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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT SCE-9001

"PWR REACTOR PHYSICS METHODOLOGY USING CASMO-3/SIMULATE-3"

SOUTHERN CALIFORNIA EDISON COMPANY

SAN ONOFRE NUCLEAR GENERATING STATION UNITS 1, 2, AND 3

DOCKET NOS. 50-206, 50-361, AND 50-362

1.0 INTRODUCTION

By letter dated October 22, 1990 (Ref. 1), the Southern California Edison Company (SCE) submitted the Topical Report SCE-9001, "PWR Reactor Physics Methodology Using CASMO-3/SIMULATE-3," (Ref. 2) for NRC review. This report documents the capability of SCE to implement and apply the stated methods to core reload physics design activities for the San Onofre Nuclear Generation Station (SONGS) Units 1, 2, and 3. 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.

SCE intends to use the CASMO-3/SIMULATE-3 programs in licensing applications, including PWR reload physics design, calculations for startup predictions, generation of physics input for safety analyses, core physics data books and set point updates for both the reactor protection and monitoring systems.

2.0 SUMMARY OF THE TOPICAL REPORT

Topical Report SCE-9001 compares the CASMO-3/SIMULATE-3 model results with measurements obtained from benchmarking data covering 15 operating cycles of the San Onofre Units 1, 2, and 3 and the initial cycle of Arkansas Nuclear One, Unit 2 (ANO-2), and from three Babcock and Wilcox critical experiments. The plant analyses were performed over a wide range of conditions from ambient temperature to full power operation. The good agreement between the measured and calculated values presented in the topical report is used to validate the

SCE application of the computer programs for analysis of the SONGS units.

SCE intends to use these methods for steady-state PWR core physics reload design and licensing applications, including fuel bundle and loading pattern analysis; for the generation of core physics databooks, startup predictions, safety analysis inputs; and for the calculation of setpoint updates for both the reactor protection system and the core monitoring system.

2.1 Overview

Section 1 of the topical report provides introductory background information and an overview of the scope of the report. It also lists the key PWR physics parameters for which uncertainty factors are determined. These are:

- Core reactivity
- Inverse boron worth
- Power coefficient
- Isothermal temperature coefficient
- Control rod worth
- Axial offset
- Assembly power peaking
- Pin peaking

A summary of the calculational bias and reliability factors is presented for each of these key parameters.

2.2 Methodology

Section 2 of the topical report describes the SCE-specific CASMO-3/SIMULATE-3 computer program package methodology, provides references for each of the individual components, and briefly outlines the procedures used by SCE for the model applications.

CASMO-3 is the Studsvik Energiteknik lattice physics code (Ref. 7) used by SCE in determining the neutronics input to SIMULATE-3 for PWR 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, that is created by an auxiliary code CASLIB (Ref. 8). Another auxiliary file management program, MOVEROD-3 (Ref. 9), is available for editing existing CASMO-3 restart-history files and rearranging individual fuel pin burnup data to allow detailed analysis of a specific reconstituted fuel assembly.

SIMULATE-3 was acquired from Studsvik of America (Ref. 10). 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 precalculated 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 albedos, 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. 11) 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. 12) for predicting best-estimate, steady-state fuel rod performance parameters for light-water reactor fuel rods. This program has been previously reviewed and approved (Ref. 13) 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 Data Sources

Section 3 of the topical report describes the four reactors which provided measured plant data from a range of plant startup and normal operation conditions.

SONGS-1 is a three-loop Westinghouse (W) PWR plant with a 14x14 fuel rod array, 157 fuel assembly core, generating 1347 megawatts-thermal (MWt) at rated power, which began commercial operation in 1968. There are 45 full-length rod cluster control assemblies containing silver-indium-cadmium (Ag/In/Cd) absorber material, clad in stainless steel. The incore flux instrumentation consists of two moveable fission chambers which can be inserted into 30 core locations. The neutron flux detector signals are processed offline with the W INCORE3 program (Ref. 14) to infer the 3D measured power distribution in the core.

SONGS-2 and SONGS-3 are two-loop Combustion Engineering (CE) PWR plants with a 16x16 fuel rod array, 217 fuel assembly core, generating 3390 Mwt at rated power, which began commercial operation in 1983 and 1984. There are 83 full-

length and eight part-length (PL) control element assemblies (CEAs) containing Inconel, Ag/In/Cd, and B_4C absorber materials, clad in Inconel-625. In addition, burnable poison absorber rods (B_4C-Al_2O_3) with Zircaloy-4 clad, are selectively used to replace fuel rods for reactivity depletion control. The incore instrumentation consists of 56 fixed strings, each containing five fixed axial rhodium detectors. The neutron flux detector signals are processed offline with the CE CECOR program (Ref. 15) to determine the measured 3D core power distribution.

ANO-2 is a two-loop CE PWR plant operated by the Arkansas Power and Light Company (AP&L) with a 16x16 fuel rod array, 177 fuel assembly core, generating 2815 Mwt at rated power, which began commercial operation in 1980. There are 73 full-length CEAs and eight PL CEAs containing the same absorber materials as SONGS 2 & 3, with the PL CEAs also containing a 58 inch water-filled segment. The burnable poison rods contain the same absorber material as SONGS 2 & 3, with the ANO-2 Cycle 1 core having part-length axial and asymmetric radial poison distributions. The incore instrumentation consists of 44 fixed Rh detector strings also identical to those in the SONGS 2 & 3 units.

2.4 Benchmark Comparisons

Section 4 of the topical report compares the calculated key parameters (listed in Section 2.1) with the measured or inferred plant data and develops both the calculational uncertainty and tolerance factors. The measured data cover the range from zero power startup testing to normal full power operations at the SONGS Units 1, 2, 3, and at the ANO-2 unit. Six cycles from SONGS-1, five cycles from SONGS-2, four cycles from SONGS-3, and the initial cycle from ANO-2 were included for a total of 16 operating cycles, covering initial and reload cores.

For each of the parameters compared, the sample mean, standard deviation, and the root-mean-square (RMS) of the observed differences were calculated using standard statistical techniques, including the ANSI standard normality tests (Ref 16), and the Bartlett test for poolability (Ref. 17). 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 18.

2.5 Pin Peaking Factor Uncertainty

Section 5 of the topical report references the previous YAEC benchmarking of the pin power reconstruction capabilities of SIMULATE-3 and presents the results of the additional SCE benchmarking effort. Three of the reported DOE-sponsored B&W critical experiments (Cores 01, 12, and 18) from Reference 19, which are closest to the SONGS lattice configurations (fuel pin and water hole dimensions), were selected for benchmarking of the pin power reconstruction capability of SIMULATE-3. The mean, standard deviation, RMS difference and the tolerance limit values were determined with the standard statistical methods previously referenced.

The SIMULATE-3 pin peaking factor uncertainties for the Planar Radial Peaking Factor (F_{xy}), the One-Pin Peaking Factor (F_Q), and for the Integrated Radial Peaking Factors (F_R, $F_{delta-h}$) were determined by combining the assembly power peaking uncertainties (S^S_{FXY}, S^S_{FQ}, and S^S_{FR}), from the benchmarking comparisons (Topical Report Section 4), with the appropriate uncertainty factor for the pin power reconstruction.

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% 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 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 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-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 for an operating PWR were compared between SIMULATE-3 and the currently accepted PDQ-7 method of pin power distribution calculations.

Critical boron concentration

The SCE SIMULATE-3 model predictions of critical boron concentration (CBC) and reactivity were compared to zero-power startup test measurements and full-power operating data for six cycles of SONGS-1, five cycles of SONGS-2 and four cycles of SONGS-3 operation. Differences between calculated and measured boron ppm data are stated in absolute terms, calculated minus measured (c-m). The SIMULATE-3 reactivity (1 - 1/Keff) is also calculated for each case.

The statistics from the zero-power comparisons quantify the model accuracy for predicting the CBC and reactivity for beginning-of-cycle (BOC), xenon-free conditions. Thirty-two measurements from 15 cycles of startup tests are included. Of these measurements, 17 are unrodded and 15 are with control rods inserted in the core. Five of the measurements were taken with the reactor critical at low temperatures during initial cycle startups.

The at-power comparison results, corrected for control rod insertion and for deviations from the full-power, equilibrium xenon conditions, are used as conservative estimates of the model uncertainty for all equilibrium power conditions with thermal feedback. There are a total of 112 measurements from 8 operating cycles. Two low-power CBC measurements, one each from Cycle 1 of SONGS 3 and Cycle 2 of SONGS 2 are also included to demonstrate that there is no significant increase in the difference at power levels less than 100%.

A three-step statistical analysis was performed on the measured versus the SIMULATE-3 calculated CBC differences and on the SIMULATE-3 calculated reactivities for the CBCs, as measured. First, the sample mean, standard deviation, and RMS were calculated for CBC and reactivity differences, respectively. These differences are due to SIMULATE-3 calculational uncertainties, variations in B-10 isotopic concentrations, and measurement (titration) uncertainties. For example, boron concentration measurement errors can be as high as 5 ppm. For conservatism, all differences are assumed due only to SIMULATE-3 calculational uncertainties.

Second, the two sample distributions were tested for normality using ANSI Standard N15.15-1974 (Ref. 16). The normality test is used since the 95/95 tolerance limit method assumes that the population has a normal distribution. The test concluded that both distributions, CBC and reactivity differences, are normal. Finally, the bias, 95/95 reliability factor and tolerance limit were calculated. The resulting 95/95 tolerance limits for zero-power CBC and reactivity predictions, for all temperatures and rodded conditions, are -7 ± 26 ppm and -0.08 ± 0.26 %delta-k/k, respectively. The 95/95 tolerance limits for all at-power and rodded or unrodded conditions for CBC and reactivity are 2 ± 34 ppm and 0.01 ± 0.35 %delta-k/k, respectively.

Isothermal Temperature Coefficient

SCE used SIMULATE-3 to calculate the isothermal temperature coefficient (ITC) over a wide range of soluble boron concentrations (145 to 2524 ppm) and of temperatures (150 to 583 degrees F), comparing with a total of 54 measurements from 14 cycles of operation. 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, (c-m), to determine the 95/95 tolerance limit for all power, moderator temperature and rodded conditions. The 95/95 tolerance limit (bias ± reliability factor) is $[0.05 \pm 0.24] \times 10^{-4}$ delta-k/k/°F.

Power coefficient

SIMULATE-3 predictions of the power coefficient were compared by SCE with seven measurements from four early cycles of SONGS Units 2 and 3 operation. The power coefficient is defined as the change in reactivity due to an incremental change in the core average power level (% power). The differences are given in absolute terms, (c-m). Due to the limited size of the database, a statistically meaningful 95/95 tolerance limit could not be derived. However, all of the differences are within 0.2×10^{-4} delta-k/k/%P, and the sample mean and standard deviation are 0.03 and 0.09, respectively. Since these differences include both the calculational and the measurement uncertainties, a conservative 95/95 tolerance limit of 0.2×10^{-4} delta-k/k/%P was assumed based on engineering judgement.

Control rod worths

The SIMULATE-3 prediction of control rod worths was compared by SCE with the BOC zero-power startup measurements for 13 operating cycles from SONGS Units 1, 2, and 3 at nominal and several off-nominal conditions.

A statistical analysis of the control rod worth differences determined the bias, standard deviation and the normality of the difference distribution. The bias and standard deviation are 1.18% and 4.89%, respectively. The uncertainty (S_{OBS}) has two components: the measurement uncertainty (S_M), and the calculational uncertainty (S_C). These two components are related to the observed uncertainty by,

$$S^2_{OBS} = S^2_M + S^2_C.$$

The measurement uncertainty is quantified by comparing the measured control rod worths from the initial startups of SONGS 2 and 3. Since the two units are duplicate plants (identical fuel enrichments, burnable absorber loading, etc.,), the measured control rod worths at the beginning of the first cycle should be the same. Therefore, the observed difference in the SONGS 2 and 3 measurements is attributable to the measurement uncertainty and manufacturing tolerances. For a total of seven rod worth measurements the standard deviation (S_D) of the difference in the measured rod worths, which includes

measurement uncertainties from the two measurements, is four percent. Therefore, the net measurement uncertainty was estimated as:

$$S^2_M = 1/2 * S^2_D = 8.00\%$$

The control rod worth calculational uncertainty can then be determined:

$$S_C = [S^2_{OBS} - S^2_M]^{1/2}$$
$$= [\ (4.89)^2 - (8.00)^2\ ]^{1/2}$$
$$= 3.99\%$$

Finally, the 95/95 reliability factor for the calculational error was calculated as:

$$\text{Reliability Factor} = K_{95/95} * S_C$$

where $K_{95/95}$ is the critical factor obtained from Reference 18 for the sample size of 54. The 95/95 tolerance limit (bias ± reliability factor) then becomes -1.2 ± 8.2%.

This tolerance limit will be applied to the SIMULATE-3 calculation of CEA worths at all power and moderator temperature conditions by,

Predicted CEA Worth = (Calculated CEA Worth) * [1 - Bias ± R. F.].

Stuck rod worth

The SIMULATE-3 capability to predict the worst stuck rod worth and the 'net' control rod worth were verified by SCE through simulating measurements performed during the initial startup of ANO-2. The net rod worth is defined as the negative reactivity worth from insertion of all control rods except for the most reactive single rod, which remains stuck out.

The comparison of the SIMULATE-3 calculated All-Rods-In (ARI), net rod worth, and the worst (most reactive) stuck rod worth with the measurements show good agreement. The observed differences for these cases are all within the 95/95 tolerance limits of -9.4% and +7.0%, established in the previous control rod worth comparisons. Therefore, it is concluded that the 95/95 tolerance limit for the rod worth is also applicable to the net worth.

Inverse boron worth

The SIMULATE-3 calculated inverse boron worths (IBW) were compared by SCE with a total of 16 measurements from 14 cycles of operation for the SONGS Units 1, 2, and 3. The IBW is calculated using:

IBW = -(CBC1 - CBC2) / (delta-Reactivity)

where CBC1 is the critical boron concentration for Statepoint 1 (SP-1), CBC2 is for Statepoint 2 (SP-2), and delta-Reactivity is the reactivity change (% delta-k/k) required to go from the SP-1 to SP-2 conditions.

This reactivity change is accomplished normally by control rod insertion or withdrawal.

The c-m differences at BOC, zero-power conditions are all within 10%. The mean and standard deviation are 2.5% and 5.6%, respectively, leading to a 95/95 reliability factor of 3.1% for the relative (percent) uncertainty in the calculation of the critical boron concentration. The 95/95 reliability factor for the control rod worth was previously found to be 8.2%. Statistically adding these values gives an IBW reliability factor of 9.3%. For conservatism the factor was rounded up to 10%, which bounds all the observed differences.

Assembly power distribution

The SIMULATE-3 calculated rhodium incore detector reaction rates and assembly power distributions were verified by SCE through comparison with direct incore signal measurements and with the inferred measured radial and axial power distributions from Cycles 1 through 4 of SONGS Units 2 and 3.

1) Incore detector signal comparisons:

   A total of 72 incore detector snapshots were taken at close to Hot-Full-Power (HFP) and All-Rods-Out (ARO) conditions from Cycles 1 through 4 of SONGS 2 and 3. 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 (F_{xy}, F_Q, and F_R) were calculated from multiplying the standard deviations by the k-value corresponding to the size of each sample, giving values of 4.80%, 4.17%, and 3.34%, respectively, for all power levels and rodded conditions.

2) Radial and axial power distributions:

   The measurements were taken at close to ARO and HFP conditions for SONGS-2 Cycles 1 through 4 and for SONGS-3 Cycle 3 with burnups near the beginning-of-cycle, middle-of-cycle, and end-of-cycle exposure points.

   The radial comparisons demonstrate that the SIMULATE-3 assembly powers agree very well with the CECOR-inferred measured powers. The RMS error for each case is within 0.020 (absolute difference). The SIMULATE-3 axial results also agree well with the CECOR measurements. The RMS values of differences, (c-m), are well below 0.05 (absolute difference). For those state-points with an RMS error greater than 0.02, the two power distributions agree very well except in the top and bottom 5% axial zones of the core where the CECOR powers are inferred using pre-calculated extrapolation distances. When these two regions are removed from the comparison, all RMS errors are below 0.02.

3) Axial offset:

A total of 12 axial power distribution measurements from the above five cycles of operation were used for the comparison of axial offset (AO), defined as:

$$AO = (P_T - P_B) / (P_T + P_B)$$

where P_T is the power in the top half and P_B is the power in the bottom half of the core.

The mean and the standard deviation for the differences, (c-m), were determined to be -0.003 and 0.005, respectively. The maximum difference is -0.011. The 95/95 reliability factor then becomes 0.014, for the sample size of 12.

4.0 SUMMARY AND CONCLUSIONS

Southern California Edison Company (SCE) has performed extensive benchmarking using the CASMO-3/SIMULATE-3 methodology. This effort consisted of detailed comparisons of the calculated key physics parameters with the measurements obtained both from 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 SCE to use the CASMO-3/SIMULATE-3 computer program package for application to San Onofre, Units 1, 2, and 3.

Based on the analyses and results presented in the topical report, the staff concludes that the CASMO-3/SIMULATE-3 methodology as validated by SCE can be applied to steady-state PWR reactor physics calculations for the SONGS units 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, core physics databooks, and reactor protection system and monitoring system setpoint updates.

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

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Date: August 10, 1992

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