GULF STATES UTILITIES COMPANY

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Gentlemen:

River Bend Station - Unit 1
Docket No. 50-458

In a letter dated May 2, 1991, (RBG-34939) Gulf States Utilities (GSU) submitted to the NRC for review and approval topical report EA-PT-91-003-M entitled "River Bend Station Plant Transient Analysis Methodology" and requested that the NRC issue a safety evaluation report (SER) for the topical by January 1992. On October 31, 1991, GSU submitted a supplement to the topical. The supplement submittal was in two parts, one proprietary (RBG-35876) and the other not (RBG-35877).

Concurrent with the NRC submittal, GSU requested from a contractor, Mr. Samuel L. Forkner, an independent review of the proprietary portion of the supplement. Attachment 1 to this letter is the executive summary of that independent review. Attachment 2 is a copy of Mr. Forkner's resume. This is provided to aid the NRC in the completion of the NRC review of this topical. If you have any questions or desire further information on this independent review, please contact Mr. L.L. Dietrich at (504) 381-4866.

Sincerely,

W.H. Odell

W.H. Odell
Manager - Oversight
River Bend Station Nuclear Group

LAE/JSM/LLD/kvm

Attachment

A001

ATTACHMENT 1

Executive Summary of the Review of
Topical Report EA-PT-91-0003-SP
River Bend Station
Plant Transient Analysis Methodology
Supplement 1
Delta CPR Methodology and Additional Benchmarks

by

Samuel L. Forkner

April 13, 1992

Executive Summary

A review was performed of the Gulf States Utilities (GSU) Company Topical Report EA-PT-91-0003-SP which describes the ΔCPR methodology to be applied for Plant Transient Analyses for the River Bend Station (RBS). The review concentrated upon the technical correctness and completeness of the report. The conclusions reached in the review are summarized below. The original review was performed on a draft version of the topical report and some comments made in the original review are not applicable since changes were made to the final report. This summary reflects the comments considered to be applicable to the final version of the report.

The key premise of the approach taken in the topical report to account for uncertainties can be summarized as:

> The fast scram reactivity insertion available for BWR/6s not only reduces the severity of the Anticipated Operational Occurrences (AOOs) to be analyzed with the methodology but also removes much of the sensitivity of the results to modeling. This allows the uncertainty of the most sensitive analysis input (scram speed) to be conservatively bounded and the evaluation of the statistical combination of uncertainties does not have to be as complex as that required for plants with slower scram reactivity insertion, while still avoiding excessively restrictive operating limits.

Review of the base cases and sensitivity studies in the topical report confirm the validity of the premise and the basic approach used in the uncertainty analysis in the topical to be reasonable and conservative.

The GSU topical methodology applies the sensitivity study standard deviation in RCPR directly to the best estimate value of RCPR to get the 95/95 RCPR value. The same standard deviation is, in effect, to be applied to future cycles since the Statistical Adjustment Factors (SAFs) will be generically used. There is some possibility that this would not be a conservative approach if future analyses were for conditions resulting in much larger RCPR values than the base case of the sensitivity study. However, future cycles for RBS are not expected to result in significantly higher RCPR values and the use of zero for negative SAFs introduces a degree of conservatism adequate to account for substantial increases in base RCPR.

An alternate assumption in applying the uncertainty results is to assume the uncertainty as a fraction of RCPR is generically applicable. Thus the 2σ RBS "Model Uncertainty" for the Load Rejection No Bypass (LRNB) e nt is $\frac{.020}{0.053} \approx 38\%$ of the best estimate value. Some previously approved methodologies have used a 2σ model uncertainty near 25% of the best estimate value. Considering the level of detail in the RBS model a 2σ uncertainty of 38% seems too large. While shortcomings in the SIMTRAN-E method of producing one dimensional cross sections may be contributing to the higher uncertainty, the major cause is probably the relatively small size of the RCPR for the base event.

The topical report obtains the 95/95 RCPR value as:

$$RCPR_{95/95} = RCPR_{mean} + 1.645 \times \sigma_{RCPR}$$

where:

$$\sigma_{RCPR} = \frac{1}{2}\sqrt{\sum_i \Delta RCPR_i}$$

with the summation over the components used in the sensitivity study which used input perturbations estimated to be 2 standard deviations at 95% confidence on each parameter. Thus σ_{RCPR} is the 95% confidence upper limit on the RCPR standard deviation. Making 95/95 variations in input parameters does not guarantee a 95/95 range of output (unless the output is a linear function of the input) and a normal distribution of RCPR is assumed by using $1.645 \times \sigma_{RCPR}$ to obtain the 95% probability upper bound. However this is not of practical concern in this case since the overall model uncertainty is composed of many components without a single dominant term. The peak excess reactivity component appears to dominate the LRNB uncertainty but this uncertainty is actually composed of a large number of independent components (cross section uncertainties, void model uncertainties, 3D to 1D collapsing uncertainties, etc.) that were not individually resolved in the analysis. Therefore, the central limit theorem of statistics would indicate that the output would tend to be normally distributed. In many earlier methodologies the scram speed uncertainty was statistically treated, since scram speed is a dominant model uncertainty the assumption of a normal distribution on RCPR was not as easily justified and typically a response surface method was employed to evaluate the $RCPR_{95/95}$ value. Use of a bounding scram speed in the GSU methodology allows the simpler treatment of uncertainty.

The 25% uncertainty assumed for peak excess reactivity is larger than typically used (other approved methodologies have been ≈13-18%) and should conservatively bound the actual uncertainty in the GSU methodology. Overall, the range on parameter uncertainties used in the RBS sensitivity study are reasonable. While it could be difficult to rigorously defend the value of any particular parameter, it is unlikely that any single parameter's uncertainty is underestimated by enough to substantially impact the margin of safety and the values are in general consistent with values used in previously approved methodologies. The parameter variations as a set provide a reasonable estimate of the uncertainty in the model and appear to ensure no key sensitivity has been overlooked.

The selection of the limiting events and initial conditions to be analyzed is reasonable and consistent with past practice. However, the Feedwater Controller Failure (FWCF) event is sometimes most limiting at slightly less than rated power, GSU should confirm that the event is most limiting from rated power for RBS or that the power dependent MCPR curve adequately bounds the behavior.

The comparison of GSU methodology results to the results of vendor calculations shows good agreement. The description of the comparisons in the report shows a sound understanding of the phenomena involved and a willingness to investigate and understand the reasons for differences.

Comparisons of the GSU methodology to measured RBS event data and the Peach Bottom turbine trip tests show good agreement and yield confidence in the accuracy of the methodology. Additional verification of the methods used to provide data to the RETRAN model could be obtained by comparing steady-state results from RETRAN to the design codes which are used to produce input to RETRAN.

Overall, the GSU methodology is a straightforward application of standard techniques similar to those that have been used in previously approved methodologies. The GSU methodology is, in general, simpler due to the use of a bounding scram speed and the reduced sensitivity to model parameters due to the fast BWR/6 scram. A large reactivity uncertainty was applied to avoiding the analysis burden required to justify a smaller uncertainty.

ATTACHMENT 2

Resume'

Samuel L. Forkner

Education:

B.S. Nuclear Engineering, University of Tennessee, 1969.
M.S. Nuclear Engineering, University of Tennessee, 1974.

Experience:

Organization: TVA, Nuclear Fuel Department, BWR Fuel Engineering
Title: Senior Engineering Specialist
Date: September, 1989 to present

Primary functions are: perform/co-ordinate engineering analyses including in-core physics and thermal hydraulic analyses, fuel management, and safety and transient analyses for the Browns Ferry reactors. Provide on shift assistance to Browns Ferry Reactor Engineers during sensitive phases of Unit 2 restart testing. Serve as Test Director for Browns Ferry Unit 3 fuel inspection. Represent TVA on industry committees. Perform Nuclear Fuel Department 10CFR 50.59 Level II reviews of fuel and core related safety evaluations for TVA reactors (both PWR and BWR). Perform check-out and validation analyses for TVA operator training simulators.

Organization: TVA, Nuclear Fuel Dept., Methods Development Branch
Title: Senior Engineering Specialist, Safety Analysis Methods
Date: September, 1988 to September 1989

Primary functions: provide engineering technical consulting to department managers and engineers in a wide range of methods and analyses including in-core physics and thermal hydraulic analyses, fuel management, and safety and transient analyses; define overall engineering requirements to implement new analysis methodologies; define interface requirements to produce integrated computer code systems for core design, simulation and safety analysis activities; and develop or direct development of new methodologies in the areas of core simulation, criticality, thermal-hydraulics and plant safety analyses.

Organization: TVA, Nuclear Fuel Branch
Title: Staff Nuclear Engineer
Date: February 1983 to September 1988

In technical charge of work performed under EPRI contract involving the development of a two-dimensional space-time diffusion code for application to BWR control rod drop accidents. Directed successful use of the methodology in preforming reload analyses for Browns Ferry Unit 3, cycle 6. Provided consulting to engineers developing method for BWR stability analysis. Developed a new three-dimensional, static, reactor core simulator code based on an advanced nodal method.

Organization: TVA, Nuclear Fuel Branch
Title: Staff Nuclear Engineer
Date: May, 1980 to February 1983

Headed a special team of six engineers responsible for developing, qualifying and obtaining regulatory approval of models based on the RETRAN program for performing licensing safety analyses for the Browns Ferry Nuclear Plant. The team goals were achieved in less than estimated time and resources.

Date: November, 1977 to May, 1980

Responsible for technical direction of TVA efforts as participants in the EPRI/Utility Systems Analysis Working Group. Efforts included developing detailed models of TVA nuclear plants for various developmental versions of the RETRAN code to determine models and features required in the final code. Also provided guidance and consulting assistance to other engineers in maintaining and extending capabilities of TVA's static reactor physics methods.

Date: November, 1973 to November, 1977

Responsible for development of the static, three-dimensional reactor core simulation code, CORE, and its qualification for use in fuel management, core design and licensing calculations.

Date: June, 1969 to November, 1973

Developed the SCOPING program for multi-cycle fuel management simulation and optimization of fuel management decisions. Worked on assignment at Gulf United Nuclear Fuels for seven months assisting in a joint development effort for LWR lattice physics and core simulation codes. Spent one year on assignment at Westinghouse PWR division participating as a member of the start-up testing team for the Ginna plant, also worked in the Nuclear Design group assisting in the preliminary core design for the Salem plant.

Organization: TVA, Nuclear Power Staff
Title: Nuclear Engineering Co-op student
Date: March, 1966 to June 1968

Assisted engineers in performing neutronics, thermal-hydraulic and economic evaluations of proposed nuclear power plants.

Other Professional Activities

Part-time consultant, current and former clients include:
General Physics, 1990 to present.
H. L. Dodds & Associates, 1990 to 1991.
SMC-ESMI, 1989,
Technology for Energy Corporation, 1979 to 1983.

Member of Electric Power Research Institute advisory committee for Reactor Physics Methods and the committee for Severe Accident Analysis Methods.

Member of Boiling Water Reactor Owners Group Reactor Committee on Thermal-Hydraulic Instability Problem Resolution.

Technical paper reviewer for Journal of Nuclear Technology, 1984-present, and for Transactions of the American Nuclear Society, 1978-1984 also for ASME Journal of Heat Transfer, 1983-1985.

Member TVA Nuclear Fuel Design Review Board.

Alternate member and consultant to TVA Nuclear Plant Safety Review Board 1979-1984.

Member of Technical Program Committee for ANS topical meeting "Advances in Fuel Management", March 2-5, 1986, Pinehurst N. C.

Reactor Analysis Session Organizer for American Nuclear Society winter meeting in 1984.

Session chairman for Second International RETRAN Meeting, 1981.

Registered engineer state of Tennessee, since 1975.

Member American Nuclear Society

Partial List of Presentations and Publications authored or co-authored:

"TVA Code Development, Topical Reports, Submittals, and NRC Meetings Related to Reload Core Licensing," solicited for Electric Power Research Institute (EPRI) / Utility Systems Analysis Meeting, New Orleans, La., March, 1987.

"BNC Code: Theory Manual, Programmers Guide, Users Guide and Applications Guide," reports prepared under Electric Power Research Institute contract RP-1761-20, May, 1984.

TVA-RLR-001, "Reload Licensing Report for Browns Ferry Unit 3, Cycle 6, Appendix A (Rod Drop Accident Methodology) and Appendix B (Stability Analysis Methodology)." Reviewed by the Nuclear Regulator Commission (NRC) and a Safety Evaluation Report (SER) issued in October, 1984.

"Qualification of a RETRAN-02 Model for Boiling Water Reactor Transient Analysis," Nuclear Technology, Vol. 61, No. 2, May, 1983.

"A Method to Determine Transient Critical Power Ratios for Boiling Water Reactors," 2nd International Topical Meeting on Nuclear Thermohydraulics, Santa Barbara, Ca., January, 1983.

"Qualification of a RETRAN-02 Model for BWR Transient Analysis," 2nd International RETRAN Conference, San Diego, Ca., (Invited) April, 1982.

TVA-TR81-01A, "BWR Transient Analysis Model Utilizing the RETRAN Program, (Models, Qualification Comparisons, Sensitivity Studies & Statistical Treatment of Uncertainties)," Submitted to NRC July, 1982, (approved for referencing in licensing and SER issued April, 1983).

"Calculation of a Generator Load Rejection Transient in a 3293MW BWR/4 with RETRAN-02," EPRI / Utility Systems Analysis Meeting, September, 1981.

"A Method to Calculate Optimum Power Profiles in LWRs," Transactions of American Nuclear Society, Vol. 32, No. 2 November, 1979.

TVA-TR79-01A, "Verification of TVA Steady State BWR Physics Methods," Licensing Topical Report submitted to NRC January, 1979 (approved for referencing and SER issued November, 1979).

TVA-TR78-03A, "Three-Dimensional LWR Core Simulation Methods," June, 1978, Licensing Topical Report submitted to NRC January, 1979 (approved for referencing and SER issued November, 1979).

TVA-TR78-02A, "Method for Lattice Physics Analysis of LWRs," June, 1978, Licensing Topical Report submitted to NRC January, 1979 (approved for referencing and SER issued November, 1979).

"Two-Group Nodal Core Simulator Based on TRILUX Style Coupling," American Nuclear Society-Canadian Nuclear Association Joint Meeting in Toronto, Ontario, June, 1976.

"An Algorithm for Determining Optimal Fuel Management with Generalized Cycle Requirements," American Nuclear Society Mathematics & Computation Topical Meeting, Ann Arbor, Mi., April, 1973.