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ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO PHILADELPHIA ELECTRIC COMPANY TOPICAL REPORT PECO-FMS-0005 "METHODS FOR PERFORMING BWR STEADY-STATE REACTOR PHYSICS ANALYSES" PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 DOCKET NOS. 50-277 AND 50-278

1.0 INTRODUCTION

By letter dated February 1, 1988 (Ref. 1), Philadelphia Electric Company (PECo) submitted for review PECO-FMS-0005, "Methods for Performing BWR Steady-State Reactor Physics Analyses." The report describes and qualifies methods used by PECo for steady-state core physics analysis of PECo Boiling Water Reactors (BWR). The information in this report was supplemented by information submitted with Reference 13 in response to questions from the NRC staff and consultants. The review by the staff of this report and supplemental information was performed with the assistance of consultants from Brookhaven National Laboratory (BNL).

PECo intends to perform the reload and transient calculations required for the operation of its reactors and, in support of this effort, has developed its own analysis methodologies. This report is one in a series describing these methodologies. It (1) describes the three-dimensional BWR steady-state coupled neutronic/thermal-hydraulic modeling used in the PECo integrated sequence of reactor analysis computer programs, (2) presents results of the benchmarking of the methods against measured Peach Bottom, Units 2 and 3 (PB-2, PB-3) data, (3) discusses the applicability of the PECo steady-state methods to the calculation of important BWR core design and licensing parameters (Maximum Average Planar Linear Heat Generation Rate (MAPLHGR), Minimum Critical Power Ratio (MCPR), Doppler and void coefficients, scram reactivity, etc.), and (4) outlines the calculational procedures used in the analysis of the rod withdrawal error (RWE), mislocated bundle loading error (MBLE) and loss of feedwater heater (LFWH) events, and standby liquid control system (SLCS) shutdown margin.

The primary computer codes used by PECo for steady-state core analyses are based on the Electric Power Research Institute (EPRI) Advanced Recycle Methodology Program (ARMP). These have been benchmarked by EPRI, and others. Versions of the codes have been submitted by other utilities and approved by the NRC. Use of these methods is common in the industry.

The principal calculational tools in the PECo steady-state physics methodology are the CASMO-1-PECO (Ref. 2) and SIMULATE-E-PECO (Ref. 3) computer codes. CASMO-1-PECO is PECo's version of the CASMO-1 multigroup two-dimensional transport theory code for assembly burnup calculations. The PECo version includes ENDF/B-V updates to the delayed neutron data. The gadolinium burnup in the fuel rods is evaluated with MICBURN (Ref. 4) using an effective macroscopic cross section for gadolinium bearing fuel rods. SIMULATE-E-PECO is the PECo BWR version of the SIMULATE-E (Ref. 5) three-dimensional coupled neutronics/thermal-hydraulics steady-state nodal analysis code which calculates power distributions and reactivity effects in light water reactors. The thermal-hydraulics calculations are performed by FIBWR (Ref. 6) which employs a detailed pressure drop analysis to determine the two-phase thermal-hydraulics in both the in-channel and bypass regions. SIMULATE-E has been modified to incorporate an option which provides for cross section dependence on moderator temperature, in order to improve the accuracy of the zero power critical calculation. Furthermore, a set of subroutines have been incorporated into SIMULATE-E to allow direct evaluation of thermal margins such as critical power ratio (CPR), linear heat generation rate (LHGR), and MAPLHGR. An automated data link between CASMO-1 and SIMULATE-E is provided by NORGE-B (Ref. 7).

2.0 EVALUATION

The evaluation of this report has included the review of (1) the calculational models and procedures used in carrying out the BWR steady-state analyses with the CASMO-1 and SIMULATE-E codes, (2) the benchmarking of the codes, (3) the input models and assumptions used, and (4) the PECo responses to specific questions raised during this review (Ref. 13). The major results of this review are summarized in the following.

Qualification

The PECo steady-state physics methods have been qualified by comparison with measured data and higher order calculations. The CASMO-1 code has been benchmarked against KENO-IV Monte Carlo calculations, PDQ-7-E calculations and the Studsvik Kritz critical measurements. The Kritz critical benchmarking was performed by the code developer. The segment of the CASMO-1 code in which the calculation of isotopic concentrations is carried out was qualified by Studsvik using measured isotopic ratios from the Yankee Rowe Core-I and Saxton Core-II experiments (Refs. 8 and 9, respectively).

The SIMULATE-E code was benchmarked against measured data from PB-2 and PB-3 spanning five recent cycles. Hot critical eigenvalues, local (four bundle average) and core-average axial power distributions were calculated and compared with measured data from 47 statepoints over the five cycles.

Comparing CASMO-1 calculations of single bundle fission rate distributions to KENO-IV results as well as data from the Kritz experiments, PECo has determined an uncertainty of approximately 2 percent in the CASMO-1 pin power prediction. Based on a statistical analysis of the comparisons of SIMULATE-E to measurement, an overall mean hot critical eigenvalue of 0.9946 with a standard deviation of 0.003 delta-K was obtained. The PECo cold physics methods were qualified using 31 recent Peach Bottom insequence, xenon-free startup criticals. A mean SIMULATE-E cold critical eigenvalue of 0.9916 with a standard deviation of 0.0035 delta-K has been determined. Comparison of predicted and measured neutron flux readings at each axial traversing in-core probe (TIP) location and axially integrated readings at all TIP locations produced overall root mean square (RMS) differences of 6.9 percent and 4.1 percent, respectively.

Doppler coefficients calculated with CASMO-1 have been compared with the results inferred from the Swedish temperature coefficient measurements (Refs. 10, 11). The comparison shows that the CASMO-1 calculated Doppler coefficient is more negative than the measured data by an overall bias of approximately -6.6 percent. The CASMO-1 Doppler standard deviation is 10.2 percent. The overall Doppler bias resulting from the combination of CASMO-1 and NORGE-B reactivity biases is -6.9 percent for unrodded nodes and 4.8 percent for rodded nodes. Determination of the combined standard deviation of the individual CASMO-1 and NORGE-B Doppler reactivity uncertainties has resulted in an overall nodal uncertainty of 10.9 percent for unrodded nodes and 10.4 percent for rodded nodes. The benchmarking of void reactivity calculations has been accomplished by comparisons of CASMO-1 void reactivity data with the KENO-IV code. The mean void coefficients from the two codes differ by approximately 1.6 percent in magnitude. At a core average void fraction of 0.4, the void coefficient fit uncertainty for unrodded nodes is 2.34×10^{-5} delta-K/K/%V which represents a NORGE-B fit uncertainty of approximately 2.1 percent in the calculated void coefficient. For rodded nodes the NORGE-B void coefficient fit uncertainty is approximately 4.8×10^{-5} delta-K/K/%V. This represents approximately a 4.4 percent uncertainty in the calculated void coefficient.

This benchmarking has been reasonably complete, appropriate comparisons have been made and results fall within expected and satisfactory ranges. This PECo qualification is acceptable.

The comparisons of measured and calculated power distributions are used to determine SIMULATE-E power distribution uncertainties. Two rejection criteria are used by PECo in selecting the data for determining the uncertainties; (1) elimination of the TIP data taken in the top and bottom 18 inches of the 150-inch active fuel region and (2) rejection of the data from symmetrically located TIP pairs if the absolute difference in the integrated TIP reading is greater than 9 percent. Since the top and bottom 18 inches of the active fuel do not include the core axial peak power locations and since the contribution of these regions to the (axially integrated) thermal limits is negligible, this deletion is acceptable. PECo's criterion for rejecting symmetric TIP data was based on the assumption of the failure of the TIP instrumentation. However, since it cannot be readily demonstrated that the cause of the 9 percent difference in the integrated readings is due to detector system failure, this rejection was found unacceptable. In response to Question-15 (Ref. 13) PECo indicated that all TIP data, including the symmetric TIP's with differences of more than 9 percent, will be included in the uncertainty analysis (Ref. 13). This is acceptable.

When all measured TIP signals are included in the calculation/measurement statistical evaluation, including previously rejected strings, and the top and bottom 18 inches of the core are excluded, the pointwise and integral RMS values are found to be 8.9 percent and 4.8 percent, respectively. PECo has evaluated the propagation of these power uncertainties into the thermal limits uncertainties and the results are as follows: The MCPR uncertainty is 5.9 percent, the MAPLHGR uncertainty is 8.9 percent and the Peak Pin Linear Heat Generation Rate (PPLHGR) uncertainty is 9.6 percent. These are reasonable values and are acceptable.

Application

The PECo physics methods have been applied to a series of core design and licensing calculations. MAPLHGR, PPLHGR, MCPR, Doppler and void reactivities, control rod scram reactivities, cold shutdown margin and other safety parameters are calculated with the PECo SIMULATE-E model. The SIMULATE-E code has been modified to include a MAPLHGR algorithm which is similar to that found in the Peach Bottom plant process computer software. MAPLHGR values calculated with SIMULATE-E using a process computer power distribution were found to be in very good agreement with the process computer MAPLHGR edits. To demonstrate adequate margin to the technical specification LHGR limit for steady state operation, PECo introduced a PPLHGR correlation in SIMULATE-E. In this correlation, PPLHGR is proportional to the product of the average fuel pin segment power and the local pin peaking factor. PECo has installed the General Electric critical power correlation, GEXL (Ref. 12), into SIMULATE-E for the evaluation of the CPR. Such CPR evaluations are applicable only to GE fuel unless justification for other fuels is submitted and approved.

Doppler reactivity is evaluated as a function of fuel temperature with the SIMULATE-E code at various exposure points in the cycle. At each cycle exposure point a SIMULATE-E converged power distribution is established. Core reactivity calculations are then performed at two neighboring statepoints in which the core average fuel temperature is perturbed by fuel delta-T and all other plant conditions are held constant, assuming no moderator feedback effects. The resulting converged eigenvalue at each average fuel temperature is used to determine the Doppler reactivity. The Doppler reactivity obtained with SIMULATE-E as a function of average fuel temperature is used in the transient analyses.

The void reactivity is determined as a function of moderator density in a manner similar to that used for the Doppler temperature evaluation discussed above.

The review of these applications have indicated that appropriate models and methods have been used, and the use of these methodologies is acceptable.

Appendices A, B, C, and D of the report summarize the calculational procedures used by PECo to evaluate the mislocated bundle loading error, rod withdrawal error, loss of feedwater heating events, and standby liquid control system shutdown margin, respectively.

The analysis of the mislocated bundle loading error event is performed with SIMULATE-E and must demonstrate that the MCPR safety limit is not violated as a result of bundle mislocation. The PECo analysis procedure assumes that a high reactivity assembly, representing the mislocated bundle, is placed at the location of the least limiting bundle in the least limiting (four-bundle) local power range monitor (LPRM) cell, using CPR as a guide, and the core is depleted to the cycle exposure at which the CPR is evaluated.

The CPR of the mislocated assembly is subtracted from the value of the correctly loaded assembly at the symmetric monitored location (to determine the maximum delta-CPR) and the difference is added to the CPR safety limit. Bundles having a CPR greater than this increased limit cannot violate the core safety limit as a result of a bundle mislocation and are excluded from the search. This step is repeated for the remaining bundles until all bundles in the core loading are shown to have a CPR greater than the safety limit in the event of a bundle mislocation. The entire procedure is repeated at exposure increments of 1,000 MWD/STU to account for depletion and burnable poison effects in the evaluation of the MBLE event.

PECo evaluated the MBLE event using SIMULATE-E results corresponding to various Peach Bottom-3 Cycle-7 exposures at rated operating conditions (power, flow, and xenon). Local conditions (bundle power, flow and xenon) were allowed to vary and the resulting delta-CPR were determined.

In order to allow generic analyses of the MBLE event, PECo has used this PB-3 data to correlate the bundle delta-CPR to the bundle initial CPR (ICPR). The determination of the core delta-CPR using this correlation assumes, however, that the limiting bundle is the mislocated bundle. In order to ensure that neighboring bundles will not exceed the MCPR limit, in response to Question-19 (Ref. 13), an additional analysis was carried out by PECo. In this analysis all assemblies within one assembly pitch of the mislocated bundle were included. A second MBLE generic evaluation (similar to that presented in Appendix B) was performed based on the adjacent assembly ICPR and delta-CPR data. This adjacent-assembly delta-CPR including a 2-sigma uncertainty and an additional 0.05 added for conservatism was found to be 0.10, a value bounded by the delta-CPR of 0.13 reported in the PECo topical. Based on this expanded MBLE analysis it is concluded that the PECo method for calculating the MBLE event is acceptable.

The rod withdrawal error event is a localized operational transient resulting in the insertion of positive reactivity, a core power increase and a spatial redistribution of power resulting in a loss of operating margin. PECo analyzes this event by selecting a high worth error rod at the maximum reactivity point in the cycle. With the core at rated xenon-free conditions the control rod pattern is adjusted to place at least one fuel bundle, located close to the (fully inserted) error rod, at or near the MCPR and LHGR operating limits. A series of calculations are performed in which the error rod is withdrawn and the MCPR, LHGR and Rod Block Monitor (RBM) responses are determined as a function of fuel type and error rod notch position.

The RBM response is based on allowable LPRM string failure combinations. In order to ensure that thermal limits will not be violated when the error rod is located in regions with less than four LPRM strings, PECo has indicated in response to Question-17 (Ref. 13) that (a) for rods using two LPRM strings, one string will be failed and (b) for rods using three LPRM strings, two strings will be failed. We find this revised PECo method for evaluating the RWE event acceptable.

Loss of feedwater heating results in a transient reduction in the feedwater temperature, an overall core power increase and a redistribution of power in the core. The transient is terminated by the increase in Doppler and void feedback reactivities or by a scram during normal plant operation.

In the analysis of the LFWH event there is some concern that system variables such as pressure, feedwater flow, steam flow, etc., at intermediate points during the transient might be more limiting than the values assumed for the initial and final statepoints. Such a case could give rise to a limiting delta-CPR during the transient which is not bounded by the final-state delta-CPR. In response to this concern PECo has indicated that changes in the feedwater flow and steam flow are accounted for in the SIMULATE-E analysis of the LFWH transient and that changes in the system pressure are expected to have a very small effect and are ignored in the LFWH evaluation. Furthermore, PECo has reported that changes in these system variables are slow and proceed in a quasi-static manner. The SIMULATE-E model can, therefore, be used to determine the delta-CPR which occurs between the initial and final equilibrium states. We therefore conclude that the PECo procedure for the analysis of the LFWH event is acceptable.

3.0 CONCLUSIONS

The staff, with assistance of consultants from BNL, has reviewed the Topical Report PECO-FMS-0005 which presents the description and qualifications of the PECo steady-state core physics methods, comprised of an integrated sequence of computer programs, and the application of these programs for design, licensing and operations calculations. The review has included the material provided both in the topical report (Ref. 1) and in response to questions (Ref. 13). The program components are principally industry standard methodologies, which generally have been reviewed in previous utility submittals. This report presents qualification of the methodology and its use by PECo by comparisons of calculations with relevant reactor data, including previous cycles from Peach Bottom, Units 2 and 3. The review has concluded that the qualification process has covered an acceptable range of comparisons and these comparisons demonstrate that the methodology is capable of satisfactory analysis of relevant reactor configuration and steady-state operating conditions. Furthermore, the review concludes that PECo is capable of using the methodology acceptably for providing acceptable safety related core parameters and relevant event analyses to support reload and other licensing actions. It is acceptable for such use by PECo. CPR evaluations are limited to GE fuel.

4.0 REFERENCES

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