ATTACHMENT

TECHNICAL EVALUATION REPORT

TECHNICAL EVALUATION OF TOPICAL REPORT EA-CA-91-0001-M
(STEADY-STATE CORE PHYSICS METHODS
FOR BWR DESIGN AND ANALYSIS)
FOR GULF STATES UTILITY COMPANY

J.F. Carew

September 18, 1992

Prepared for the
Office of Nuclear Regulatory Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NRC FIN A-3868, Task# 30

Reactor Analysis Group
Advanced Technologies Division
Brookhaven National Laboratory
Upton, Long Island, New York 11973

1.0 INTRODUCTION

By letter dated January 30, 1991 (Reference-1), Gulf States Utilities (GSU) has submitted the topical report EA-CA-91-0001-M, entitled "Steady State Core Physics Methods for BWR Design and Analysis", for NRC review and approval. The topical report describes the methods and models proposed for Gulf States Utilities application to the reload analyses of the River Bend Station (RBS). The report includes a description of the computer codes and models, and the qualification of these methods via comparison to steady-state operating reactor benchmarks and River Bend plant data. The GSU methods for performing the licensing analyses of the loss of feedwater heating, rod withdrawal and fuel misloading events are also provided together with the basis for the RBS shutdown margin and standby liquid control system technical specifications.

The purpose of this review is to evaluate the GSU methods for performing RBS licensing analyses. This includes both an evaluation of the codes and benchmarking as well as the conservatism of the modeling and analysis employed in the evaluation of the licensing events. The topical report and reload methods are summarized in Section-2 and the technical evaluation is presented in Section-3. The technical position is given in Section-4.

2.0 SUMMARY OF THE TOPICAL REPORT

2.1 Core Analysis Methods

The GSU core analysis methodology employs CASMO (Reference-2) to perform the

detailed lattice physics calculations for the RBS. CASMO is used to determine the homogenized nodal cross-sections, pin-wise power distributions and TIP instrument response factors. CASMO solves for the energy-dependent neutron flux distribution in the two-dimensional fuel assembly geometry using a transport theory response matrix/transmission probability method. The nuclear properties are initially determined over a fine spatial mesh using slowing down theory. These fine-mesh cross-sections are then collapsed from the standard 25-groups down to a few-group representation, which is used to calculate the node-wise cross-sections for SIMULATE-E (Reference-3). Delayed neutron fractions and inverse velocities are also calculated for input to transient analyses.

The gadolinia (Gd_2O_3) burnable absorber rods are calculated with MICBURN. MICBURN performs a one-dimensional multi-region depletion calculation of the gadolinia rod and determines effective fine-group cross-sections for input to CASMO. The spatial model consists of six macro-regions within the fuel rod, each of which is subdivided into twenty subregions for the gadolinia depletion.

In order to provide the nuclear data required for the core statepoint analyses, CASMO fuel assembly calculations are performed over the expected

range of fuel burnup at 0.0, 0.40 and 0.70 void fraction, and for selected fuel temperatures and control states.

The core performance analyses are performed with the SIMULATE-E three-dimensional coupled neutronics/thermal-hydraulics code. The GSU methodology uses the modified coarse mesh diffusion theory option together with thermal leakage correction factors and flux discontinuity factors. The GSU version of SIMULATE-E was modified to include (1) the GEXL-PLUS critical power correlation (Reference-4) and (2) a mechanistic incore TIP response.

The core flow and void distributions are calculated with the FIBWR (Reference-5) BWR steady-state thermal-hydraulics code. FIBWR solves the mass, momentum and energy conservation equations within each fuel assembly, using a known power distribution and a total core flow or pressure drop boundary condition, to determine the assembly-wise mass flow rates and the bypass flow. The power in the peripheral assemblies is calculated using neutron-current albedo boundary conditions and thermal-leakage correction factors. These factors are determined using the ABLE program (Reference-6) and are normalized to operating reactor measurements.

The SIMULATE-E core depletion is performed on a node-wise basis and selected isotopic inventories are tracked. Sequential cycles are depleted by reshuffling the fuel assemblies and corresponding nodal exposure arrays at the end of each cycle.

The core kinetics data required for the RBS transient analyses is obtained using SIMTRAN-E (Reference-7). SIMTRAN-E performs a two-dimensional spatial integration of the local CASMO data to determine the kinetics input for the one-dimensional RETRAN-02 model.

2.2 Methods Qualification

As qualification for the steady-state physics methods and models, GSU has performed benchmark comparisons to measured TIP readings at several plants and to the assembly-wise and rod-wise gamma scan measurements performed at Quad Cities-1 at the end of Cycle-2 (Reference-8). The comparison of the QC-1 SIMULATE-E TIP readings for the Cycle 1 and Cycle 2 statepoints indicated agreement to within 8.6%[1] for the nodal readings and 7.0% for the TIP integrals. Comparisons to the Quad Cities-1 gamma scan bundle powers indicated a SIMULATE-E accuracy of 2.9%, and the axial and nodal power comparisons indicated an accuracy of 4.3% and 2.8%, respectively. SIMULATE-E benchmarking comparisons were also made to the Peach Bottom-2 (PB-2) Cycle-1 and Cycle-2 test data (References-9 and 10). Comparison of the SIMULATE-E predictions to the PB-2 measured TIP integrals indicated agreement to within 4.9%.

In addition to this generic benchmarking, GSU has performed plant-specific qualification via comparison to River Bend measurement data for Cycles 1-3. Comparisons of the SIMULATE-E predictions to measured RBS nodal TIP readings

---

[1]All % differences are one standard deviation unless indicated otherwise.

and TIP integrals indicated an accuracy of 7.1% and 3.0%, respectively. Comparisons were also made for the hot and cold critical eigenvalues. SIMULATE-E reproduced the hot eigenvalue for 208 statepoints, including a range of operating conditions, to within 0.2%, and the cold eigenvalue for 27 statepoints to within 0.3%.

2.3 Licensing Applications

2.3.1 Loss of Feedwater Heating Event

The loss of feedwater heating (LFWH) transient is considered to be quasi-static and is analyzed with SIMULATE-E. The initial core power is taken to be 102% of rated thermal power, and the core flow and pressure are at rated conditions. In the RBS Cycle-3 analysis presented, a 100°F reduction in feedwater temperature is assumed and the SIMULATE-E calculated ΔCPR is 0.08.

2.3.2 Control Rod Withdrawal Error Analysis

SIMULATE-E is used to calculate the quasi-static increase in core power level and shift in the power distribution toward the error rod location, for the control rod withdrawal error (CRWE) event. The calculation is performed at the core-average exposure resulting in the cycle peak hot excess reactivity, with the MCPR bundle typically located in a diagonally adjacent (or closer) control-cell, and the highest worth rod being withdrawn in a four-rod gang.

The event is terminated by the control system or a flow-biased scram. The SIMULATE-E calculated MCPR for the RBS Cycle-3 CRWE event analysis presented is 1.16.

2.3.3 Fuel Loading Error Analysis

The mislocation or misorientation of a fuel bundle can result in increased local powers and a potential approach to thermal limits. The event ΔCPR depends on the specific mislocated/misoriented bundle, the core fuel loading and the cycle burnup. GE has analyzed this event for sixteen BWR reload designs, and for annual, eighteen-month and control-cell equilibrium cycles (References-11 and 12). GSU bases their evaluation of the fuel loading error event on the GE generic analysis and does not perform a cycle-specific analysis for this event.

2.3.4 RETRAN-02 Kinetics Input

The steady-state methodology provides the physics input to the GSU RETRAN-02 systems analysis model. SIMULATE-E is used to deplete the core to the statepoint required for initialization of the RETRAN-02 model. Both base-case and perturbed SIMULATE-E calculations are performed to determine the cross section dependence on the local void fraction, fuel temperature, and control rod insertion. SIMTRAN-E is used to integrate the cross section data, weighted with the product of the SIMULATE-E direct and adjoint flux solutions, over the horizontal plane and determine the one-dimensional (axial) cross

section data for RETRAN-02.

3.0 SUMMARY OF TECHNICAL EVALUATION

The steady-state physics methodology topical report describes the GSU methods for performing RBS steady-state core design calculations and selected licensing analyses, and the associated methods benchmarking. This review focused on the completeness and quality of the benchmarking, and the conservatism of the event-specific methods and assumptions employed in the licensing analyses. The initial review of the report raised several questions and concerns which were transmitted to GSU in Reference-13. This evaluation included the material presented in the topical report as well as the GSU responses provided in References-14 and 15. The evaluation of the major issues raised during this review is summarized in the following.

3.1 Methods Qualification

The Quad Cities-1 end-of-cycle-2 (EOC-2) gamma scan measurements provide an excellent benchmark for the qualification of BWR neutronics methods. The La^{140} 1.6 MeV gamma scan measurements allow an accurate determination of both the recent pin-wise and assembly-wise power distributions. GSU has made comparisons of the QC-1 EOC-2 gamma scan measurements and the power distributions predicted by the SIMULATE-E model. In Response-4 of Supplement-2 (Reference-15), GSU has identified an improvement in the calculation of the benchmarks which results in a reduction in the pin power calculation-to-

measurement standard deviation from 3.6% to 3.1%. This reduced standard deviation is consistent with the agreement obtained using similar methodologies.

In addition to the QC-1 power distribution benchmarking, as part of the qualification of the SIMULATE-E cold shutdown margin calculational methods GSU has calculated the QC-1 cold critical measurements. In Response-15 of Supplement-1 (Reference-14), GSU has provided an additional model improvement for use in the SIMULATE-E cold critical calculations. With this revised model, the SIMULATE-E cold critical eigenvalue predictions are 0.998 ± .003 for the in-sequence criticals and 0.996 ± .002 for all (local and in-sequence) criticals. This SIMULATE-E model improvement should be applied, as appropriate, in future GSU cold critical calculations.

As plant-specific benchmarking of the SIMULATE-E cold shutdown margin calculations, GSU calculated the cold critical eigenvalues for a series of River Bend Cycle 1-3 statepoints. The SIMULATE-E predicted cold eigenvalues indicated a positive bias of ~ 0.5% for Cycle-2 relative to Cycles 1 and 3. GSU has indicated in Response-4 (Reference-14) that when only the appropriate cold critical statepoints (having a moderator temperature below 200°F and a negligible xenon concentration) are included, the Cycle-2 cold critical eigenvalue bias is well within the calculation-to-measurement standard deviation.

The benchmarking of the GSU core analysis methods is based on comparisons to

operating data from the Quad Cities-1, Peach Bottom-2 and River Bend plants. The comparisons presented demonstrate the accuracy and reliability of the GSU methodology in performing core performance analyses for the fuel designs and core loadings included in this measurement data base. If future RBS reload fuel designs and/or loadings involve core designs that differ from those included in this benchmark data base, a requalification of the GSU methodology will be required.

3.2 Licensing Applications

3.2.1 Control Rod Withdrawal Error Analysis

The CRWE event is relatively slow and is treated as a sequence of steady-state SIMULATE-E calculations. The highest worth rod is withdrawn in a four rod gang to the rod withdrawal limiter (RWL) setpoint. The calculated ΔCPR is used to verify the applicability of the technical specification $MCPR_p$ curves. There are several bounding or conservative assumptions included in the GSU analysis. The rod pattern controller and rod worth limiter are defeated so that a control rod pattern with a high worth control rod may be implemented. In order to bound all cycle statepoints, the analysis is performed at the point of maximum hot excess reactivity. The reactivity worth of each control rod is determined and the strongest rod is chosen for the withdrawal. As an additional bounding assumption, the limiting bundle is (typically) assumed to be within a diagonally adjacent control cell of the error rod. This methodology is based on the approved fuel vendor methodology, with the primary

difference being the computer codes used to perform the calculations (Response-9, Reference-14).

Comparison of the fuel vendor and GSU OLMCPR calculated for the CRWE event indicated a GSU (nonconservative) underprediction of 0.02 (Supplement-1, Reference-14). In Supplement-2 (Reference-15) GSU has indicated that this underprediction is due to the use of the coarse mesh diffusion theory (CMDT) rather than the modified coarse mesh diffusion theory (MCMDT) option in SIMULATE-E. When the MCMDT SIMULATE-E option is used, the GSU and fuel vendor OLMCPRs are in exact agreement. It is concluded that in order to provide an appropriately conservative CRWE analysis, the SIMULATE-E MCMDT option should be used in determining the OLMCPR for the control rod withdrawal error event.

3.2.2 Fuel Loading Error Analysis

GSU intends to use the fuel vendor analysis of the fuel loading error (FLE) event in RBS reload licensing evaluations. GE has performed a generic analysis of the FLE event for a range of reload core designs which includes typical RBS fuel loadings. If a future RBS reload core includes fuel designs or is loaded in a manner which is outside the approved generic analysis a reevaluation of the FLE event will be required.

3.2.3 Loss of Feedwater Heating Analysis

The loss of feedwater heating event results in a substantial increase in core inlet subcooling, an increase in core power, and a reduction in the CPR of the

limiting bundles. GSU calculates the resulting change in core power distribution and the reduction in MCPR using the SIMULATE-E model. The application of the steady-state model is consistent with presently approved licensee methods for analyzing the LFWH transient. The initial core inlet subcooling is determined by a heat balance, using the assumed initial conditions together with plant measurement data. The core is assumed to be initially at rated pressure and flow.

The GSU analysis includes several bounding or conservative assumptions. These include an initial 100°F reduction in feedwater temperature, an initial power level of 102% of rated, and the failure of the simulated thermal power trip which would normally terminate the transient before the limiting power is reached. In addition, the event is analyzed at the exposure statepoint that results in the closest approach to thermal limits (Reference-14). This methodology is based on the approved fuel vendor LFWH analysis, with the primary difference being the selection of the codes used to perform the calculations.

3.2.4 Standby Liquid Control System

The GSU methodology includes an analysis of the shutdown margin capability provided by the standby liquid control system (SLCS). The calculations are performed with the SIMULATE-E cold model which has been benchmarked against both local and in-sequence cold critical measurements at several plants including River Bend. In order to determine the minimum shutdown margin, the

calculations are performed at 68°F, xenon-free and with all control rods withdrawn. The SIMULATE-E model includes a conservative estimate of the boron reactivity worth (Response-5, Reference-14). The SLCS shutdown margin calculation assumes the technical specification minimum boron concentration of 660 ppm which is substantially smaller than the design value of 825 ppm. In addition, the SLCS analysis is performed at the statepoint of maximum cold excess reactivity to insure a determination of the minimum shutdown margin.

4.0 TECHNICAL POSITION

The GSU steady-state physics methodology topical report EA-CA-91-0001-M and supporting information provided in References-13 and 14 have been reviewed in detail. The topical report documents the GSU method and related benchmarking, for performing steady-state core performance calculations and selected licensing event analyses. Based on this review it is concluded that the proposed methods are acceptable for performing reload licensing analyses for the River Bend Station, under the conditions stated in Section-3 of this evaluation and summarized in the following.

1. Methodology Benchmarking

A requalification of the GSU steady-state methodology will be required for River Bend reload core designs that differ from those included in the EA-CA-91-0001-M benchmark data base (Section-3.1).

2. Analysis of the Control Rod Withdrawal Error Event

In order to provide a conservative analysis of the control rod withdrawal error event, the SIMULATE-E modified coarse mesh diffusion theory should be used to determine the OLMCPR for the CRWE event (Section-3.2).

3. Analysis of the Fuel Loading Error Event

A reevaluation of the fuel loading error event will be required for River Bend core designs that are outside the range of the approved generic analysis (Section-3.2).

REFERENCES

1. "River Bend Station - Unit 1, Docket No. 50-48," Letter RBG-34407, W.H. ODell (GSU) to USNRC, dated January 30, 1991.

2. A. Ahlin, M. Edenius, and H. Haggblom, "CASMO, A Fuel Assembly Burnup Program; User's Manual," AE-RF-76-4158 (Rev ED), Studsvik Energiteknik AB (1978).

3. D.M. Ver Planck, W.R. Cobb, R.S. Borland, and P.L. Versteegen, "SIMULATE-E: A Nodal Core Analysis Program for Light Water Reactors; Computer Code User's Manual," EPRI NP-2792-CCM, Electric Power Research Institute (1983).

4. "GEXL-Plus Correlation Application to BWR/2-6 Reactors, GE6 Through GE8 Fuel", NEDC-31598P, General Electric Company (1988); proprietary.

5. B.J. Gitnick, R.R. Gay, R.S. Borland, and A.F. Ansari, "FIBWR: A Steady-State Core Flow Distribution Code for Boiling Water Reactors; Computer Code User's Manual," EPRI NP-1924-CCM, Electric Power Research Institute (1981).

6. B.L. Darnell, B. Morris, M.L. Zerkle, "ABLE - An Albedo and Boundary Leakage Evaluation Program for Light Water Reactors-Computer Code User's Manual," Science Applications Inc. (1984).

7. J.A. McClure and G.C. Gose, "SIMTRAN-E: A SIMULATE-E to RETRAN-02 Datalink," EPRI NP-5509-CCM, Electric Power Research Institute (1987).

8. N.H. Larsen, G.R. Parkos, O. Raza, "Core Design and Operating Data for Cycles 1 and 2 of Quad Cities 1," EPRI NP-240, Electric Power Research Institute (1976).

9. N.H. Larsen, "Core Design and Operating Data for Cycles 1 and 2 of Peach Bottom 2," EPRI NP-563, Electric Power Research Institute (1978).

10. L.A. Carmichael and R.O. Niemi, "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2," EPRI NP-564, Electric Power Research Institute (1978).

11. Letter, R.A. Engel (GE) to T.A. Ippolito (NRC), "Change in General Electric Methods for Analysis of Misloaded Bundle Accident," November 4, 1980.

12. Letter, R.E. Engel (GE) to D.B. Vassallo (NRC), "Change in General Electric Methods for Analysis of Misloaded Bundle Accident," March 23, 1982.

13. "Request for Additional Information Concerning Topical Report EA-CA-91-0001-M, Rev.-0, 'Steady-State Core Physics Methods for BWR Design and Analysis,'" Letter, D.V. Pickett (USNRC) to James C. Deddens (GSU), dated December 9, 1991.

14. "River Bend Station-Unit 1, Docket No. 50-48," Letter RBG-36222, W.H. Odell (GSU) to USNRC, dated January 9, 1992.

15. "River Bend Station-Unit 1, Docket No. 50-48," Letter RBG-36667, W.H. Odell (GSU) to USNRC, dated March 26, 1992.