectively, leading to a 95/95 reliability factor of 8.58 ppm/% delta-rho for the uncertainty in the calculation of the boron worth.

Control rod worths

The SIMULATE prediction of control rod worths was compared by EOI at BOC zero-power startup conditions for sixty measurements from 20 operating cycles using the boron dilution technique, and for forty measurements from 10 cycles using the rod swap technique. A statistical analysis of the control rod worth differences determined the bias and standard deviation and the normality of the difference distribution. The bias and standard deviation are -0.005 and 0.058 % delta-rho for the boron dilution and -0.011 and 0.073 % delta-rho for the rod swap technique, respectively. The observed uncertainty has two components: the measurement uncertainty and the calculational uncertainty.

The measurement uncertainty is not estimated, so the control rod worth calculational uncertainty is equal to the observed uncertainty. The 95/95 reliability factors for the calculational error were calculated as 0.117 % delta-rho for boron dilution and 0.155 % delta-rho for rod swap, respectively. The 95/95 tolerance limit (bias ± reliability factor) then becomes (-0.005 - 0.117) = -0.122 for boron dilution and (-0.011-0.155) = -0.166 % delta-rho for rod swap, respectively. This tolerance limit will be applied to the SIMULATE calculation of CEA worths at all power and moderator temperature conditions.

Isothermal temperature coefficient

EOI used SIMULATE to calculate the isothermal temperature coefficient (ITC) at HZP, all rods out over a range of soluble boron concentrations (800 to 1750 ppm), comparing with a total of 20 measurements. The ITC is defined as the change in reactivity due to a incremental (one degree F) change in the core average moderator and fuel temperature. A statistical analysis was performed on the ITC difference, (m-p)distribution. The sample mean and standard deviation was -0.077 and 0.035 x 10^{-4} delta-rho/°F, respectively. The resulting reliability factor was 0.084 x 10^{-4} delta-rho/°F.

Isotopic compositions

EOI used INTERPIN/CASMO to model the depletion of five fuel rods from two Zion Unit-1 fuel assemblies (C63 & C64) from the high burnup irradiation program reported in Reference 8. Fuel rod burnups ranged from 23.5 to 51.7 GWd/MtU. Total U and Pu calculated number densities were compared to the measurements, as well as isotopic ratios. The mean of the relative differences (m-c)/c and reliability factors were determined for 14 isotopics. The root-mean-square (rms) errors were also compared to the reported YAEC-1363 results. All measured data was bounded by the upper and lower tolerance limits of the calculated isotopics.

Doppler coefficient

Previous benchmarking against industry standard Hellstrand measured resonance integrals and Doppler Coefficients has shown CASMO to be within 3% of the Hellstrand results (Ref. 18). EOI has also evaluated proprietary SOA comparisons between CASMO and MCNP-3A to verify that they are within the 10% multiplicative Doppler coefficient reliability factor approved in Reference 7.

Delayed neutron parameters

EOI re-examined the total uncertainty of the core effective delayed neutron fraction (β_{eff}) and the effective neutron lifetime (l^*) to verify that the currently approved 3% reliability factors from Reference 7 are still bounding.

Power distributions

The SIMULATE calculated rhodium incore detector reaction rates and assembly power distributions were verified by comparison with direct incore background-corrected signal measurements for ANO-1, ANO-2 and WSES-3. The key power distribution parameters are the:

1) Total peaking factor (F_q), the maximum local peak pin power to average power ratio;

2) Integrated radial peaking factor (F_r), the maximum axially integrated peak pin power to core average power ratio; and

3) Planar radial peaking factor (F_{xy}), the maximum planar peak pin power to planar power ratio.

A total of 276 incore detector snapshots were taken from 73% power to Hot-Full-Power (HFP) statepoint conditions from Cycles 7 through 11 of ANO-1, Cycles 1 through 10 of ANO-2 and Cycles 1 through 6 of WSES-3. The predicted reaction rates were compared with the measured signals to determine the bias (mean) and standard deviation for the observed differences. The measurements were taken at close to steady-state conditions and the results shown are for burnups near the BOC, MOC, and EOC exposure points. Radial and axial comparisons at each statepoint demonstrate that the SIMULATE/RHOBURN predicted detector reaction rates agree well with the measured values.

The local peaking factor uncertainty was quantified by CASMO/SIMULATE comparisons to ten critical experiments documented in References 19 and 20. These experiments were based on advanced PWR lattice designs using gadolinium (Gd) and erbium (Er) as burnable poisons (BP) within the UO_2 fuel rod matrix, as well as unpoisoned lattices. Two core types were benchmarked, 15x15 fuel pin arrays with 17 small water holes (similar to B&W designs), and 16x16 fuel pin arrays with 5 large water holes (similar to ABB/CE designs). Fuel enrichments ranged from 1.94 to 4.02 wt% U-235, with zero to 4 wt% Gd_2O_3 loading. The mean, standard deviation, rms difference and the tolerance limit values were determined with the standard statistical methods previously referenced. The variances and standard deviations of the absolute differences between the measured and calculated pin powers were determined for the group of all lattices and for subgroups by core type. Since the data from the different core-type categories were not poolable by Bartlett's test (Ref. 16), the largest relative standard deviation for the local peaking error (0.01261) was used to determine the combined tolerance limit for the nodal and local peaking uncertainties.

The 95/95 reliability factors for assembly peaking factors (F_q, F_r and F_{xy}), based on detector reaction rate signal comparisons, were calculated by multiplying the standard deviations by the k-value corresponding to the size of each sample, giving values of 0.0458, 0.0362, and 0.0440 for ANO-1; 0.0392, 0.0318, and 0.0353 for ANO-2; and 0.0363, 0.0300, and 0.0332 for WSES-3, respectively. Converting from detector signal (reaction rate) to detector

'power' the 95/95 reliability factors become 0.0500, 0.0399, and 0.0481 for ANO-1; 0.0403, 0.0323, and 0.0366 for ANO-2; and 0.0402, 0.0303 and 0.0348 for WSES-3.

Core monitoring code Inputs

The CECOR program, (Ref. 9) used for ANO-2 and WSES-3, synthesizes three-dimensional power distributions from fixed incore detector readings. The detector signals (reaction rates) from the five axially spaced detectors in each string are background- and depletion-corrected and converted to detector powers. Coupling coefficients are used to calculate pseudo-detector powers in uninstrumented assemblies or assemblies with failed detectors. A five term Fourier fit constructs the assembly axial shape, based on the five detector powers. Calculation of the maximum 1-pin assembly peaks are done using 1-pin peaking library coefficients, which are functions of burnup, control rod position, and axial detector location (Ref. 14). The information necessary to generate the CECOR data library comes from three-dimensional, quarter core, full power, SIMULATE nodal code depletion calculations and from CASMO lattice physics calculations.

4. CONCLUSIONS

Entergy Operations, Inc. has performed extensive benchmarking using the CASMO/SIMULATE methodology. This effort consisted of detailed comparisons of calculated key physics parameters with measurements obtained both from the EOI operating PWRs and from the relevant critical experiments. These results were used to determine the set of 95/95 (probability/confidence) tolerance limits for application to the calculation of the stated key PWR physics parameters. This effort also demonstrated the ability of EOI to use the CASMO/SIMULATE computer program package for application to their PWR units.

Based on our review of the experimental data base, the calculations performed, and the methods used to determine the components of uncertainty and the combination of these components into overall uncertainties, we find the methodology described in ENEAD-01-P acceptable for use by EOI for the ANO-1, ANO-2 and WSES-3 plants. Based on the analyses and results presented in the topical report, the staff concludes that the CASMO/SIMULATE methodology, as validated by EOI, can be applied to steady-state PWR 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 PWR reload physics analysis, generation of safety analysis inputs, startup predictions, and core physics databooks .

As in the earlier approvals, application of the approved package is limited to the fuel configuration and core design parameters verified in this and other referenced topical reports; introduction of significantly different fuel designs would require evaluation by the licensee and may require further validation of the models.

5. REFERENCES

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