Responses to NRC Questions on River Bend Station Plant Transient Analysis Methodology, Supplement 1, EA-PT-91-0003-SP

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Thermal/Hydraulic Analysis Group
Engineering Analysis Section
Engineering Department
River Bend Nuclear Group
Gulf States Utilities

Contributing Engineers

T. W. Oliphant
D. R. Swope

Reviewer: J. L. Thompson — Date: 2/17/93
J. L. Thompson
Supervisor - Thermal/Hydraulic Analysis

Approval: [signature] for JS Miller — Date: 2/17/93
J. S. Miller
Director - Engineering Analysis

1.0 Summary

Gulf States Utilities (GSU) submitted topical report EA-PT-91-0003-M[1], entitled "River Bend Station Plant Transient Analysis Methodology," to the U. S. Nuclear Regulatory Commission (NRC) in May, 1991. This report describes GSU's computer programs, system models, and methods used for plant transient analysis at River Bend Station (RBS). The report also provides comparisons to Peach Bottom 2 transients and selected River Bend transients.

A topical report supplement, EA-PT-91-0003-SP[2], entitled "River Bend Station Plant Transient Analysis Methodology, Supplement 1," was submitted to the NRC in October, 1991. The supplementary report describes GSU's ΔCPR methodology, hot channel model, and uncertainty analysis methodology. The supplement also presents applications of the RBS methodology to calculation of thermal limits. Additional comparisons with RBS transients and revised Peach Bottom 2 transient comparisons were provided.

In October, 1992, the NRC requested[3] additional information on the transient methodology topical report EA-PT-91-0003-M. GSU's responses to the NRC questions are documented in EA-PT-0003-S1[4] which was submitted to the NRC in December, 1992.

An NRC request[5] for additional information on the topical report supplement, EA-PT-91-0003-SP, was received in December, 1992. Section 2.0 of this report provides responses to those questions. The format for Section 2.0 consists of a restatement of the NRC question followed by the GSU response. References are provided in Section 3.0.

2.0 Responses to NRC Questions on River Bend Station Plant Transient Analysis Methodology, Supplement 1, EA-PT-91-0003-SP

Question 1

Describe the Chapter 9 revision to the core physics procedures which required the regeneration of the Peach Bottom 2 kinetics data. Why does this change only have a significant effect on the TTI predictions?

Response

The revision to the core physics procedures involved the use of ENDF/B-V kinetics tables (delayed neutron fractions, betas, and fast and thermal neutron velocities, V) instead of ENDF/B-III tables or modified versions. The cross sections remained ENDF/B-III. There were also minor changes in the SIMULATE-E runs due to corrected rod patterns in several depletion steps and other minor changes.

The change in procedure was used for all three Peach Bottom trip events and resulted in slight increases in calculated power over the calculations made before the previous revision. In addition, some changes to thermal/hydraulic data were made for all three Peach Bottom trip events. The decrease in peak power seen in the TT1 calculation was due to changes in thermal hydraulic data resulting from sensitivity studies with a criteria of improved calculation of measured parameters.

The most significant of these changes was a redistribution of separator inertia between inlet and outlet. TT1 was most affected because it has the smallest total separator inertia and saw the largest percentage change at the separator inlet, which has the largest impact on the results. The response to Question 7 provides a description of the separator inertia changes.

Question 2

Provide the basis for the 95/95 uncertainty values used for the core leakage flow, total core pressure losses, jet pump M-ratio, TCV stroke time, initial thermal power, and maximum runout time.

Response

To quantify the methodology uncertainty, ΔCPR sensitivity studies were performed which perturbed the components of uncertainty by an amount judged to be two standard deviations from their nominal value at a 95 percent confidence level. Engineering

judgment was used to select and quantify these components of uncertainty, supplemented by information available from technical references and test data reports. In addition, License Topical Reports previously submitted by fuel vendors[6,7], other licensees[8,9], and national laboratories[10] were reviewed. The standard deviations quoted by these prior uncertainty analyses were compared to the values derived for River Bend specific systems and conditions, and helped guide our selection process. For many of the low significance uncertainty components, bounding values were used as the 2σ uncertainty level. The bases for the specific uncertainty components requested are provided in Table I below.

TABLE I

BASES FOR UNCERTAINTY COMPONENTS

Component	Uncertainty (2σ)	ΔCPR Sensitivity	References for Bases
Core Leakage Flow	10%	Low	8, and 9
Total Pressure Losses	10%	Low	6, 8, and 9
Jet Pump M Ratio	5%	Medium	11
TCV Stroke Time	20%	Medium	Plant Data
Initial Thermal Power	2%	Insignificant	7
Feedwater Maximum:			
Runout Flow	5%	Low	8, and 9
Ramp Rate	50%	Insignificant	Bounding

Question 3

The Table-6.8 values of $\delta RCPR_{95/95}$ and resulting statistical adjustment factors have been determined for specific transients having relatively small values of RCPR. Will the Table-6.8 values of $\delta RCPR_{95/95}$ be applied in the RBS licensing analyses on a percent basis (i.e. δRCPR/RCPR is assumed constant)? If not, provide justification for the method used to apply these $\delta RCPR_{95/95}$ uncertainty values.

Response

The $RCPR_{95/95}$ values will not be used in the GSU licensing analysis on a percent basis. The $RCPR_{95/95}$ value is used to determine the statistical adjustment factor (SAF) for

each transient (e.g. load rejection without bypass, pressure regulator downscale failure). Specifically,

$$SAF_N = (RCPR_{95/95})_N - (RCPR_{\text{design basis calc.}})_N$$

where N is the transient type.

In turn, the SAF for each transient is applied to determine the required operating limit MCPR (OLMCPR) for each transient as follows:

$$OLMCPR_N = \frac{SLMCPR}{1 - [(RCPR_{\text{design basis calc.}})_N + SAF_N]}$$

where SLMCPR is the safety limit MCPR.

Since transient specific SAFs are used to account for uncertainties in the calculation of transient specific OLMCPRs, RCPR ratios are not needed to account for uncertainty differences for various transients. Section 6 of Reference 2 contains a description of this process.

RCPR is defined as ΔCPR/ICPR as described in Section 5.0 of Reference 2. The $RCPR_{95/95}$ values presented in Table 6.8 are not delta values, but represent the 95% probability ΔCPR/ICPR value at a 95% confidence level for a specific transient. ΔRCPRs, given in Tables 6.4 through 6.6, are used in the RBS methodology for calculating the standard deviations which in turn are used in calculating the $RCPR_{95/95}$ values presented in Table 6.8.

The RCPR values for the three events are typical of values calculated for rapid transients terminated by a reactor protection system (RPS) trip in a BWR/6. The BWR/6 design has a higher control rod insertion rate than the BWR/3, 4, and 5 designs. This feature reduces the severity of the pressurization events (e.g. load rejection, feedwater controller failure, and pressure regulator downscale failure).

Question 4

The scram and doppler reactivity contributions to the excessive reactivity are relatively small initially, but ultimately increase sufficiently to dominate the excess reactivity and terminate the transient. How does the calculated reactivity compare with the reactivity inferred from the Peach Bottom 2 measurements, as a function of time, up to the point of minimum CPR? Justify the use of the uncertainty in the peak excess reactivity for the scram and doppler reactivity uncertainties.

Response

Core power is strongly dependent upon excess reactivity, so changes in total reactivity are mirrored by changes in power. In References 1 and 2, GSU chose to compare the time dependent calculations to the directly measured parameter, core power, for each of the Peach Bottom 2 turbine trip tests. In Section 6.1.2 of Reference 2, the peak reactivity of the Peach Bottom turbine trip tests was inferred from the point reactor kinetics equation, and is therefore proportional to (1/Power)(dPower/dt). A comparison of the calculated and inferred reactivities for TT1 as a function of time is shown in Figure 1.

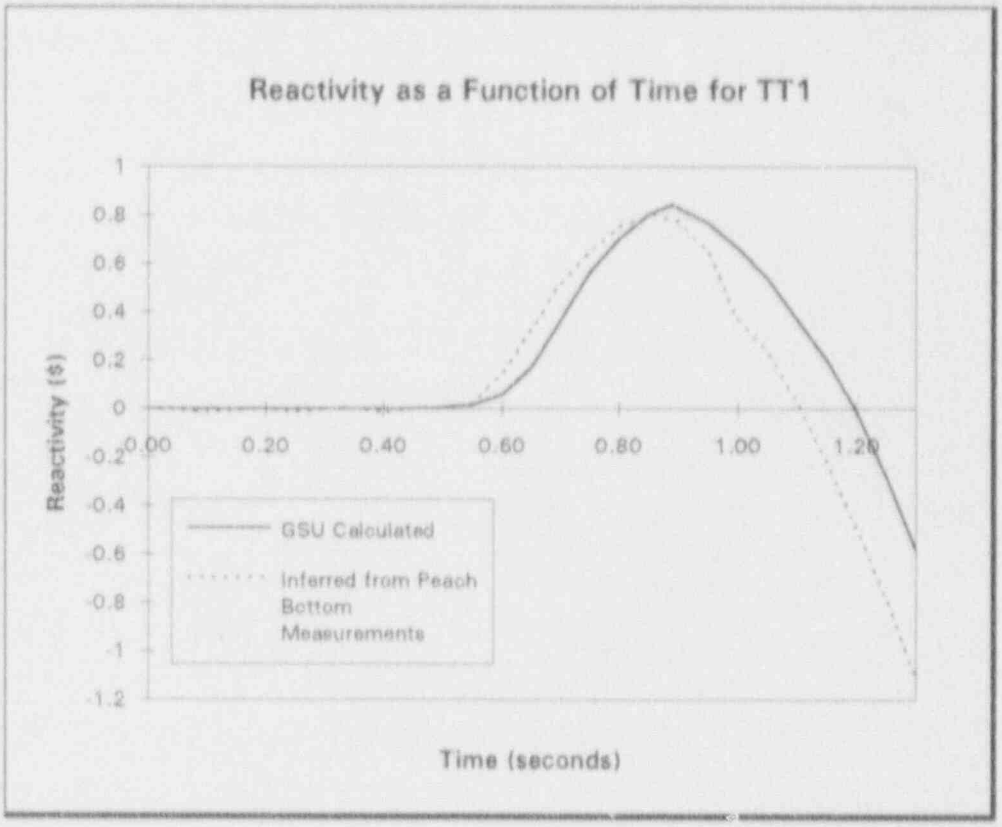

Figure 1 Reactivity as a function of time for TT1.

The point reactor kinetics equation was used to help quantify the uncertainty in kinetics parameters in units of reactivity. GSU compared the inferred reactivity from each of the Peach Bottom tests to calculated excess reactivity. The standard deviation in peak excess reactivity was used as a convenient single measure of the adequacy of the neutronics data (i.e., two group cross sections, delayed neutron fractions, decay

fractions, and mean neutron generation times). Use of the peak excess reactivity uncertainty is justified for two reasons:

1) Being proportional to the derivative of the power spike rate, peak excess reactivity is a good indicator of the transient response including void and Doppler feedbacks, just prior to rod insertion, and

2) Although scram worth is a strong feedback, its contribution to the transient uncertainty is small since once an axial portion of the core is fully controlled, power generation in that portion is dramatically reduced. The major uncertainty associated with scram is the timing of the scram, not the rod worth. Scram time uncertainty is treated separately in the GSU methodology.

Question 5

Compare the GSU and vendor methodologies for determining the flow dependent $MCPR_f$ limit, and justify any differences. How do the GSU and vendor predictions of the $MCPR_f$ limit compare?

Response

The vendor and GSU methods are essentially the same. Both use a steady-state thermal-hydraulics program (FIBWR in the case of GSU). Both of these programs are similar in nature, and use equivalent methods and correlations in the solution of flow distribution within the BWR core, as well as MCPR determination.

The method to be employed, is to perform calculations at the 105%-of-rated steam flow control rod line, at the maximum core flow rate attainable with the recirculation flow control valves at the electronically limited valve position. The purpose of these calculations is to adjust the bundle relative power until the calculated bundle MCPR is equal to the safety limit MCPR. Keeping the relative bundle powers and axial power shapes intact, a series of calculations are performed at different core flows along the 105%-of-rated steam flow control line. The calculated MCPR at the given flow point is the $MCPR_f$. This is the same approach as outlined in the Technical Specification[12], BASES 3/4.2.3. The GSU and vendor calculated $MCPR_f$ curves are similar.

Question 6

Discuss the difference between the control rod insertion curves for the pressurization and overpressurization events.

Response

The scram function of the control rods is accomplished by scram accumulators located at the hydraulic control unit of each control rod. The accumulator is partially filled with water, and is maintained at a relatively high pressure by means of a nitrogen blanket. Upon receipt of a scram signal, the scram valves open, admitting the water in the accumulator to below the drive piston, which causes the control rod to rapidly insert against reactor pressure. Therefore, the rate of insertion is dependent upon the reactor pressure.

The overpressurization event results in higher vessel pressures during the insertion of control rods than pressurization and non-pressurization events. Therefore, the rate at which the control rods are inserted is slower for the overpressurization transient. The insertion rates used by GSU are the same as those used by the fuel vendor. Figure 2 below shows the nominal scram rates used for evaluation of non-pressurization, pressurization, and overpressurization events. Also shown is the Technical Specification scram speed requirement for reactor dome pressure of 1040 psia. Note that all scram curves used for non-pressurization, pressurization, and overpressurization events are more conservative than the Technical Specification requirement.

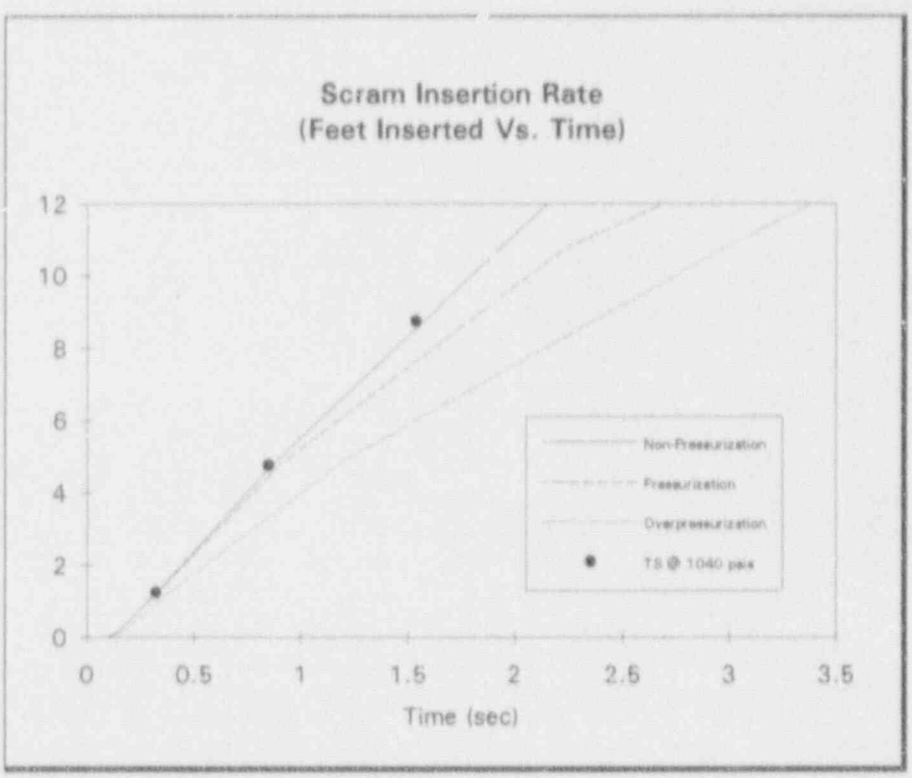

Figure 2 Scram insertion rate used for various events.

Question 7

Has the RBS model been adjusted to improve agreement with the Peach Bottom 2 measurements and, if so, what is the effect of this adjustment on the inferred peak excess reactivity?

Response

The only adjustment made, other than to match specific initial conditions for each of the three turbine trip tests, was the manner of allocating separator inertia to the separator inlet, standpipe inlet, and separator return junctions.

The total inertia was determined as a function of separator inlet quality from vendor data as described in Section 7.3.2 of Reference 1. The allocation to specific junctions, also described in Reference 1, has been revised to put a constant value at the separator return junction, half of that part attributed to the standpipes at the standpipe inlet and the remainder of the total at the separator inlet. As separator inlet quality and total separator inertia vary, only the separator inlet inertia changes. The separator outlet, which is almost all liquid, has constant inertia. This gave improved power response for TT1 without significantly affecting the other Peach Bottom events.

The power curve, and therefore the peak excess reactivity, was found to be sensitive to the separator inertia distribution. A larger difference between calculated and measured powers will indicate a larger difference between calculated and inferred peak excess reactivity.

Question 8

How were the standard deviations of Table 6.8 determined? Do the $RCPR_{95/95}$ values of Table 6.8 include a two standard deviation allowance as indicated in Figure 6.1?

Response

Table 6.4 through 6.6 in Reference 2 present transient specific uncertainty values for each component of uncertainty evaluated. Since the initial perturbations were two standard deviation (2σ) values, the uncertainties are also 2σ values. The square root of the sum of the squares (SRSS) of the ΔRCPR values was calculated to determine an overall uncertainty in CPR units for the transient at the 2σ level. The one standard deviation (σ) value for the transient is then half of the SRSS of the ΔRCPR values.

The $RCPR_{95/95}$ values shown in Table 6.8 do not include a two standard deviation allowance. Instead, the $RCPR_{95/95}$ value is the mean RCPR value plus 1.645 standard deviations.

Question 9

What is the difference between the $\Delta RCPR_{95/95}$ data of Table 6.8 and Tables 6.4 through 6.7?

Response

As discussed in the response to Question 3, the $RCPR_{95/95}$ values in Table 6.8 are not delta values, but represent the 95/95 ΔCPR/ICPR value for a specific transient. The $RCPR_{95/95}$ values are used to calculate the transient specific SAFs.

The ΔRCPR values shown in Tables 6.4 through 6.6 represent the transient specific uncertainty at the 2σ level for a particular component of uncertainty (e.g. initial thermal power). The ΔRCPR values for a transient are combined statistically and then used to calculate the RCPR standard deviation which is in turn used to calculate the transient specific $RCPR_{95/95}$.

Table 6.7 does not contain ΔRCPR data.

Question 10

What is causing the ~20% GSU underprediction (relative to the vendor) of the ΔCPR for the pressure downscale failure (PDR) and feedwater controller failure (FWCF) events? Note that this difference is outside the expected $\Delta CPR_{95/95}$ given in Table 6.8 for these events.

Response

The ΔCPR agreement between the GSU and vendor methods is very good. Due to the very small ΔCPR for the pressure regulator failure and the feedwater controller failure events, the percentage difference is large. However, the magnitude of this difference is not significant. A difference of 0.01 in ΔCPR between the vendor and GSU calculations is to be expected given the potential for subtle modeling differences, and round-off.

The values in Table 6.8 are based on an uncertainty analysis for the GSU methodology which is used to calculate SAFs. The SAFs are used to adjust the OLMCPR for

uncertainties in the GSU computer codes and analytical models. Since the statistical values in Table 6.8 are based entirely on GSU calculated results, differences between the vendor and GSU calculated values do not have any relationship to the expected uncertainty given in Table 6.8.

Question 11

What initial conditions (e.g., core flow, cycle exposure, power distribution, feedwater flow and temperature) will be assumed in the LRNB, PDF, FWCF and MSIV closure licensing analyses and are these conditions conservative relative to expected operating conditions.

Response

Tables 7.1 and 7.2 of Reference 2 list initial conditions for pressurization and overpressurization events, respectively. Tables II and III list the values used for initial conditions and parameters in addition to those listed in the table referenced above. Pressurization events are initiated at rated conditions with uncertainty accounted for in the statistical adjustment factor (SAF). The SAF provides a conservative estimate of MCPR. Overpressure events are initiated at conditions consistent with a conservative bounding initial condition of 102% power.

TABLE II

INITIAL CONDITIONS AND PARAMETERS FOR PRESSURIZATION EVENTS

Parameter	Value	Justification
Core Flow	84.5 Mlb/hr	Rated Core Flow
Cycle Exposure	End of Cycle	Bounding, provides conservative void and scram feedback.
Power Distribution:		
System Model	Haling	This is the optimized end of cycle power distribution target for operation.
Hot Channel	1.4 peaked chopped cosine	Conservative.
Feedwater Flow	12.45 Mlb/hr	Consistent with 100% power heat balance.
Feedwater Temperature	420 °F	Consistent with 100% power heat balance.

TABLE III

INITIAL CONDITIONS AND PARAMETERS FOR
THE OVERPRESSURIZATION EVENT

Parameter	Value	Justification
Core Flow	84.5 Mlb/hr	Rated Core Flow
Cycle Exposure	End of Cycle	Bounding, provides conservative void and scram feedback.
Power Distribution:		
System Model	Haling	This is the optimized end of cycle power distribution target for operation.
Hot Channel	N/A	Hot Channel not used in ASME overpressurization analysis.
Feedwater Flow	12.74 Mlb/hr	Consistent with 102% power heat balance.
Feedwater Temperature	422 °F	Consistent with 102% power heat balance.

Question 12

Will the bypass flow be modeled as a negative fill in the FWCF analysis to ensure there is no overshoot of the bypass flow capacity?

Response

While GSU believes that some overshoot of bypass flow is realistic, our design basis methodology conservatively models the bypass flow using a negative fill in all cases where the bypass valves are predicted to open. The Peach Bottom turbine trip transients were best estimate calculations with a modification consisting of piping with heat sink from the bypass valves to a time dependent condenser. The intention was to simulate those transients as closely as possible, including any overshoot that may have occurred. However, the RBS design basis model, which models the bypass as a negative fill for the FWCF, is intentionally conservative in this regard relative to best estimate models.

Question 13

How are calculational uncertainties accounted for in the MSIV closure overpressurization analysis.

Response

A bounding approach is used for the MSIV overpressurization analysis to account for calculation uncertainties. The initial power is assumed to be 102% of rated thermal power, rather than 100% of rated thermal power which was assumed for the pressurization events. Other initial conditions (dome pressure, steam flow, feedwater temperature, etc.) were determined from a heat balance for the 102% power and 100% core flow condition.

Question 14

Describe the modeling of the closing of the MSIVs and opening of the SRVs in the RBS overpressurization analysis. Does this treatment conservatively bound the performance of these systems?

Response

The MSIVs are assumed to close linearly over a three second period, which is the fastest speed the valves may close by design. This also is the fastest rate allowable by Technical Specifications[12].

The pressure relief system at River Bend Station employs sixteen dual acting safety/relief valves (SRVs). The SRVs are actively opened in the relief mode by an air actuator when sensed pressure in the reactor vessel exceeds the relief set pressure. The SRVs are passively opened in safety mode when steam line pressure exceeds the SRV spring set pressure.

Following the present licensing basis, eight SRVs with the highest relief set pressures are allowed to open in the relief mode. The remaining eight SRVs are those with the highest safety set pressures and are allowed to open in safety mode. The mode selected for each of the SRVs to operate results in the highest peak pressure for the overpressurization event. System performance is conservatively modeled using design logic delays, and valve opening characteristics. Mass flow through the valve is reduced by decreasing the valve flow area from the best estimate value by 10%, which is consistent with the ASME code requirements.

The above measures ensure that the MSIVs close in a conservatively fast manner, and that the predicted safety relief flow is conservatively low prior to reaching the peak pressure.

Question 15

Provide the predicted peak pressure and margin to limits for each of the reactor coolant pressure boundary components.

Response

Peak pressure in the reactor vessel is limited to 110% of the design pressure of 1250 psig, or 1375 psig. Table IV below gives the pressures, limits, and margin for a variety of locations within the reactor coolant pressure boundary.

TABLE IV

MARGIN TO PRESSURE LIMITS FOR REACTOR PRESSURE BOUNDARY COMPONENTS

Location	Peak Pressure (psig)	Limiting Pressure (psig)	Margin (psi)
Vessel Bottom Head	1247	1375	128
Steam Line	1218	1375	157

The peak pressure at the bottom of the vessel (peak pressure location) reported above is slightly greater than the value reported in Reference 2. Since the issuance of Reference 2, it has been discovered that the reactor tripped approximately 0.226 seconds early due to a minor input error. The difference in pressure is small, and is still well within the acceptance limit of 1375 psig.

Question 16

In the MSIV closure overpressurization analysis, how many SRVs are considered to be inoperable and is this consistent with the maximum allowed in the technical specifications?

Response

The overpressurization analysis reported in Reference 2 was performed with all SRVs operable (eight in safety mode and eight in relief mode as discussed in the response to Question 14). This is consistent with the analysis provided by the fuel vendor with each reload. As discussed in the response to Question 15, the GSU overpressurization analysis shows a peak pressure well below the acceptance limit of 1375, with somewhat greater margin than shown for 16 SRVs operable in the analysis presented in the RBS Updated Safety Analysis Report[13] (USAR). of Reference 13.

The RBS Technical Specifications require five SRVs to be operable in the safety mode, and an additional four SRVs to be operable in the relief mode. The requirement for nine operable SRVs is based on the USAR analysis. USAR Figure 5.2-4 shows the results of the analysis for eight through sixteen SRVs operable. The analysis shows that nine operable SRVs are sufficient to prevent reactor vessel pressure from exceeding the acceptance limit, with larger margin for more valves operable, as expected.

The GSU calculations confirm that the worst cycle conditions result in a wider margin to the pressure limit with 16 SRVs operable than do the original USAR calculations. It is expected that a GSU calculation with only nine SRVs operable would also show a greater margin than does the original USAR analysis and therefore would bound the Technical Specifications.

Question 17

How is the uncertainty in the time-dependent hot-channel radial peaking factor accounted for in the licensing analyses?

Response

The uncertainty in the hot channel radial peaking factor is accounted for by choosing a value which bounds the highest value expected to occur during steady-state. One of the major assumptions of one-dimensional (axial) transient analysis is that the radial power distribution remains constant throughout the transient. Most radial power fluctuations which occur as a result of a rapid core-wide power spike will tend to flatten the initial power distribution, hence the constant radial peaking factor assumption is conservative.

The trend observed in the local power range monitor (LPRM) measurements taken during the Peach Bottom turbine trip tests indicate little change in the relative bundle powers during the transient. These LPRM measurements further substantiate the adequacy of the constant radial peaking factor assumption.

Question 18

What effect does the GSU core physics procedures revision have on the inferred (calculation-to-measurement) peak reactivity differences and the determination of $RCPR_{95/95}$?

Response

The core physics procedure revision occurred prior to the completion of the sensitivity studies and uncertainty analyses. Since that time, the revised procedure has been used for all Peach Bottom core physics calculations. No $RCPR_{95/95}$ calculation results were submitted using the old procedure. As the requested analyses were never performed with the obsolete method, we are unable to generate them without a substantial effort.

3.0 References

1. "River Bend Station Plant Transient Analysis Methodology," **EA-PT-91-0003-M**, April 1991, transmitted to the NRC by RBG-34,939 dated May 2, 1991.

2. "River Bend Station Plant Transient Analysis Methodology; Supplement 1 Delta CPR Methodology and Additional Benchmarks", **EA-PT-91-0003-SP**, October 1991, transmitted to the NRC by RBG-35876 dated October 31, 1991.

3. Letter from D. V. Pickett (NRC) to J. C. Deddens (GSU) dated October 30, 1992, River Bend Station, Unit 1 - Request for Additional Information Re Topical Report EA-PT-91-0003-M, "River Bend Station Plant Transient Analysis Methodology," (TAC No. M80315), RBC-42927.

4. "Responses to NRC Questions on River Bend Station Plant Transient Analysis Methodology EA-PT-91-0003-M," **EA-PT-91-0003-S1**, December 1992, transmitted to the NRC by RBG-37,930 dated December 18, 1992.

5. Letter from E. T. Baker (NRC) to P. D. Graham (GSU) dated December 22, 1992, River Bend Station, Unit 1 - Request for Additional Information Regarding Topical Report EA-PT-91-0003-M, Supplement 1, "River Bend Station Plant Transient Analysis Methodology" (TAC No. M80315), RBC-43190.

6. "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors NEDO-24154 and NEDE-24154-P, Volumes I, II, and III", USNRC, June 1980.

7. "Analytical Model for Loss-of-Coolant Analysis in Accordance With 10CFR50 Appendix K," **NEDO-20566A**, General Electric Company, September 1986.

8. "BWR Transient Analysis Model Utilizing the RETRAN Program," **TVA-TR-81-01**, Tennessee Valley Authority, December 1981.

9. "PECo Methods for Performing BWR System Transient Analysis," **PECo-FMS-0004**, September 1987.

10. "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," USNRC, June 1980.

11. "Testing of Improved Jet Pumps for the BWR/6 Nuclear System", **NEDO-10602**, General Electric Company, June 1972.

12. River Bend Station Technical Specifications.

13. River Bend Station Updated Safety Analysis Report (USAR).