Maria Korsnick
Site Vice President

R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, New York 14519-9364
585.771.3494
585.771.3943 Fax
maria.korsnick@constellation.com

Constellation Energy
Generation Group

December 9, 2005

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

ATTENTION: Document Control Desk

SUBJECT: R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Response to Requests for Additional Information Regarding Revised Loss-of-Coolant-Accident Analyses

This letter is in response to the October 28, 2005 "Request for Additional Information Regarding Revised Loss-of-Coolant-Accident Analyses, R. E. Ginna Nuclear Power Plant (TAC No. MC6860)". Ginna committed to provide responses within 45 days, but acknowledged in a December 18, 2005 public meeting on this subject that, due to the comprehensive level of analysis being performed, certain responses would require additional time. Attachment 1 provides the majority of the responses to the October 28 request. The balance of the responses, denoted as "Post-LOCA Long-Term Cooling" RAIs #2, #3, and #5 will be submitted by January 16, 2006.

These responses do not include any other regulatory commitments. If you have any questions, please contact George Wrobel at (585) 771-3535 or george.wrobel@constellation.com.

Very truly yours,

Mary Korsnick

Mary G. Korsnick

A001

1001448

STATE OF NEW YORK :
: **TO WIT:**
COUNTY OF WAYNE :

I, Mary G. Korsnick, being duly sworn, state that I am Vice President – R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this response on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Mary G. Korsnick

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Monroe, this 9 day of December, 2005.

WITNESS my Hand and Notarial Seal:

Sharon L. Miller
Notary Public

My Commission Expires:
12-21-06

SHARON L. MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 2006

Attachments

Cc: S. J. Collins, NRC
P. D. Milano, NRC
Resident Inspector, NRC

Mr. Peter R. Smith
New York State Energy, Research, and Development Authority
17 Columbia Circle
Albany, NY 12203-6399

Mr. Paul Eddy
NYS Department of Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223-1350

Attachment 1

Response to October 28, 2005 Request for Additional Information

By letter dated April 29, 2005 (Agency wide Documents Access and Management System Accession No. ML051260239), R.E. Ginna Nuclear Power Plant, LLC (the licensee) submitted an application to amend the technical specifications (TSs) for the R.E. Ginna Nuclear Power Plant. Specifically, the licensee proposed changes that would reflect the revised analyses performed in support of the planned extended power uprate. To complete its review, by letter dated October 28, 2005, the NRC staff requested the following information:

Effects Of Post-LOCA Analysis on Containment Sump pH

1. In order to complete its evaluation, the staff needs to review the general assumptions and calculations used by the licensee to prove that the containment sump pH will be maintained above 7 throughout the duration of the accident.

   Describe the procedure utilized for calculating pH of the containment sump water during the 30 day period after a loss-of-coolant accident (LOCA). If the calculations were performed manually, describe the methodology and provide sample calculations. If a computer code was used, provide the input to the code and the results calculated by it.

   Response:

   Plant design criteria require that the containment sump be maintained within a certain pH range during post-accident operation. This ensures the long term availability of the Safety Injection System (SIS), and prevents iodine from re-entering the containment atmosphere. The addition of boron during Containment Spray and Safety Injection makes the initial sump pH in the acidic range. To neutralize the sump pH, sodium hydroxide (NaOH) is added from the spray additive tank (SAT) via the Containment Spray System. Sump pH is a function of both boron and NaOH concentrations.

   In this evaluation, the pounds of boron and the pounds of NaOH in the sump is determined by considering both minimum and maximum delivered volumes and concentrations from boration sources and the SAT. The boration sources include the Refueling Water Storage Tank (RWST), Reactor Coolant System, and the SIS Accumulators. The pH is determined for each case using boric acid/NaOH titration curves. Although a computer code is used to facilitate the calculation, the results can be easily verified with a hand calculation.

   For example the computer code inputs with respect to the minimum calculated sump pH are as follows:

Mode 1 RCS Mass (lbm):	2.70165E5
RCS boron concentration (ppm):	2069 (BOL Max.)
RWST mass injected (lbm):	2.47825E6 (Max.)
RWST concentration (ppm):	3050 (Tech. Spec. Max.)
Accumulators mass injected (lbm):	1.4218E5 (Max.)
Accumulator concentration (ppm):	3050 (Tech. Spec. Max.)

SAT minimum injected volume (cu ft): 140.1

SAT NaOH concentration (wt percent): 30 (min.)

Based on the above inputs the code simply determines the average concentration of boron and NaOH in the sump post accident. Accordingly for the average boron concentration in the sump the code adds the masses injected at each boron concentration and divides by the total sump mass. And for the average NaOH concentration in the sump the code divides the total pounds of NaOH injected by the total sump mass. These results are used in conjunction with boric acid / NaOH titration curves to determine sump pH.

For the minimum sump pH case the code analysis results are as follows:

Sump boron concentration (ppm):	2950
Sump NaOH concentration (ppm):	1208
Sump pH:	7.8

Best-Estimate Large-Break LOCA (LBLOCA) Analysis

1. In order to show that the referenced, generically approved LOCA analysis methodologies apply specifically to Ginna, provide a statement that confirms that Ginna LLC and its vendor have ongoing processes to assure that the ranges and values of the input parameters for the Ginna LOCA analysis conservatively bound the ranges and values of the as-operated plant parameters.

Response:

Both Ginna LLC and its analysis vendor (Westinghouse) have ongoing processes which ensure that the values and ranges of the Best Estimate Large Break LOCA analysis inputs for peak cladding temperature and oxidation-sensitive parameters bound the values and ranges of the as-operated plant for those parameters.

The Ginna LLC process consists of documenting input parameters and rounding ranges along with the bases for the values in a "DBCOR" document. The "DBCOR" becomes the new "Accident Analysis Assumptions" when EPU is implemented. Any modifications to Ginna are required to follow the modification configuration control process. This process requires the effect of the modification on any Accident Analysis Assumptions to be addressed and resolved. Also, the Instrument Uncertainty program ensures the instrumentation can operate within the ranges assumed in the accident analysis (BELOCA).

2. If the plant-specific analyses are based on the model and/or analyses of any other plant, provide the justification showing that the model or analyses applies to Ginna.

   Response:

   The Best Estimate LBLOCA analysis and associated model to support the Ginna EPU are both Ginna plant-specific.

Over-Pressure Protection - Safety Valve Capacity

1. In its July 7 application with supporting documentation, descriptions of the provisions to address over-pressure protection were included for Ginna when operating at the uprated power. The NRC staff is reviewing continued sufficiency of the design margin of the safety valve capacity at the uprated power. The information provided in the application only addresses the change to the pressurizer safety valve lift setting and does not address the adequacy of the safety valve capacity. Although Table 5.2-1 in the Ginna Updated Safety Analysis Report does refer to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, "Nuclear Vessels," 1965, the application does not provide details about analyses that were done at the uprated power to demonstrate the adequate relief capacity and to show that sufficient design quantify margin remains.

   Westinghouse Report WCAP-7769, Revision 1, "Topical Report Overpressure Protection for Westinghouse Pressurized Water Reactors," dated June 1972, does provide demonstration of compliance for Ginna with Article NM-7000, "Protection Against Overpressure," in Section III of the ASME Code. However, WCAP-7769 assumed that Ginna operating at 1518.5 megawatts thermal (MWt).

   Provide the analysis results, determined using methods consistent with those in WCAP-7769 (including credit for the second (or later) safety grade trip from the reactor protection system), to show sufficiency margin in the design capacity for the Ginna pressurizer and steam line safety valves, with Ginna operating at the uprated power of 1775 MWt.

   Response:

   The Ginna EPU overpressure analyses are consistent with the requirements of SRP 5.2.2. SRP 5.2.2 requires that the second safety grade reactor trip signal be credited for safety valve sizing calculations. This is consistent with the safety valve sizing procedure discussed in Section 2 of WCAP-7769. WCAP-7769 states, "For the sizing, main feedwater flow is maintained and no credit for reactor trip is taken." This analysis is typically performed prior to construction of the plant to provide a basis for the capacity requirements for the safety valves and the requirement of SRP 5.2.2 provides a conservative basis for the number and design of the valves.

   However, WCAP-7769 goes on to say, "After determining the required safety valve relief capacities, as described above, the loss of load transient is again analyzed for the case where main feedwater flow is lost when steam flow to the turbine is lost. For this case, the bases for analysis are the same as described above except that credit is taken for Doppler feedback and appropriate reactor trip, other than direct reactor trip on turbine trip." This describes the analysis performed in Chapter 15 of the UFSAR which verifies that the overpressure limits are satisfied with the current/latest design.

The analyses performed in support of the Ginna EPU Program are not safety valve sizing calculations - no changes are being made to the safety valves as a result of this uprating. The Loss of External Electrical Load / Turbine Trip analysis performed for the EPU Program, presented in Section 2.8.5.2.1, demonstrates that the safety valves have adequate capacity to maintain peak primary pressure below 110% of design which satisfies the requirements of GDC-15. GDC-15 applies to "any condition of normal operation, including anticipated operational occurrences" which does not include a common mode failure of the first safety grade reactor trip signal.

The Loss of External Load / Turbine Trip RCS overpressure analysis is performed to demonstrate that, in the event of a sudden loss of the secondary heat sink, the associated increase in reactor coolant system temperature does not result in overpressurization of the RCS system.

Small-Break LOCA (SBLOCA) Analysis

RAI #1: Provide the full set of transient parameters for the 1.5, 2, and 3-inch break sizes that includes the following:

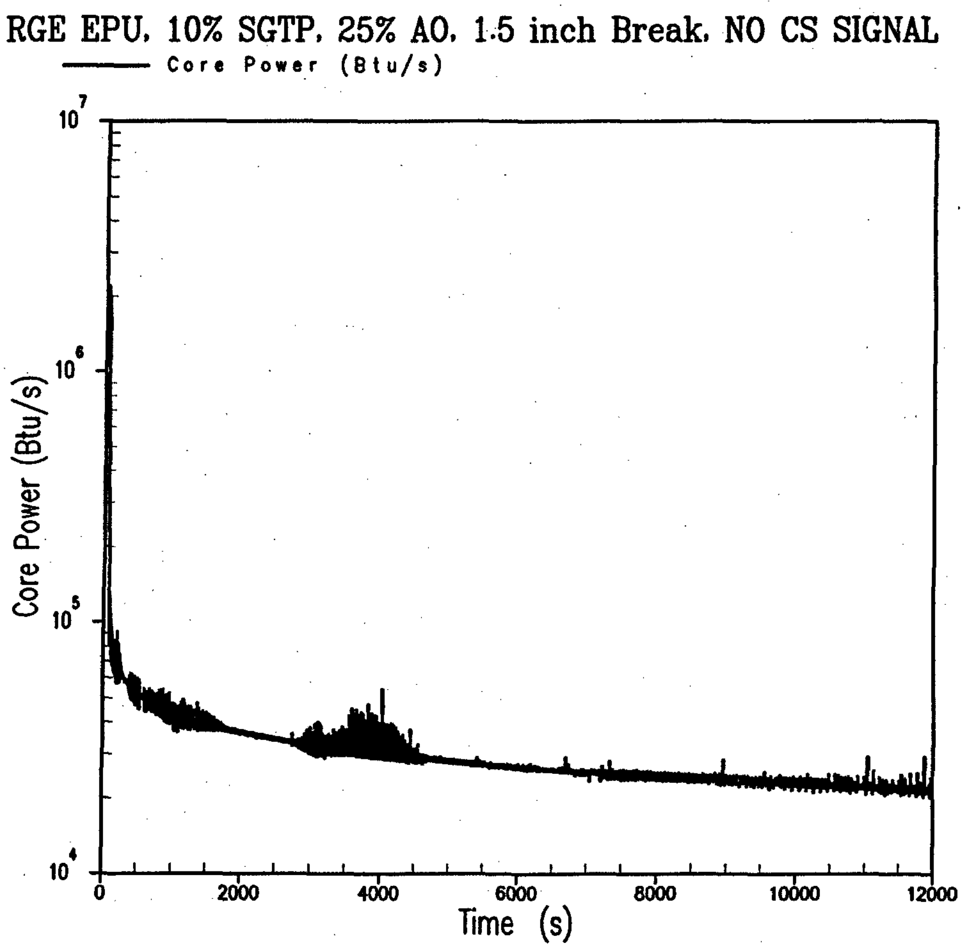
- 1.1 core power
- 1.2 core inlet mass flowrate
- 1.3 break mass flow rate
- 1.4 break quality
- 1.5 pressurizer pressure
- 1.6 inner vessel or core two-phase level
- 1.7 clad temperature at peak clad temperature (PCT) location
- 1.8 steam temperature at hot spot
- 1.9 heat transfer coefficient at hot spot
- 1.10 injection mass flow rate vs time (pumped should be separate from accumulator injection)
- 1.11 condensation rate in cold legs
- 1.12 void fractions in each core node versus time

Response:
The following plots address SBLOCA RAI #1, requesting transient parameters for the 1.5-, 2.0-, and 3.0-inch break sizes.

1.1 Figures 1.1-1 through 1.1-3 show the core power for the entire length of the transient for the 1.5-, 2.0- and 3.0-inch breaks.

Attachment 1
Response to October 28, 2005 Request for Additional Information

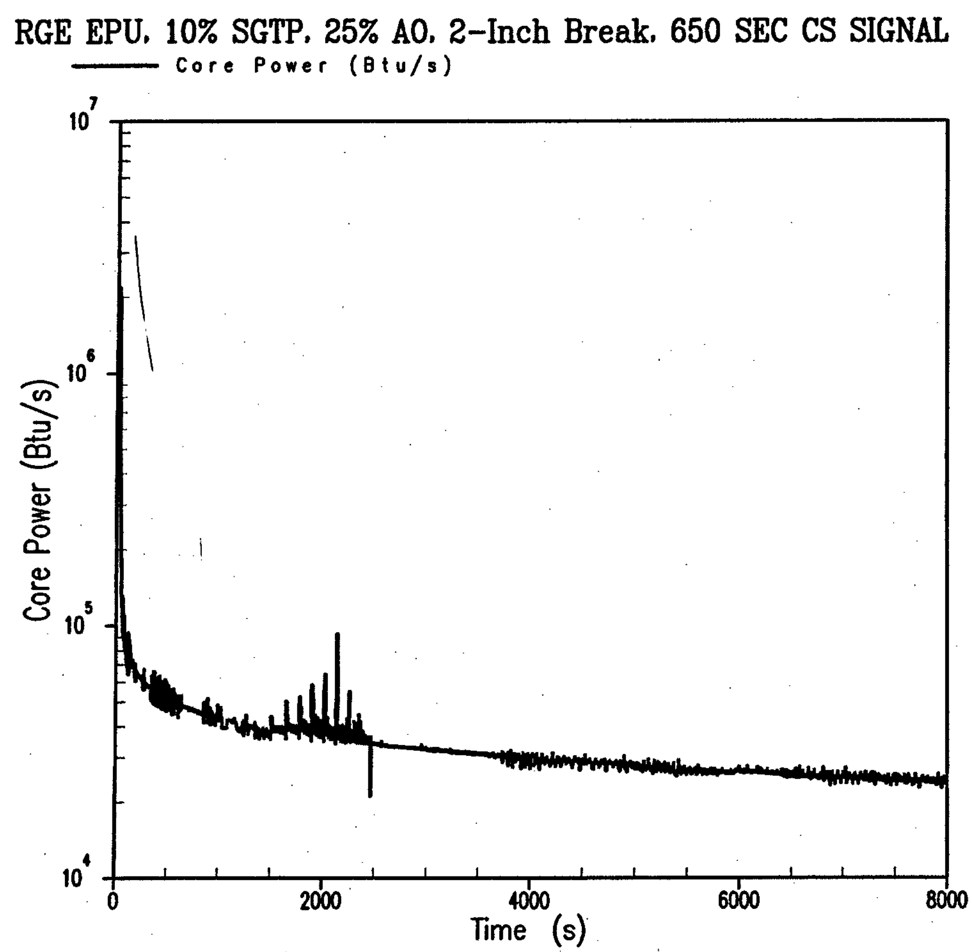
Figure 1.1-1: 1.5-Inch Core Power (Btu/s)



Attachment 1

Response to October 28, 2005 Request for Additional Information

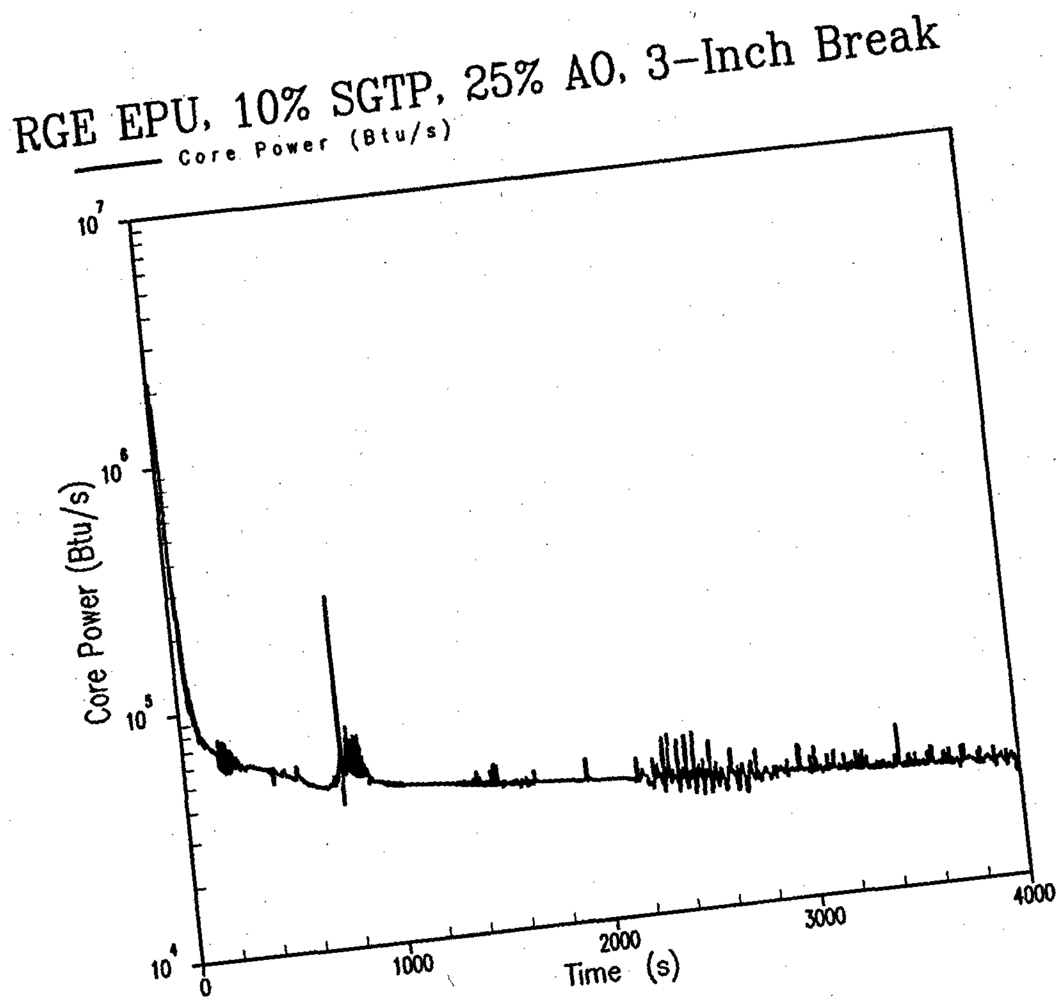
Figure 1.1-2: 2.0-Inch Core Power (Btu/s)



Attachment 1

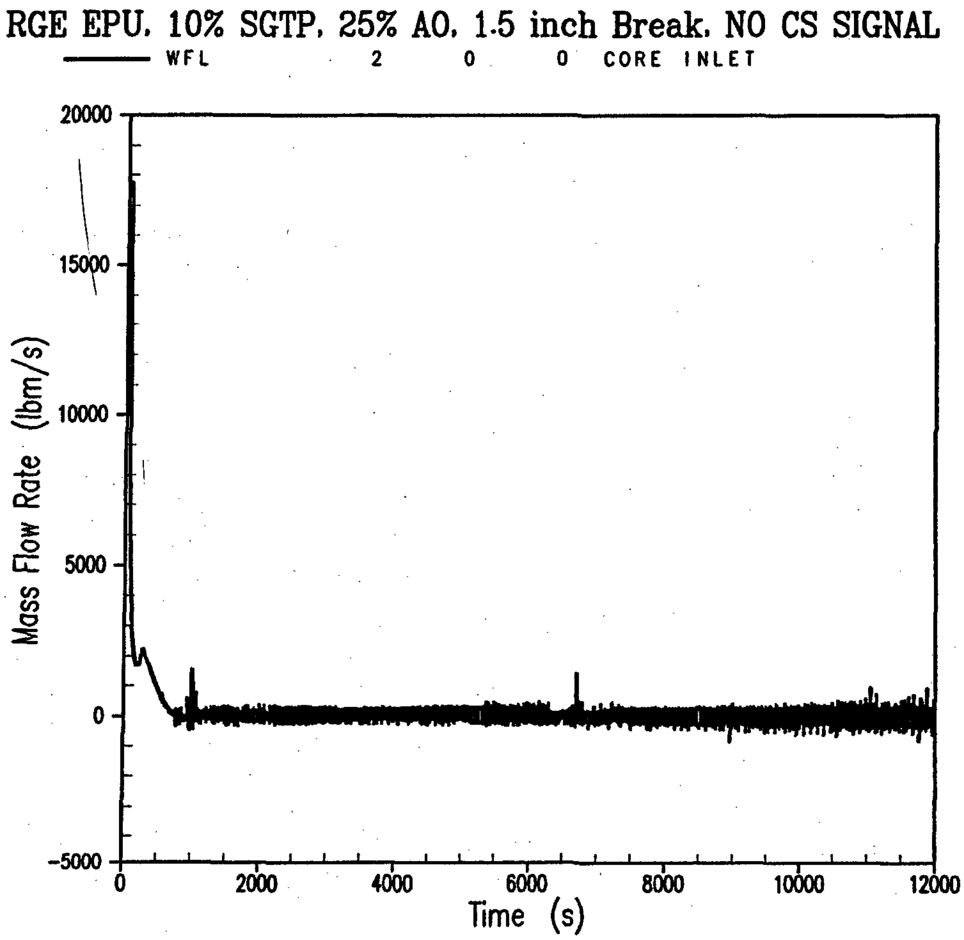
Response to October 28, 2005 Request for Additional Information

Figure 1.1-3: 3.0-Inch Core Power (Btu/s)

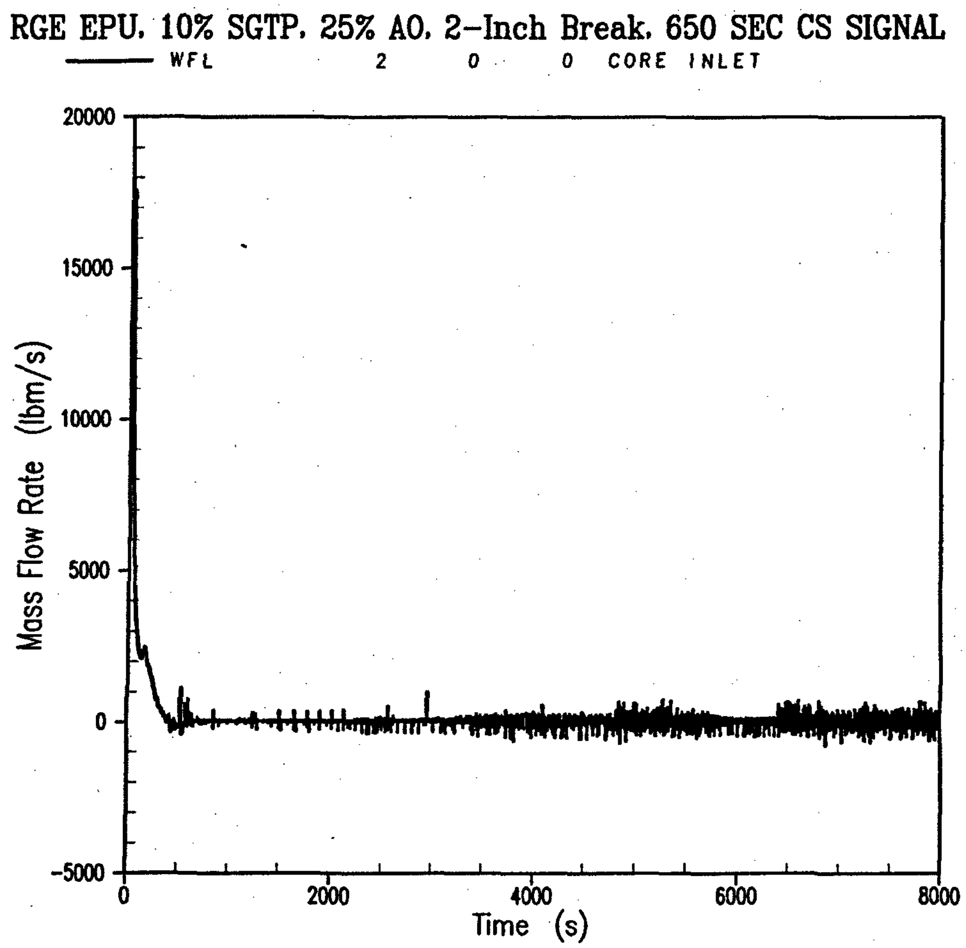


1.2 Figures 1.2-1 through 1.2-3 show the core inlet mass flowrate for the 1.5-, 2.0- and 3.0-inch cases.

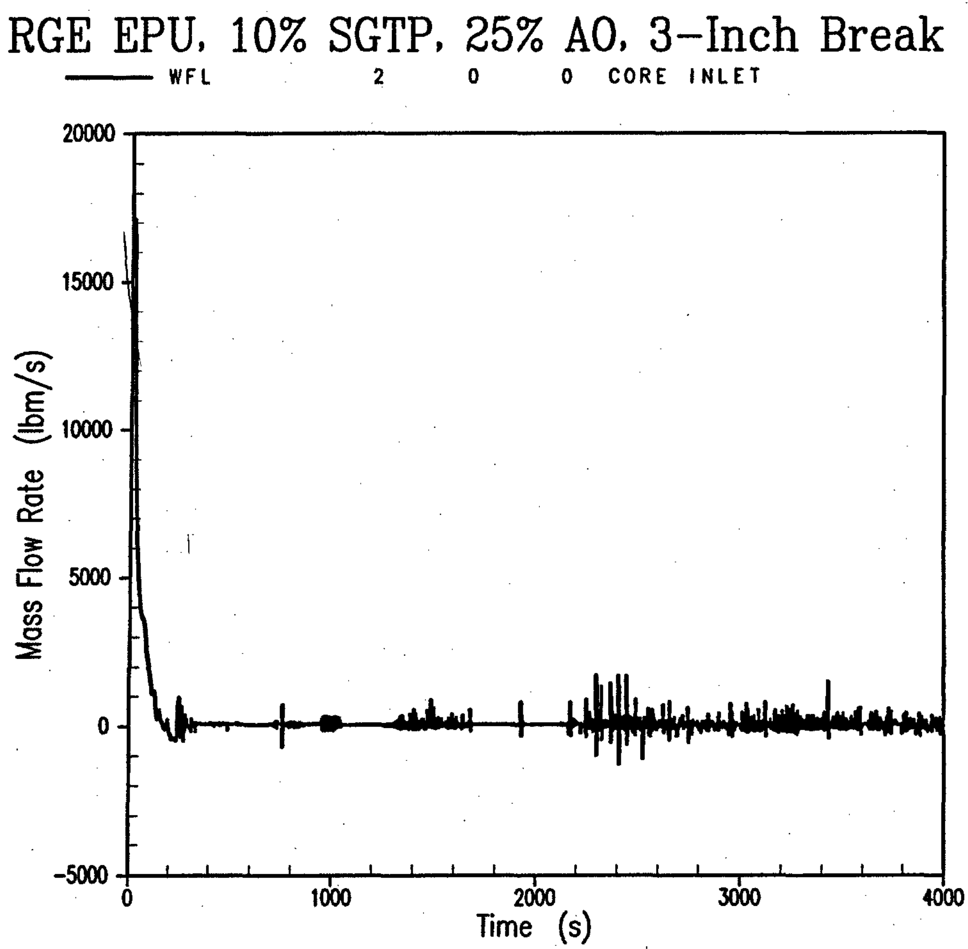
Figure 1.2-1: 1.5-Inch Core Inlet Mass Flowrate



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.2-2: 2.0-Inch Core Inlet Mass Flowrate



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.2-3: 3.0-Inch Core Inlet Mass Flowrate



1.3 Figures 1.3-1 though 1.3-3 address the request for break mass flowrate for the 1.5-, 2.0- and 3.0-inch transients.

Figure 1.3-1: 1.5-Inch Break Mass Flowrate

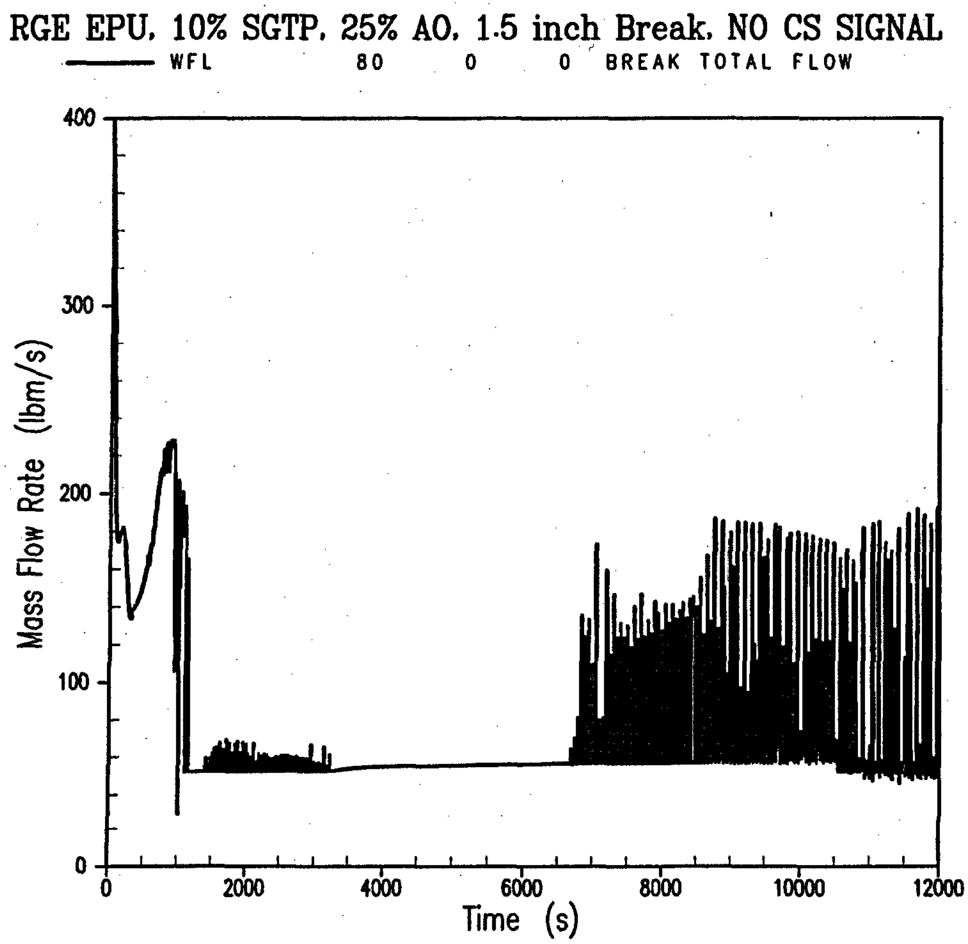


Figure 1.3-2: 2.0-Inch Break Mass Flowrate

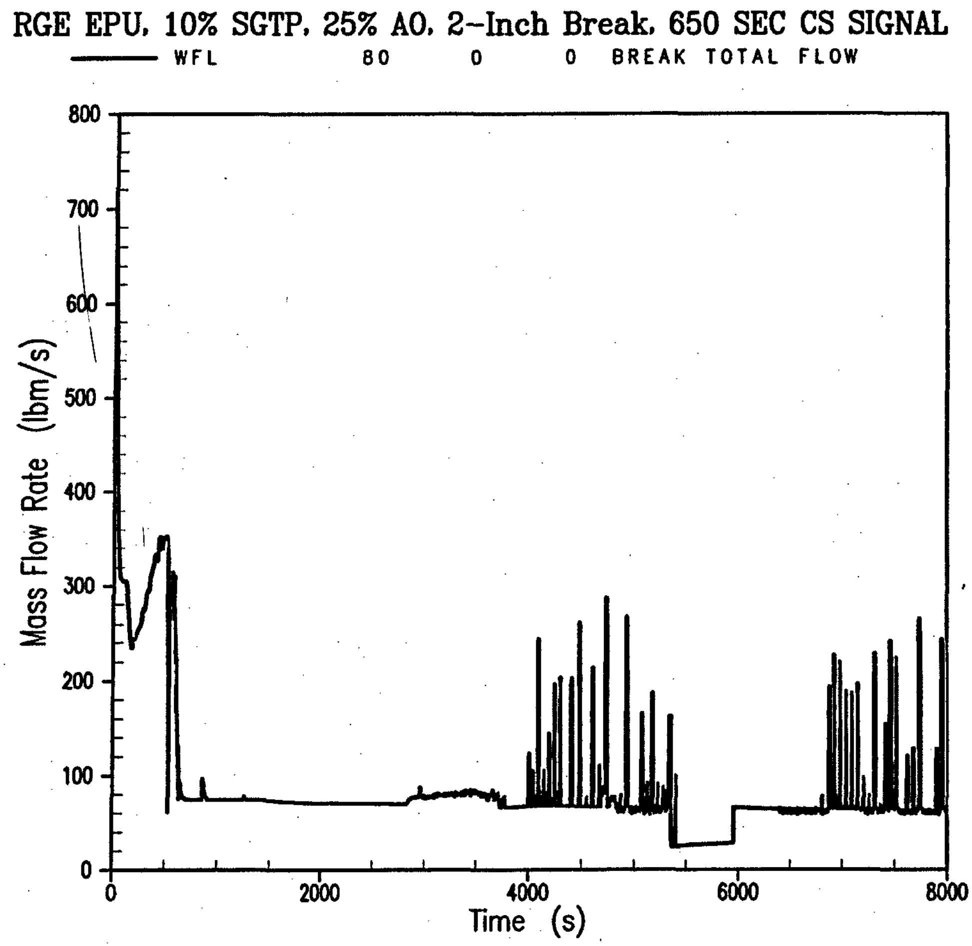
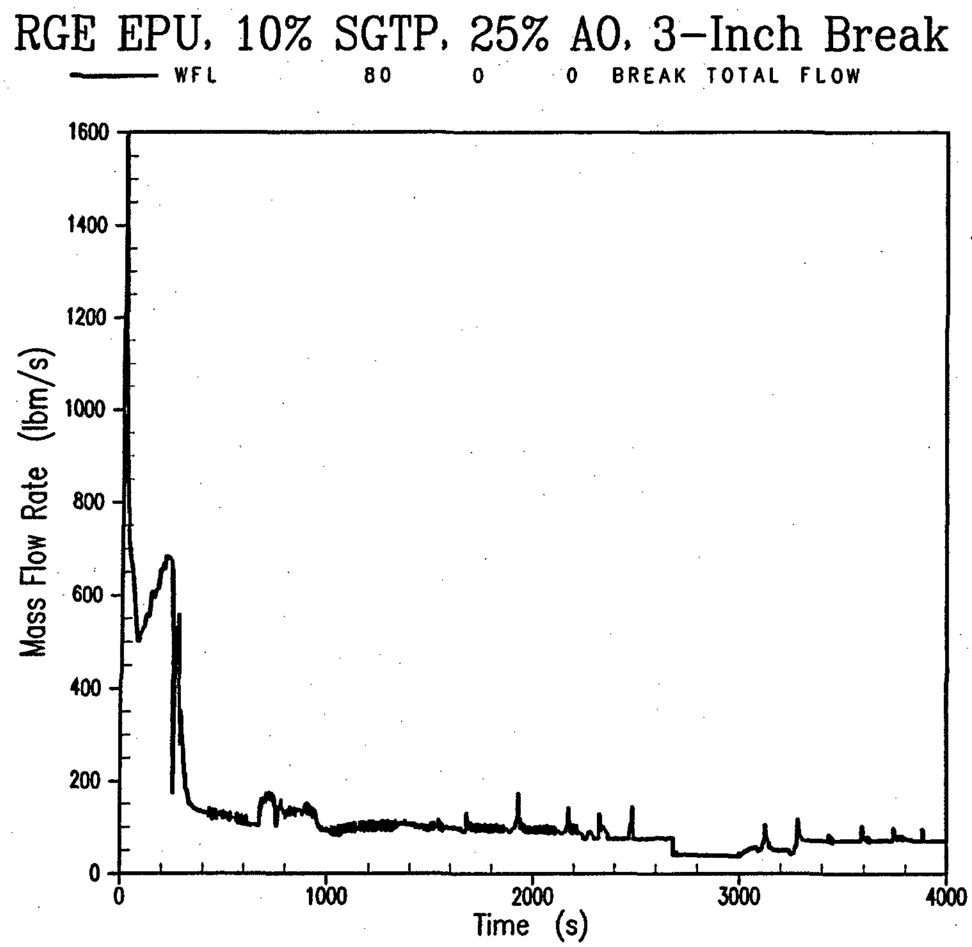


Figure 1.3-3: 3.0-Inch Break Mass Flowrate



1.4 Figures 1.4-1 through 1.4-3 show the break quality for the 1.5-, 2.0- and 3.0-inch break cases.

Figure 1.4-1: 1.5-Inch Break Quality

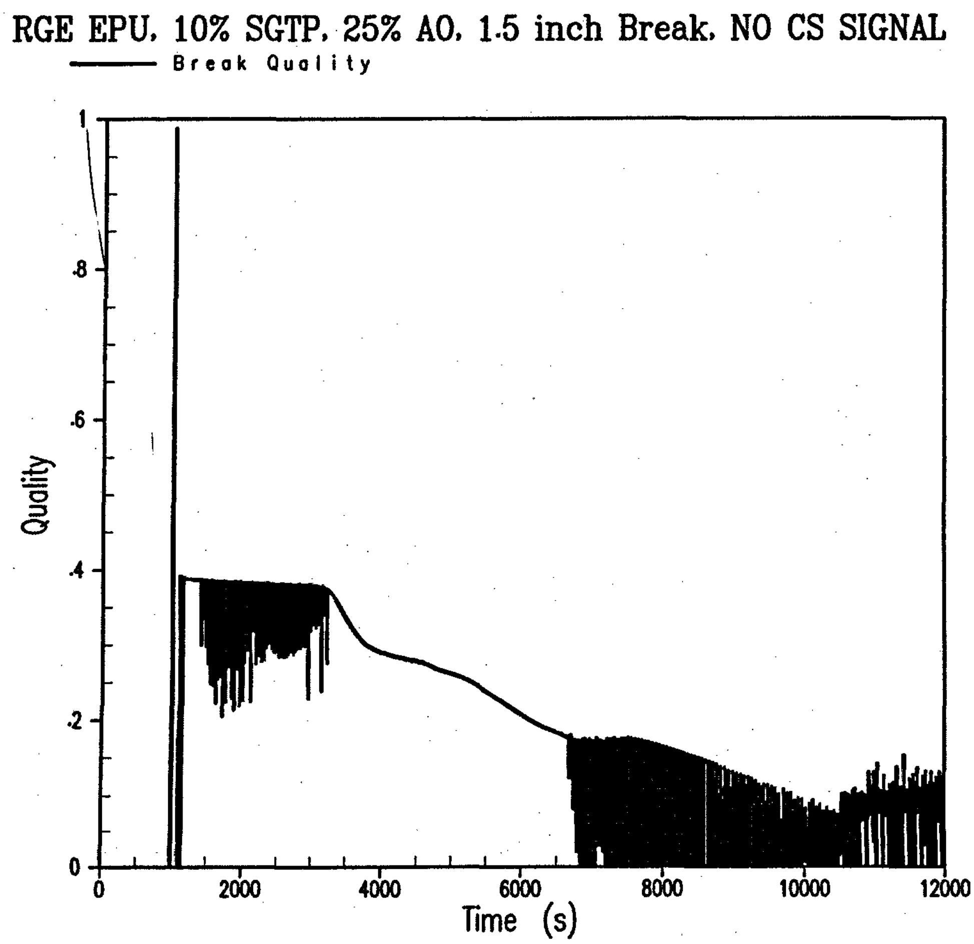
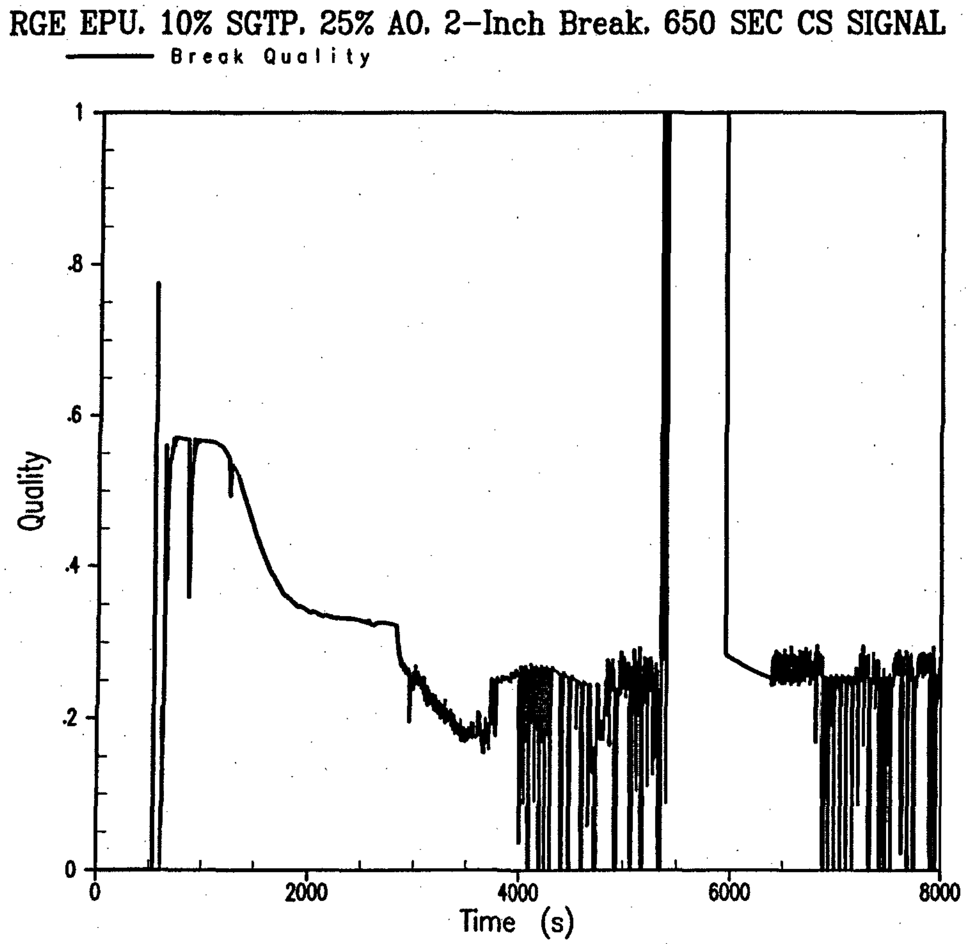
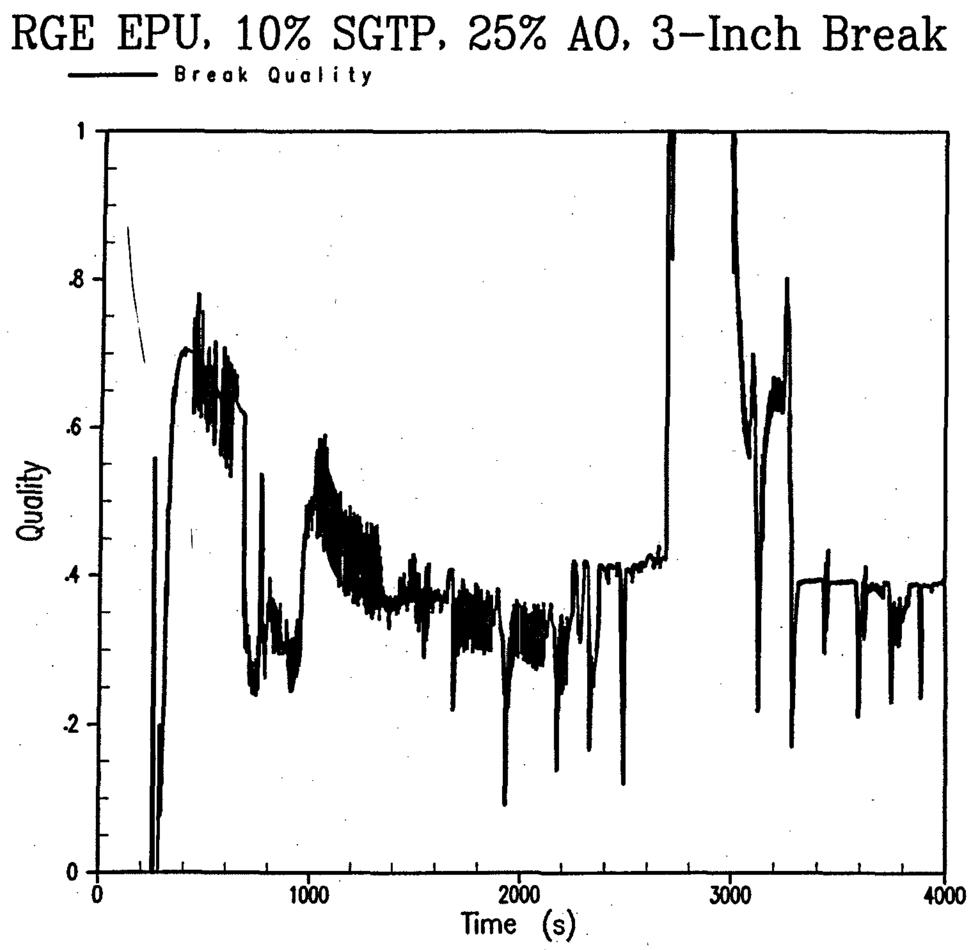


Figure 1.4-2: 2.0-Inch Break Quality

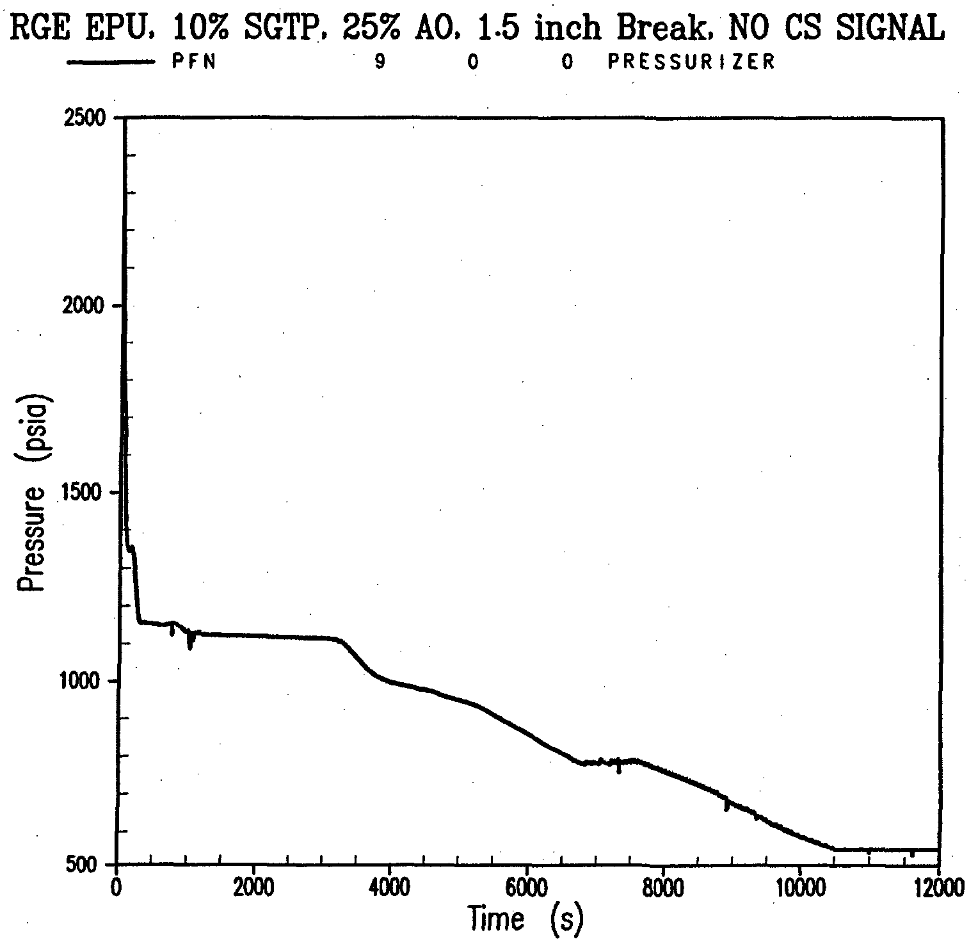


Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.4-3: 3.0-Inch Break Quality

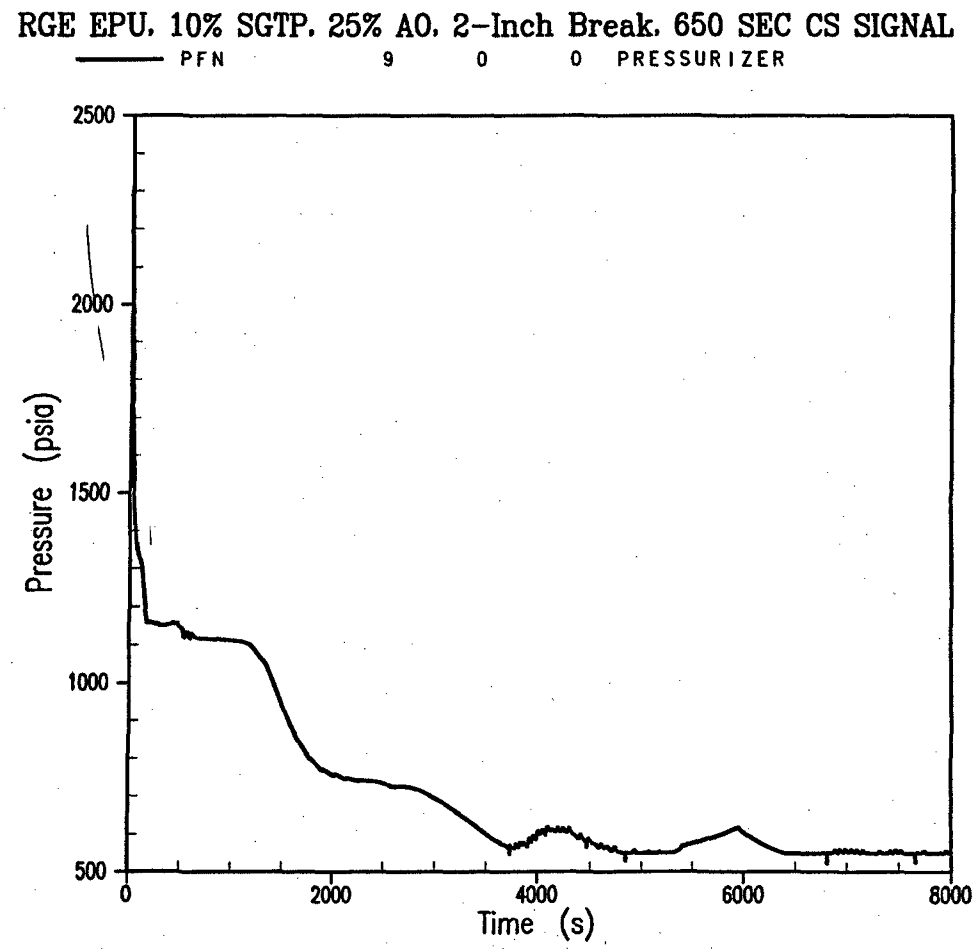


Figures 1.5-1 though 1.5-3 show the pressurizer pressure transient for the 1.5-, 2.0- and 3.0-inch breaks.

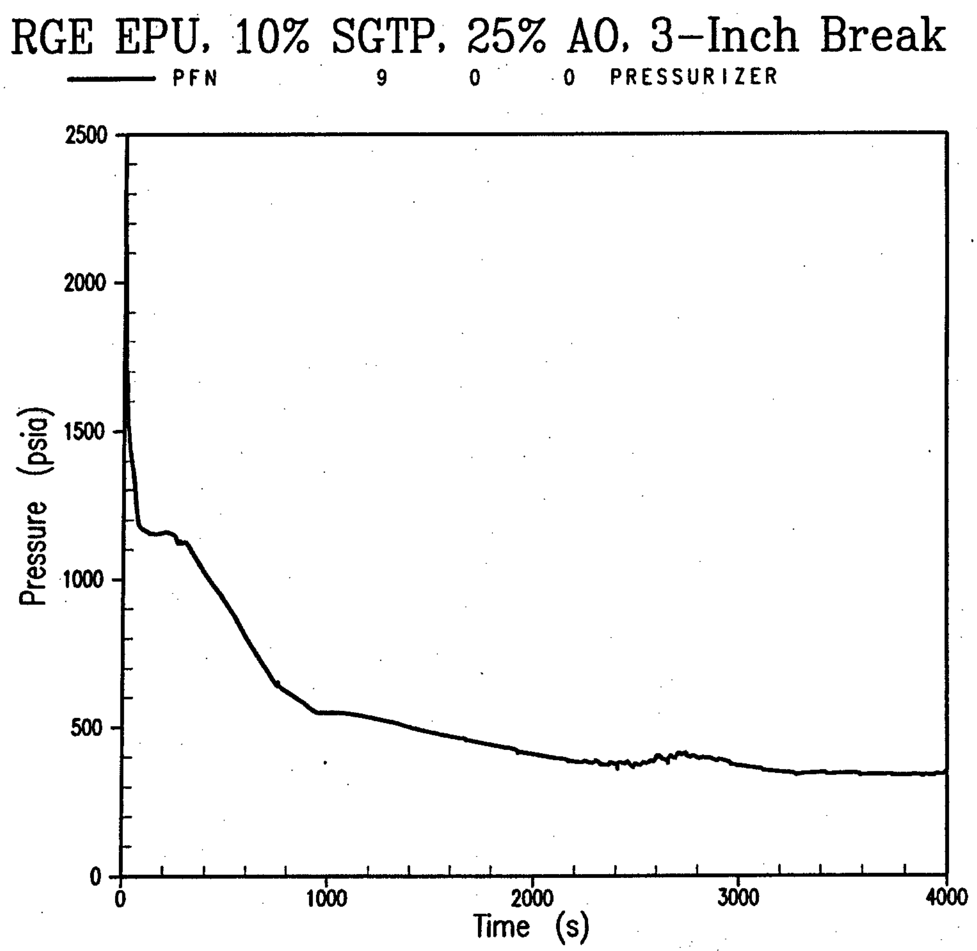
Figure 1.5-1: 1.5-Inch Pressurizer Pressure



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.5-2: 2.0-Inch Pressurizer Pressure



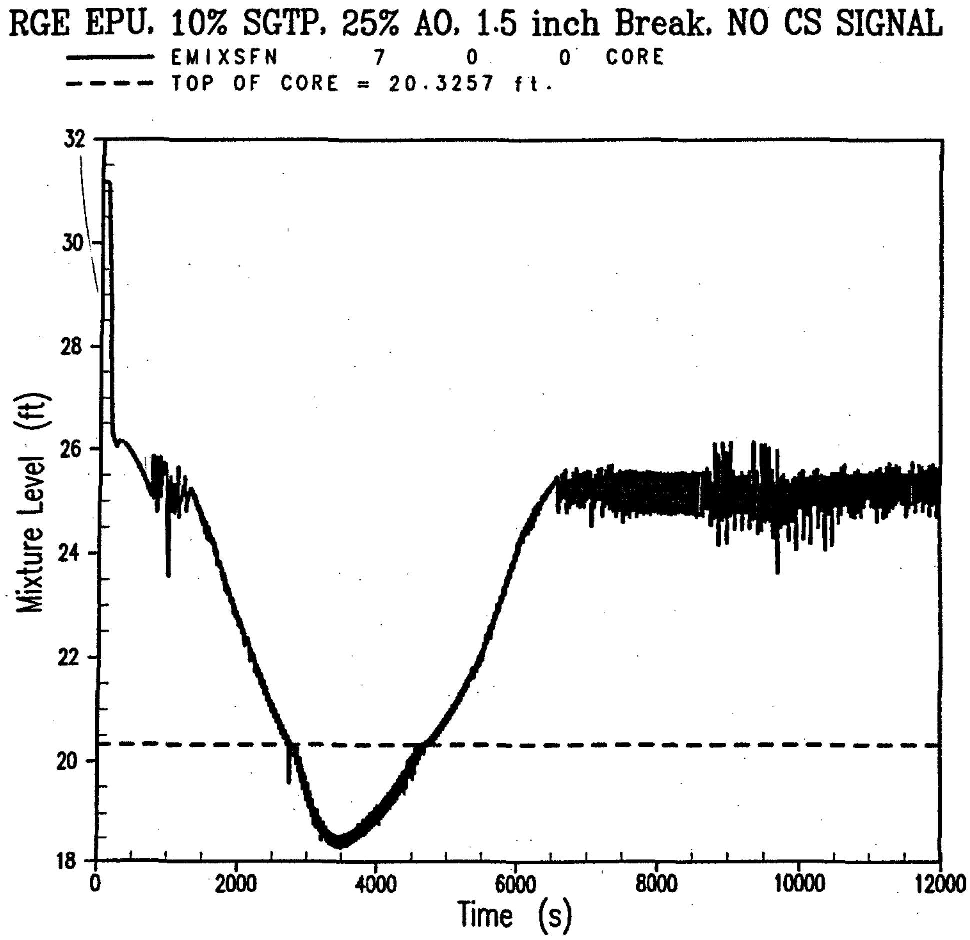
Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.5-3: 3.0-Inch Pressurizer Pressure



Attachment 1
Response to October 28, 2005 Request for Additional Information

1.6 Figures 1.6-1 through 1.6-3 show the core mixture level for the 1.5-, 2.0- and 3.0-inch breaks.

Figure 1.6-1: 1.5-Inch Core Mixture Level



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.6-2: 2.0-Inch Core Mixture Level

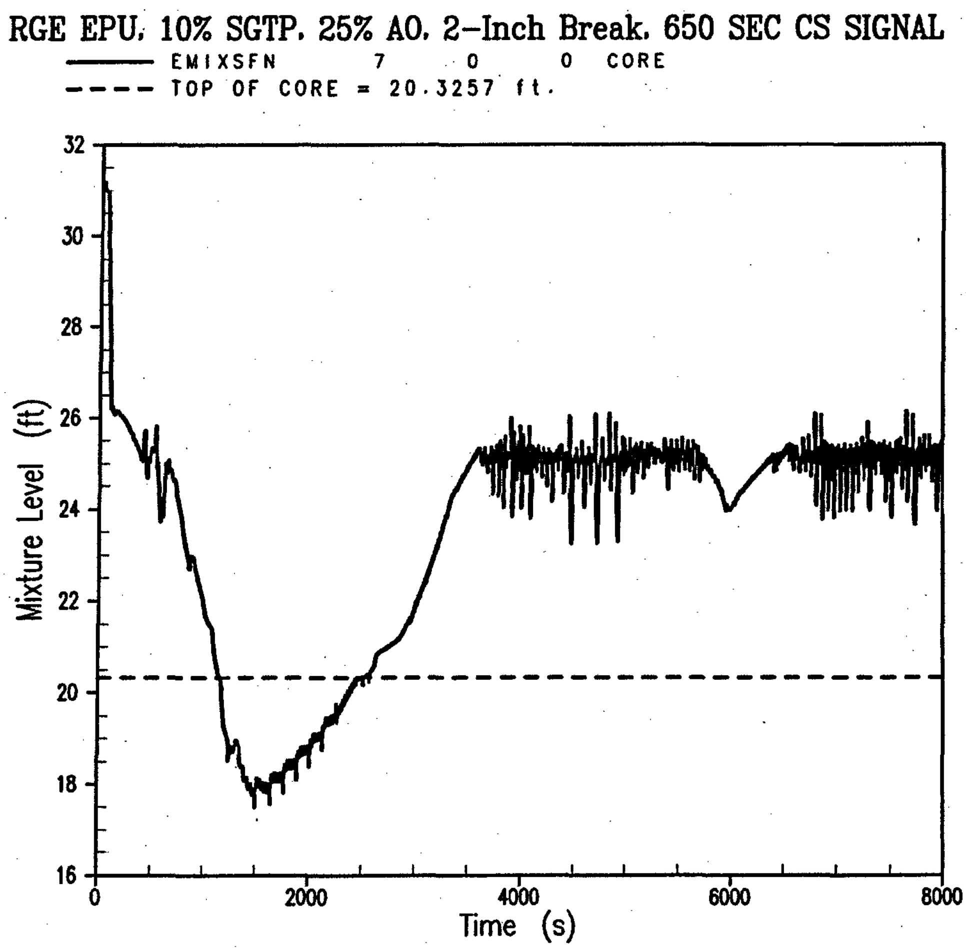
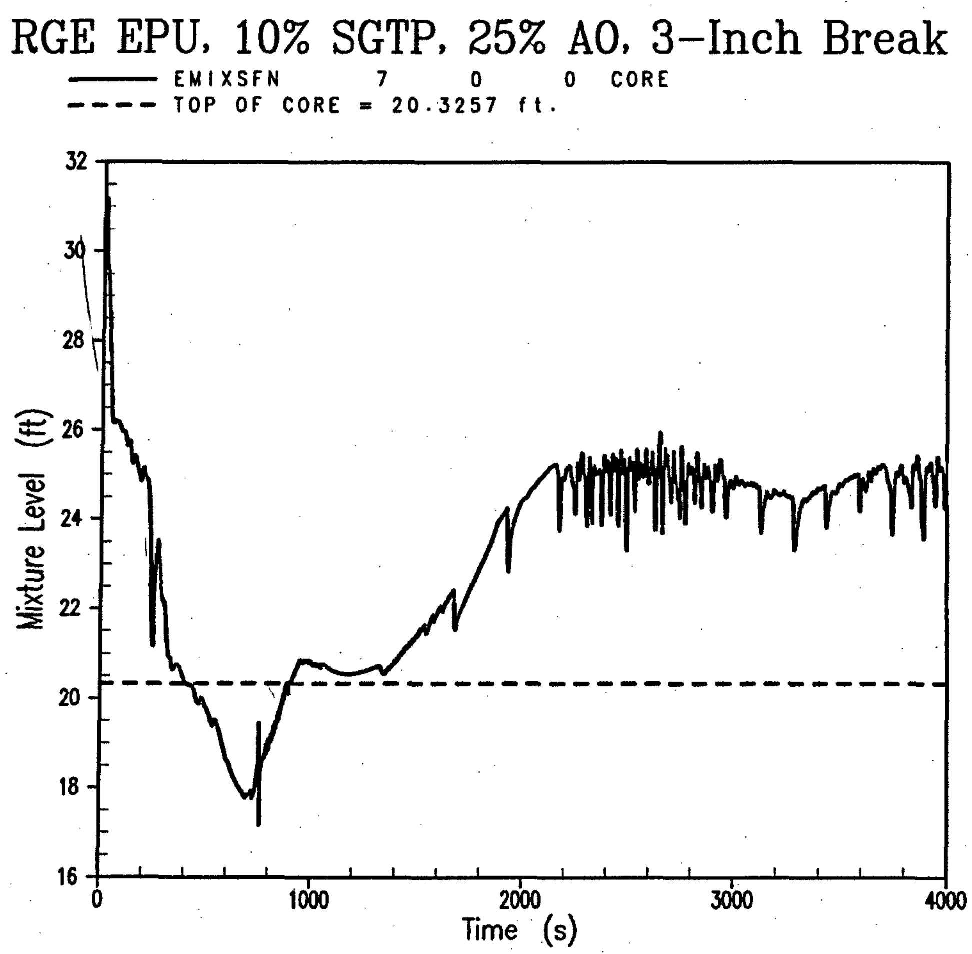
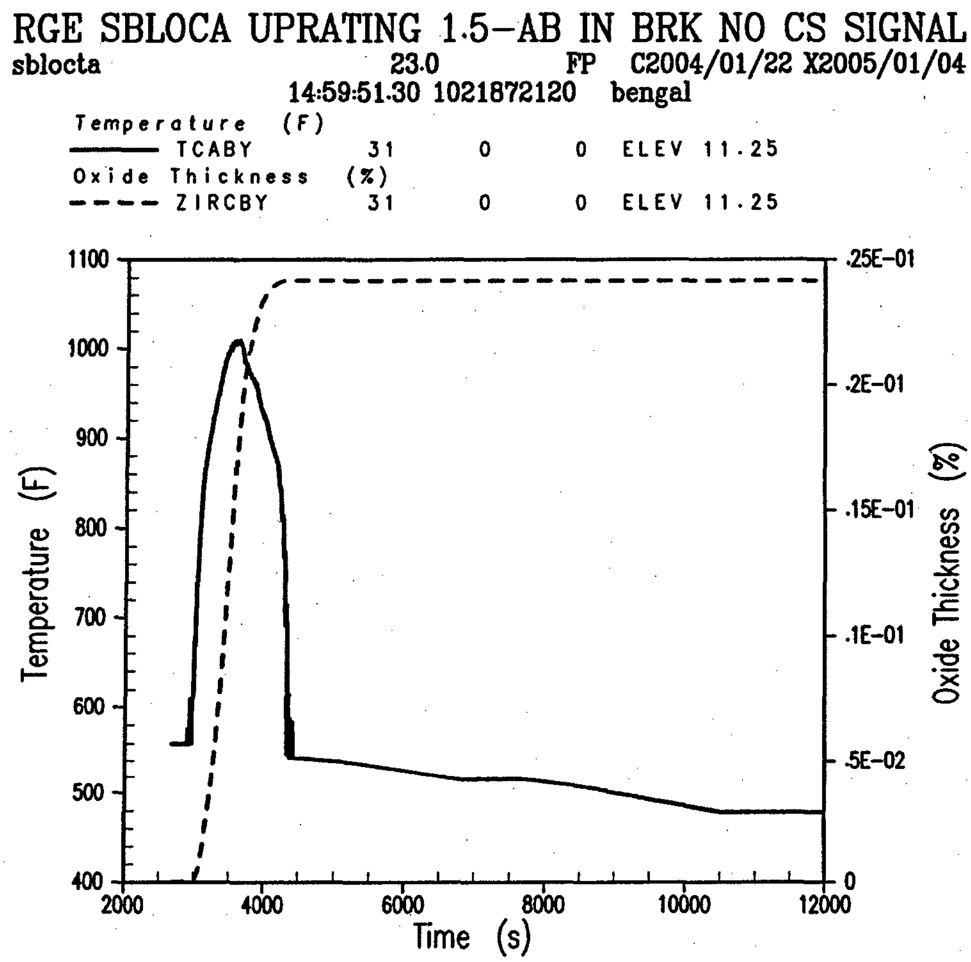


Figure 1.6-3: 3.0-Inch Core Mixture Level



1.7 Figures 1.7-1 though 1.7-3 show the clad temperature at the peak clad temperature location for the entire transient as well as the maximum local oxidation at the maximum local oxidation location for the 1.5-, 2.0- and 3.0-inch break sizes.

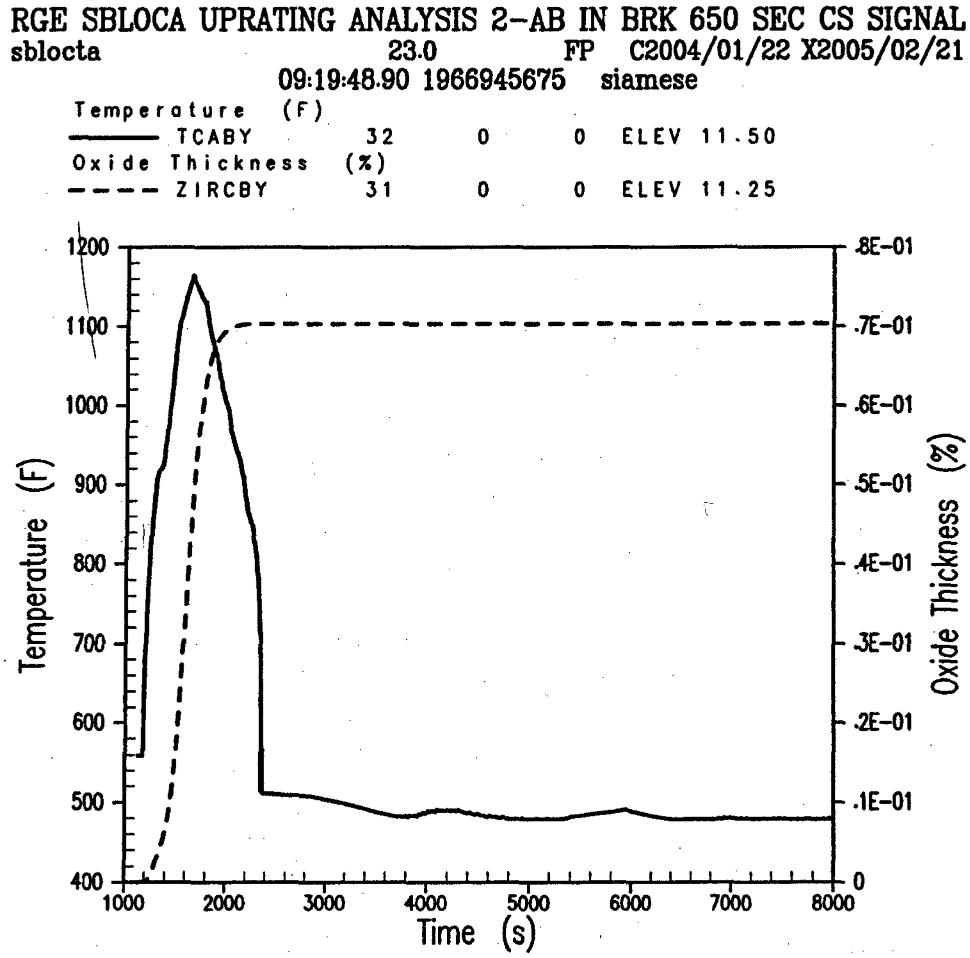
Figure 1.7-1: 1.5-Inch Clad Temperature at PCT Elevation and Maximum Local Oxidation



Attachment 1

Response to October 28, 2005 Request for Additional Information

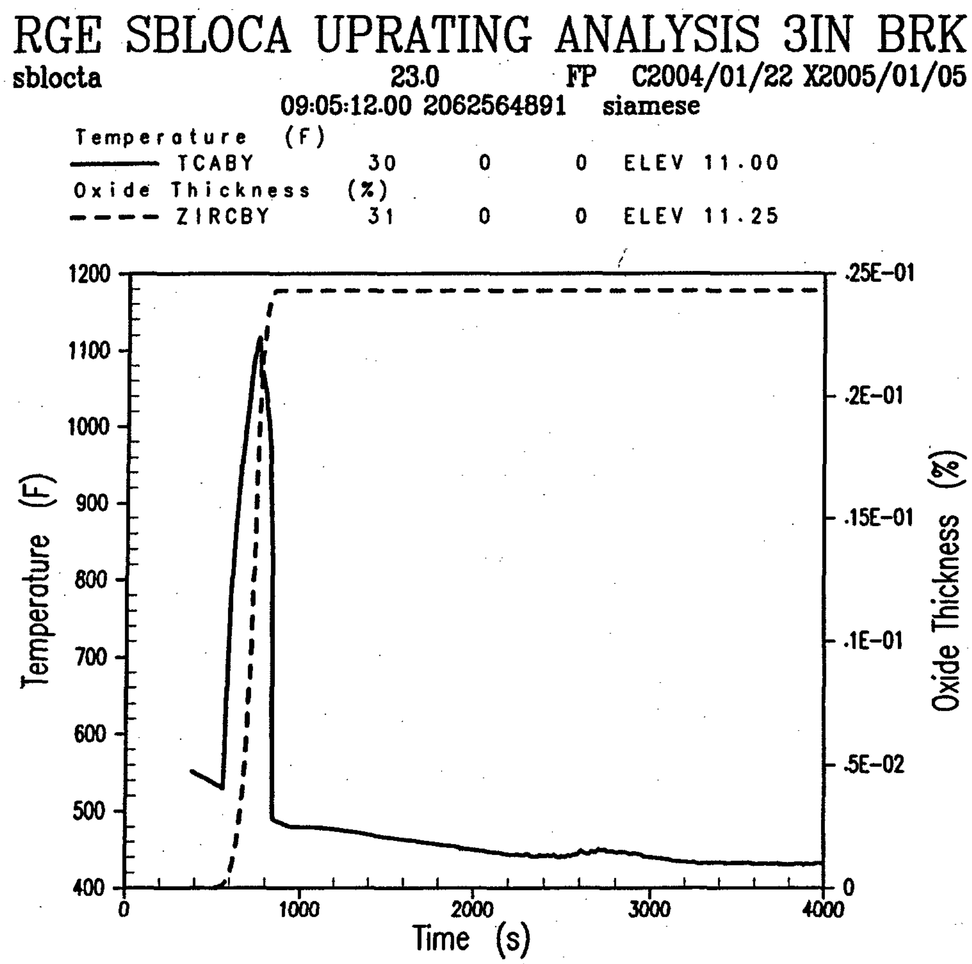
Figure 1.7-2: 2.0-Inch Clad Temperature at PCT Elevation and Maximum Local Oxidation



Attachment 1

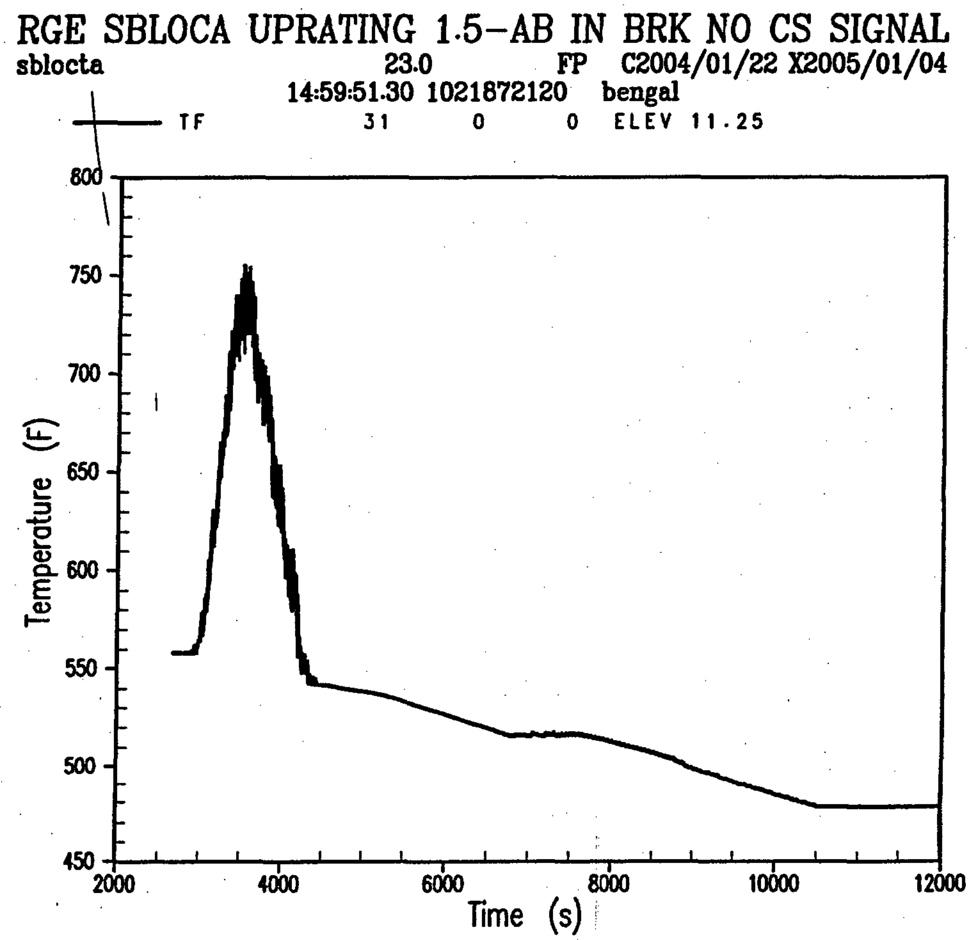
Response to October 28, 2005 Request for Additional Information

Figure 1.7-3: 3.0-Inch Clad Temperature at PCT Elevation and Maximum Local Oxidation

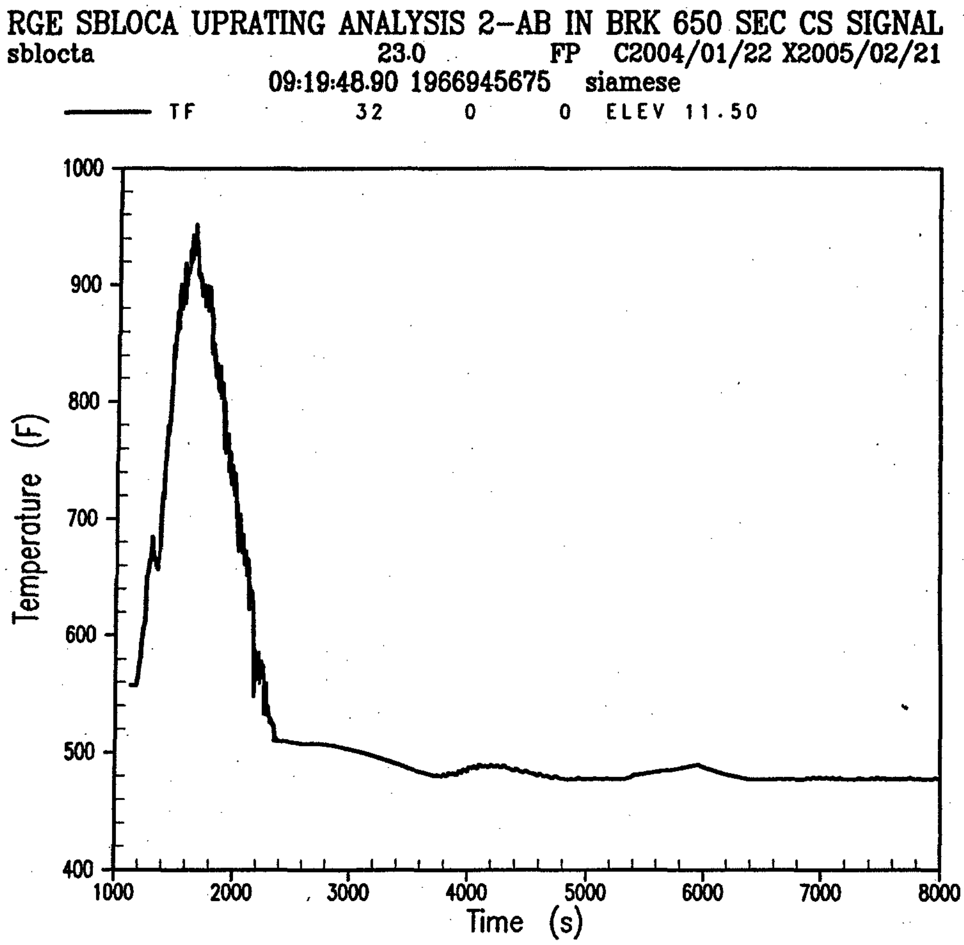


1.8 Figures 1.8-1 though 1.8-3 show the steam temperature at the hot spot for the 1.5-, 2.0- and 3.0-inch break cases.

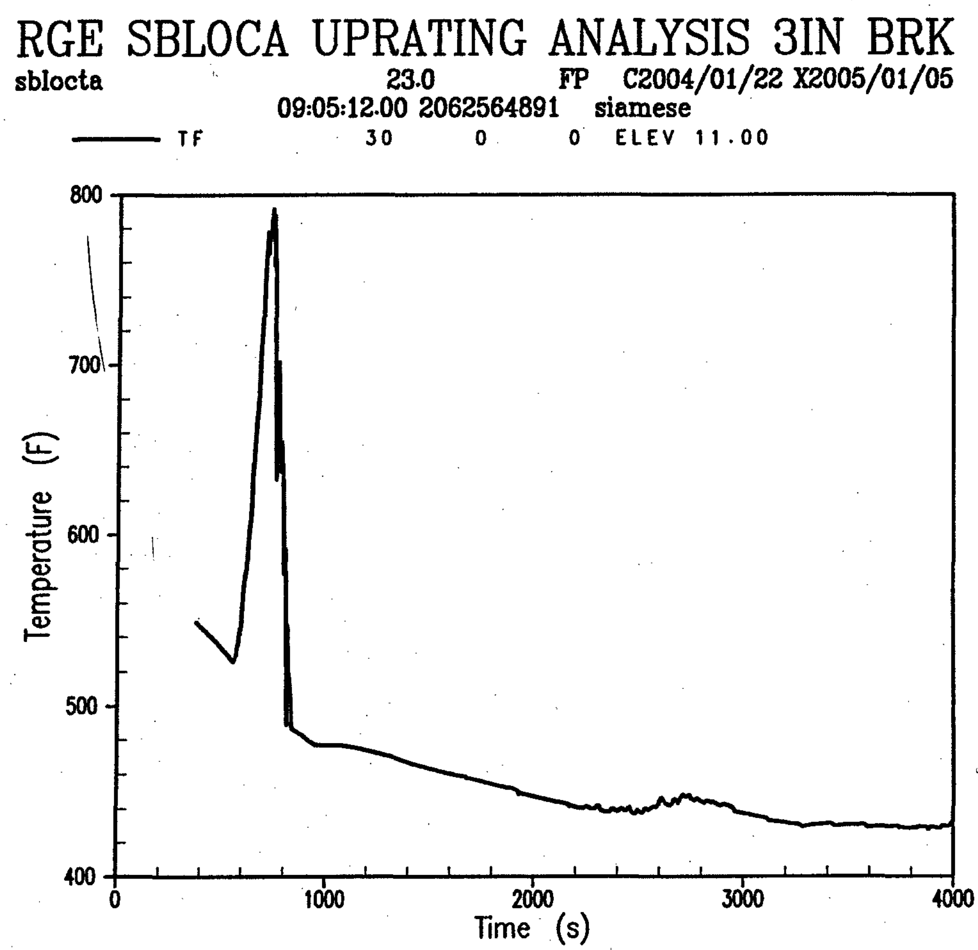
Figure 1.8-1: 1.5-Inch Steam Temperature at the Hot Spot



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.8-2: 2.0-Inch Steam Temperature at the Hot Spot

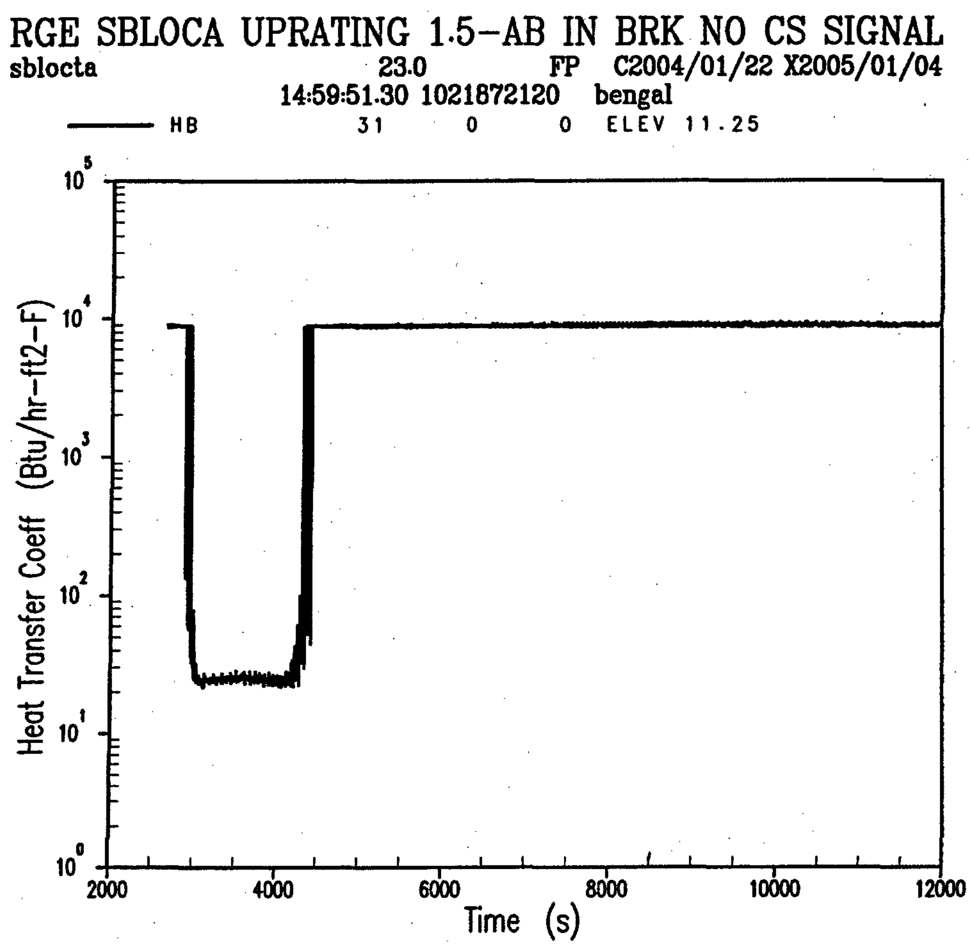


Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.8-3: 3.0-Inch Steam Temperature at the Hot Spot

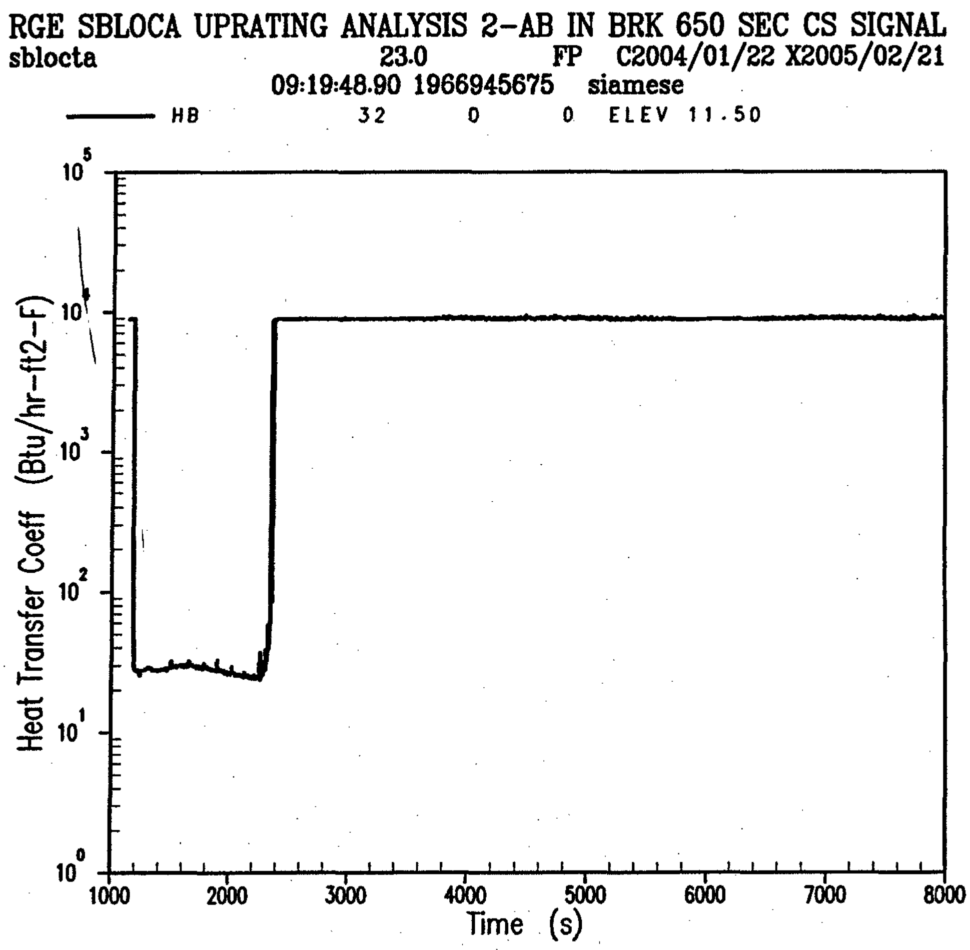


1.9 Figures 1.9-1 though 1.9-3 show the heat transfer coefficient at the hot spot (PCT location) for the 1.5-, 2.0- and 3.0-inch break cases.

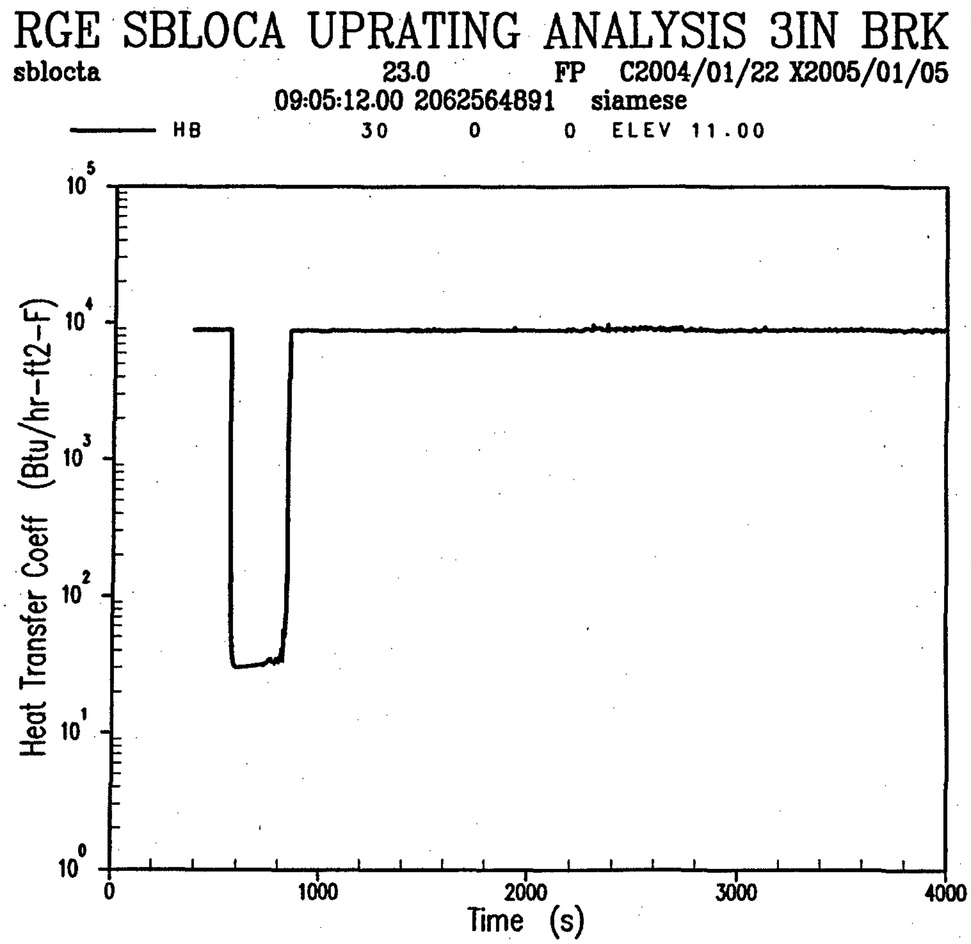
Figure 1.9-1: 1.5-Inch Heat Transfer Coefficient



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.9-2: 2.0-Inch Heat Transfer Coefficient

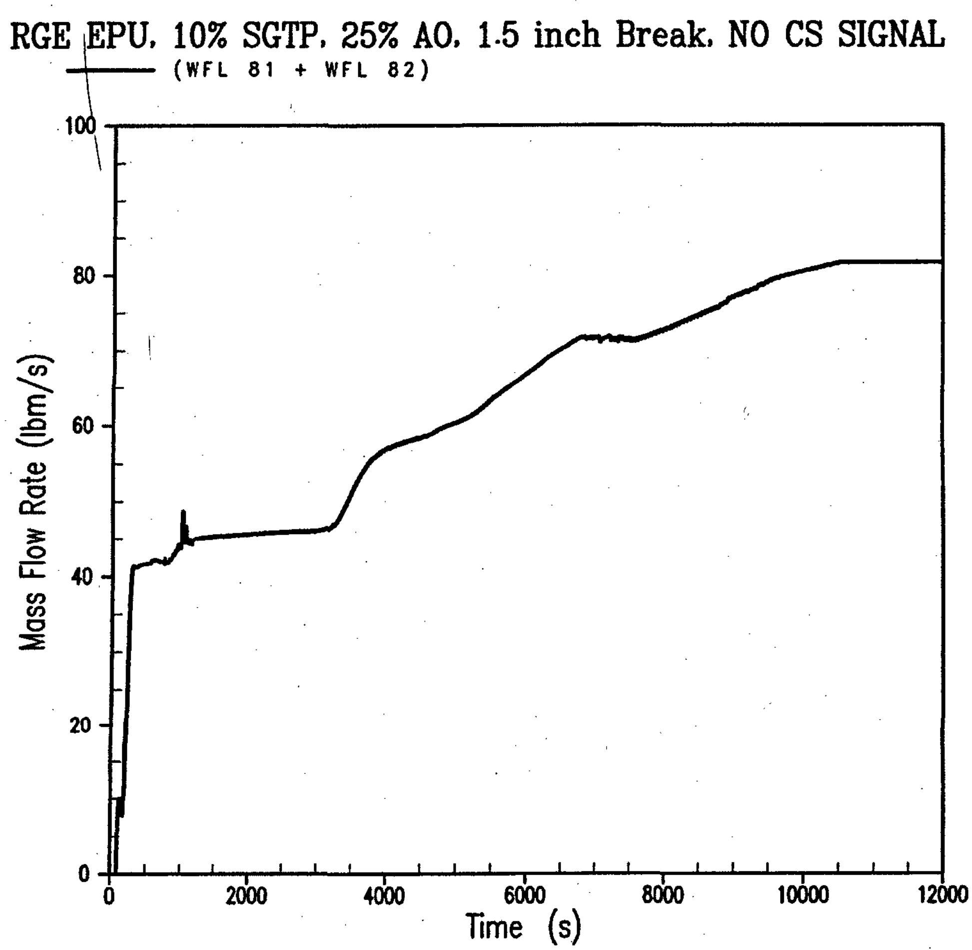


Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.9-3: 3.0-Inch Heat Transfer Coefficient

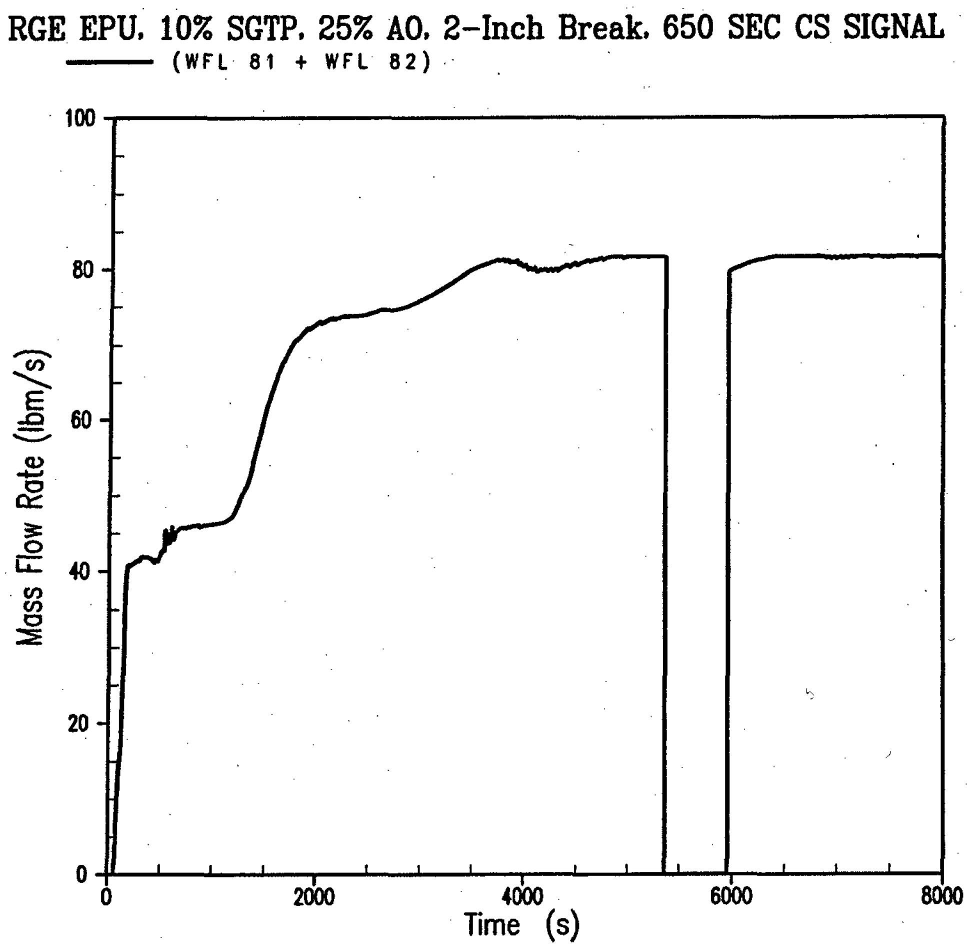


1.10 Figures 1.10-1 through 1.10-3 address the request for the safety injection (SI) mass flow rate versus time for the 1.5-, 2.0- and 3.0-inch break transients. Figures 1.10-4 through 1.10-6 address the request for accumulator injection mass flow rate versus time for the 1.5-, 2.0- and 3.0-inch break transients.

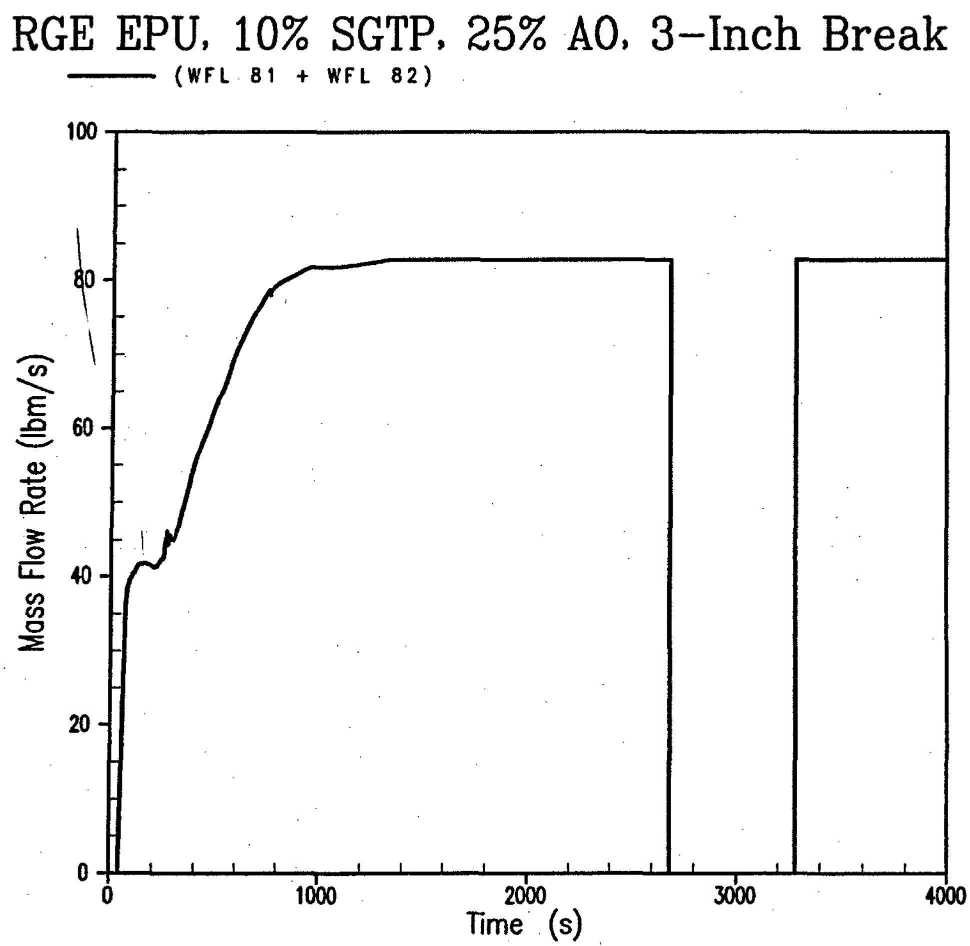
Figure 1.10-1: 1.5-Inch SI Mass Flow Rate



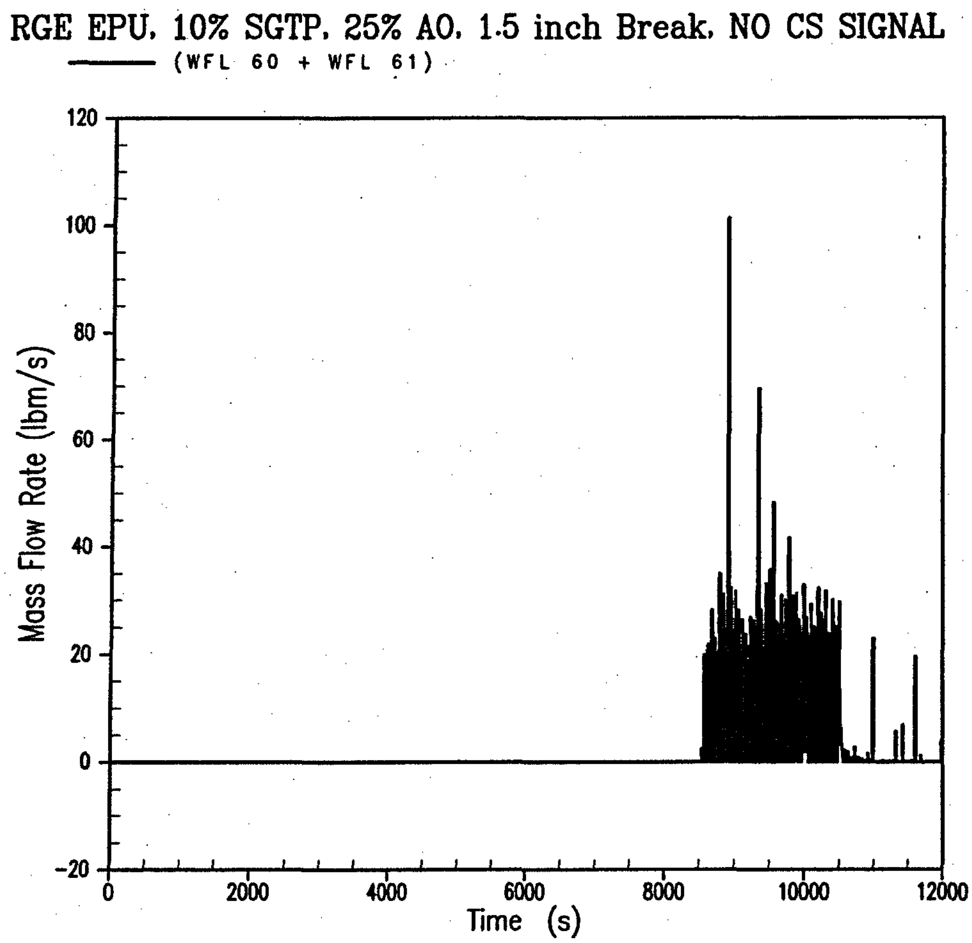
Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.10-2: 2.0-Inch SI Mass Flow Rate



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.10-3: 3.0-Inch SI Mass Flow Rate



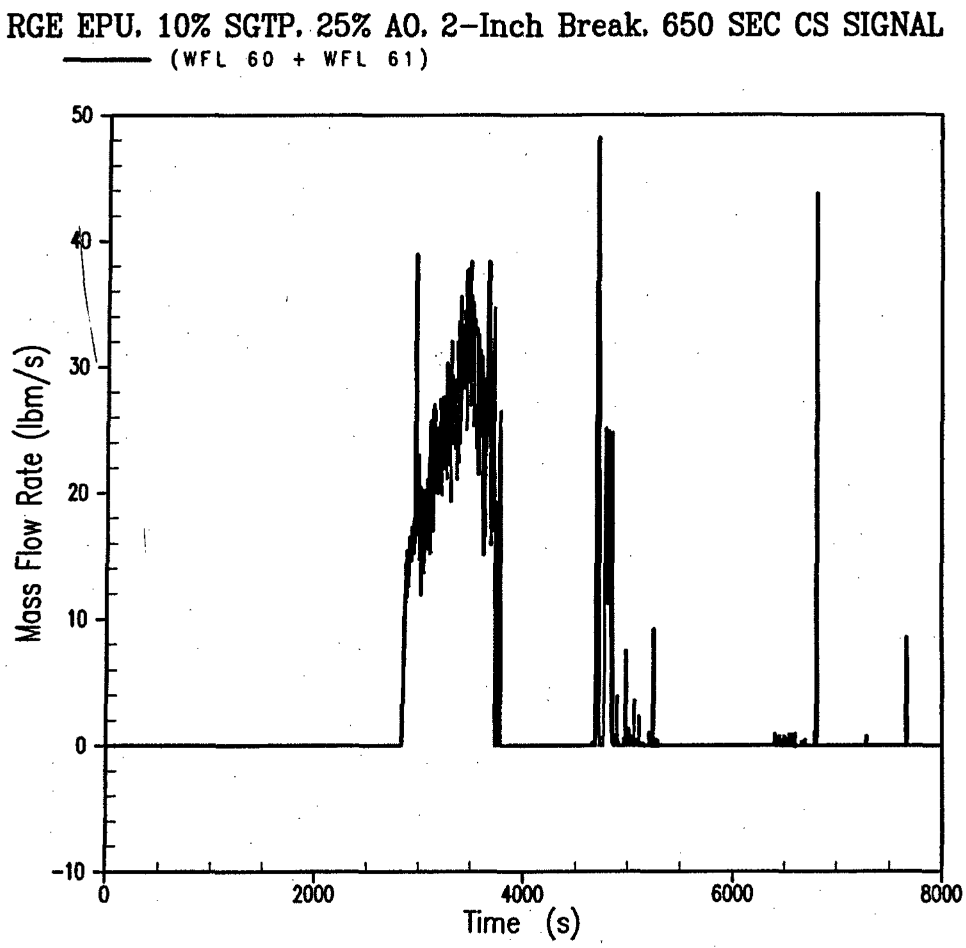
Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.10-4: 1.5-Inch Accumulator Injection Mass Flow Rate



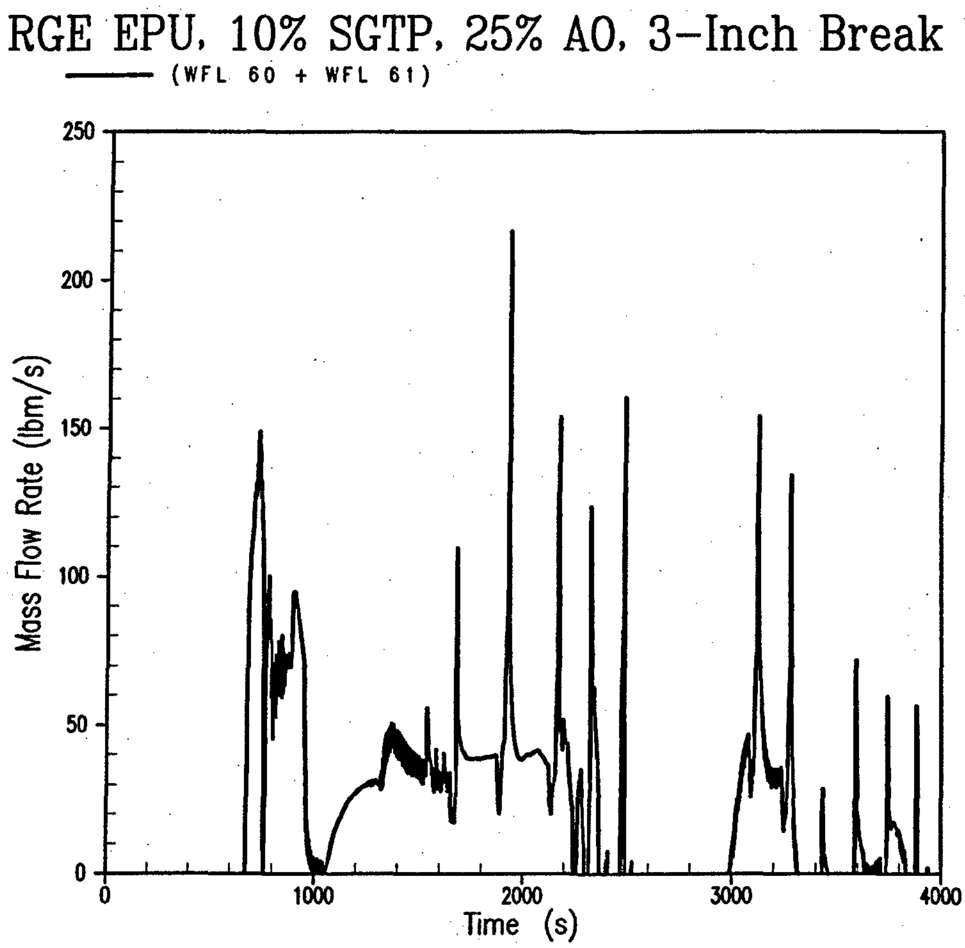
Attachment 1

Response to October 28, 2005 Request for Additional Information

Figure 1.10-5: 2.0-Inch Accumulator Injection Mass Flow Rate

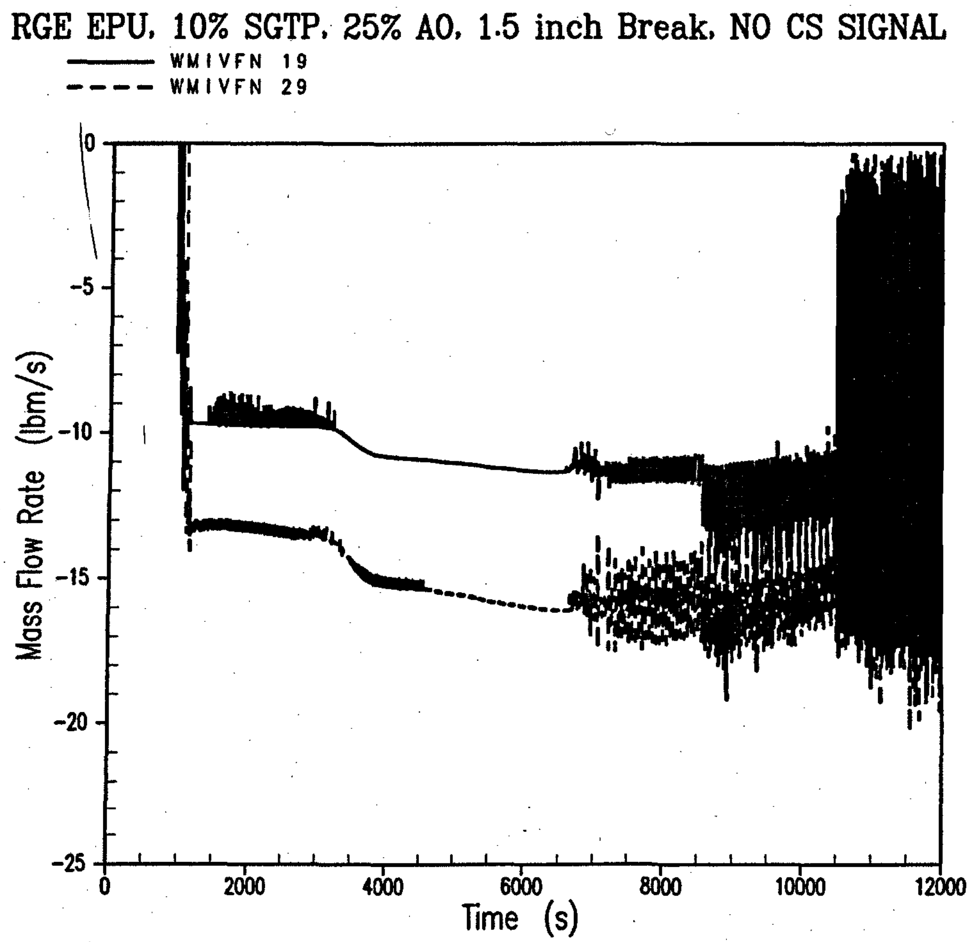


Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.10-6: 3.0-Inch Accumulator Injection Mass Flow Rate

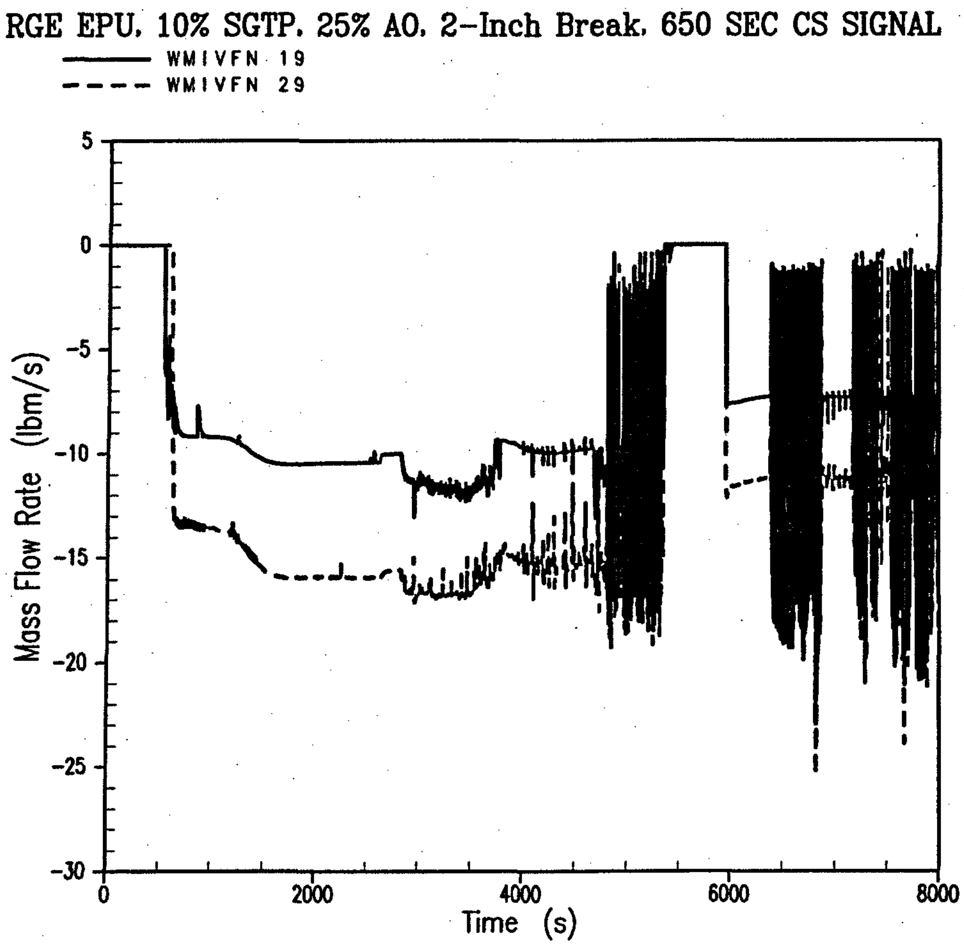


1.11 Figures 1.11-1 though 1.11-3 show the condensation rate in the cold legs for the 1.5-, 2.0- and 3.0-inch break cases.

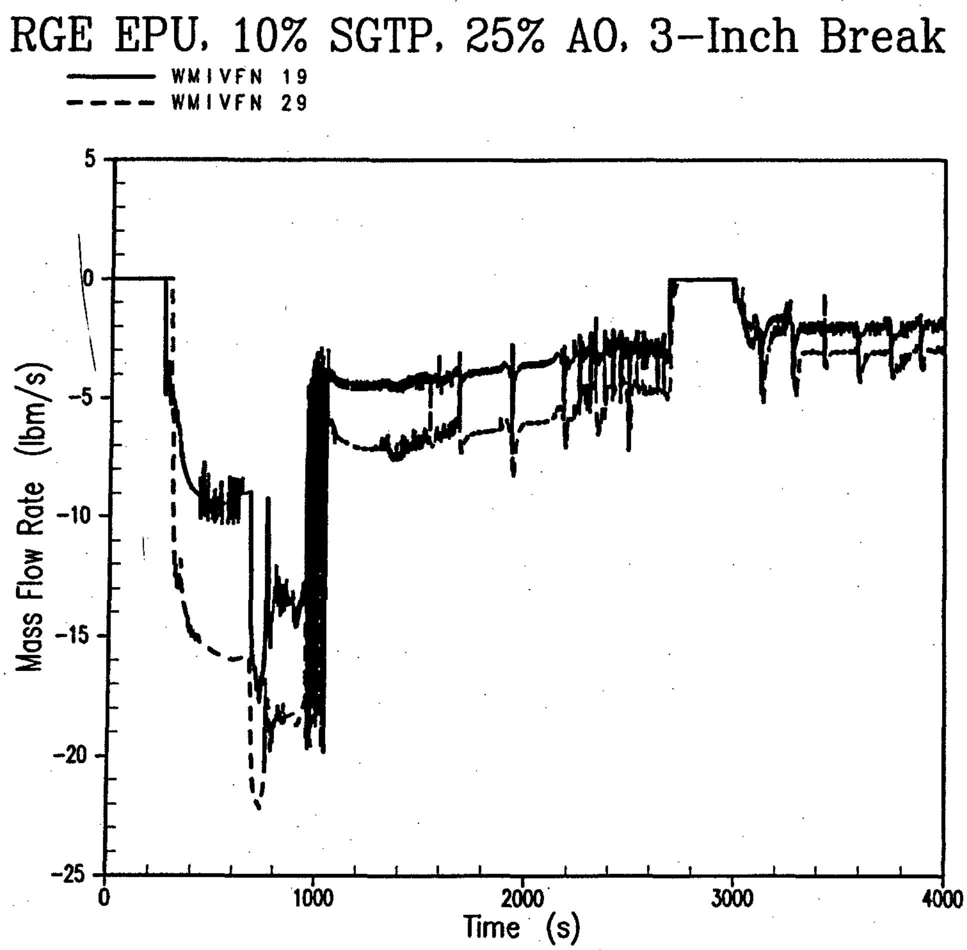
Figure 1.11-1: 1.5-Inch Condensation Rate in Cold Legs



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.11-2: 2.0-Inch Condensation Rate in Cold Legs

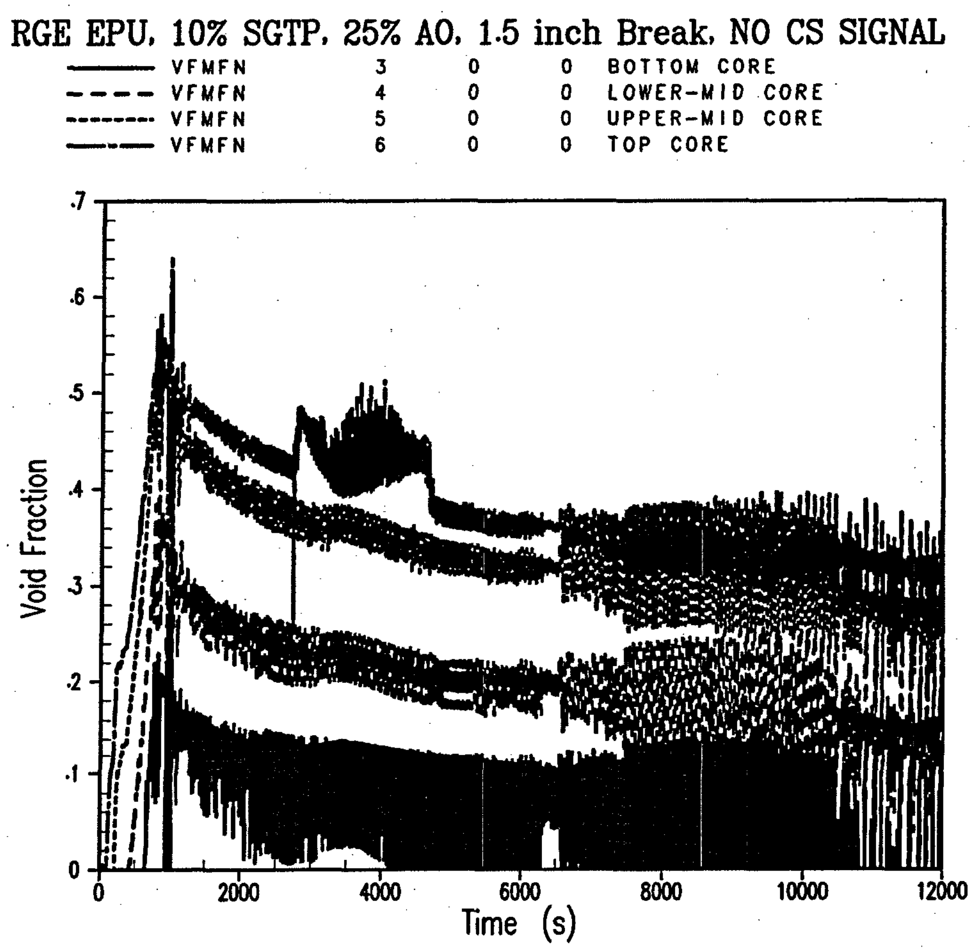


Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.11-3: 3.0-Inch Condensation Rate in Cold Legs

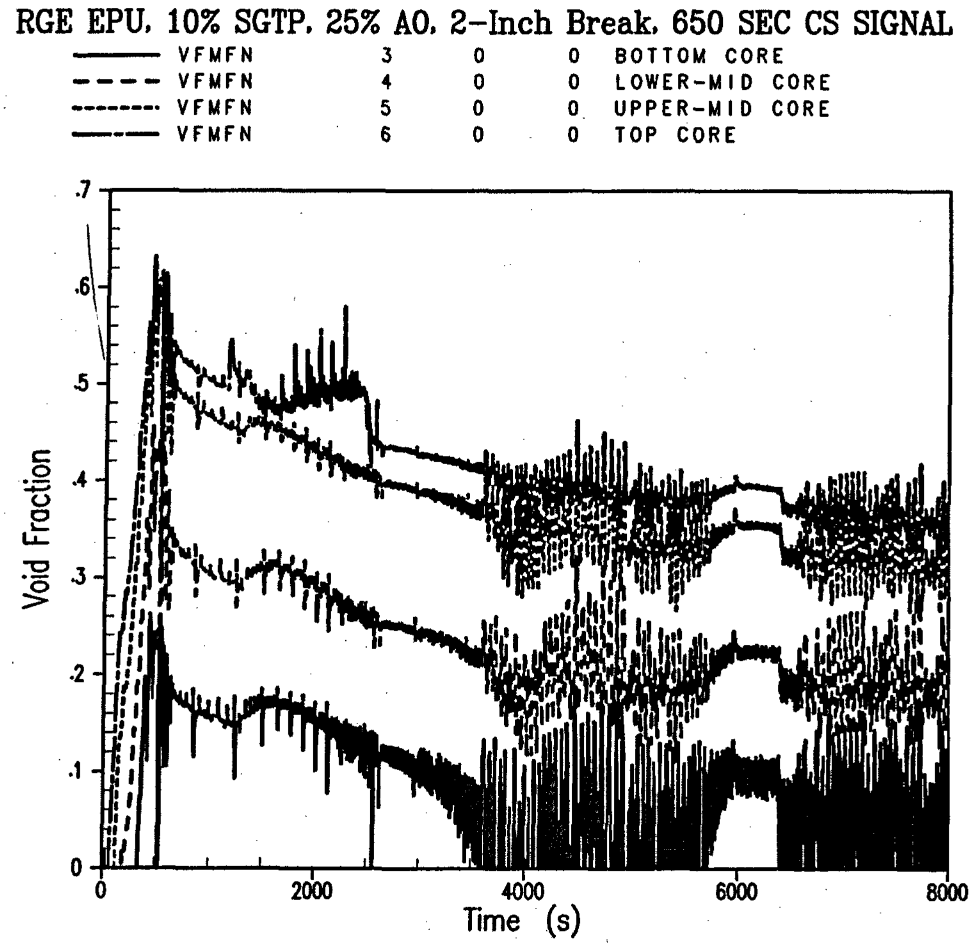


1.12 Figures 1.12-1 though 1.12-3 show the void fraction in each core node versus time for the 1.5-, 2.0- and 3.0-inch break transients.

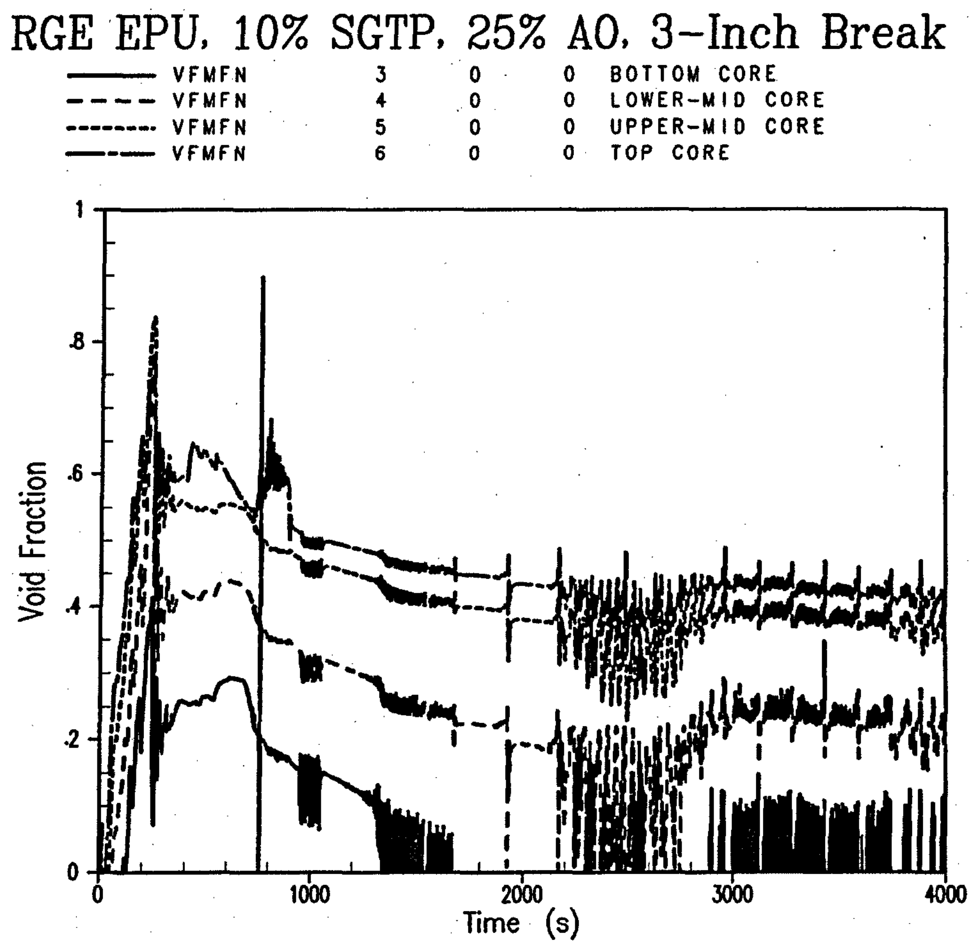
Figure 1.12-1: 1.5-Inch Void Fraction in Core Nodes



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.12-2: 2.0-Inch Void Fraction in Core Nodes



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 1.12-3: 3.0-Inch Void Fraction in Core Nodes



Attachment 1

Response to October 28, 2005 Request for Additional Information

RAI #2: What is the bottom elevation of the suction leg piping and the top elevation of the core? Also, provide the top elevation of the cold-leg discharge pipe.

Response:

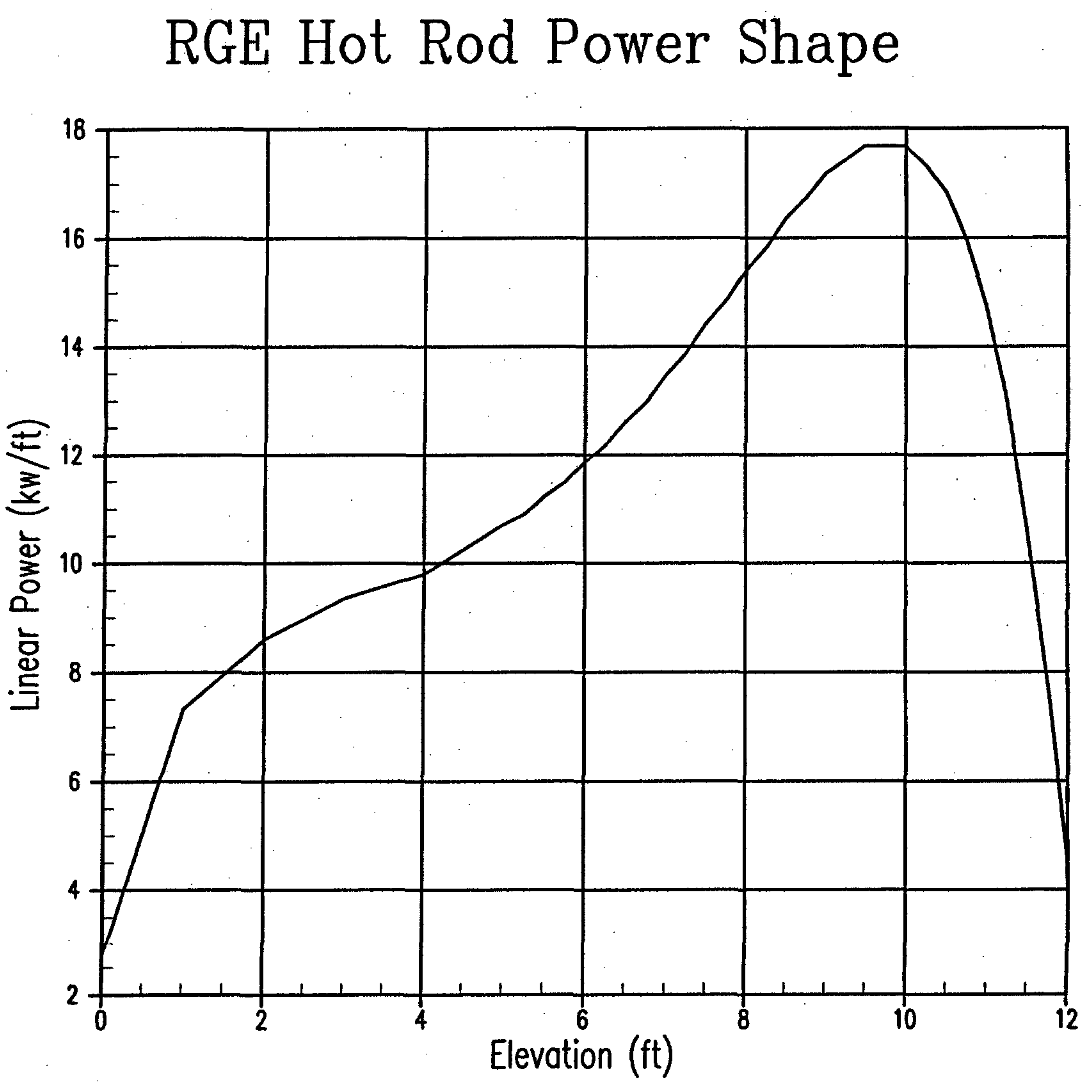
The bottom elevation of the suction leg piping is: 15.5336 ft
The top elevation of the core is: 20.3257 ft
The top elevation of the cold-leg discharge pipe is: 26.5108 ft
Note: All elevations are with respect to the bottom of the reactor vessel.

RAI #3: Provide the limiting top peaked axial power shape used in the analysis.

Response:

Figure 3-1 shows the hot rod limiting top peaked axial power shape used in the Reference 1 analysis. The power shape takes into account 10% SGTP and 25% AO SBSH.

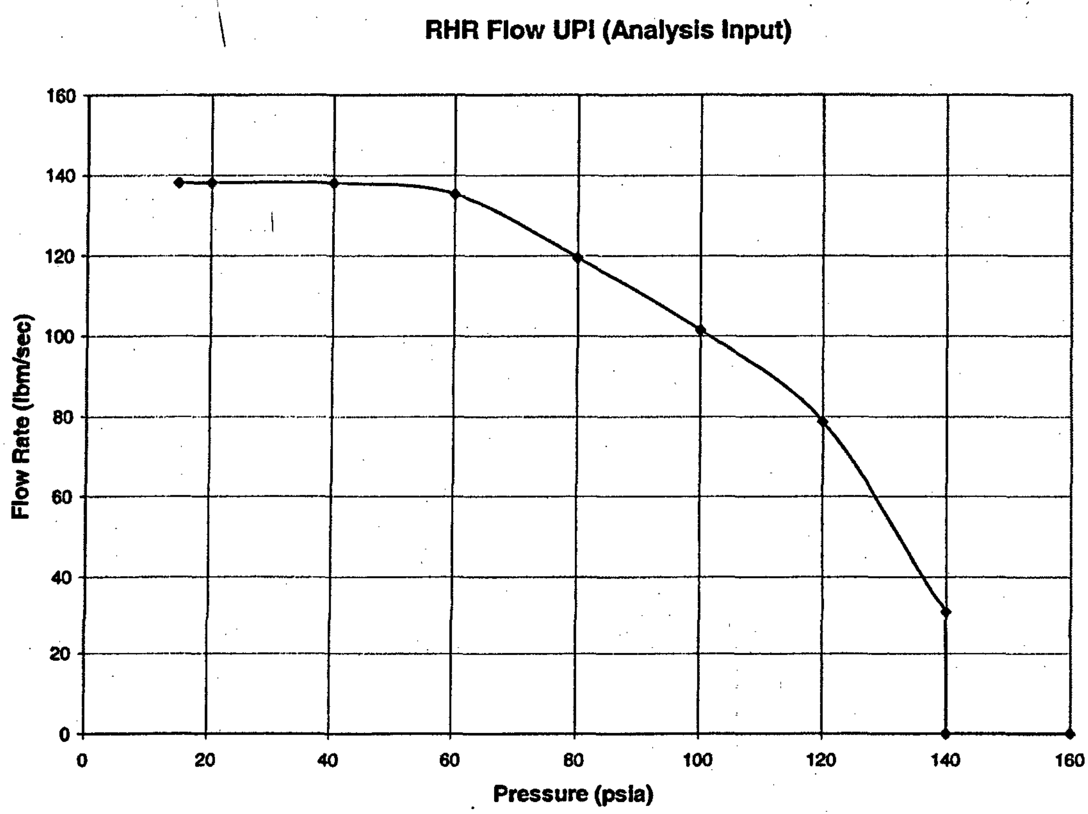
Figure 3-1: Hot Rod Limiting Power Shape



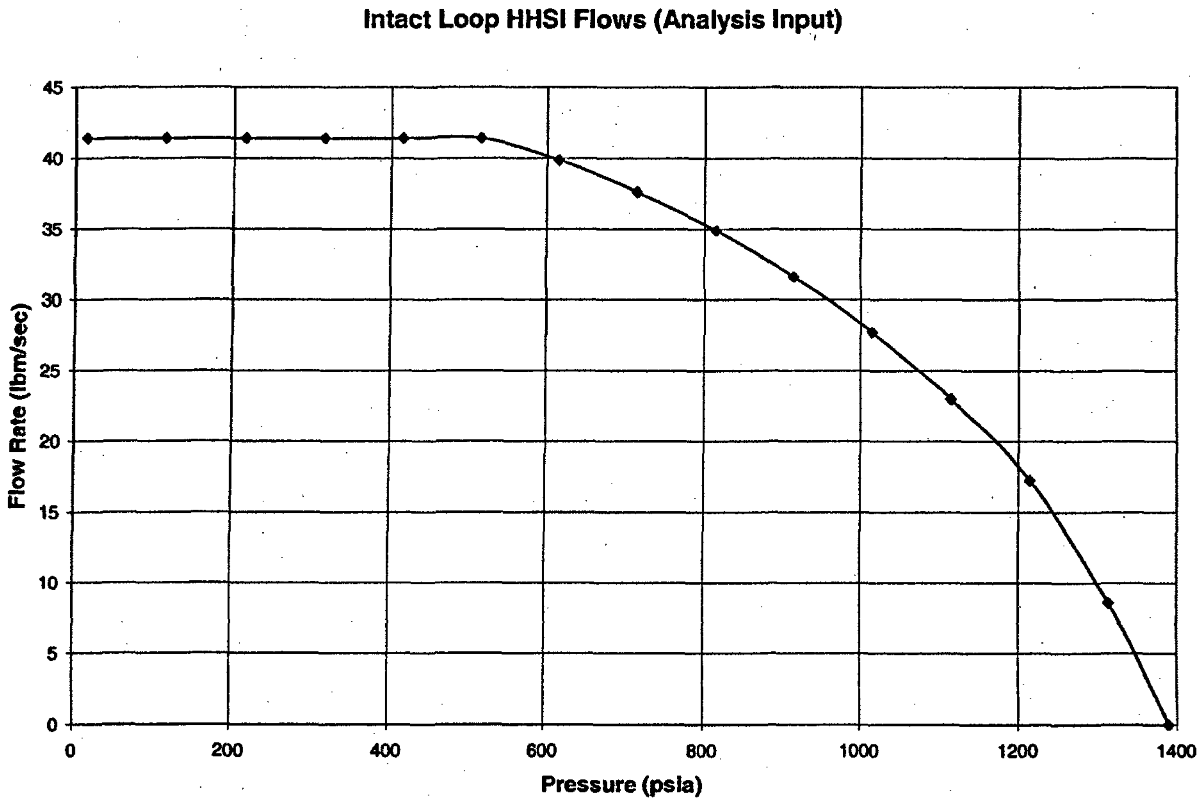
RAI #4:
4.1 Provide the head flow curve for the pumped safety injection (SI) system for the severed emergency core cooling injection line.
4.2 Provide a set of plots for this break (see item 1).

Response:
4.1 A severed emergency core cooling injection line is calculated as an 8.75-inch diameter break. Figure 4.1-1 through 4.1-3 show the head flow curves for the pumped SI system for this break size.

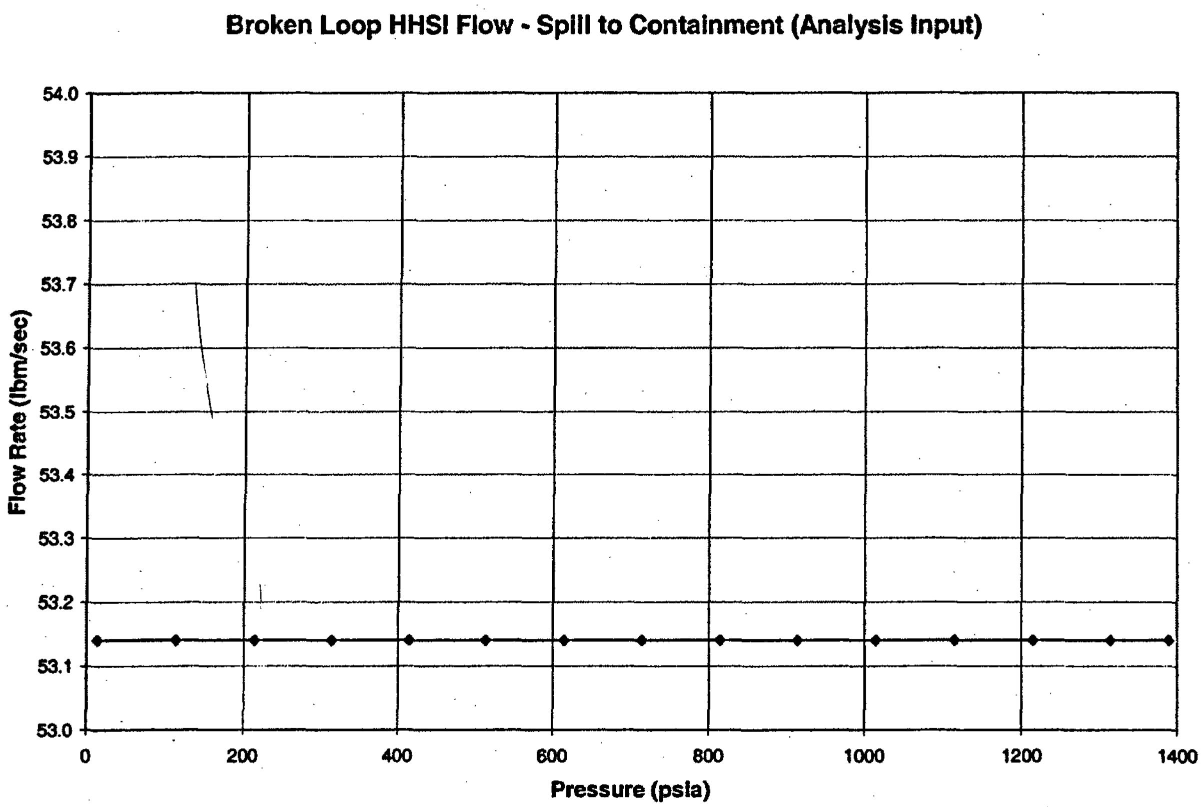
Figure 4.1-1: RHR Flow Curve for 8.75-Inch Break



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 4.1-2: Intact Loop HHSI Flow Curve for 8.75-Inch Break



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 4.1-3: Broken Loop HHSI Flow Curve for 8.75-Inch Break



Response:

4.2 Figure 4.2-1 through 4.2-12 show the set of plots requested under RAI #1 for the 8.75-inch break.

A 600 second SI interruption is assumed during switchover to recirculation for all SBLOCA cases analyzed for R.E. Ginna. Note that this is a conservative assumption since it terminates SI flows completely during switchover. In reality, based on R.E. Ginna's EOP ES-1.3, the SI flows will not be completely terminated during switchover. There will be flow from either the HHSI pumps or the LHSI pumps while the other pumps are being realigned should the RCS pressure allow. In particular, this assumption would affect the larger break size cases, including the injection line break (8.75 inches), since the RCS pressure will be below the LHSI cut in pressure. As such, it would allow for continuous SI injection (either HHSI or LHSI) during the switchover period rather than the modeled 600 second interruption for the larger break sizes.

Note that in addition to the above assumption, for the injection line break case the NOTRUMP model assumes the following:

1. Broken loop SI flow is spilled to the containment
2. No flow is assumed from the broken loop accumulators. Therefore, Figure 4.2-9 shows the flow from the intact loop accumulator only.
3. No flow is assumed from the HHSI pumps following switchover to recirculation.

For the 8.75 inch break case minimal core uncovery is seen during this transient. Figure 4.2-12 shows that the void fraction in the core reaches a value of 1.0 (indicating core dryout during that time) for very short period of time during the initial blowdown period. This is short-lived in duration and has a negligible effect on PCT for this case.

Based on the above discussion, the rod heatup code was not run for the 8.75-inch break. Therefore, the plots requested in RAI #1 with respect to the peak clad temperature location and the hot spot are not included in this evaluation (Items 1.7 through 1.9).

Figure 4.2-1: 8.75-Inch Core Power (Btu/s)

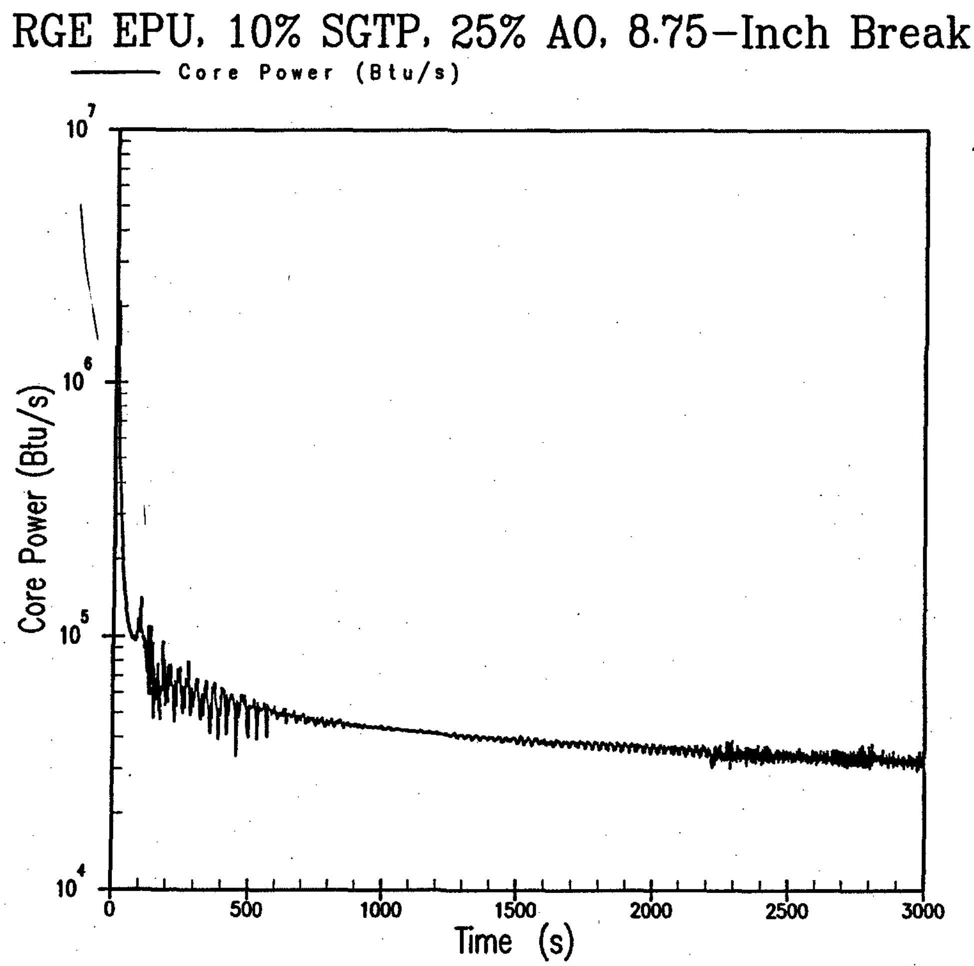


Figure 4.2-2: 8.75-Inch Core Inlet Mass Flowrate

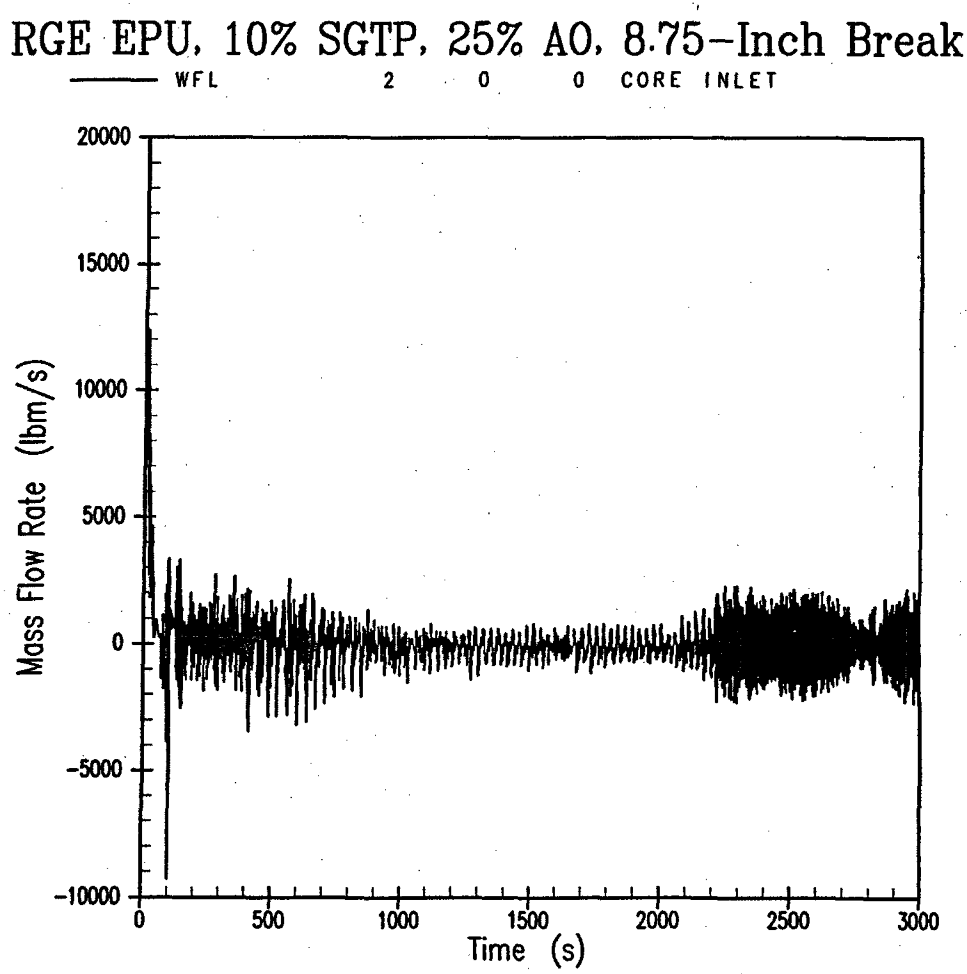


Figure 4.2-3: 8.75-Inch Break Mass Flowrate

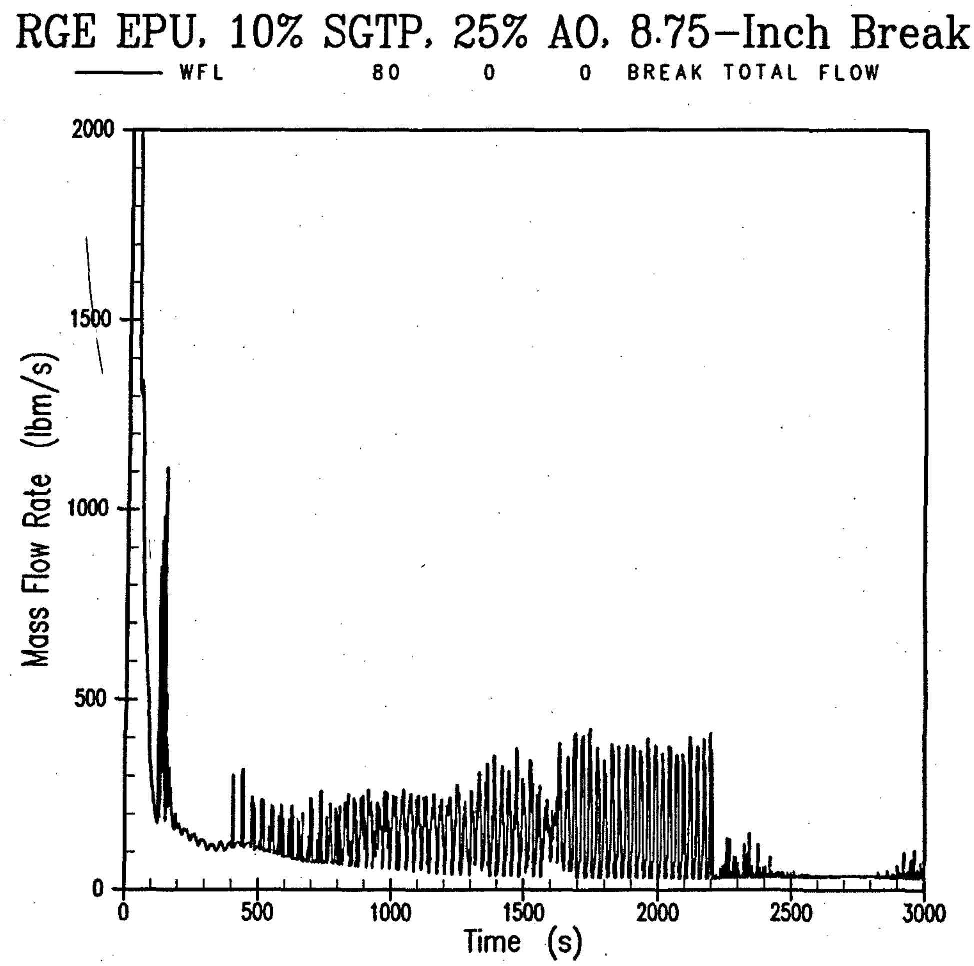


Figure 4.2-4: 8.75-Inch Break Quality

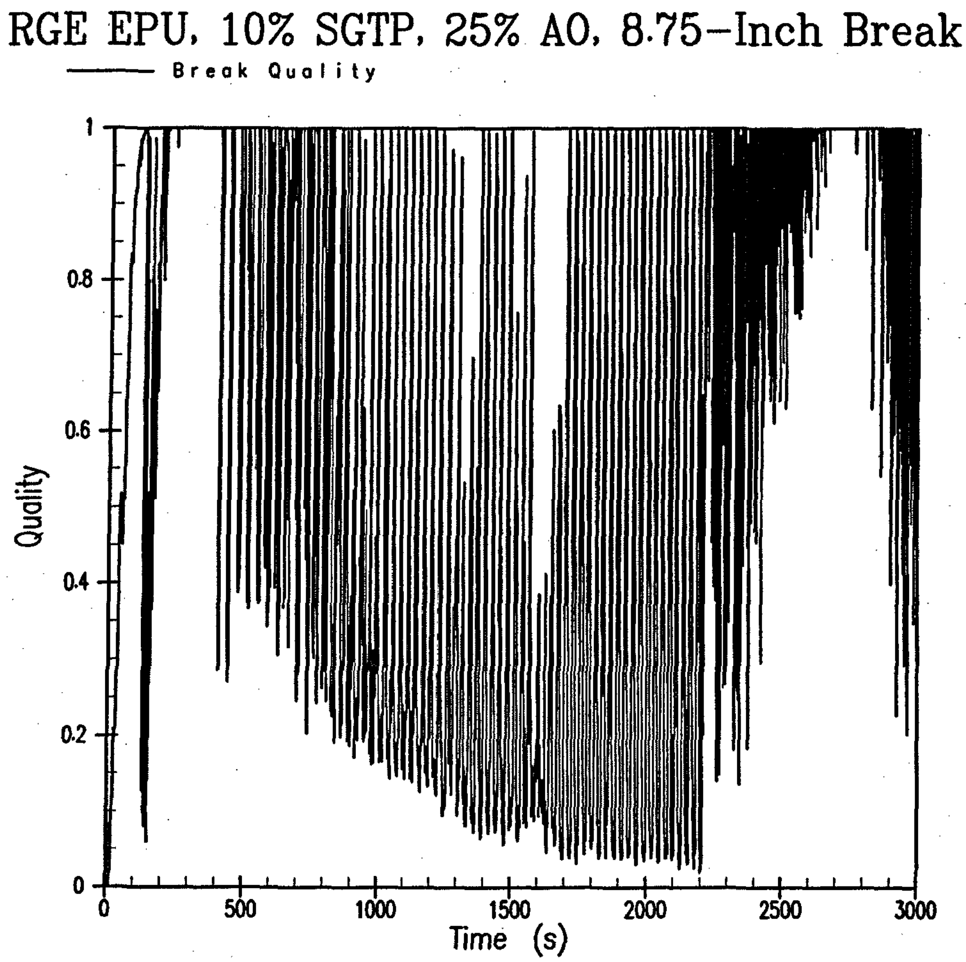


Figure 4.2-5: 8.75-Inch Pressurizer Pressure

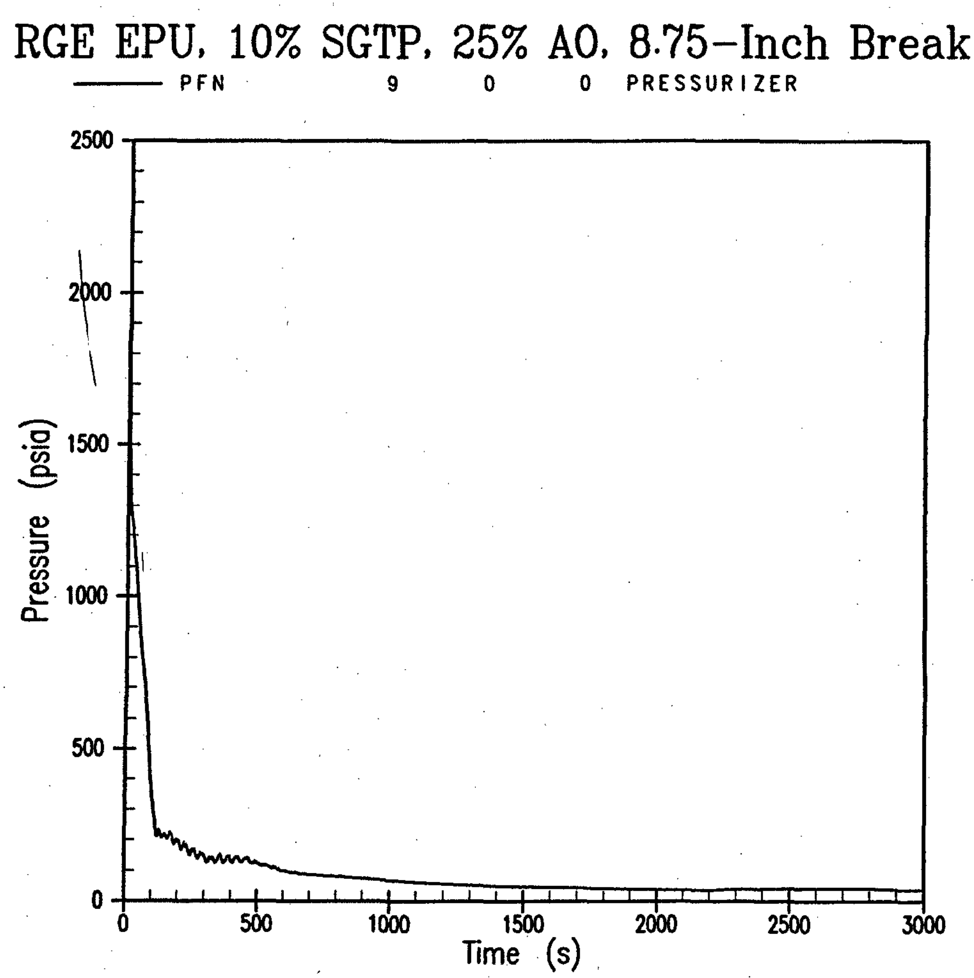


Figure 4.2-6: 8.75-Inch Core Mixture Level

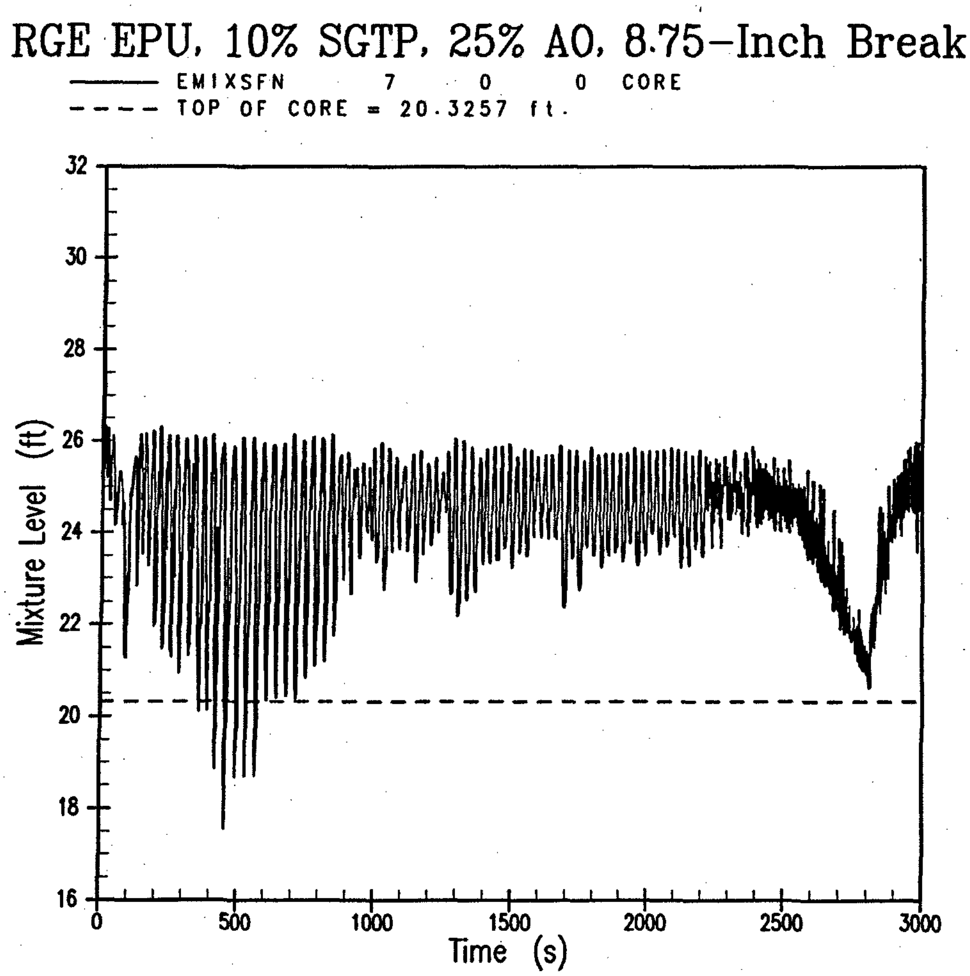


Figure 4.2-7: 8.75-Inch Broken Loop High Head SI Mass Flow Rate (Spill to Containment)

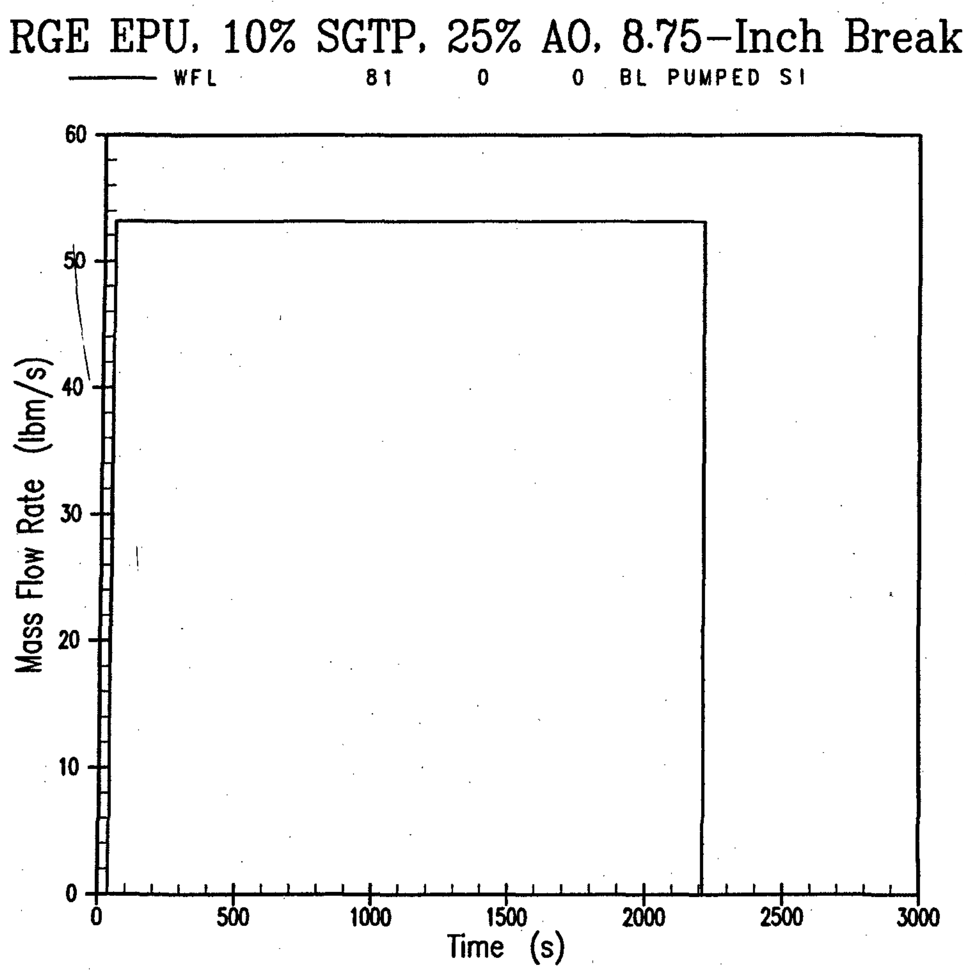


Figure 4.2-8: 8.75-Inch Intact Loop High Head SI Mass Flow Rate

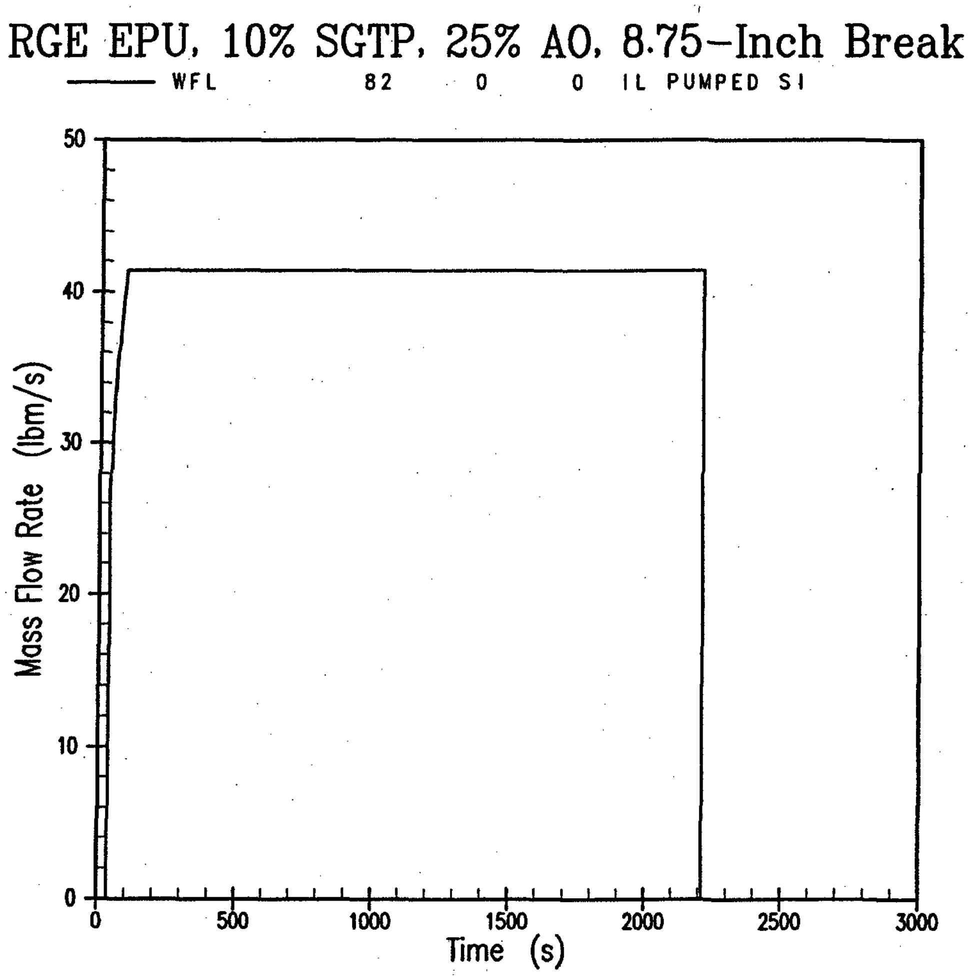


Figure 4.2-9: 8.75-Inch Intact Loop Low Head SI Mass Flow Rate

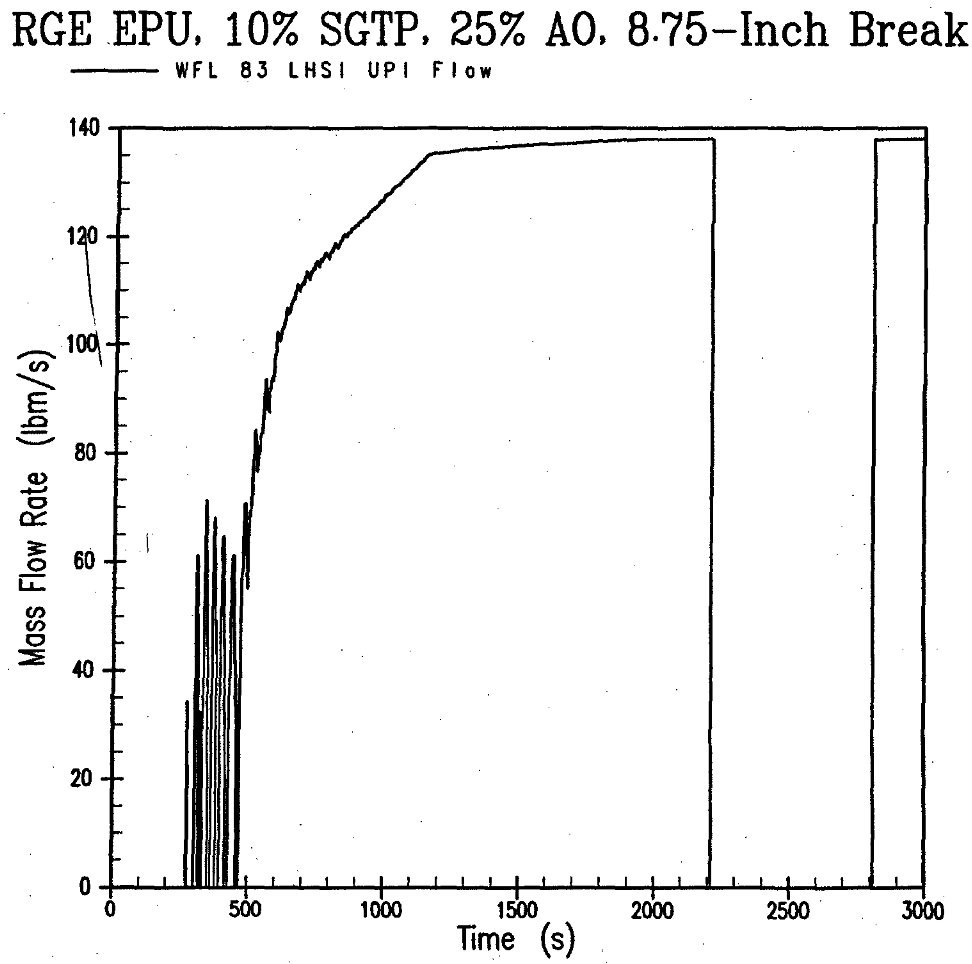


Figure 4.2-10: 8.75-Inch Intact Loop Accumulator Injection Mass Flow Rate

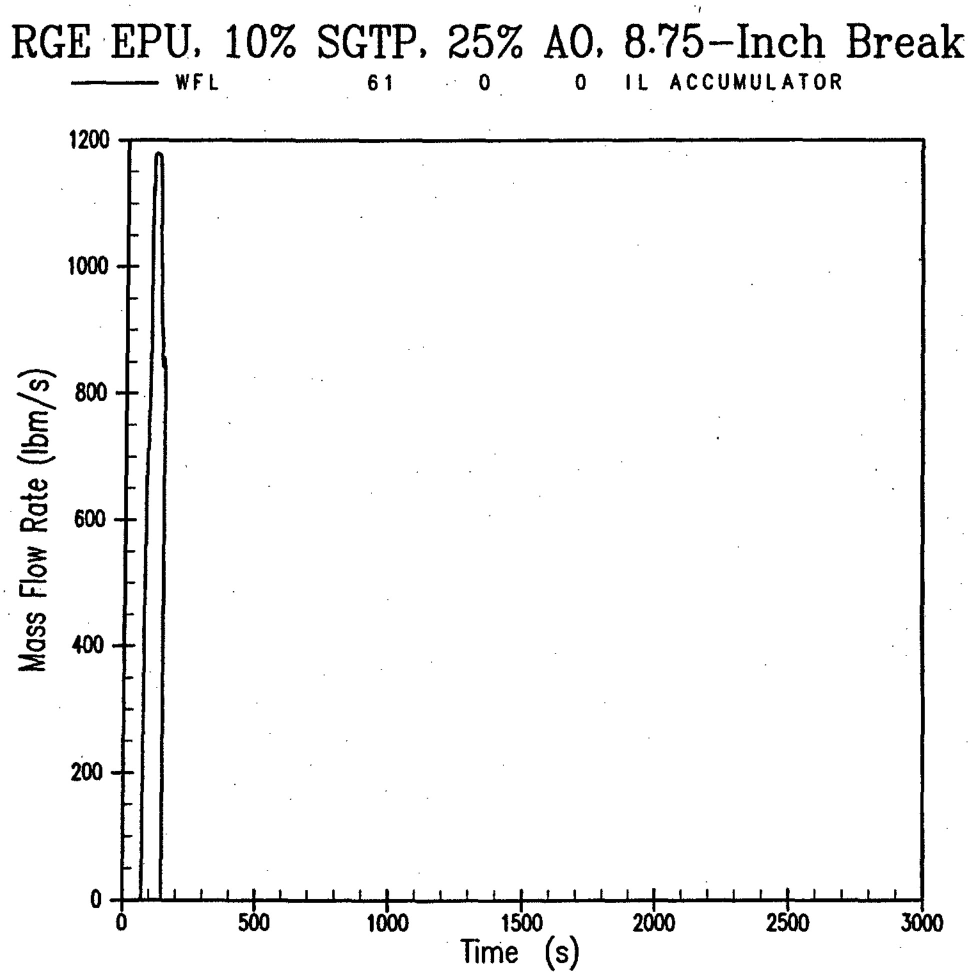


Figure 4.2-11: 8.75-Inch Condensation Rate in Cold Legs

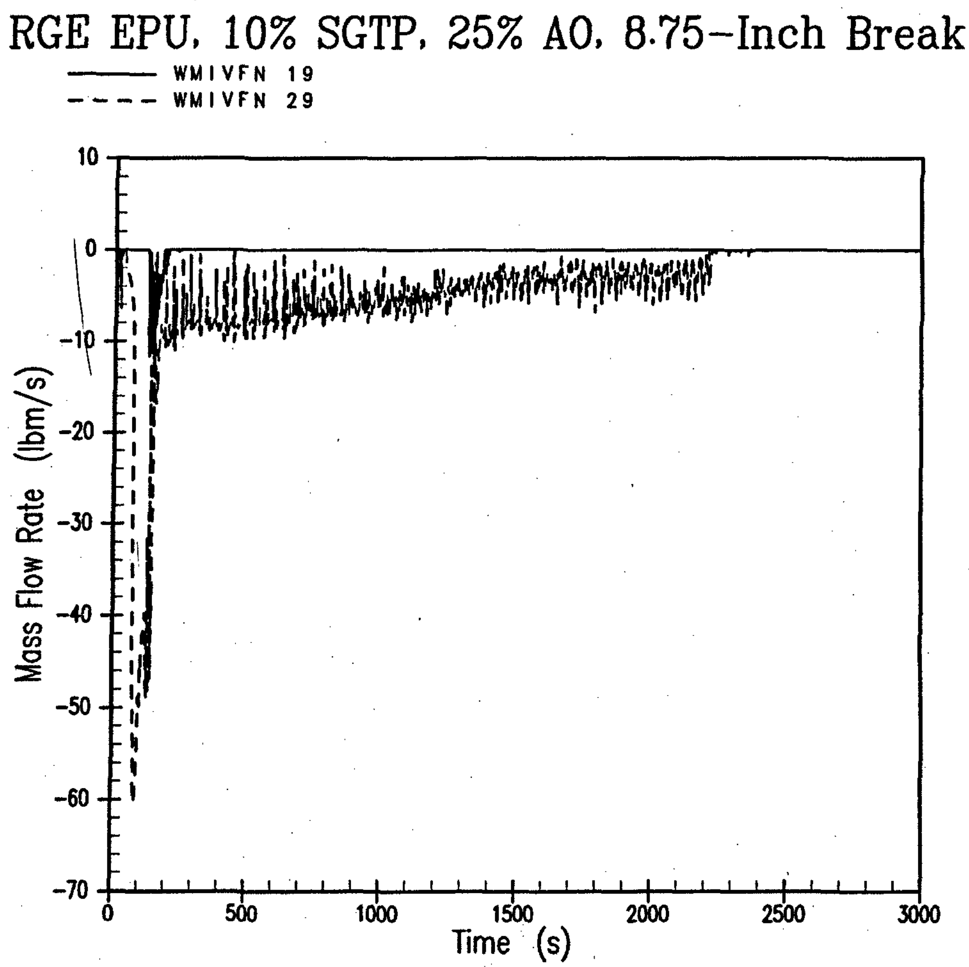
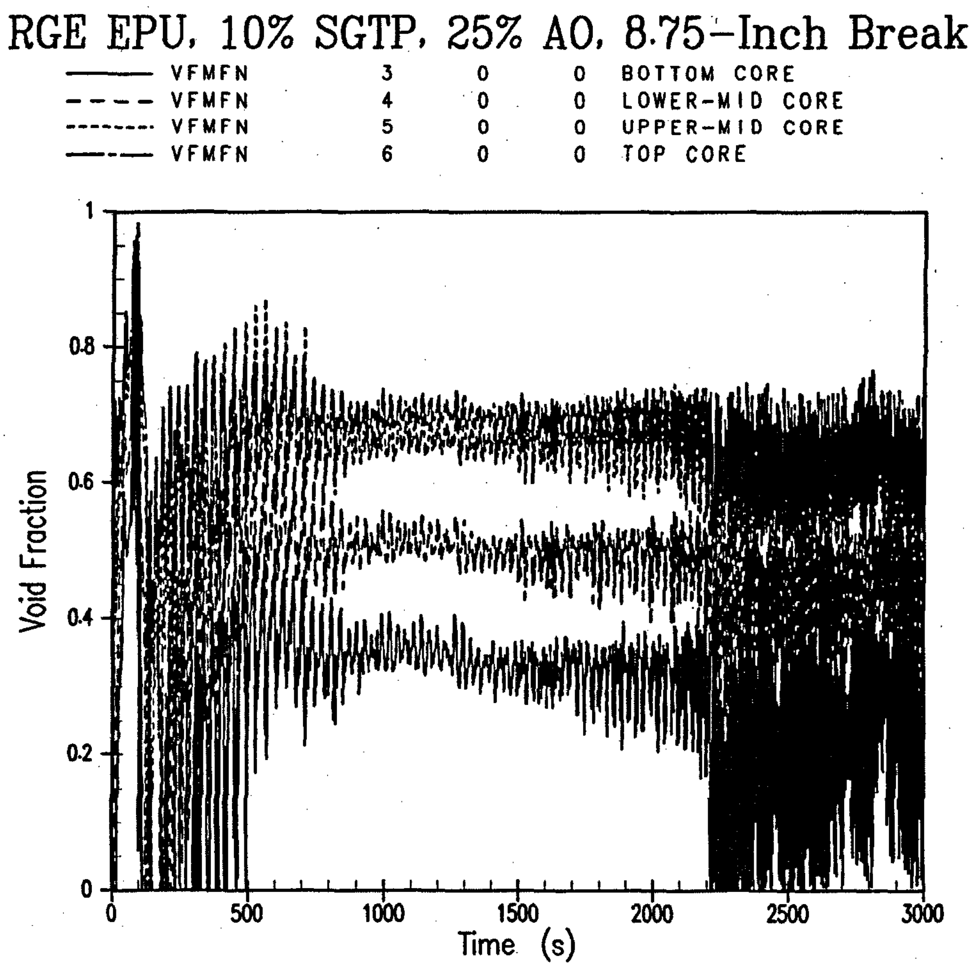


Figure 4.2-12: 8.75-Inch Void Fraction in Core Nodes



RAI #5: Breaks larger than 3 inches were not provided. Provide information to demonstrate that breaks as large as 0.5 ft^2 are not limiting and, as such, the worst-case break has been identified.

Response:

Break sizes larger than 3 inches in diameter were simulated using the NOTRUMP computer code. These included cold leg breaks 4-, 6-, 8.75- and 9.75 inches in diameter. The 8.75 inch break represents the severed injection line break and the 9.75 inch break represents the 0.5 ft^2 break.

Figures 5-1 – 5-4 show the core mixture level plots for the above breaks. These plots show that there is no core uncovery for the 4-and 6-inch breaks. Figure 5-3 and 5-4 also indicates that there is no significant sustained uncovery for the 8.75 and the 9.75-inch breaks. Therefore, for these cases it was not necessary to perform a rod heat up calculation. Based on this these breaks are not limiting. Therefore it can be concluded that breaks larger than 3 inches will not be limiting and the 2-inch break continues to be the limiting small break case for R.E. Ginna.

Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 5-1: 4.0-Inch Break Core Mixture Level

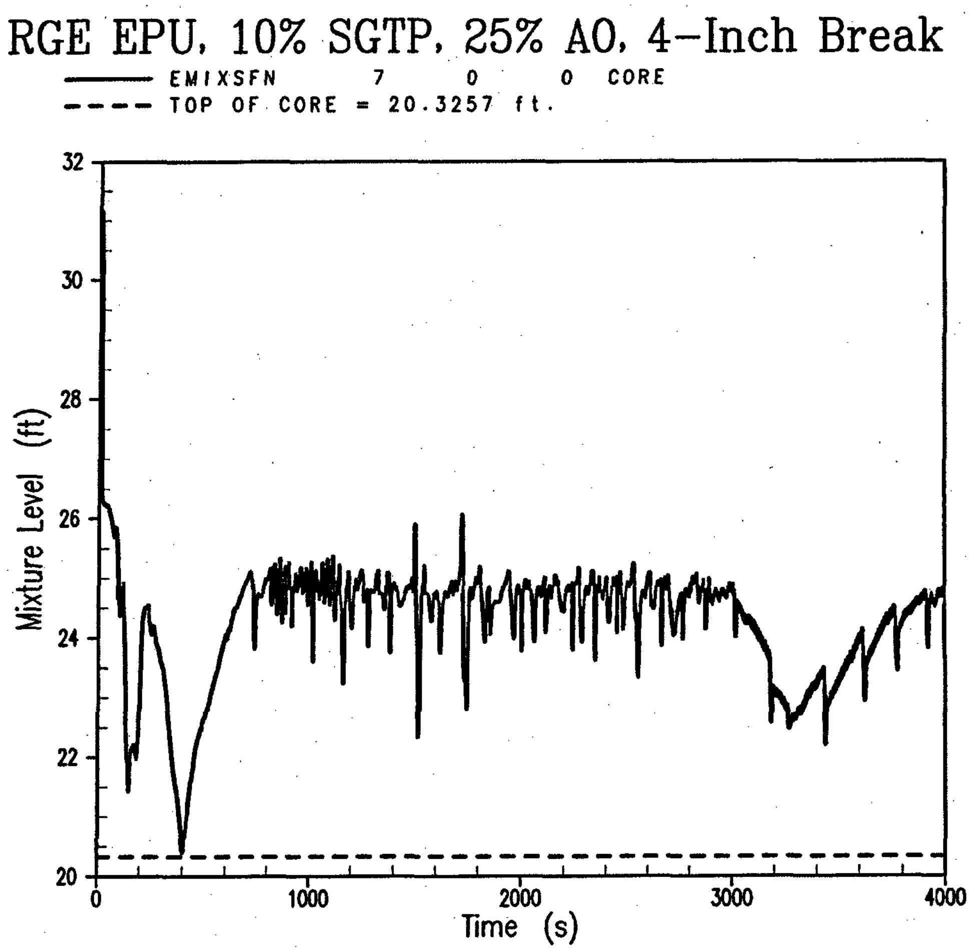


Figure 5-2: 6.0-Inch Break Core Mixture Level

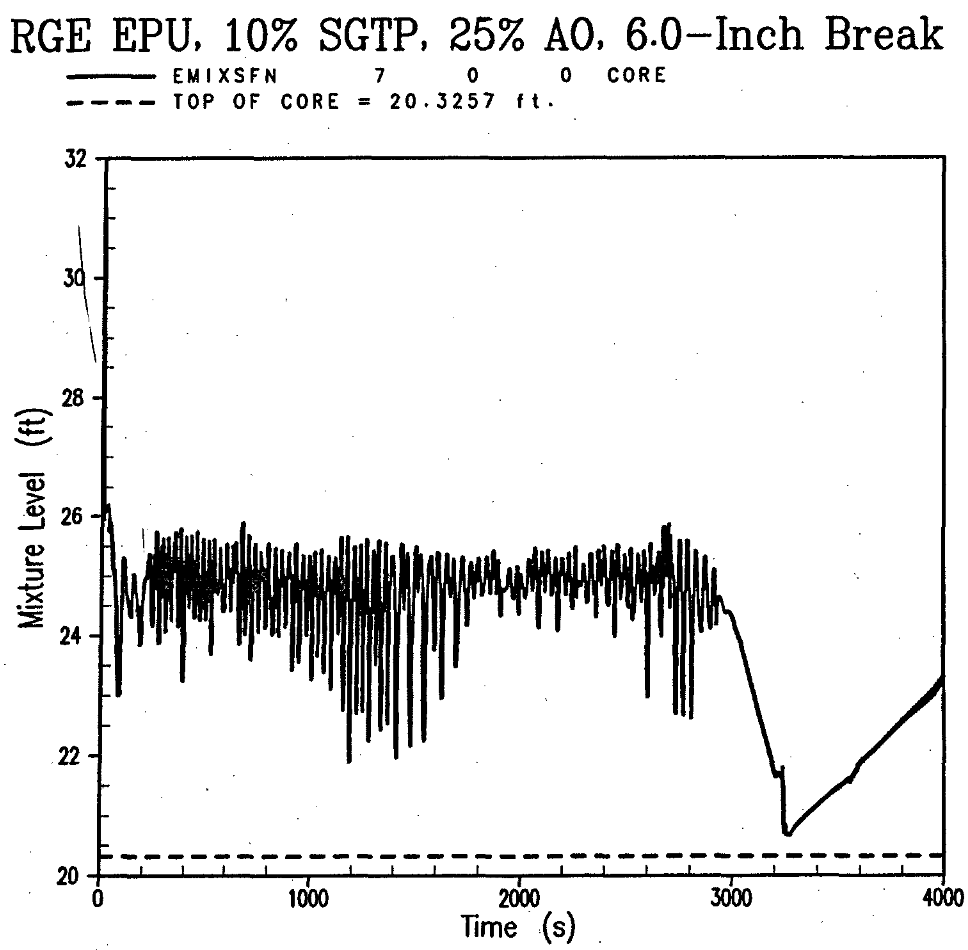


Figure 5-3: 8.75-Inch Break Core Mixture Level

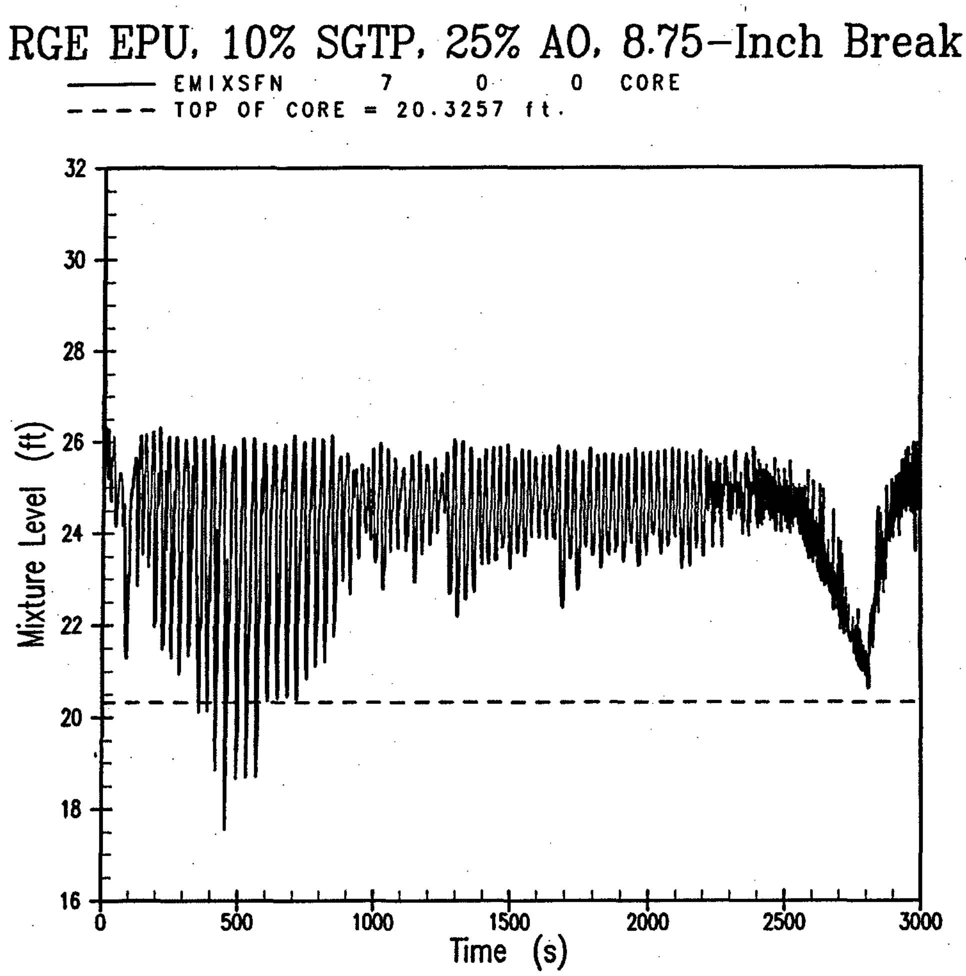


Figure 5-4: 9.75-Inch Break Core Mixture Level

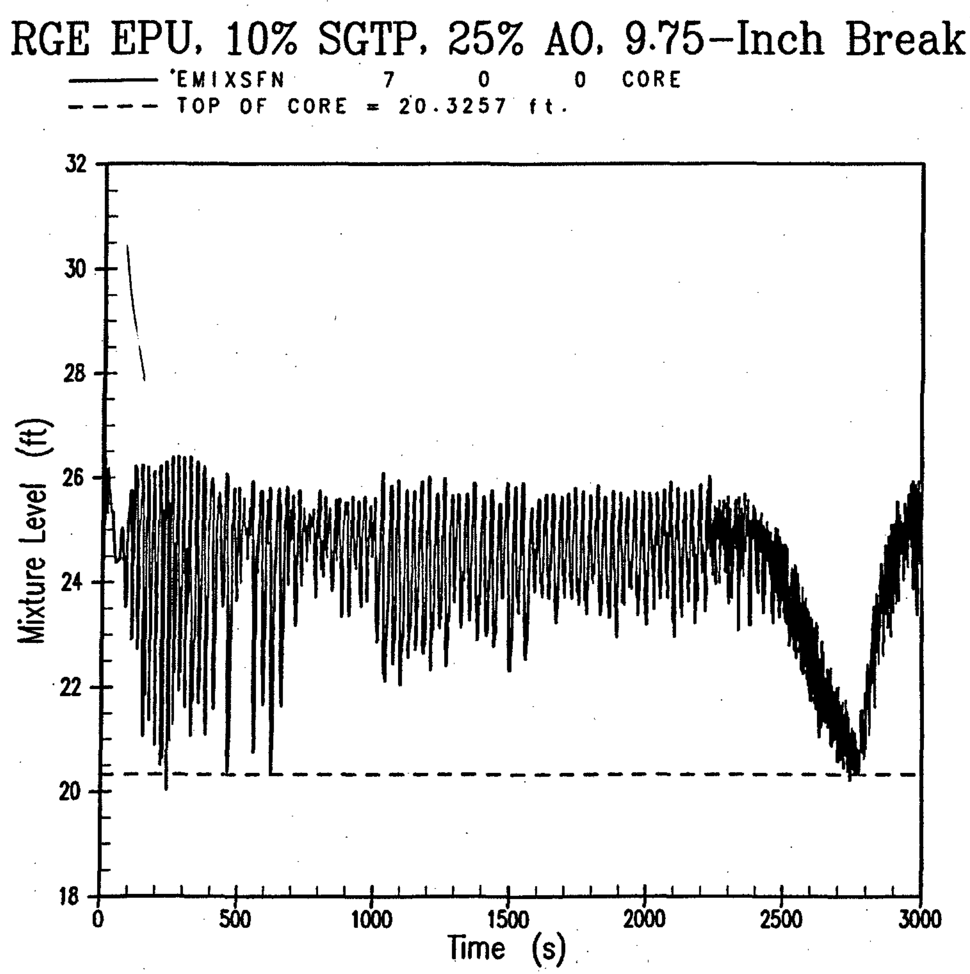


Figure 5-5: 4.0-Inch Break RCS Pressure

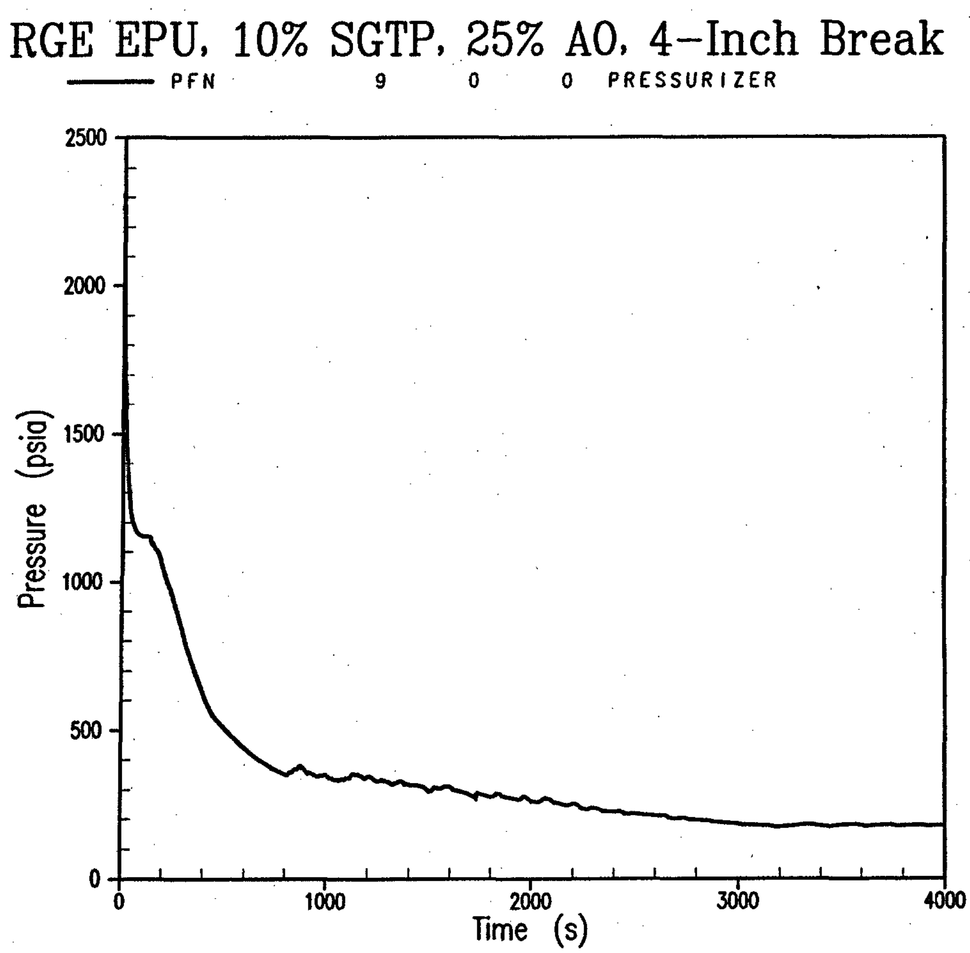


Figure 5-6: 6-Inch Break RCS Pressure

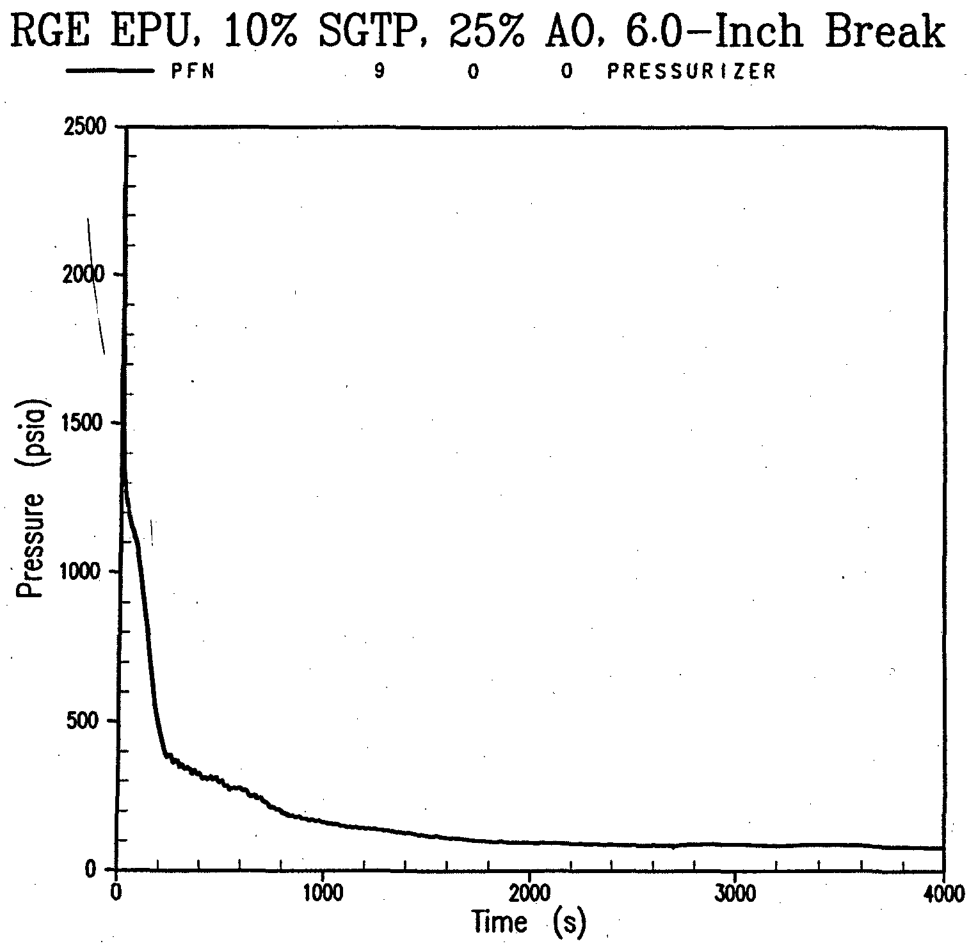


Figure 5-7: 8.75-Inch Break RCS Pressure

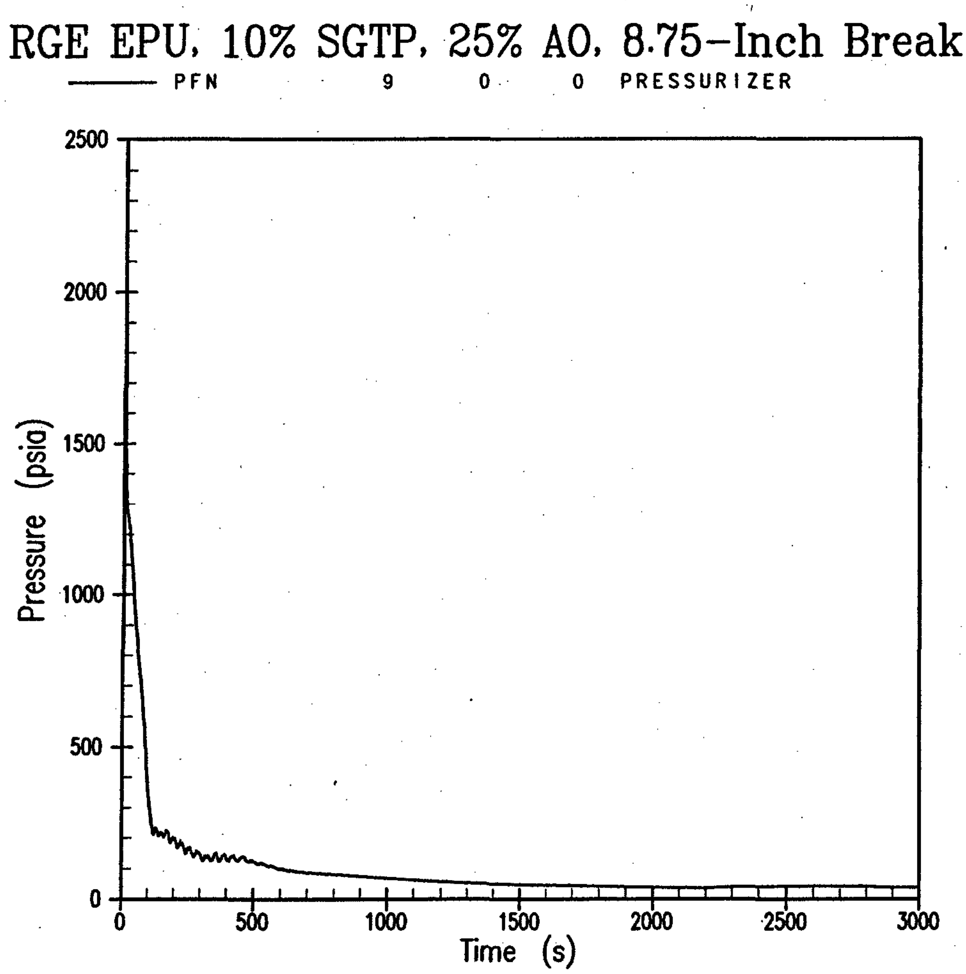
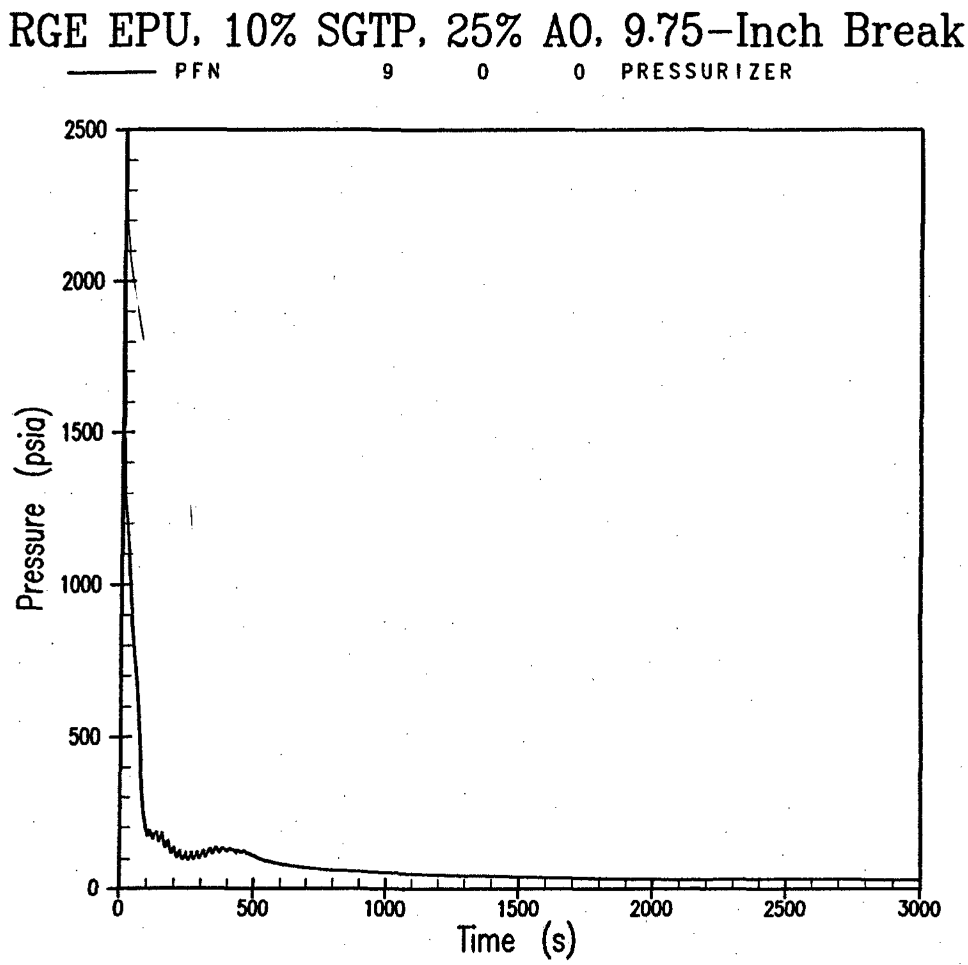


Figure 5-8: 9.75-Inch Break RCS Pressure



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 5-9: 4.0-Inch Break Mass Flowrate

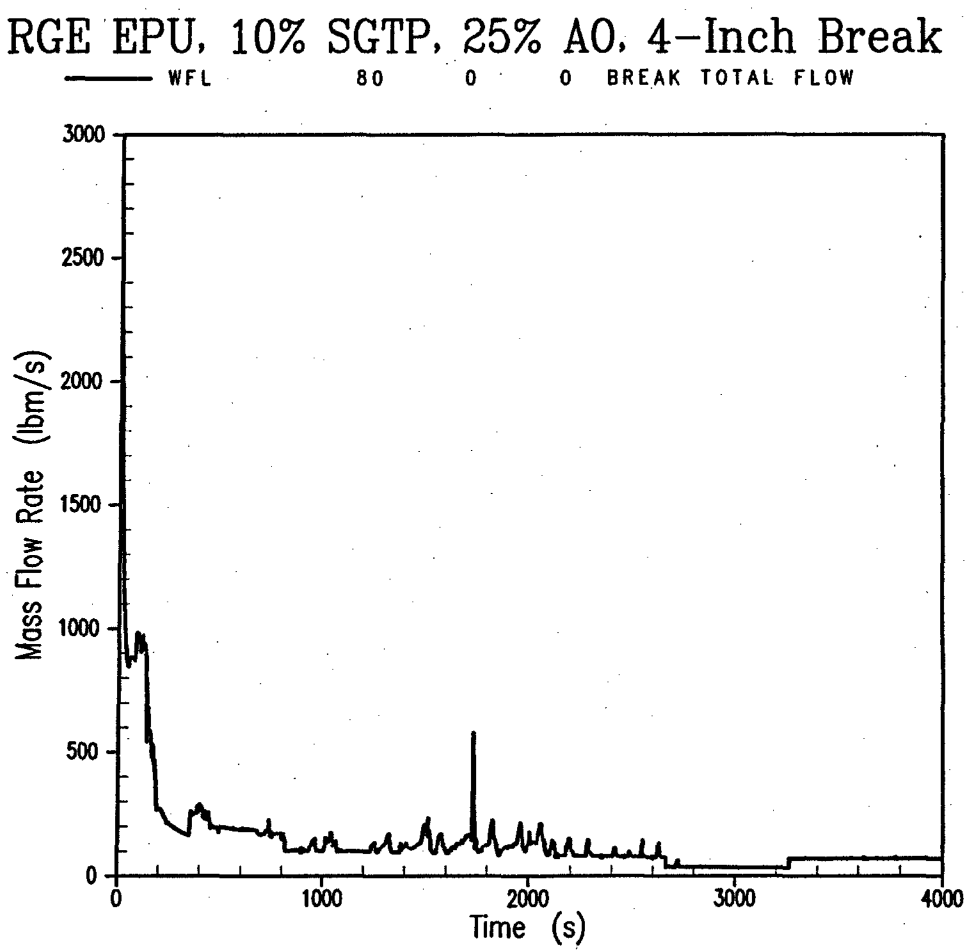


Figure 5-10: 6.0-Inch Break Mass Flowrate

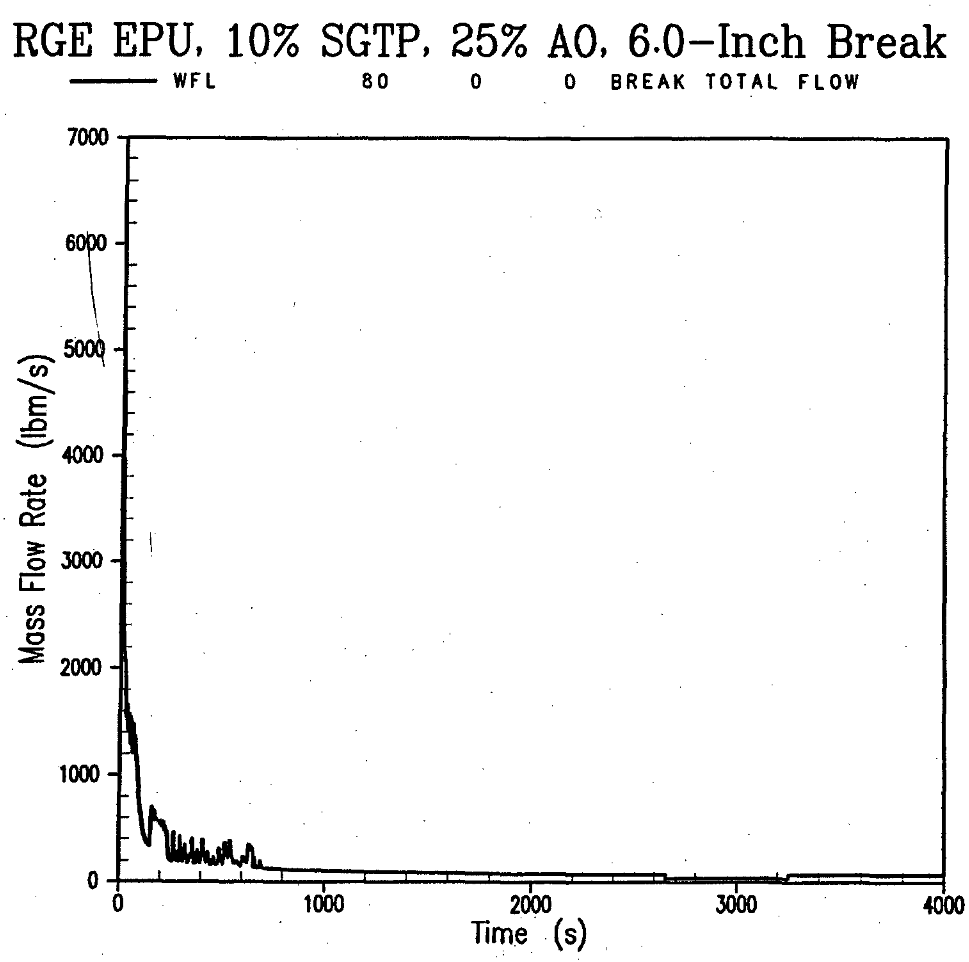


Figure 5-11: 8.75-Inch Break Mass Flowrate

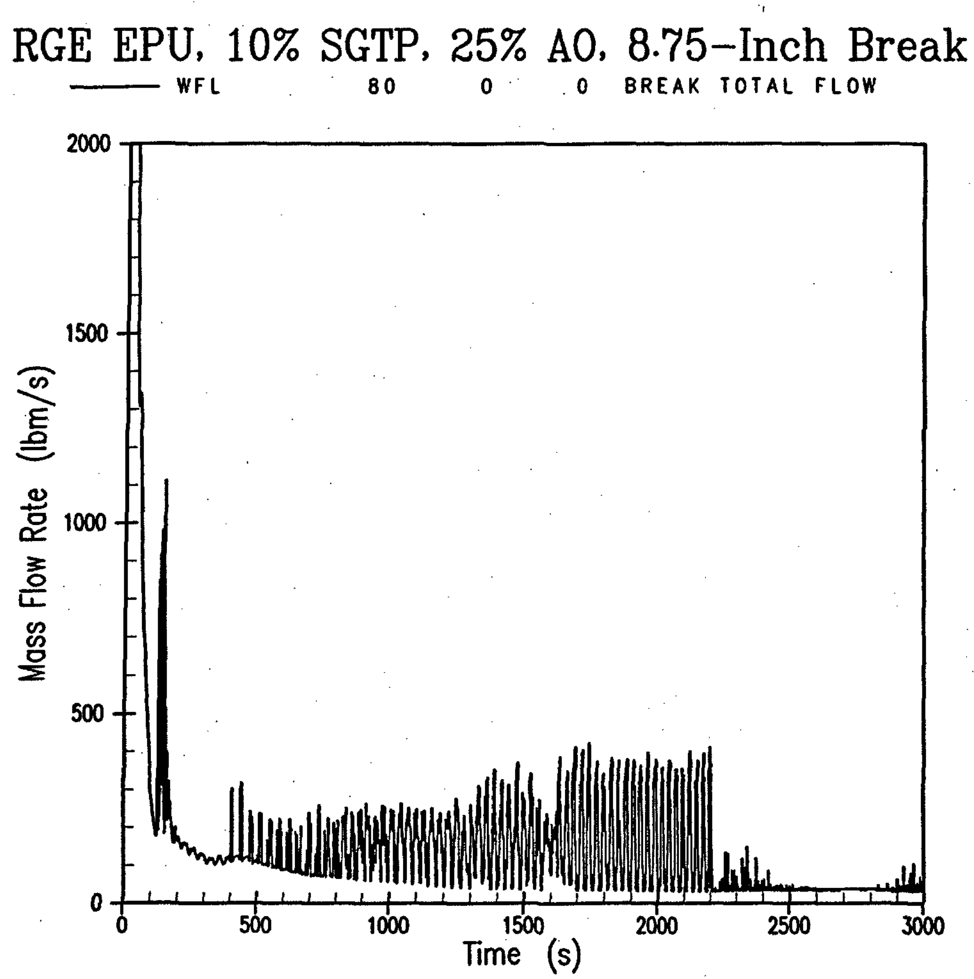
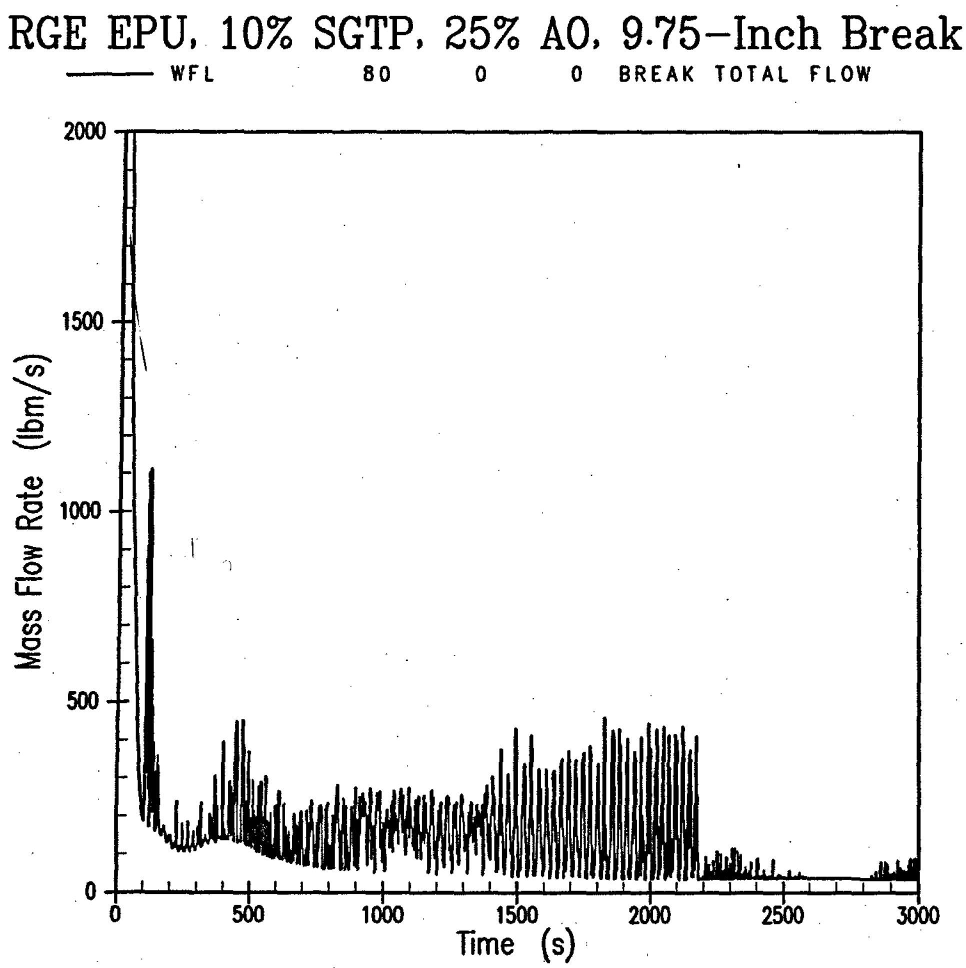


Figure 5-12: 9.75-Inch Break Mass Flowrate



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 5-13: 4.0-Inch Break Quality

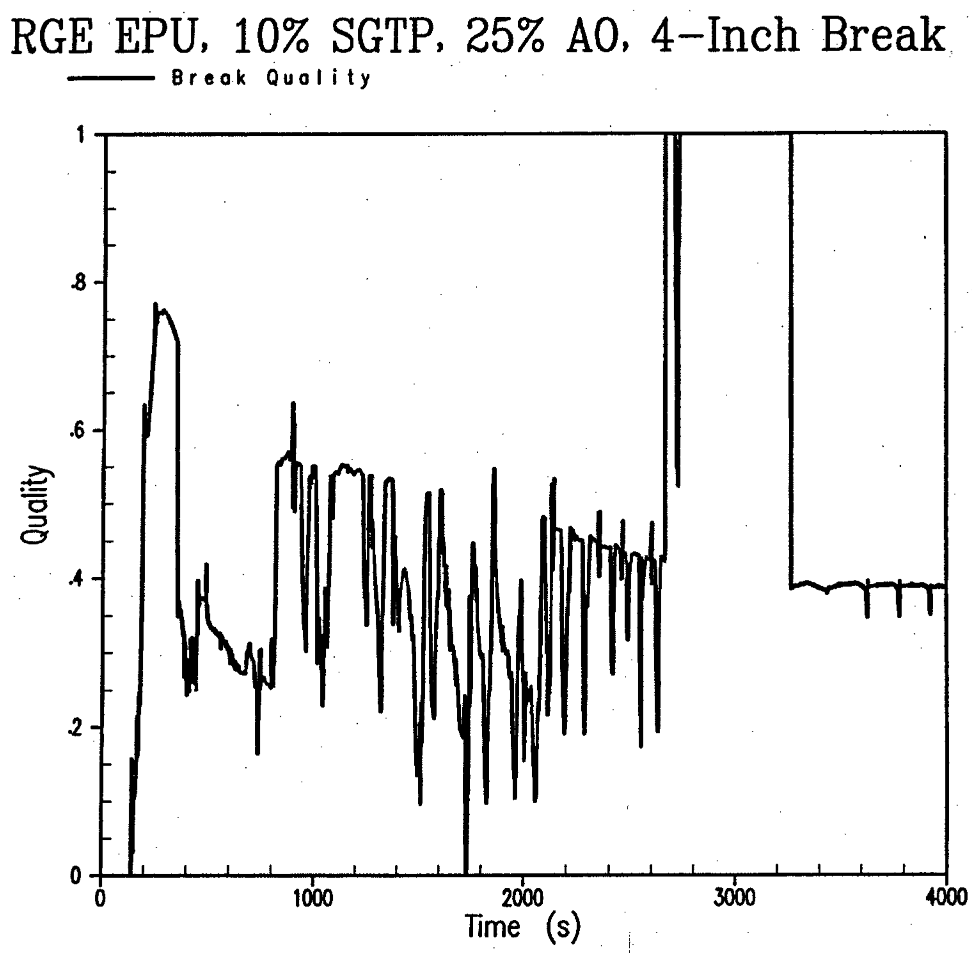


Figure 5-14: 6.0-Inch Break Quality

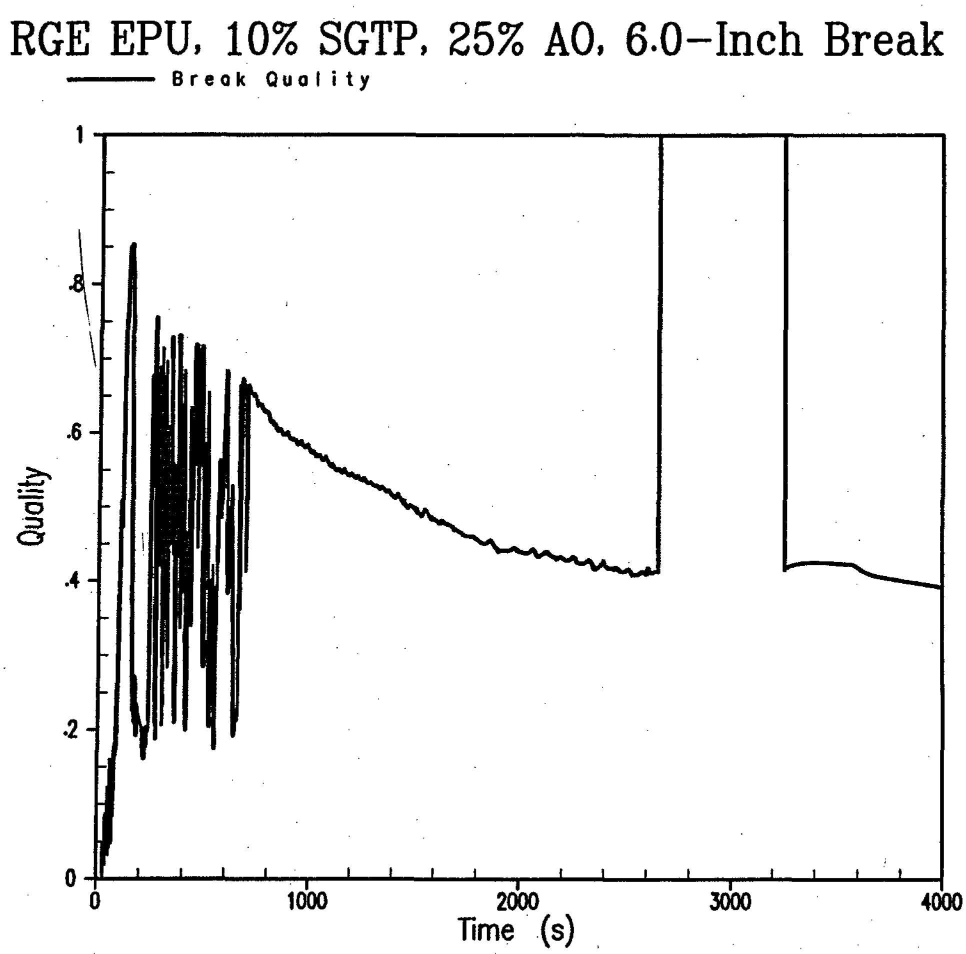


Figure 5-15: 8.75-Inch Break Quality

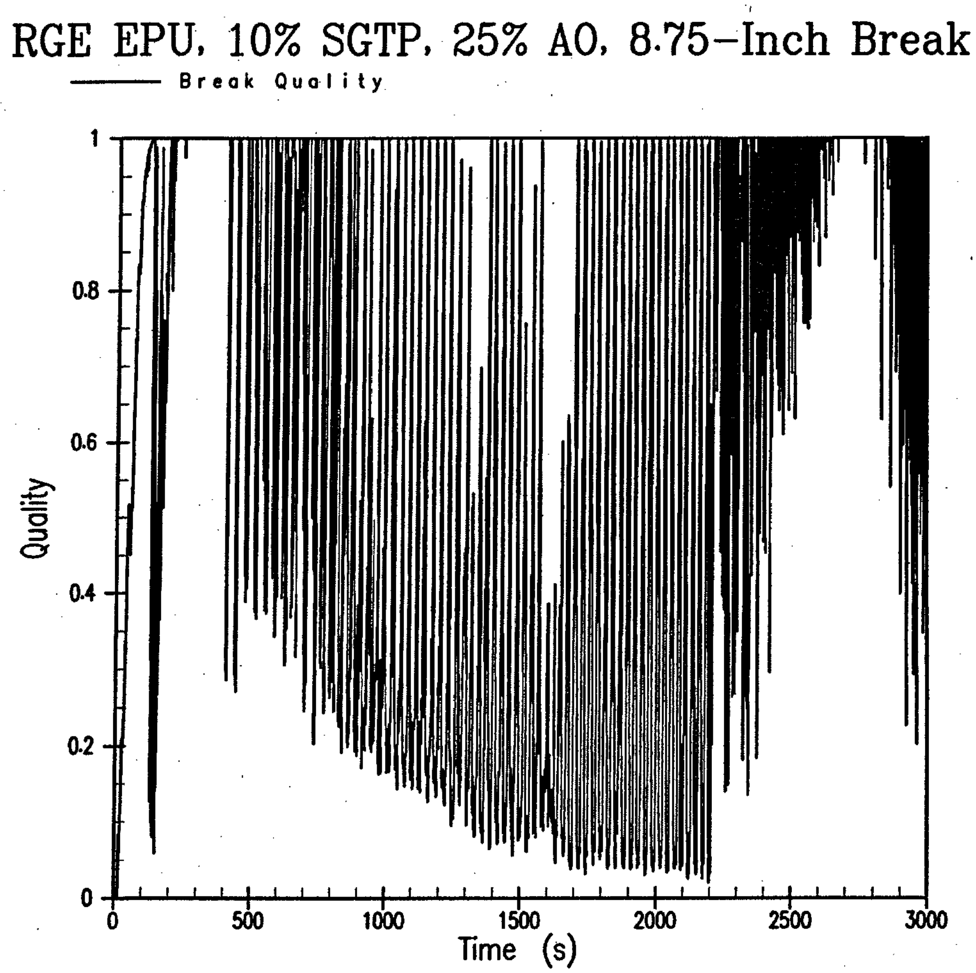


Figure 5-16: 9.75-Inch Break Quality

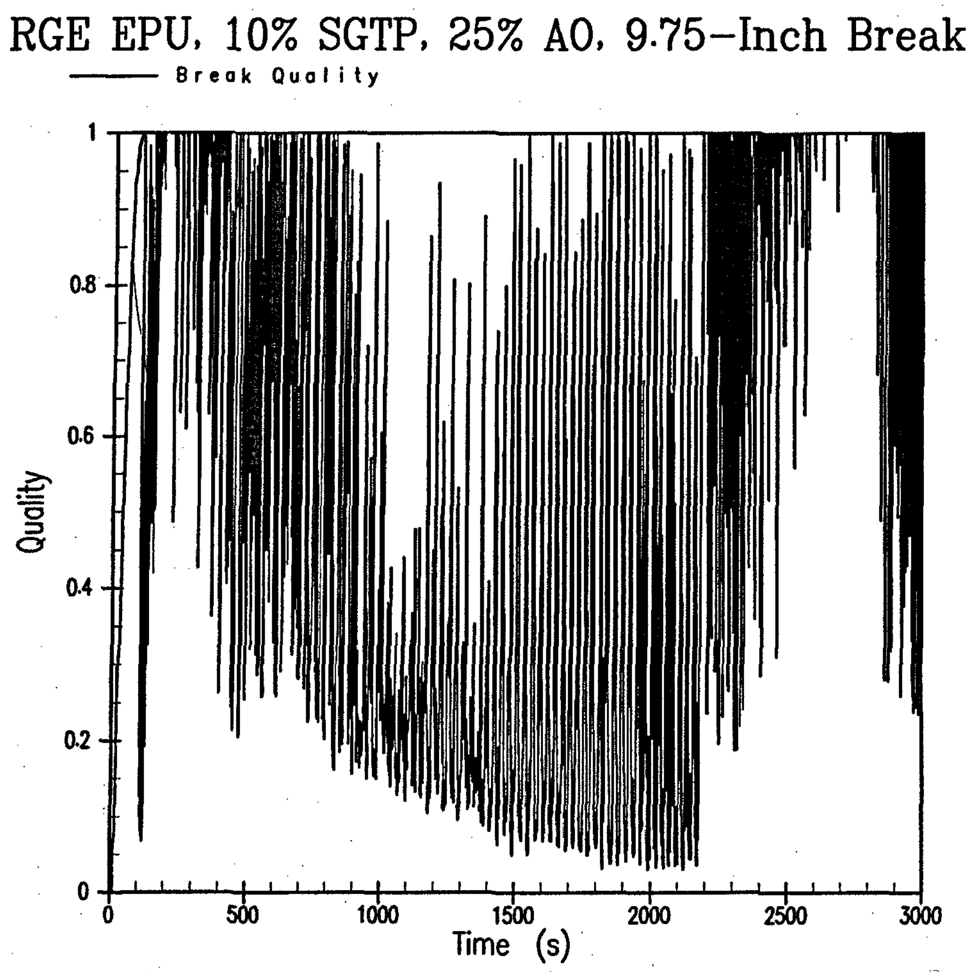


Figure 5-17: 4.0-Inch Break Total SI Flow

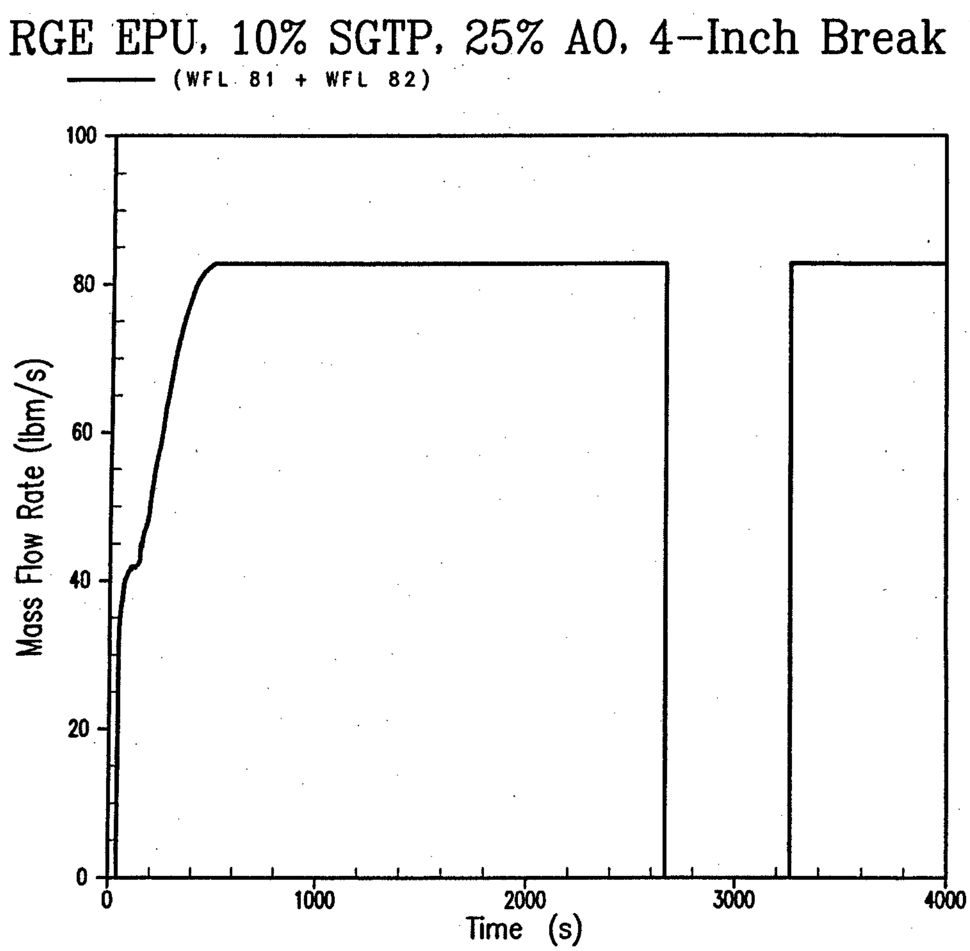


Figure 5-18: 6.0-Inch Break Total SI Flow

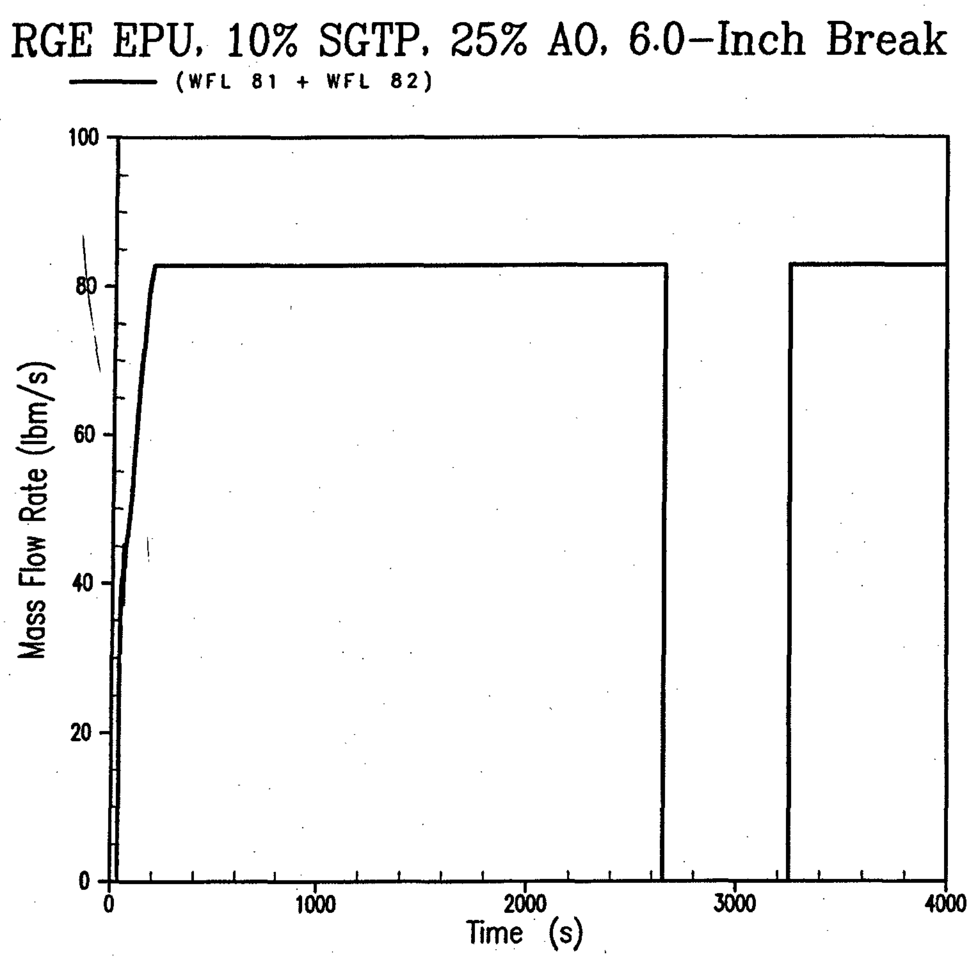


Figure 5-19: 8.75-Inch Broken Loop High Head SI Mass Flow Rate (Spill to Containment)

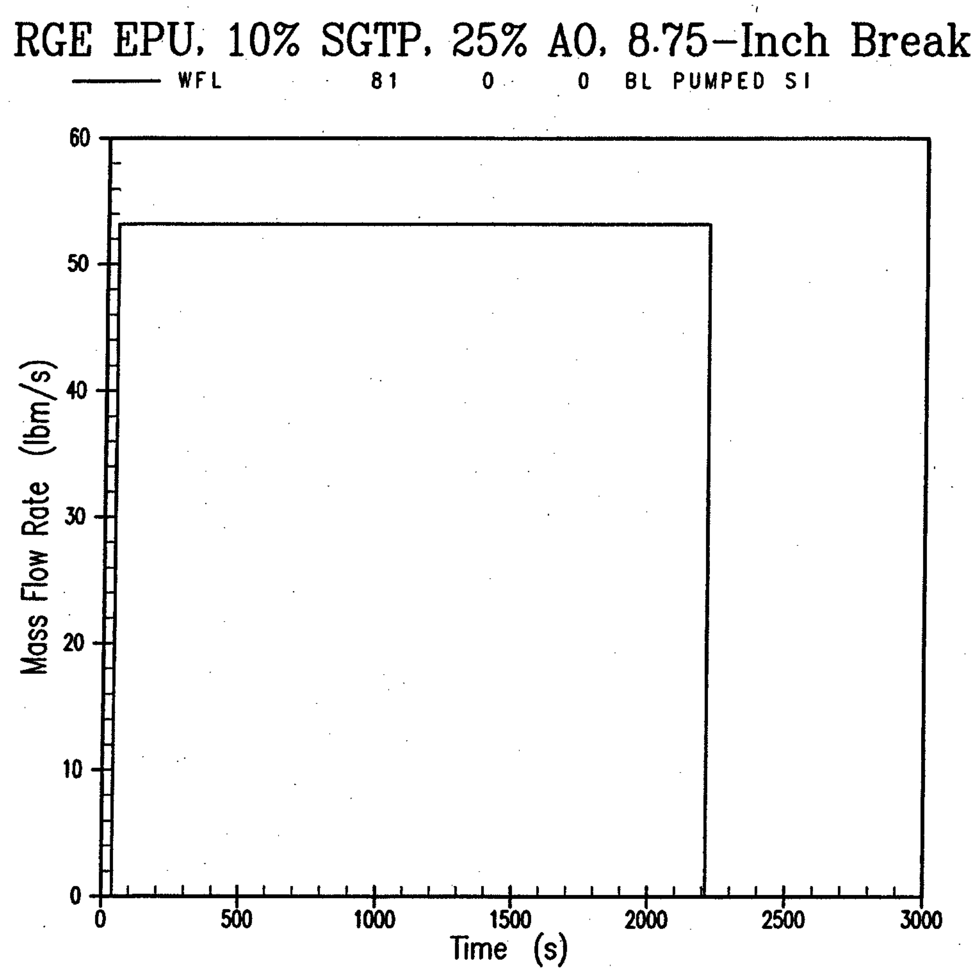


Figure 5-20: 8.75-Inch Intact Loop High Head SI Mass Flow Rate

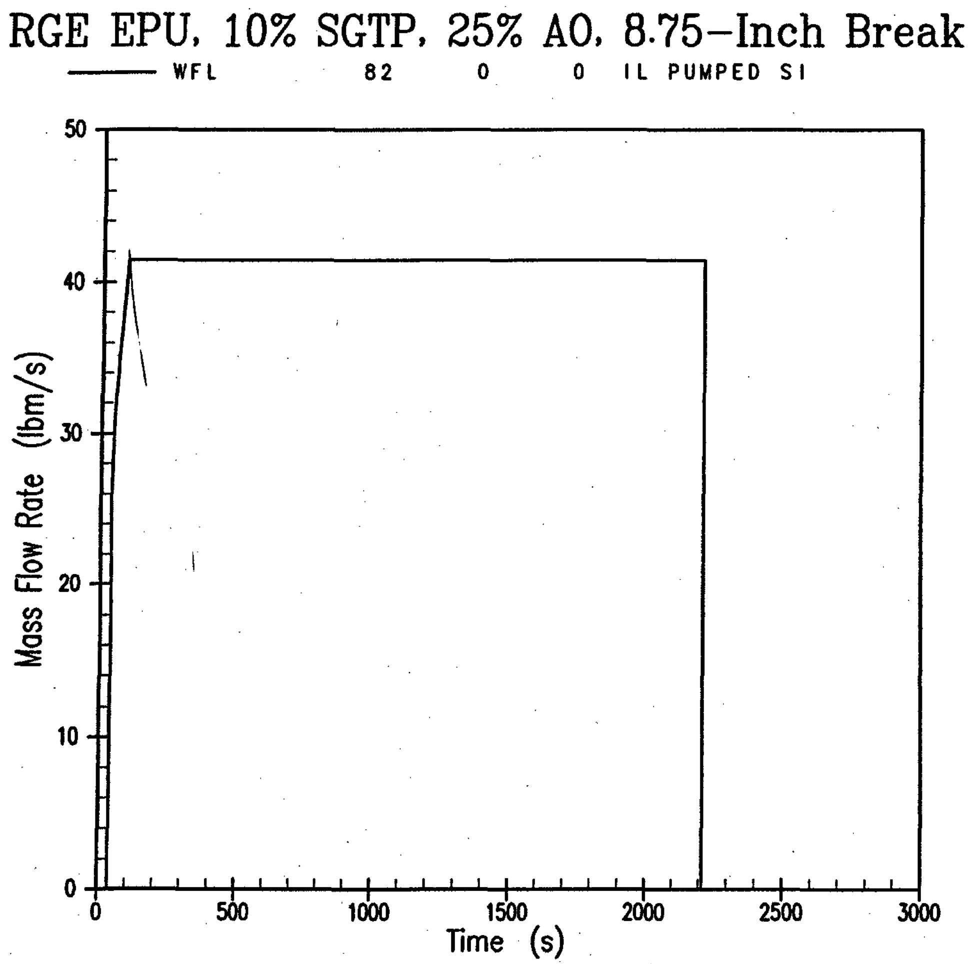


Figure 5-21: 8.75-Inch Intact Loop Low Head SI Mass Flow Rate

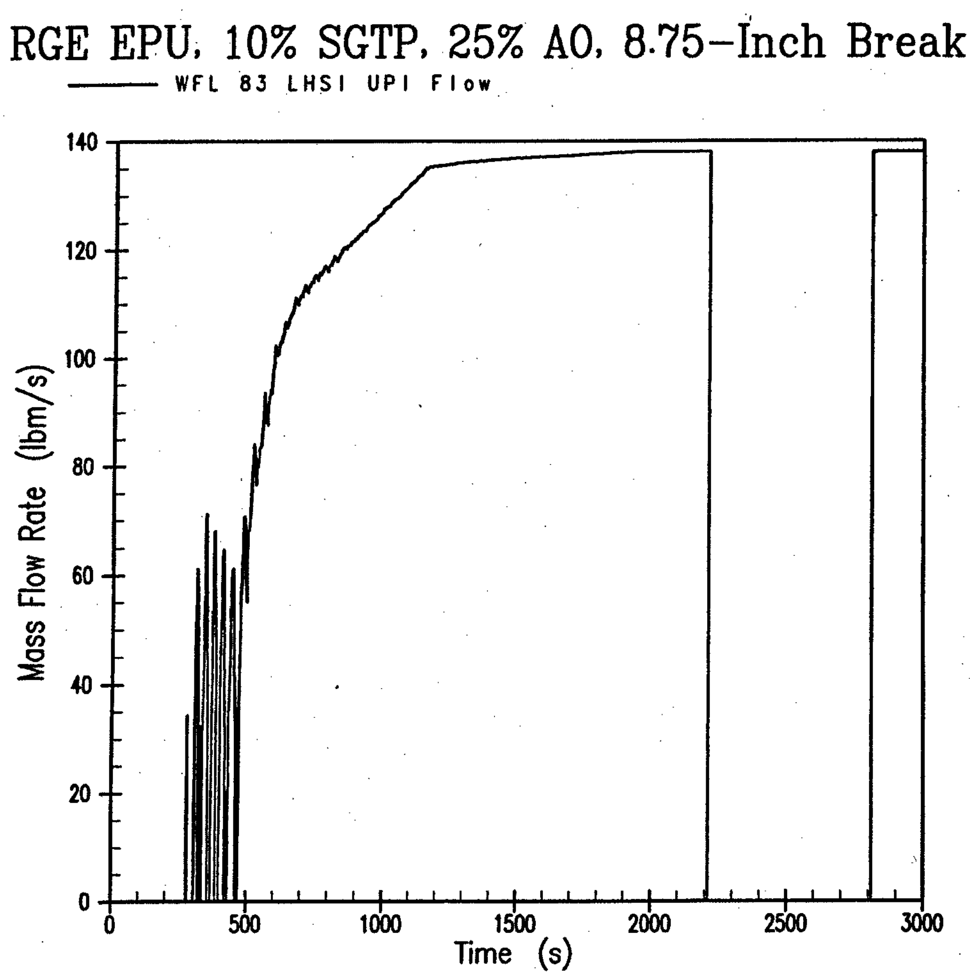


Figure 5-22: 9.75-Inch Broken Loop High Head SI Mass Flow Rate (Spill to Containment)

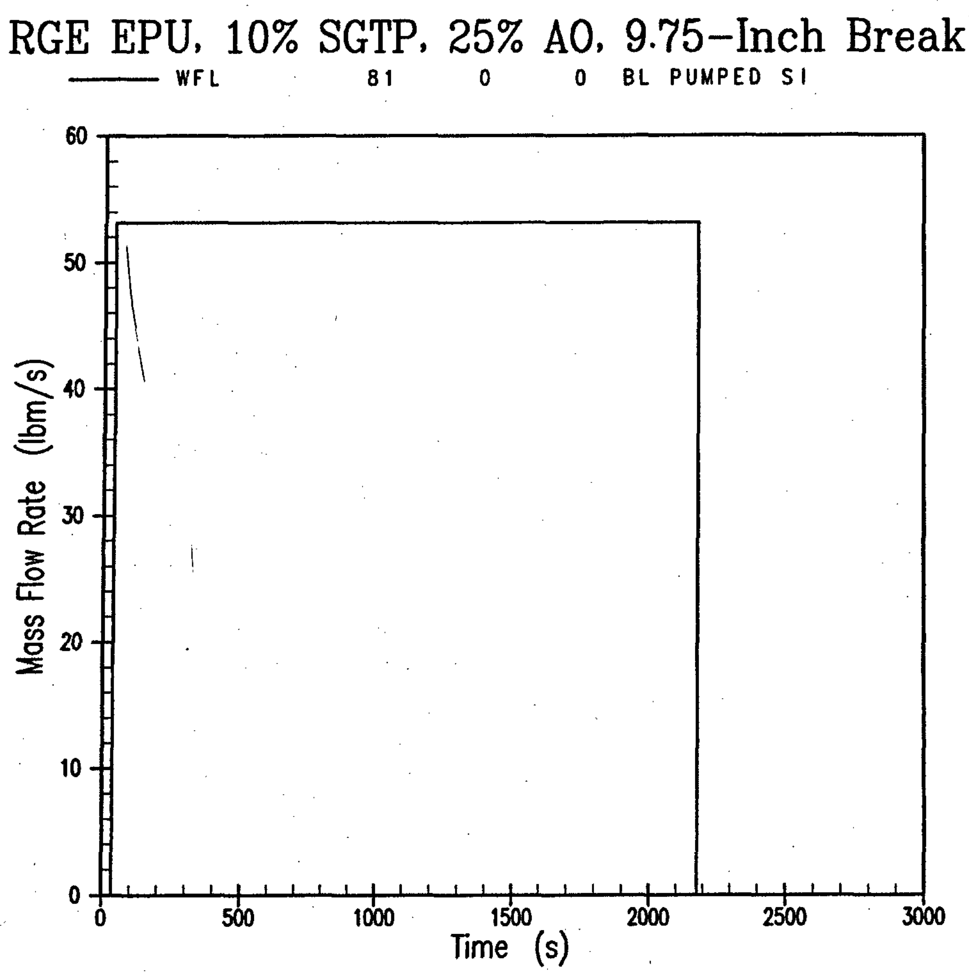


Figure 5-23: 9.75-Inch Intact Loop High Head SI Mass Flow Rate

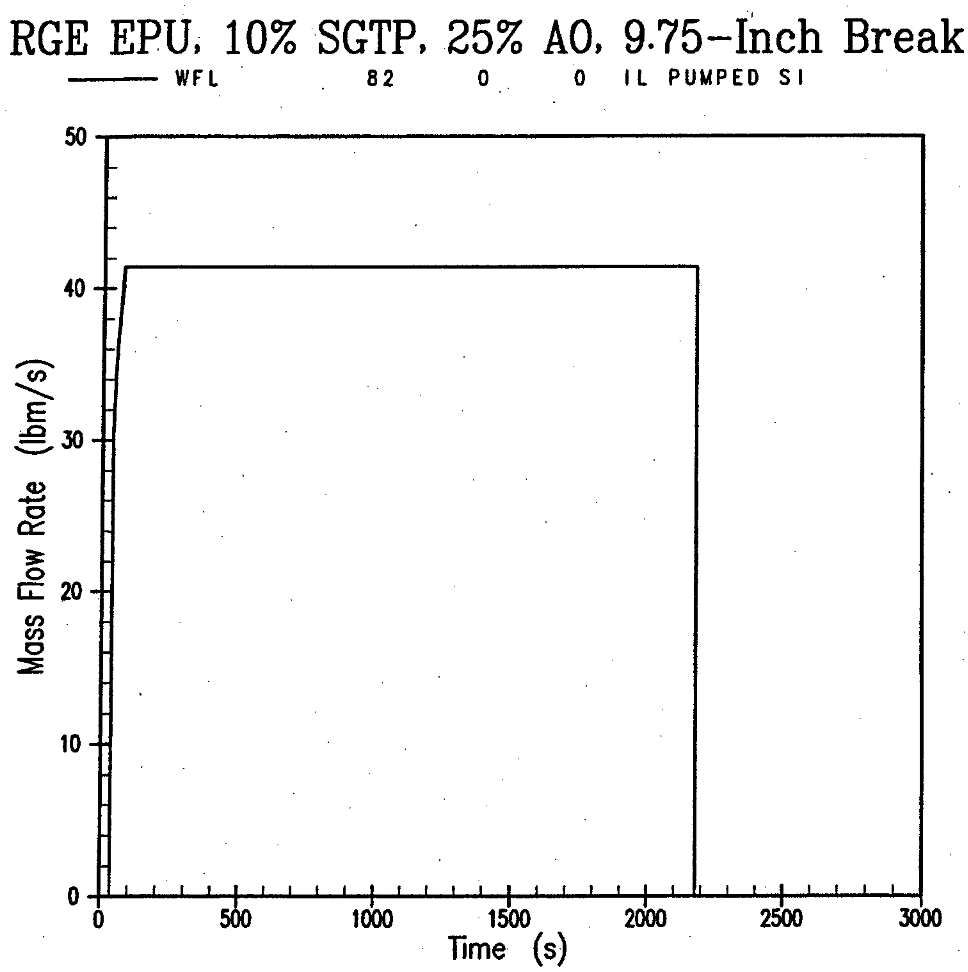
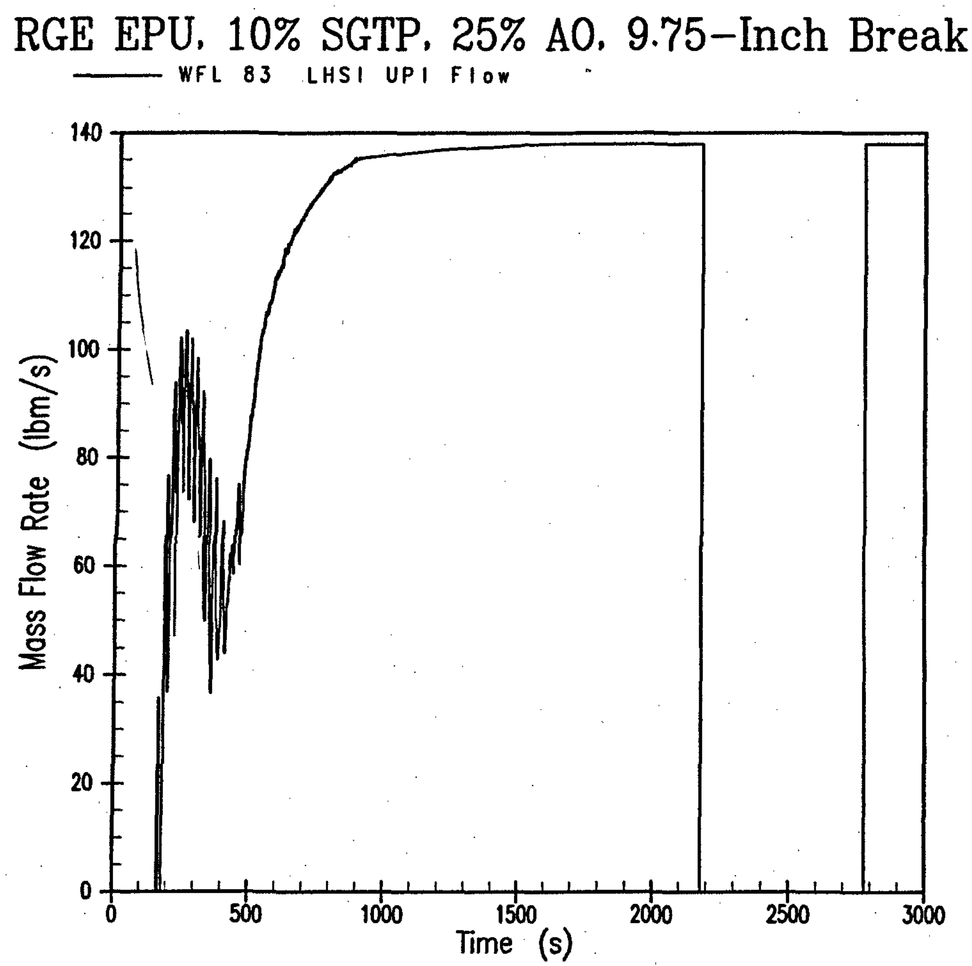


Figure 5-24: 9.75-Inch Intact Loop Low Head SI Mass Flow Rate



Attachment 1

Response to October 28, 2005 Request for Additional Information

RAI #6:

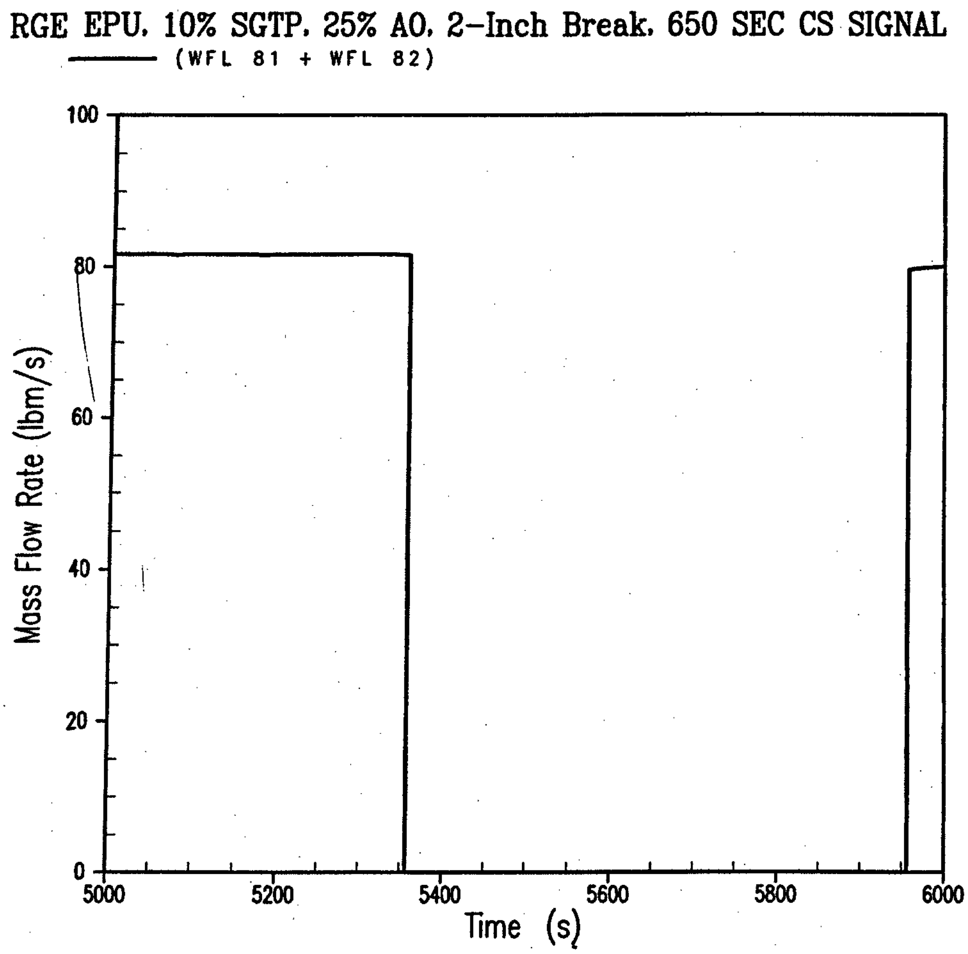
6.1 The 2-inch break shows an interruption in SI flow for about 500 seconds while the 3-inch break shows an interruption of about 600 seconds.

6.2 For the 3-inch break, the two-phase level in the upper plenum shows very little recession when the SI flow has been terminated from 2700 to 3300 seconds. Explain and verify the insensitivity of the two-phase level due to the termination of SI flow. Provide an alternate means to verify core uncovery does not occur for all small breaks.

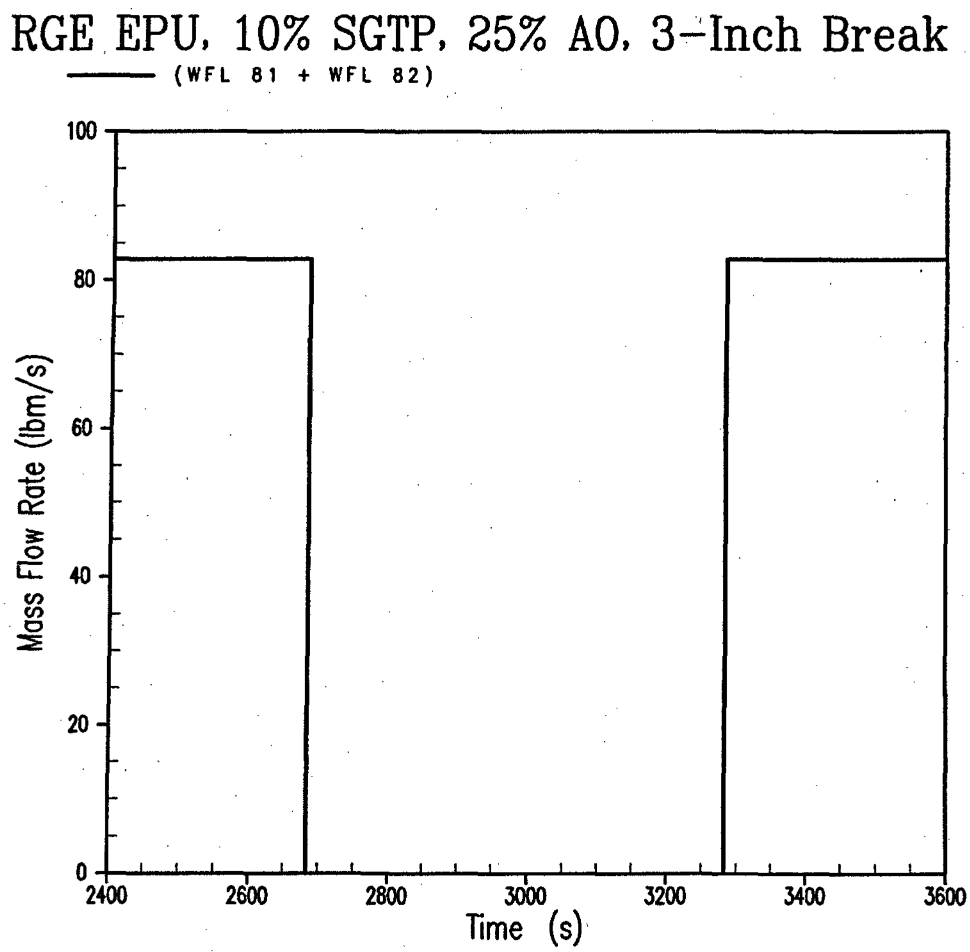
Response:

6.1 The SI flow interruption for the 2- and 3-inch break cases were assumed at 600 seconds in the NOTRUMP runs. Figures 6.1-1 and 6.1-2 present a close up view of the SI flows for the 2- and 3-inch break cases during the time of SI interruption. From these figures it can be seen that the 600 seconds SI flow interruption was correctly modeled for both the 2- and 3-inch breaks.

Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 6.1-1: 2.0-Inch Break SI Mass Flow Rate



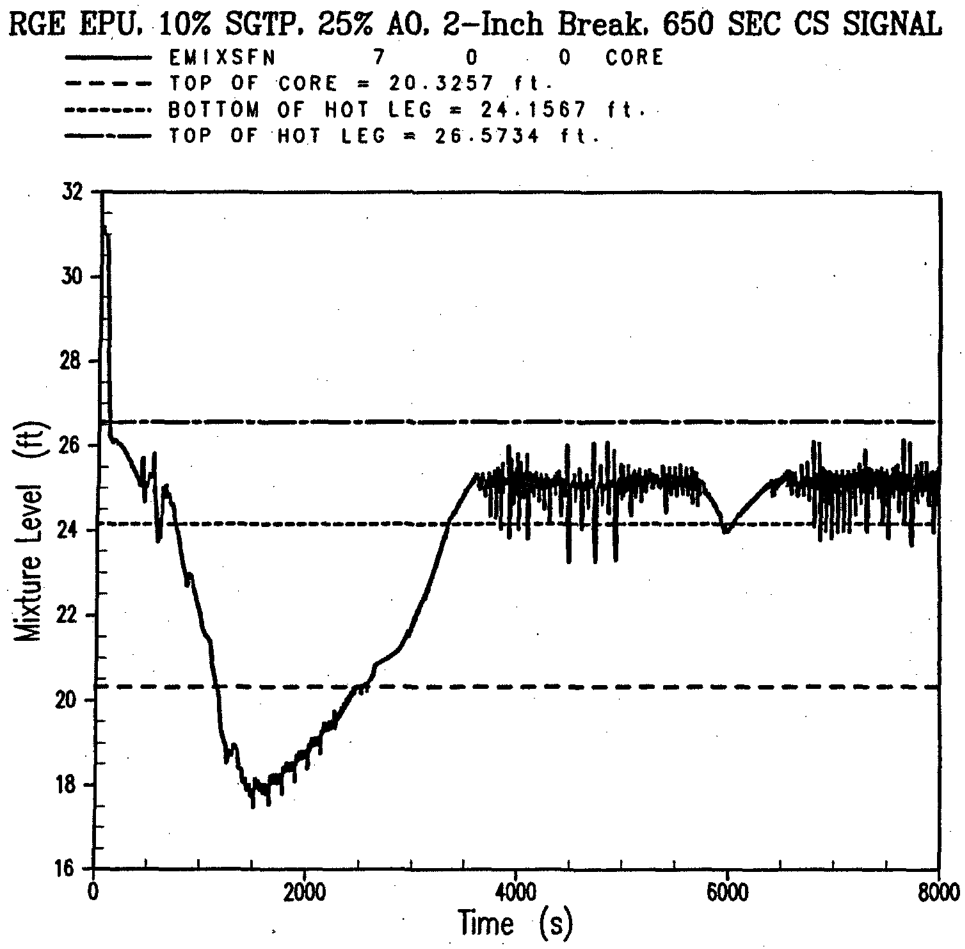
Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 6.1-2: 3.0-Inch Break SI Mass Flow Rate



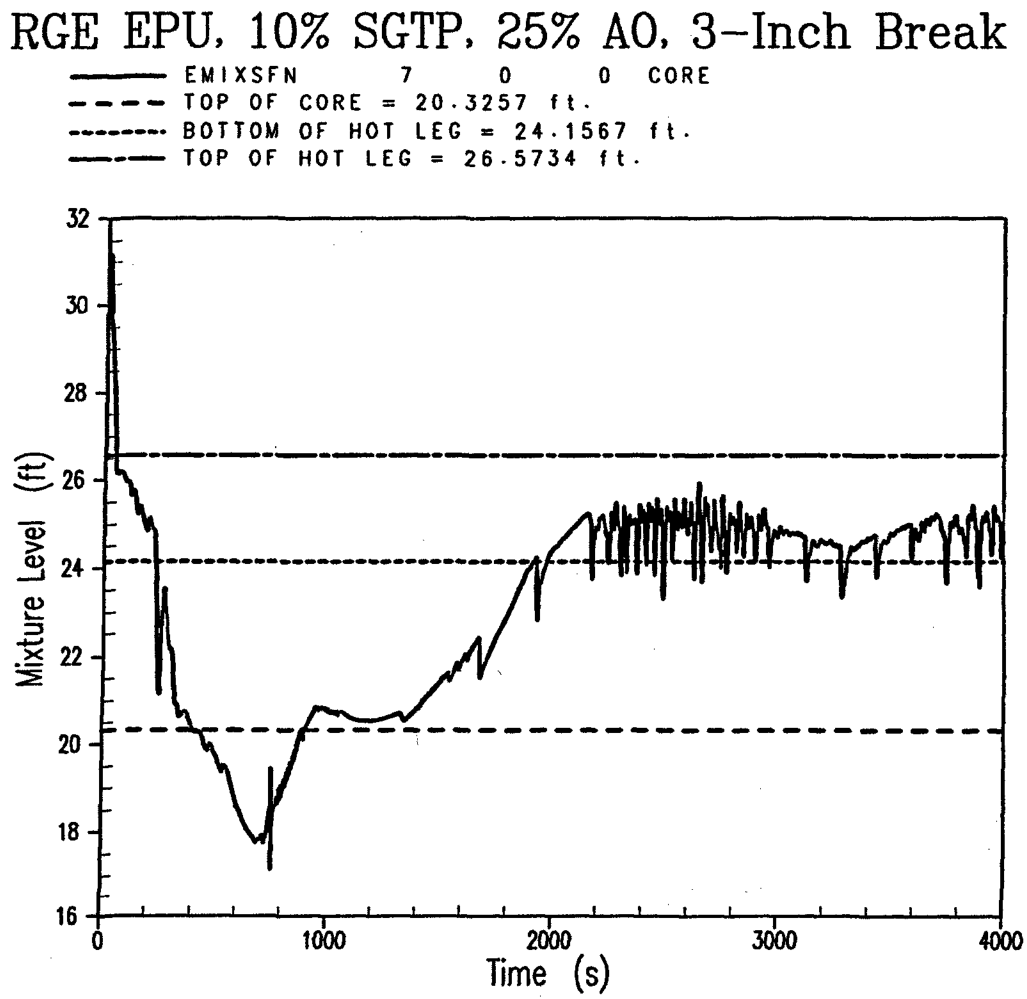
Response:

6.2 Figures 6.2-1 and 6.2-2 show the two-phase core mixture level for the 2- and 3-inch break cases. These figures show that prior to the SI interruption, the two-phase mixture level is in the hot leg region. The mixture level begins to decrease when the SI flow interruption begins and continues to decrease to approximately the bottom of the hot legs when the switchover interruption ends after 600 seconds. Figures 6.2-3 – 6.2-4a show the integrated flow between the hot legs and the upper plenum. From these figures it can be seen that there is backward flow from the hot legs into the upper plenum during the interruption period (Figures 6.2-3a and 6.2-4a). Figures 6.2-5 and 6.2-6 show a comparison of the collapsed liquid levels in the active fuel and downcomer regions for the 2- and 3-inch breaks. As can be seen, there is also considerable inventory available in the downcomer region during the SI interruption period. These two factors would account for the minimal reduction in the core mixture level observed for the 3-inch break case.

Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 6.2-1 2.0-Inch Break Mixture Level in the Core



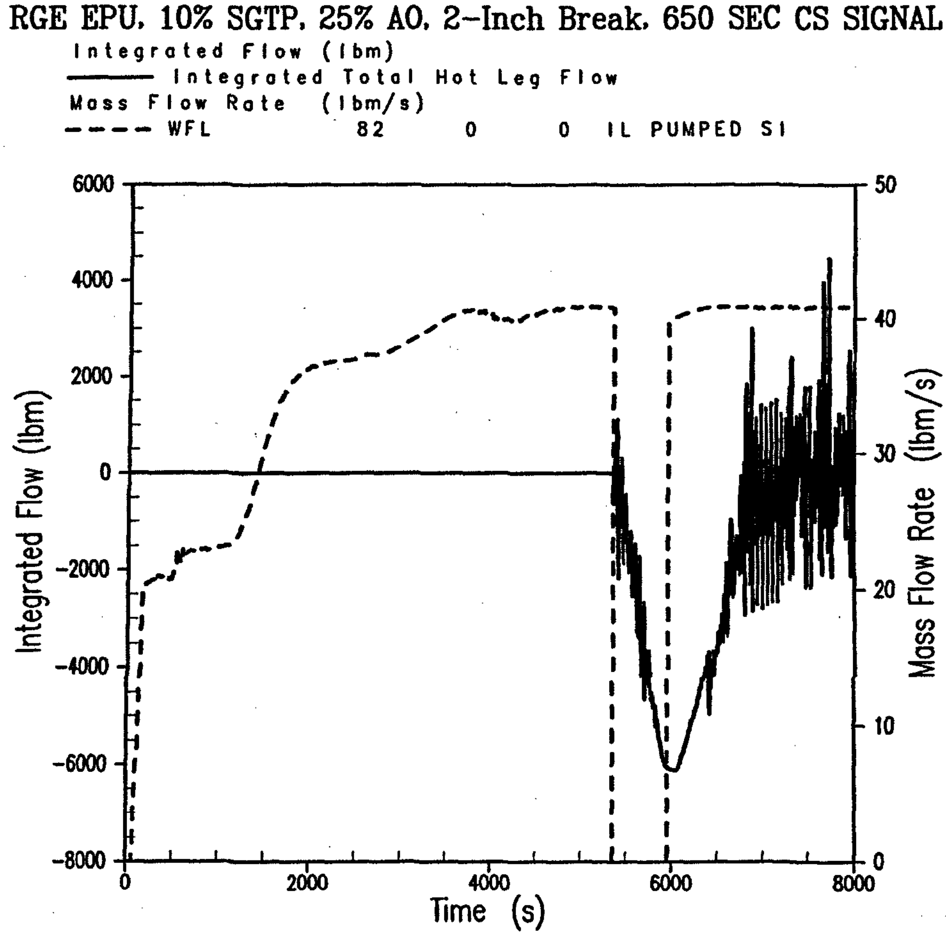
Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 6.2-2 3.0-Inch Break Mixture Level in the Core



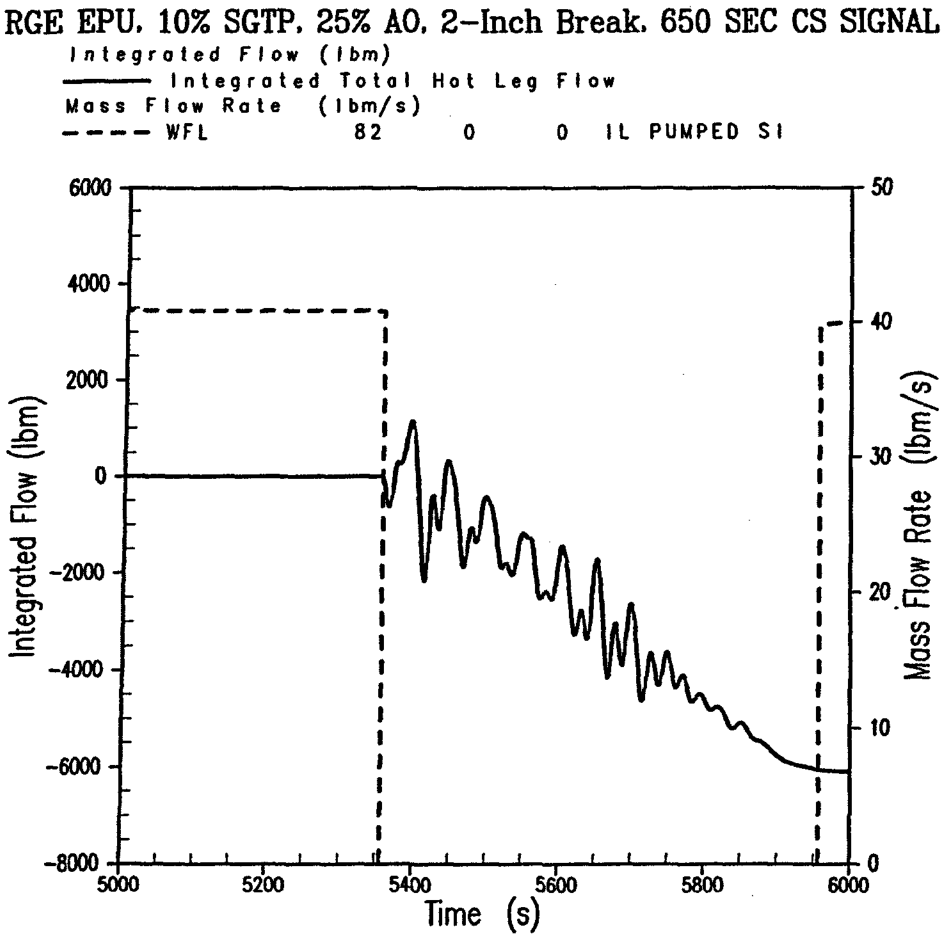
Attachment 1

Response to October 28, 2005 Request for Additional Information

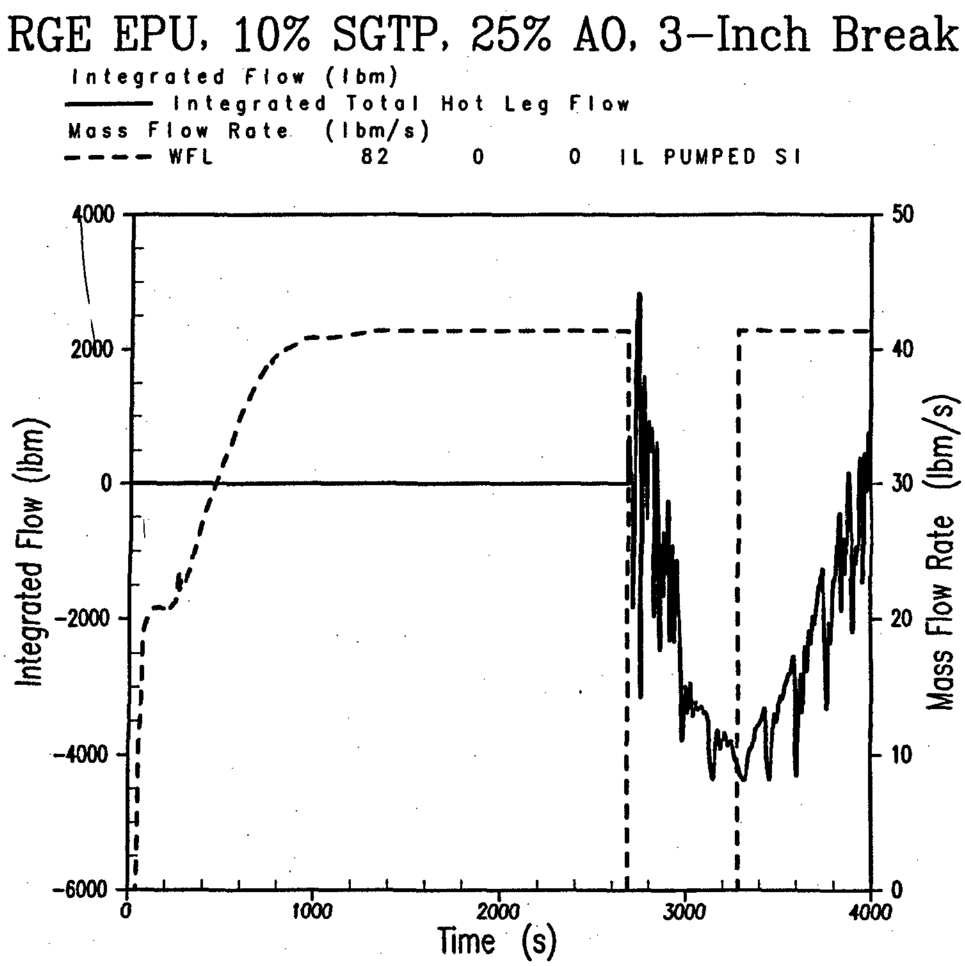
Figure 6.2-3 2.0-Inch Break Integrated Hot Leg Flow vs. SI Flow



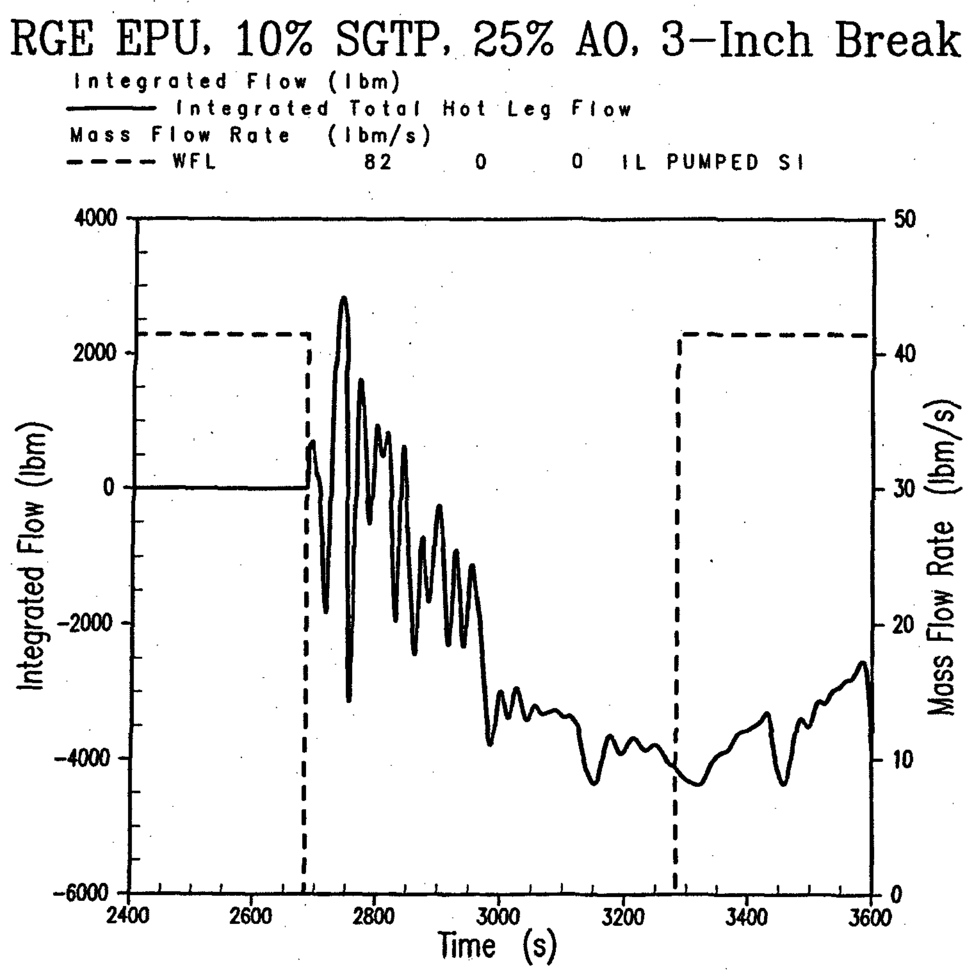
Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 6.2-3a 2.0-Inch Break Integrated Hot Leg Flow vs. SI Flow



Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 6.2-4 3.0-Inch Break Integrated Hot Leg Flow vs. SI Flow



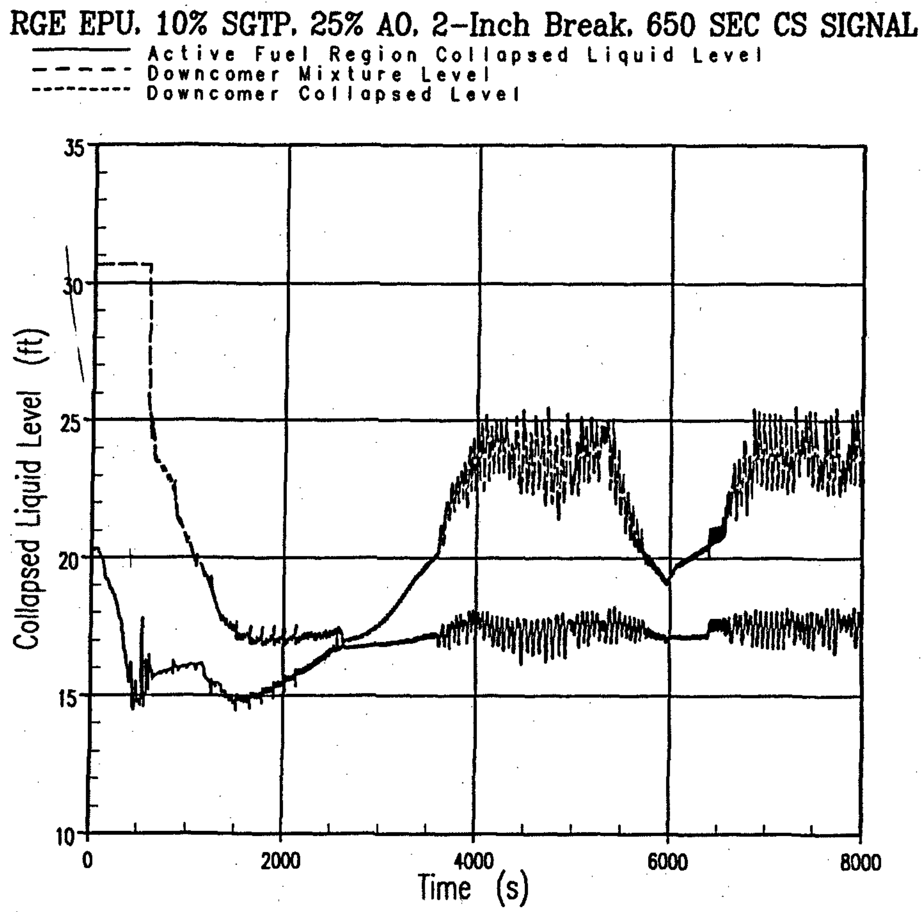
Attachment 1
Response to October 28, 2005 Request for Additional Information
Figure 6.2-4a 3.0-Inch Break Integrated Hot Leg Flow vs. SI Flow



Attachment 1

Response to October 28, 2005 Request for Additional Information

Figure 6.2-5 2.0-Inch Break Active Fuel Region/Downcomer Collapsed Liquid Level

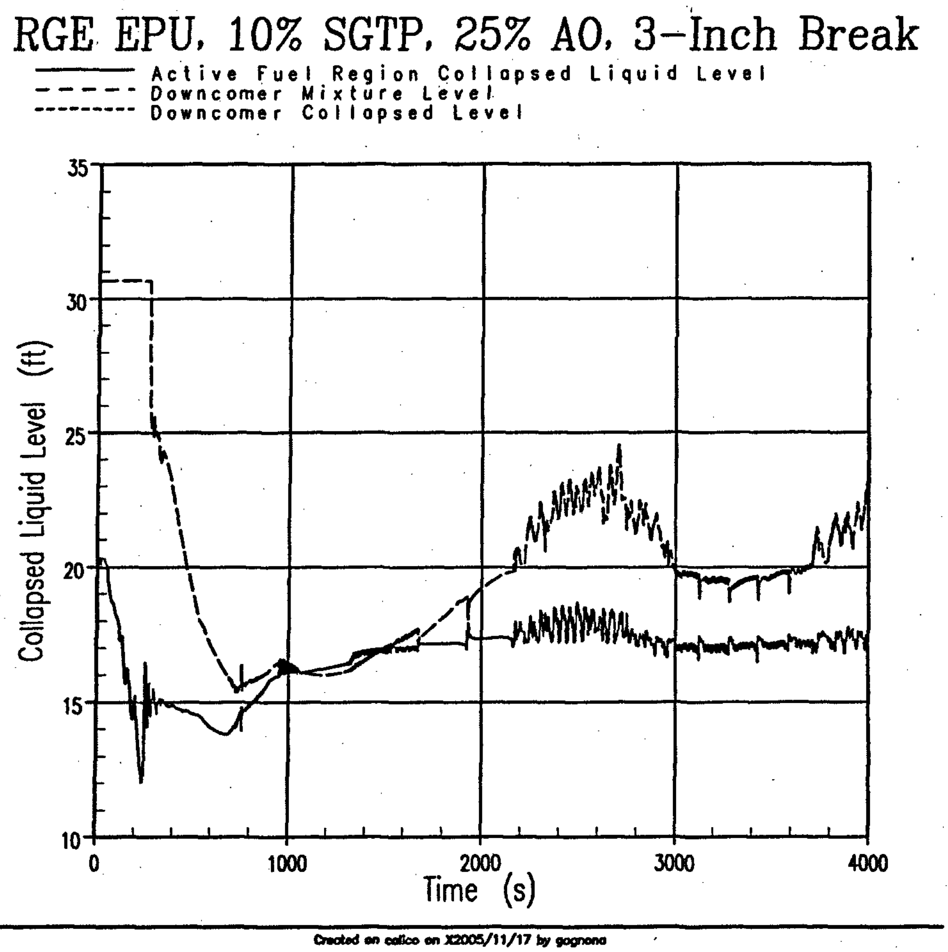


Created on calico on X2005/11/17 by gognono

Attachment 1

Response to October 28, 2005 Request for Additional Information

Figure 6.2-6 3.0-Inch Break Active Fuel Region/Downcomer Collapsed Liquid Level



RAI #7: For larger breaks, the termination of SI flow is expected to have a more significant impact. Identify the impact of SI flow termination on the largest SBLOCA and the limiting LBLOCA.

Response:

The largest SBLOCA case simulated for R.E. Ginna is the 0.5 ft^2 break. The results of the 0.5 ft^2 break are discussed in the response to question 5. In particular Figure 5-4 shows insignificant core uncovery for this case and hence no rod heat up calculation was necessary for this case. Therefore, it can be concluded that the SI flow termination does not have a significant impact on the NOTRUMP results.

Attachment 1

Response to October 28, 2005 Request for Additional Information

Post-LOCA Long-Term Cooling

1. The long-term cooling analysis and boric acid precipitation analysis are based on a 1975 Westinghouse letter that the NRC staff does not consider acceptable. Submit a new analysis that contains the following considerations for performing the long-term cooling analyses:

   a. The mixing volume must be justified and the void fraction must be taken into account when computing the boric acid concentration.

   b. Since the mixing volume is a variable quantity that increases with time, the boric acid concentration just prior to the switch to simultaneous injection should reflect the variable size of the mixing region set by the pressure drop in the loop. The fluid static balance between the downcomer and inner vessel region (i.e. lower plenum, core, and upper plenum regions of the vessel) can then be performed taking into account the loop pressure drop at a given steaming rate to compute the mixture volume in the core and eventually the upper plenum regions. The concentration in the resulting mixing volume just prior to expansion into the upper plenum (which will cause a sudden decrease in concentration due to the large area change in the upper plenum) must be shown to remain below the precipitation limit.

   c. The precipitation limit must be justified, especially if containment pressures greater than 14.7 psia are assumed or additives are contained in the sump water.

   d. The decay heat multiplier, as required by Appendix K to 10 CFR 50.46, must employ a *multiplier of 1.2 for all times. 10CFR50.46(b)(5) states that "decay heat shall be removed* for an extended period of time required by the long lived radioactivity remaining in the core." Appendix K , (I)(A) (4) entitled "Fission Product Decay" states that "the heat generation from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time."

Response:

New boric acid analyses were performed that include the considerations listed above. Specifically two boric acid analyses were performed; large break LOCA and small break LOCA. A detailed discussion of results of these analyses will be presented in the response to RAI #2 and RAI #3. Below is a summary of the analyses relative to items "a" thru "d" considerations listed above.

For large hot leg breaks, cold leg injection will provide flushing flow during the injection mode. After re-alignment to sump recirculation cold leg injection will be terminated if system pressure is below UPI cut-in pressure. Once cold leg injection is terminated, boric acid will begin to build up in the core. The large break LOCA boric acid analysis determines the appropriate time for restoring cold leg injection.

In regards to the "a" thru "d" considerations listed above, the following methodology was used for the large break LOCA boric acid analysis.

a. Mixing volume / void fraction / core boiloff was extracted from WCOBRA/TRAC hot leg break case.

b. Mixing volume was varied with time as predicted by WCOBRA/TRAC. This mixing volume reflects system effects such as loop resistance.

c. The boric acid solubility limit was based on atmospheric conditions. Neither containment overpressure nor sump additives were credited.

d. Appendix K decay heat was used in all calculations.

Small Break LOCA Boric Acid Analysis

For small cold leg breaks where the system pressure stays above UPI cut-in pressure, boric acid will begin to build up in the core immediately after blowdown. Within 1 hour, operators will begin to depressurize the reactor coolant system in accordance with Emergency Procedure ES-1.2, Post LOCA Cooldown and Depressurization. So long as the system can be depressurized prior to reaching the boric acid atmospheric solubility limit, there is no potential for boric acid precipitation even with rapid inadvertent system depressurization using the pressurizer PORVs. The small break LOCA boric acid analysis determines how long the boric acid concentration remains under atmospheric solubility limit. This time is available for system depressurization.

In regards to the "a" thru "d" considerations listed above, the following methodology was used for the small break LOCA boric acid analysis.

a. Mixing volume / void fraction / core boiloff was extracted from NOTRUMP 4 inch break case as this break requires no operator action to depressurize.

b. Mixing volume was varied with time as predicted by NOTRUMP. This mixing volume reflects system effects such as loop resistance.

c. The boric acid solubility limit was taken at atmospheric conditions. Neither containment overpressure nor sump additives were credited.

d. Appendix K decay heat was used in all calculations.

2. Small breaks were not addressed. The boric acid concentration for the limiting SBLOCA needs to be evaluated. Provide a summary of the results to show that the boric acid concentration is not sufficient to cause precipitation should the operators inadvertently depressurize the reactor coolant system (RCS) in a rapid manner.

Response:

As described in the cover letter, this request will be responded to by January 16, 2006.

3. Provide information to show that for the largest break that does not actuate upper plenum injection (UPI) (where a cooldown is required) that there is sufficient time to perform this function given an appropriate precipitation time based on consideration of the four items in item 1 above.

Response:

As described in the cover letter, this request will be responded to by January 16, 2006.

4. What is the temperature of the SI water entering the core at the time of SI re-initiation at 6 hours?

Response:

For large break and intermediate break LOCAs, cold leg SI would be terminated is at the end of the realignment to sump recirculation. Cold leg SI would be re-established after 6 hours, in accordance with the Emergency Operating Procedures. At this point, the temperature of the injected SI would be the temperature RHR heat exchanger outlet temperature. For small break LOCAs, cold leg SI would not be terminated until the system is sufficient such that natural circulation will be established.

Regardless of the precise temperature of the SI injected into the cold legs, it is expected that the water entering the core region via the lower plenum would be heated to near saturation temperature. In addition to the stored energy in the metal structures in the downcomer (vessel wall, barrel, thermal shields or neutron pads), there are two other mechanisms to heat the injected SI water as it travels to the core region. They are as follows;

a. Heat transfer from the core and upper plenum regions into the downcomer water through the core-former structure or through the wall separating the upper downcomer from the upper plenum.

b. Steam flowing in the path from the upper plenum to the break will heat incoming water. This would apply to water injected into either the hot legs or cold legs.

These mechanisms provide a means of heating incoming SI water making it highly unlikely that water in the lower plenum would be significantly below saturation temperature.

5. Once UPI initiates, at what time following a large break LOCA would the core steaming rate be insufficient to entrain the hot-side injection?

Response:

As described in the cover letter, this request will be responded to by January 16, 2006.

6. What are the guidelines for depressurizing the RCS below 140 psia? Describe the emergency operating procedure (EOP) requirements to accomplish this? Is there a time constraint for initiating a cooldown? Does the cooldown consider a failure of the steam generator atmospheric dump valves (ADVs)?

Response:

Operators are trained and procedures are structured to rapidly depressurize the reactor coolant system during a LOCA. Emergency Procedure E-0, Reactor Trip or SI, will be implemented immediately after the reactor trip. In this procedure operators verify operation of Engineered Safety Features (ESF) equipment and diagnose the LOCA event. By experience in the simulator, after 10 minutes operators will transition to Emergency Procedure E-1, Loss of Reactor or Secondary Coolant, where additional equipment is verified operational to enhance the cooldown (charging, instrument air, service water, etc.) and the need to cooldown and depressurize is identified. Again, based on simulator experience, operators will transition to Emergency Procedure ES-1.2, Post LOCA Cooldown and Depressurization, in approximately 30 minutes. While in ES-1.2 operators will commence the plant cooldown and depressurization. Experience with simulator training crews indicates that the RCS cooldown will be started within one hour of the break occurring. There is no time constraint for initiating a cooldown and one is not necessary. Operators are aware of the importance of depressurizing the RCS in order to stop the loss of inventory and a time constraint for commencing a cooldown could be a distraction.

Emergency Operating Procedure ES-1.2 requires cooldown of the RCS at the maximum cooldown rate allowed by Technical Specifications. Operators are instructed to use both atmospheric dump valves (ADVs) to maximize the cooldown rate should the normal condenser dump valves not be available. Operators are also instructed to operate the ADVs locally if an active equipment failure causes remote operation to fail.

7. If the RCS refills early during the cooldown for very small breaks and hot water is trapped in the pressurizer with a saturation temperature above the entry temperature to start RHR, how is the pressurizer eventually cooled to initiate residual heat removal (RHR)? Explain the method to reduce RCS pressure under these conditions. Can cooldown be accomplished before the condensate storage tank supply is exhausted?

Response:

For very small breaks, the pressurizer is cooled using normal spray if reactor coolant pumps are operating. If normal spray is not available, a PORV is opened to depressurize and cool the pressurizer. If no PORV is available, auxiliary spray would be used with charging pumps available. Assuming none of the aforementioned methods are available, the Technical Support Center would be consulted and would likely recommend raising and lowering the pressurizer level to mix the hot water in the pressurizer and the cooler water of the RCS. These actions are taken concurrent with continuing the RCS cooldown by steaming from the steam generators.
The capacity of one Condensate Storage Tank (CST) is adequate to maintain the RCS in hot standby for two hours. Although two CSTs should be available, if only one tank is available it is

possible that the water supply from that tank could be exhausted. In this case lake water is used as the ultimate heat sink to supply makeup water using the standby auxiliary feedwater pump.

8. What precipitation limit is used for LBLOCAs and intermediate-break LOCAs? Explain whether debris in the sump water affects this limit and the time variation in boric acid concentration.

Response:

The boric acid precipitation limit used in both the large break LOCA and small break LOCA boric acid analyses is the experimentally determined solubility limit at the boiling point of a saturated boric acid and water solution at atmospheric conditions. The increased boric acid solubility that would result from containment overpressure or the increased boric acid solubility that would result from sump additives was not credited.

While there are no known comprehensive industry studies of the effect of sump debris on boric acid precipitation characteristics, some relevant observations were made in the Fauske solubility tests discussed in Fauske Report, FAI/05-67, "Increase in Solubility Limit as a Result of Sodium Hydroxide (NaOH) in the Containment Sump Water," dated June 2005. The Fauske test summary report indicated that powdered impurities introduced into a saturated boric acid water solution did not cause boric acid to precipitate out of solution. Questions concerning this subject have been discussed with the NRC and the NRC has concurred that questions concerning this subject are not directly related to the uprating of the plant. The NRC has also indicated that an immediate response to these questions is not required for the safe operation of the plant. The low safety significance of questions related to boric acid precipitation is based on the following safety assessment.

Safety Assessment of Generic Questions Regarding the Effect of Sump Debris on Post-LOCA Boric Acid Precipitation

During recent PWR Extended Power Uprating license amendment requests, discussions between the NRC, licensees and safety analysis vendors have raised generic questions regarding the effects of sump debris on the potential for boric acid precipitation after a LOCA. These questions do not constitute a significant safety concern based on the following:

- There is low probability of a large break LOCA where conditions leading to significant boric acid accumulation may be encountered. Small break LOCA scenarios are less likely to result in boric acid precipitation due to the higher boric acid solubility limit at higher pressures and the beneficial effect of cooldown procedures on reducing core boiloff.

- Some of the transient behavior uncertainties that are treated conservatively, if fully understood and incorporated into boric acid precipitation analyses, would have a beneficial effect on the results. These include; liquid entrainment around the loop, boron carryover in the steam, containment overpressure, and mixing in regions outside the core, upper plenum and portions of the lower plenum.

- If best estimate or realistic assumptions are used in the analyses, the predicted boric acid precipitation times would be long after the typical EOP action times specified in plant's EOPs. The most significant realistic assumptions are best estimate decay heat and nominal boron concentrations for containment sump boration and dilution sources.

- Observations from a test facility that represented a typical Westinghouse 3-loop PWR indicate that boric acid precipitation would not occur for at least 24 hours after a large break LOCA (WCAP-16317-P, "Review and Evaluation of MHI BACCHUS PWR Vessel Mixing Tests," November 2004). These observations are indicative of the potential for boric acid precipitation in 2-Loop and 4-Loop plant designs as well.

- The EOPs for Westinghouse-designed PWRs provide the necessary steps and logic to implement actions that will mitigate the potential for boric acid precipitation in the core. EOP actions are particularly relevant to small break LOCA scenarios where plant depressurization and cooldown is a priority. EOP actions would reduce or eliminate the potential for boric acid precipitation for many of the scenarios considered in the boric acid analyses.

- In the event that boric acid precipitation should occur, it is unlikely that core cooling would be compromised. It is expected that the boric acid precipitate would tend to plate out on the colder structures, accumulate in the lower plenum, or would collect at the top of liquid mixture level (Fauske Report FAI-05-13, "Solubility of Boric Acid in Water with TSP Added," 02-04-05 and CE Report LOCA-75-127, "Post LOCA Boric Acid Mixing Experiment," 10-06-75). It would take significant boric acid precipitate to totally block water from getting to the core.

9. For intermediate breaks that produce RCS pressures above the UPI shut-off head, the SI pumps are secured if the need to switch to recirculation should occur. Explain the procedure for assuring RHR can be actuated if the SGs have to be cooled down, especially with the loss of offsite power and failure of one of the ADVs. What is the timing for cooldown of the SGs to assure RHR will be operating when the switch to recirculation is made?

Response:

Operators will depressurize both steam generators using both ADVs. The single failure of an active component is overcome by local manual control of the ADVs as necessary. A high-head recirculation flow path is established upon entering the recirculation mode if the RCS pressure is above the shutoff head of the RHR pumps. The high-head recirculation flow path involves lining up the discharge of the RHR pumps to the SI pump suction. This lineup produces adequate injection pressure to assure continued injection flow at elevated RCS pressure.

As previously stated, operators will maximize efforts to cool and depressurize the RCS. Given high-head recirculation, it is not necessary to depressurize the RCS below the RHR pump shutoff at the time of recirculation in order to continue injection.

10. Following LBLOCAs, borated water is entrained in the steam exiting the core, which can enter the SG tubes. Since the secondary side is at high temperatures, the borated water can be boiled-off leaving behind the boric acid. What happens to the boric acid in the SGs? Can boric acid build-up sufficiently to increase the loop resistance and depress the two-phase level in the core?

Response:

There are no known industry studies of the potential for boric acid plating out on spacer grids, fuel alignment plate, or any structures in the upper plenum prior to the boric acid solubility limit being reached. Once residual heat is removed from the non-heat-source structures (such as spacer grids and fuel nozzles), there would be insufficient surface boiloff to create large amounts of deposition prior to hot leg switchover time. Concerning the possibility of plating out boric acid on the fuel rods, any plating on the fuel rods prior to fuel quench would likely be a boron compound other than orthoboric acid (discussed on page 220 of P. Cohen, P., 1969, Water Coolant Technology of Power Reactors, Chapter 6, "Chemical Shim Control and pH Effect,") since the melting point of orthoboric acid is 340°F. After core quench, boron compounds that come out of solution might be expected to return to solution quickly since locations below the mixture level would be exposed to a dilute core region solution. Boron compounds that return to solution and flow back into the vessel would be consistent with the boron acid assumptions used in the calculations (i.e. boric acid contained in core boiloff remains in the core region).

Boric acid plate-out in the SG tubes could occur only during the period where there is significant liquid entrainment around the loop and only when the SGs act as a heat source. Under these conditions, the rate at which boric acid accumulates in the core would be greatly reduced for two reasons; the liquid entrainment passing through the SGs that is not vaporized would remove boric acid from the core region, and the boric acid plate-out on the inside of the tubes would be removing boric acid from the core region. Once the SGs act as a heat sink, condensation on the inside of the tubes would return any boric acid to solution. Similarly, boric acid plate-out on other structures would return to solution when the residual heat in the structure is removed and the plated surfaces are exposed to a low-quality 2-phase mixture or liquid dilute solution. In response to this RAI, a review of the Ginna large break boric acid analysis was made in order to estimate the volume of boric acid that might be deposited in the SGs during the period when entrainment around the loop would be significant. Calculations have shown that 1 hr. 12 min. is the time after which the core steaming rate is insufficient to support hot leg entrainment. At 1 hr. 12 min. a total of about 1643 lbm of boric acid would be left behind in the core as the result of boiloff. If it is assumed that 5% of that boric acid is entrained around the loops and left in the steam generators, the total boric acid per steam generator is 1643 * 0.05 / 2 = 41 lbm. Assuming a density of boric acid of 50-100 lbm/ft^3, the resulting volume per steam generator is less that 1 cubic foot. If the 1 cubic foot of boric acid were to be deposited over a 10 ft length of steam generator tubing, the thickness would be 1 ft^3 / ((4765 tubes x 0.664 in ID / 12 in/ft x π) * 10 ft) x 12 in/ft = 0.0014 inches thick. This is not a sufficient deposition to cause a significant increase in loop resistance.

11. Following a SBLOCA, the RCS can boil for an extended period of time. While the boric acid will remain in solution at the high temperatures, the sudden need to depressurize the RCS rapidly could cause an inadvertent precipitation. Explain what guidelines or EOP directives are available to the operators to assure this does not happen.

Response:

Analysis using the NOTRUMP computer code demonstrates that for smaller breaks the RCS will be filled, natural circulation will start and core boiling will cease before the boric acid concentration exceeds the precipitation limit for atmospheric pressure. For larger breaks the RCS will be depressurized to the point where simultaneous injection will begin, also before the boric acid concentration exceeds the precipitation limit for atmospheric pressure. Boron concentration will at no time exceed the saturation limit for a depressurized state and there is no concern that rapid depressurization of the RCS could cause boron precipitation.

12. Explain how the EOPs guide the operators to assure them that they can refill the RCS for all small breaks and re-establish natural circulation to flush the boric acid from the vessel.

Response:

Operator training and procedures establish and maintain a high priority on depressurizing the RCS, returning normal level to the pressurizer and maintaining subcooling. If the break size is small enough to support refilling the RCS, natural circulation will begin and core boiling and boron buildup will cease. If the break size is too large to support refilling the RCS, the RCS will be depressurized to the point where simultaneous injection will prevent boric acid buldup.

RAI #13: What is the size of the bottom mounted instrument tubes? Are failed instrument tubes in the bottom of the head part of the design basis? If so, was a failed tube analyzed at extended power uprate conditions? Also, is operator action required to assure the core remains below 2200F with one tube assumed failed?

Response:

No plant specific analyses were performed for the R. E. Ginna Nuclear Power Plant (RGE) as part of the Extended Power Uprate (EPU) program with regards to failures of bottom mounted instrumentation (BMI) tubes. However, in response to the Davis Besse and South Texas Unit 1 events, a comprehensive Westinghouse Owners Group (WOG) program for both traditional Westinghouse and Combustion Engineering System 80 reactor vessels was developed several years ago to assess the impacts of a postulated leak or failure of one or more BMI nozzles. The WOG program included the following tasks.

- *Historical information review to determine the extent to which BMI breaks have been analyzed and* to determine the effort required to address the potential consequence of a BMI failure.
- Small Break LOCA analyses to evaluate the potential effect of various failures of BMI tubes. These results are then utilized to support a probabilistic risk assessment of BMI failures.
- A materials assessment which evaluates the potential for failure based on phenomenological considerations. This includes Failure Modes and Effects Analysis (FMEA)
- Evaluation of the effectiveness of the Emergency Response Guidelines (ERGs) in dealing with this postulated scenarios and provisions for recommending modifications to the guidance.

During the execution of this program, various organizations discussed the benefits of providing a coordinated fleet-wide response to BMI related issues. As such, a joint effort between the WOG, B&W Owners Group (BWOG) and MRP was developed to provide this response. The effort culminated in the development of internal documentation which supports the various conclusions reached in regards to these issues. A meeting to present the WOG and BWOG results to the NRC was held on September 30 of this year. A summary of the observed LOCA response is provided below:

- Different plant groups demonstrate similar responses to the BMI small LOCA event. Evaluated thermal hydraulic analysis cases representative of RGE show that a Bottom Mounted Nozzle (BMN) break of approximately 1.0 inch equivalent diameter can be withstood under timely operator action (45 minutes) to depressurize without core uncovery. (Note that the Ginna bottom-mounted instrument tubes are 0.375 inches ID.)

14. Explain the impact of the refilling of the loop seals (for breaks on the side of the cold leg) on the mixing volume and boric acid concentration.

Response:

Loop seal refilling would temporarily increase the loop pressure drop and would depress the mixture level in the core. Loop seal refilling would be significant to the calculations only if the loop seal closure was sustained. Neither LOCA ECCS Evaluation Models (EMs) nor observations during the ROSA tests (discussed in Letter from Westinghouse to NRC, NSD-NRC-97-5092, "Core Uncovery Due to Loop Seal Re-Plugging During Post-LOCA Recovery, March 1997) predict sustained loop seal closure, but instead predict cyclic loop seal refilling and clearing. Cyclic loop seal refilling/clearing would promote mixing in the vessel by forcing liquid from the core region to the lower plenum and downcomer. Effective mixing resulting from this type of oscillatory behavior was observed in the modified VEERA facility tests (J. Tuunanen, et al., Experimental and analytical studies of boric acid concentrations in a VVER-440 reactor during the long-term cooling period of loss-of-coolant accidents, Nuclear Engineering and Design, Vol. 148, 1994, pp. 217-231).