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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO "NUCLEAR PHYSICS METHODOLOGY FOR RELOAD DESIGN" REPORT

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION

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1 INTRODUCTION

By letter dated September 3, 1991 (Ref. 1), the South Carolina Electric & Gas Company (SCE&G) submitted a report entitled, "Nuclear Physics Methodology for Reload Design," (Ref. 2) for NRC review. This report documents the capability of SCE&G to perform in-house core reload nuclear design analyses for the Virgil C. Summer Nuclear Station (VCSNS) using standard Westinghouse Electric Corporation (W) methodologies previously approved by the NRC.

SCE&G intends to use the currently approved W methodology and computer programs for pressurized water reactor (PWR) reload applications; specifically, for steady-state reload physics design. This reload physics design includes (1) fuel bundle and core reloading pattern analysis; (2) calculations of estimated critical predictions; (3) generation of physics and kinetics input for transient and safety analyses; and (4) generation of physics input for the plant reactivity computer, for calculation of trip setpoints, for use by the reactor protection and monitoring systems, and for inclusion in the Core Operating Limits Report (COLR) and the plant physics data books.

SCE&G states that they will continue to upgrade the nuclear reload design methodology to remain current with the latest approved calculational techniques being employed by W.

2 SUMMARY OF THE REPORT

The report references the current W computer programs and physics models used by SCE&G to analyze the VCSNS core and compares the model-predicted results with measurements obtained from benchmarking data covering VCSNS operating cycles 3, 4, and 5. The plant analyses were performed over a range of conditions from hot zero power (HZP) to hot full power (HFP) operation. The agreement between the measured and calculated values presented in the report is used to validate the SCE&G application of the computer programs for analysis of the VCSNS PWR unit.

2.1 Overview

Section 1 of the report provides introductory background information and gives an overview of the scope of the report.

2.2 Physics Methodology

Section 2 of the report briefly describes the approved W PWR methodology licensed and used by SCE&G, references the Appendix for a description of each of the individual computer codes, and briefly outlines the procedures used by SCE&G for the model applications.

2.3 Physics Model Applications

Section 3 of the report describes the application of the previously referenced W physics methodology in four major areas:

- core power distributions at steady-state conditions
- axial power distribution control limits
- core reactivity parameters
- core physics parameters for transient analysis input

The licensee also states here that they will continue to upgrade the methodology to remain current with the latest approved calculational techniques being employed by W.

2.4 Physics Model Verification

Section 4 of the report describes the three operating cycles of VCSNS which provided measured plant data from a range of plant startup and normal operation conditions. VCSNS is a three-loop (W) PWR plant with a 17x17 fuel rod array, 157 fuel assembly core, generating 2775 megawatts-thermal (Mwt) at rated power, which began commercial operation in 1984. There are 48 full-length rod cluster control assemblies. The in-core flux instrumentation consists of moveable fission chambers which can be inserted into multiple core locations. The neutron flux detector signals are processed off-line to infer the three-dimensional (3D) measured power distribution in the core.

The topical report lists the key PWR physics parameters for which comparisons of predicted to measured or inferred plant data were performed to provide verification of SCE&G's ability to apply the W methodology to VCSNS-specific reload designs. The measured data cover the range from zero power startup testing to normal full power operations at the SCE&G VCSNS unit. Three operating cycles from VCSNS were included, covering different reload core design strategies. The four parameters measured during zero power physics tests are:

- critical boron concentrations
- isothermal temperature coefficient
- control rod worths
- differential boron worth

For each of these four parameters, the observed differences were compared to a set of startup test review criteria, based on previously approved W uncertainty analyses, which represent the maximum expected deviation between prediction and measurement.

The parameters measured or inferred during at-power operation include:

- boron letdown curves,
- power peaking factors, F_Q and $F_{\Delta H}$,
- average assembly radial power distributions,
- core average axial power distributions, and
- axial offset.

3 TECHNICAL EVALUATION

3.1 Background

SCE&G has entered into a technology exchange agreement with the Commercial Nuclear Fuel Division of the Westinghouse Electric Corporation through which the relevant physics design methodology and associated computer programs have been obtained, beginning in May 1987. The licensee states that all methods employed and described in this topical report (including model development, computer programs, measured data processing, etc.) are standard W methods and reflect current practices. The licensee further states that Westinghouse has thoroughly reviewed the topical report and all supporting calculations and has concluded that SCE&G correctly applied the methodologies. PHOENIX-P is a PWR version of the W two-dimensional multi-group transport theory code (Ref. 3) described in a meeting between the W Fuel Division and the NRC staff on October 1984 (Ref. 4) that has been qualified and approved (Ref. 5) for use in calculating PWR lattice physics parameters and determining neutronics input for the two-group diffusion theory code ANC. PHOENIX-P uses a 42-energy group cross section library derived from the standard ENDF/B-V cross-section set.

ANC is an approved W three-dimensional two-group diffusion theory nodal model (Ref. 6) that was also qualified for use with PHOENIX-P by Reference 5. The code is based on coarse mesh nodal (4 nodes/assembly) diffusion theory using the non-linear nodal expansion method, with coupled thermal hydraulic and Doppler feedback. The code includes the following modeling capabilities: solution of the two group neutron diffusion equation, equivalence theory cross section homogenization, cross section depletion, explicit baffle/reflector modeling, and a rod power recovery model.

The two group model solves the neutron diffusion equation in three dimensions, with assembly homogenization. In order to preserve the flux and current continuity at nodal interfaces, ANC uses flux assembly discontinuity factors that are obtained from the PHOENIX-P 2D detailed lattice analysis. ANC also employs flux discontinuity correction factors to combine the global (nodal) flux shape and the assembly heterogeneous flux distribution for the rod power recovery model. The use of an explicit baffle/reflector cross section representation eliminates the need for user-supplied albedoes, normalization, or other adjustment at the core/reflector interface.

The fuel depletion model uses macroscopic and microscopic cross sections to account for fuel exposure without tracking the individual nuclide concentrations. ANC 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. ANC also calculates control rod worth and moderator, Doppler and xenon and samarium feedback effects.

APOLLO is a W one-dimensional (1D) axial two-group diffusion theory code (Ref. 7), currently under NRC review, which uses radially homogenized flux and volume weighted cross sections from the 3-D ANC model. The 1D APOLLO model is normalized to the 3D ANC model results by performing an elevation dependent radial buckling search at each burnup step (Ref. 8). APOLLO is an advanced version of the approved PANDA code (Ref. 9) and was also described in the October 1984 meeting of Reference 4 between W and the NRC.

FIGHTH is a W computer code, derived from previous LASER (Ref. 10) and REPAD (Ref. 11) models, which has been accepted (Ref. 12) for predicting steady-state fuel rod temperatures for low-enriched sintered UO_2 fuel rods. This program is currently used only for calculating fuel and cladding effective temperatures for input to the PHOENIX-P code as a function of burnup, linear heat generation rate, moderator temperature and flow rate.

The standard W INCORE computer program (Ref. 13) is used to process the neutron flux measurements made by the movable incore fission chambers to determine the core power distribution, as required by the VCSNS Technical Specifications. The measured flux values are combined with power-to-reaction rate ratios analytically generated with the PHOENIX-P/ANC models in order to infer a 'measured' three-dimensional power distribution. This standard W technique allows use of the previously established measurement uncertainties. Since all methods employed are stated to be standard licensed methods, the W calculational uncertainties (Ref. 14) for the nuclear hot channel factors are also used by SCE&G.

SCE&G has used the W methodology package described above to model the VCSNS operating Cycles 1 through 5 and has performed detailed comparisons to measured operating data for Cycles 3, 4, and 5. An evaluation of these comparisons is presented below for the key PWR physics parameters to be generated by the licensee.

3.2 Critical Boron Concentrations

Critical boron concentrations (CBC) were measured at HZP conditions by acid-base titration with all rods out (ARO) and with the reference bank B fully inserted. The SCE&G ANC 3D model predictions of CBC were compared to zero-power startup test measurements and full-power operating data for cycles 3, 4, and 5 of VCSNS operation. Differences between calculated and measured boron parts-per-million (ppm) data are stated in absolute terms, measured minus predicted (m-p).

The results from the HZP comparisons qualify the model for predicting the CBC and reactivity for beginning-of-cycle (BOC), xenon-free conditions. Six measurements from the three cycles of startup tests are included. All differences are within the ±50 ppm review criteria.

3.3 Isothermal Temperature Coefficient

The isothermal temperature coefficient (ITC) is defined as the change in reactivity due to an incremental change in the core average moderator and fuel temperature. ITCs were measured by making small changes in the reactor coolant system temperature and determining the corresponding change in reactivity with the plant reactivity computer. SCE&G used the 3D ANC model to calculate the ITC by uniformly varying the moderator temperature by ±5 degrees Fahrenheit (F) and by determining the Doppler temperature effect using the fitting coefficients from the FIGHTH calculations. The measured and predicted ITCs compare within the review criteria of ±3 pcm/°F from the three cycles of operation. Note that 1 pcm is equivalent to 1×10^{-5} percent delta-k/k.

3.4 Total Power Coefficient

The power coefficient is defined as the total change in reactivity due to an incremental change in the core average power level (% power), representing a combination of the moderator and fuel (Doppler) temperature effects. ANC is used to calculate the power coefficient by varying the core power level input value around the reference level and by changing the core inlet temperature in accordance with the VCSNS plant specific inlet program. Comparisons with direct measurements are not given.

3.5 Control Rod Worths

Control rod worth is the reactivity difference (pcm) between different control rod configurations. The worth of the reference control rod bank B (the bank of highest worth) was measured by boron dilution, using step-wise bank insertion and summing the differential worths obtained from the reactivity computer. The remaining rod banks (A, C, D, SA, and SB) were then individually fully inserted while holding boron constant and withdrawing bank B to maintain criticality. The integral worth of each inserted bank is inferred from the equivalent worth of the reference bank measured critical position, corrected for the presence of the inserted bank. This is consistent with the W Rod Swap Technique (Ref. 15) which was approved in 1982. The 3D ANC model was used for the prediction of the individual control rod bank worths and was compared by SCE&G with the BOC zero-power startup measurements for the three operating cycles. All relative differences [(m-p)/p] in percent were within the test review criteria of ±10% on the reference bank worth and ±15% (or 100 pcm) on the swapped bank worths. Note that the ANC model is also used to generate the analytical correction factors which account for the effect of the inserted bank on the partial integral worth of the reference bank.

3.6 Differential Boron Worths

Measured differential boron worths (pcm/ppm) were inferred by dividing the measured reference bank worth by the difference between the critical boron concentrations at ARO and at the step-wise reference bank positions. The 3D ANC model was used to calculate the worth of a ±25 ppm change around the HZP measured ARO critical boron concentration. The measured and predicted boron worths were compared by SCE&G with measurements from the three cycles of operation. All relative differences were within the test review criteria of ±15% [(m-p)/p].

3.7 Boron Letdown Curves

Critical boron concentrations from measured HFP, equilibrium xenon and samarium conditions were compared to the 3D ANC model predicted boron letdown curves for the three operating cycles. These at-power comparison results, corrected for control rod insertion and for deviations from the full-power, equilibrium xenon and samarium conditions, are used as estimates of the model uncertainty for all equilibrium power conditions with thermal feedback. There are a total of 51 measurements from three operating cycles, taken at the time of INCORE power distribution measurements. The mean difference between measured and predicted CBC for all three operating cycles is 12 ppm, with a standard deviation of 19 ppm. These differences are due to ANC calculational uncertainties, variations in B-10 isotopic concentrations, and measurement (titration) uncertainties. For conservatism, all differences are assumed due only to calculational uncertainties.

3.8 Power Peaking Factors

Measured values of the primary power peaking factors, the heat flux hot channel factor (F_Q) and the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) were inferred using the W INCORE program. The predicted power peaking factors were obtained from the 3D ANC model results at the closest depletion interval. For F_Q, the mean difference between the measured and predicted values for 51 measured statepoints over the three cycles was 2.07% [(m-p)/p] with a standard deviation of 1.47%. For $F_{\Delta H}$, the mean difference is 0.58% with a standard deviation of 1.11%. This is within the W uncertainty values from Reference 14.

3.9 Radial Power Distributions

The measured radial power distributions are inferred by the INCORE procedure, after the flux map measurements are performed using the moveable incore neutron flux detector system. The predicted power distributions are interpolated from the 3D ANC depletion step results at HFP, ARO operating conditions. The mean absolute difference between measured and predicted assembly relative powers is less than 0.014 with a standard deviation less than 0.017.

3.10 Axial Power Distributions and Axial Offset

A total of 18 axial power distribution measurements from the above flux maps over the three cycles of operation were plotted with the 3D ANC model-predicted values at similar depletion points. The measured axial offset (AO), defined as the percent difference between the relative power in the top half of the core and that in the bottom half of the core, is also inferred by INCORE and is compared with the predicted values from ANC at 51 flux map statepoints. The mean difference between measured and predicted values for the three cycles is 0.7% with a standard deviation of 0.9%.

4 SUMMARY AND CONCLUSIONS

South Carolina Electric & Gas Company has performed benchmarking for three cycles of operating data from the V. C. Summer Nuclear Station (VCSNS) using currently accepted W reload design methodologies. The benchmarking effort consisted of detailed comparisons of the calculated physics parameters with the measurements obtained from the VCSNS PWR. The results demonstrate that the VCSNS plant-specific agreement is within the W determined uncertainty analysis for the stated PWR physics parameters. This effort also demonstrates the capability of SCE&G to use the W computer program package for application to the VCSNS unit. SCE&G intends to use these methods for steady-state PWR core physics reload design applications, including fuel bundle and loading pattern analysis, safety analysis inputs and calculation of trip setpoint updates.

Based on the analyses and results presented in the topical report, the staff concludes that currently approved W methodologies, as validated by SCE&G, can be applied to steady-state PWR reactor physics calculations for the VCSNS reload design applications discussed in the above technical evaluation. The accuracy of this methodology has been demonstrated to be sufficient for use in design applications, including PWR reload physics analysis, generation of transient analysis inputs, startup predictions, core physics databooks, plant reactivity computer inputs and COLR trip setpoint updates.

Application of the approved package is to be limited to the range of fuel bundle and core reload design parameters verified in the topical report. Any change which introduces significantly different fuel designs may require further validation by the licensee.

5 REFERENCES

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