ANALYTICAL JUSTIFICATION FOR THE
TREATMENT OF REACTOR COOLANT PUMPS
DURING ACCIDENT CONDITIONS

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1. INTRODUCTION

The criteria for resolution of TMI Action Plan Item II.K.3.5, "Automatic Trip of Reactor Coolant Pump," were stated in letters[1,2] dated February 8, 1983 from Mr. Darrel G. Eisenhut (NRC) to all applicants and licensees with B&W-designed nuclear steam systems (NSSs). Those letters requested each utility to provide an individual submittal or reference to a generic submittal providing the technical justification for the treatment of reactor coolant (RC) pumps during transients and accidents.

This report provides the technical justification for the treatment of RC pumps during transient and accident conditions. This report presents the generic analyses of steam generator tube rupture for the 177-fuel assembly (FA) lowered-loop design and SBLOCA results of both raised- and lowered-loop 177-FA plant designs. These analyses support the tripping of all four RC pumps manually on indication of "loss of subcooling margin."

Generic aspects of the B&W-designed NSSs as they relate to the treatment of RC pumps are discussed in this report with the intention that this report will be supplemented by each utility member of the B&W Owners Group in providing a response to NRC Generic Letter 83-10.[1,2] Where an individual utility commitment is requested, it will be addressed on a plant-specific basis by each utility, and is not included in this report.

2. SUMMARY AND CONCLUSION

It is the position of the B&W Owners Group and B&W that the tripping of all four RC pumps is recommended following indication of a small break loss-of-coolant accident (SBLOCA) and that it can be achieved safely and reliably by the operator. This report demonstrates that the concept "loss of subcooling margin" is an appropriate signal to alert the operator of the need for pump trip and meets the intent of the criteria identified in Generic Letter 83-10.

- A loss of subcooling margin will occur for those small break LOCAs where a pump trip is required to show compliance with 10 CFR 50.46.
- As a result of best estimate SBLOCA analyses, it is concluded that times in excess of 10 minutes are available for manual operator action to trip the RC pumps following indication of a loss of subcooling margin. Furthermore, within the 10-minute time frame, the most limiting break size/trip time combination yields acceptable peak cladding temperatures far below the limits of 10 CFR 50.46. This conclusion can be compared to the minimum 2-minute RC pump trip time predicted for the limiting break size analysis using conservative methods with Appendix K assumptions. The interrelationship between break size and RC pump trip time, which determines the critical region for conservatively predicted unacceptable consequences is shown in Figure 5-2.
- Adequate subcooling margin is maintained during steam generator tube rupture events for ruptures up to and including the double-ended rupture of a single tube, to ensure forced circulation throughout the event, if the operator follows procedures based on the Abnormal Transient Operating Guidelines (ATOG).
- Reducing the need to trip the RC pumps for more likely non-LOCA events such as mild overcooling events is ensured by a judicious determination

of the subcooling margin setpoint. Procedures based on ATOG provide guidance for pump restart for those events where an unnecessary pump trip might occur. Consequently, reliance on the PORV for depressurization is unlikely.

- The RC pump trip criteria based on loss of subcooling margin precludes operation of the RC pumps in a highly voided system.

3. BACKGROUND

The treatment of RC pumps during accidents and transients has received extensive attention over the past several years. The B&W Owners Group has performed analyses[3] in response to IE Bulletin 79-05B evaluating the effect of a delayed RC pump trip using Appendix K assumptions during the course of a small break LOCA accident and has determined that an early trip of RC pumps is required to show conformance to 10 CFR 50.46 for a range of break sizes. Therefore, to be consistent with the conservative analyses performed, it is the position of the B&W Owners Group that all four RC pumps should be tripped if indications of a small break LOCA exist.

The B&W Owners Group maintains that it is highly desirable to maintain RC pump operation during non-LOCA events, as an aid in the mitigation of the transient. Consistent with this philosophy, the concept of subcooling margin was chosen as an indication for the need to trip all four RC pumps. It is the intention of the report to demonstrate that this concept is consistent with the B&W Owners Group philosophy for handling RC pumps during transient conditions and complies with the intent of the criteria stated in Generic Letter 83-10. The symptom approach of subcooling margin, developed as part of the Abnormal Transient Operating Guidelines (ATOG) Program, is intended to replace the present guidelines of tripping solely on the presence of a low RC pressure engineered safety features actuation system (ESFAS) signal.

This report is based on the above positions and demonstrates that the concept of subcooling margin is an appropriate indicator of the need to trip all four RC pumps, yet still allows continued operation for steam generator tube ruptures less than or equal to a single double-ended rupture. Justification is also provided for manual initiation of RC pump trip on loss of subcooling.

4. SIGNAL SELECTION/TRIP SCHEME PHILOSOPHY

The concept of subcooling margin was chosen as an indication for the need to trip all four RC pumps during transient conditions. No partial or staggered RCP trip schemes are considered except for the extreme case where mechanical damage to the pump is likely as this adds to increased decision making on the part of the operator during transient conditions.

A primary objective of the parameter and setpoint selection is the avoidance of RC pump trip for non-LOCA events particularly steam generator tube rupture (SGTR). Realistic operator actions in accordance with the ATOG procedures are shown in section 5.3 to avoid loss of subcooling and the need to trip the RC pumps for this event. Furthermore, since subcooling margin would be quickly regained following makeup or HPI initiation, without loss of natural circulation even if the operator failed to take actions to prevent RCP tripping and ESFAS actuation, restart of the pumps would be allowed. Consequently, reliance on the PORV for depressurization is unlikely.

A loss of subcooling will always occur for small breaks that have the potential to uncover the core and violate 10 CFR 50.46 criteria if the RCPs are tripped under certain two-phase conditions. This is demonstrated in section 5.2. Hence, the loss of subcooling margin can be used as a key indicator for RC pump trip.

For most small break LOCAs (those larger than 0.05 ft^2) rapid depressurization of the RCS with little or no decrease in the RCS temperature causes the RCS pressure to quickly decrease to the saturation pressure. As shown in Table 4-1, for 177-FA plants, a loss of subcooling occurs within 17 seconds after initiation of the LOCA for small breaks 0.05 ft^2 and larger.

The RCP trip on loss of subcooling margin will exclude extended RC pump operation in a voided system. The use of loss of subcooling is a sufficient indicator to assure that the RC pumps will be tripped for all losses of primary coolant in which RC pump trip is considered necessary.

Table 4-1. RC Pump Trip Results for 177-FA LL Plants (Conservative Appendix K Analysis)

Break size, sq ft	RCS subcooled margin before accident	Time to reach indication, s: Low RCS press. RPS reactor trip setpoint (1900 psia)	Saturation in RCS	Conservative low RCS press. ESFAS setpoint (1365 psia)	70% void fraction in RCS
0.3	45.46	0.46	<0.1	7.75	130
0.2	45.46	0.46	<0.1	9.37	180
0.1	45.46	0.95	6.0	12.14	420
0.075	45.46	1.94	9.0	13.51	640
0.05	45.46	9.54	17.25	17.68	920
0.025	45.46	19.83	46.3	44.08	--

5. JUSTIFICATION FOR THE TREATMENT OF RC PUMPS

Analyses of certain small break LOCAs, combined with the assumption of only one HPI train available, have demonstrated the potential for exceeding 10 CFR 50.46 limits if RC pumps are tripped while the RCS is in a highly voided (>70%) condition. Consistent with these results the B&W Owners Group has adopted the position of tripping all four RC pumps on indication of a loss-of-coolant accident. It is also concluded that there is a wide range of transients, especially SGTR, and LOCAs where it is desirable for an operator to maintain forced circulation cooling and mixing through operation of the RC pumps.

This section addresses the need for tripping the RC pumps upon indication of a loss of subcooling margin which is indicative of a small break LOCA and the appropriate timing of this action. The ability of the signal to discriminate between a small break LOCA of the appropriate size range of interest and a more likely event, the steam generator tube rupture, is demonstrated by a realistic analyses of the design basis steam genenator tube rupture event.

A summary of the SBLOCA analysis performed in response to IE Bulletin 79-05C, evaluating the effect of delayed RC pump trip is provided in section 5.1. Section 5.1 includes model assumptions and conclusions obtained from that analysis. Section 5.2.1 describes the analytical methods and assumptions used in the best estimate SBLOCA analysis for the generic 177-FA lowered-loop plant type. The results of the best estimate SBLOCA analysis are provided in sections 5.2.2 and 5.2.4. Section 5.2.3 discusses the break size versus RCP trip times sensitivity relationships. A qualitative discussion of the applicability of the generic 177-FA lowered-loop results to the raised-loop plant is provided in section 5.2.5. Section 5.2.6 provides the salient conclusions from the SBLOCA analyses.

5.1 Conservative Analysis Performed to Evaluate the Effect of Delayed RC Pump Trip on SBLOCA

In response to Item 2 of IE Bulletin 79-05C, a conservative evaluation was performed to determine the effect of delayed RC pump trip during the course of small break LOCAs and concluded that early trip of the RC pumps was required to show conformance to 10 CFR 50.46.

A spectrum of small break LOCAs was analyzed for a 177-FA plant, both raised- and lowered-loop designs using Appendix K assumptions. The break sizes ranged from 0.025- to 0.30-ft^2. A summary of timing results is shown in Table 4-1.

The analysis method used for this evaluation is basically that described in section 5 of BAW-10104, Rev. 3, "B&W's ECCS Evaluation Model," and the letter from J. H. Taylor (B&W) to S. A. Varga (NRC), dated July 18, 1978, which is applicable to the 177-FA lowered-loop plants for power levels up to 2772 MWt. A simplified 6-node CRAFT2[5] model similar to that shown in Figure 5-1 was used for the analysis. Although this 6-node model is simplified compared to that described in the above referenced letter, it does maintain RCS volume and elevation relationships which are important to properly evaluate the system response during a small LOCA with RC pumps running. Studies were performed to benchmark the 6-node model against the 23-node small break LOCA evaluation model (referenced letter), and the results indicate that the simplified model is acceptable for the RC pump trip analyses.

The following conclusions can be drawn from the previously described analysis:

- If the RC pumps remain operative, core cooling is assured regardless of system void fraction.
- For breaks greater than 0.025 ft^2, the RCS may evolve to system void fractions in excess of 90%.
- At 40 minutes, the 0.025-ft^2 break has evolved to only a 47% void fraction. Thus, a delayed RC pump trip for breaks less than 0.025-ft^2 will not result in core uncovery.

- The potential for high cladding temperatures for a small break transient with delayed RC pump trip is restricted to a time period between that time where the system has evolved to a high void fraction and the time of LPI actuation.
- Even with two HPI pumps available, tripping of the RC pumps at the worst time (90% void fraction) results in a core uncovery period which cannot be shown to comply with 10 CFR 50.46, if Appendix K assumptions are utilized.
- There exists a combination of break sizes and RC pump trip times which resulted in violation of 10 CFR 50.46 limits. A plot of break size versus RC pump trip time which results in unacceptable consequences is shown in Figure 5-2. This curve indicates that a prompt RC pump trip upon receipt of a low pressure ESFAS signal (which is approximately the same as loss of subcooling margin) will provide compliance to 10 CFR 50.46. The minimum time available for pump trip is approximately 2 minutes and is determined by extrapolation beyond the 0.2-ft^2 case.

5.2 Best Estimate SBLOCA Analysis

A re-analysis of the small break LOCA spectrum for 177-FA raised- and lowered-loop plant designs, discussed in section 5.1, was performed from a "best estimate" approach to evaluate the impact on required times for RC pump trip under realistic conditions. As expected, the time available for tripping the RC pumps following indication of loss of subcooling margin increased. The best estimate analyses described in section 5.2.1 is an extension of the conservative analyses described in section 5.1. Realistic assumptions, described in section 5.2.1, were substituted along with a detailed methodology for evaluating the clad temperature response.

5.2.1. SBLOCA Method of Analyses (Best Estimate)

The analytical model used for the best estimate evaluation is similar to that described in section 5.1. A simplified 6-node CRAFT2 model, as shown in Figure 5-1, was used for evaluating the RCS thermal-hydraulic responses to a SBLOCA. Node 1 contains the cold leg pump discharge piping, downcomer, and RV lower plenum. Node 2 is the primary side of the steam generator and the pump suction piping. Node 3 contains the core, RV upper plenum

and the hot legs. Node 4 is the pressurizer and Nodes 5 and 6 represent the containment and the steam generator secondary side, respectively. This model, although simplified compared to those utilized in small break LOCA evaluation analyses, maintains RCS volume and elevation relationships that are important to properly evaluate the system response during a SBLOCA with the RC pumps running. Key assumptions which differ from those described in section 5.1 are discussed below.

RC Pump Model

The MIT two-phase pump model based on the steam-water test data was used in the CRAFT2 model for two phase degradation calculations. This model provides less severe flow degradation than that used in the previous analysis. Thus, it enhances heat removal by the steam generators during the forced circulation phase of transients, and results in a lower leak fluid quality. But the new pump model has a limited effect on the overall system response during the forced circulation phase of transients.

HPI Flow

The HPI system flows for generic analyses are shown in Table 5-1 and Figure 5-17. For the best estimate analyses, two HPI pumps are assumed with 30% of one HPI flow going directly out the break.

Leak Discharge Model

The selected discharge model is the orifice equation for subcooled flow and the HEM for saturated flow both with a discharge coefficient of 0.85. This is a best estimate discharge model developed for the revised SBLOCA evaluation model[6] in compliance with NUREG 0565. The conservative analyses described in section 5.1 used an orifice (subcooled) - Moody (saturated) discharge model with a discharge coefficient of 1.0. This latter model has significantly higher leak flowrates than that of the best estimate model.

Equipment Availability

Both analyses assumed that the RC pumps remain operative after the receipt of the RCS loss of subcooling margin indication. The RC pumps were assumed to trip when the RCS has evolved to high void fractions. Also, the best estimate analysis assumed that two HPI pumps and AFW are available for core

cooling as opposed to one HPI and AFW for the previous analysis. Approximately 30% of one HPI flow is assumed to spill out the break. The AFW level is raised to 50% of the operating level immediately after the pump trip.

Power Level and Decay Heat

An operating power level of 2772 MWt is assumed for all analyses, which is bounding for all 177-FA plants. The best estimate analyses were performed using the more realistic 1.0 times ANS decay heat curve instead of 1.2 times ANS decay curve, which was used in the conservative Appendix K analysis. The decay heat has no impact on the peak cladding temperature (PCT) during the early phase of transients when the RC pumps are running, but will have a significant impact on PCT after the pumps are tripped and the core is uncovered.

FOAM Calculations

Following a RC pump trip at a relatively high RCS void fraction, steam-water separation occurs and the core is uncovered. In the conservative analyses described in section 5.1 the core is assumed to undergo adiabatic heatup until the liquid level reaches the 9-ft level (equivalent to approximately 12 ft mixture level). The PCT calculations were based on an axial power peak above the core midplane obtained from a power shape encountered during normal operation (Figure 5-3) and on a heatup period from the RC pump trip to the time of 9-ft core recovery. Use of an adiabatic heatup assumption neglected any credit for the steam cooling that will occur during the core refill phase and for the radiation heat transfer. Therefore, the maximum clad temperature was overestimated and reflected in the available time for a manual RCP trip.

This best estimate analysis, however, utilized a more detailed method to determine the maximum clad temperature. This method included a FOAM2[7] analysis to determine the inner vessel mixture height. The FOAM calculation included major sources of steam production within the vessel, i.e., steam production due to decay heat and flashing. The steaming rates due to flashing in the lower plenum are assumed to provide steam cooling of the core during refill of the downcomer. Once the mixture level rises into the core region, the core cooling is by pool boiling and steam cooling in the uncovered portion of the core.

The axial power shape shown in Figure 5-3 was used in the FOAM calculation and was implemented with a radial peaking factor of 1.0. Thus, the resultant mixture height is representative of the average channel conditions. This method is conservative since a higher peaking factor for the hot channel would result in higher froth levels, thus, faster core recovery.

Heatup Calculations

The heatup calculation was performed using the THETA[8] code in the manner described in section 5 of BAW-10104.[12] The following additional assumptions were utilized in the THETA evaluation:

1. The power shape shown in Figure 5-3 was used with a radial power factor of 1.65. This maximizes steam superheating and sets the peak local power at approximately 7 ft core elevation.
2. Coolant flow and mixture level were taken directly from the FOAM calculations for the core region. Steam flow from flashing in the lower plenum was used during the refill of the lower plenum.
3. The fuel data generated by the TACO2[9] code were used in this analysis. TACO2 includes mechanistic fuel densification and fission gas release models, which predict realistic volumetric average fuel temperatures and pin pressures as a function of burnup. The fuel input data used in this analysis were calculated at the time of maximum fuel densification (~90 MWd/mtU burnup) at which the volumetric average fuel temperature is at a maximum.

5.2.2. Best Estimate SBLOCA Spectrum Results

The break sizes examined for this analysis ranged from 0.1 to 0.3 ft^2 in area and are located in the cold leg pump discharge piping. This break location has been shown to bound other locations in determining core uncovering times and associated cladding temperature excursions. Breaks of this size do not result in a rapid system depressurization to LPI actuation pressure before the system evolves to a high void fraction. In addition, based on the conservative Appendix K analysis as discussed in section 5.1, breaks in this range yield the minimum amount of time available for manual RC pump trip following a SBLOCA. The following paragraphs discuss the RCS behavior during the transient for the break size analyzed.

0.1-ft^2 Pump Discharge Break

The average system void fractions for the break spectrum analyzed is shown in Figure 5-4. The average system void is defined as:

$$\text{Average system void (\%)} = \frac{V_1 - V_2}{V_1} \times 100$$

where V_1 = total primary system liquid volume excluding the pressurizer at time = 0,

V_2 = total primary system liquid volume excluding the pressurizer at time = t.

This parameter was utilized to show the primary system liquid inventory since the primary coolant tends to be homogeneously mixed with the RC pumps running. Following the RC pump trip at 1000 seconds, the steam-water separation occurs. This results in steam discharge out the break and increases system depressurization as shown in Figure 5-5. As the primary system further depressurizes, the ECCS injection rate increases. Thus, the average system void begins to decrease approximately 200 seconds after the RC pump trip.

As seen in Figure 5-6, the liquid volume in the reactor vessel increases rapidly following the RC pump trip as a result of water in the pump suction side flowing into the reactor vessel during the period of pump coastdown. As the primary loop flow ceases, the core refill rate decreases to a rate equal to ECCS injection less core boil-off. The core is refilled to the 9-ft level with collapsed liquid approximately 140 seconds after the assumed pump trip. Once the core liquid level reaches the 9-ft level, the core is expected to be covered by a two-phase mixture and the cladding temperature excursion would be terminated. Even with the conservative assumption of adiabatic heatup during the 140-second core uncovery period, the cladding temperature remains below the limit of 10 CFR 50.46. From the results of this break size and the Appendix K analyses, it can be concluded that, for breaks 0.1 ft^2 and smaller, more than 10 minutes time is available for manual pump trip following a receipt of the loss of subcooling margin indication.

0.2 ft^2 Pump Discharge Break

A comparison of the primary system pressure responses in Figures 5-5 and 5-7 indicates that the primary system depressurization is basically independent of break size during the first few minutes into the transient when the RC pumps are running. This is because the forced circulation of reactor coolant provides adequate heat transfer to the steam generators; the primary system thus depressurizes to a pressure approximately equal to the setpoint of the SG safety valves. The primary system depressurization rate increases following a pump trip at 400 seconds (80% void fraction) as the steam is vented out the break. As shown in Figure 5-8, as a result of leak flow and boiloff greater than the HPI flow, the liquid volume decreases from the pump coastdown at 420 seconds until the CFTs are actuated at approximately 550 seconds. The collapsed liquid level rises rapidly and reaches 9-ft level at 630 seconds. A partial core uncovering time of approximately 230 seconds necessitates a detailed heatup calculation to evaluate the consequences of a delayed RC pump trip for this break size. The heatup calculations are provided in section 5.2.3 and 5.2.4.

0.25-ft^2 Pump Discharge Break

The primary system pressure response and the RV liquid volume are shown in Figures 5-9 and 5-10, respectively. The RC pumps are tripped at 300 seconds (80% void fraction). The core refill rate is slightly higher than that of the 0.2 ft^2 break. Conservative calculations show acceptable results of a pump trip at 300 seconds. However, similar to the 0.2-ft^2 case, the extended RC pump trip time will require detailed heatup calculations or justification which follows in section 5.2.3.

0.3-ft^2 Pump Discharge Break

The 0.3-ft^2 break was analyzed in order to assure an upper bound for the SBLOCA spectrum. The primary system pressure response and liquid volume in the reactor vessel are shown in Figures 5-11 and 5-12, respectively. The RC pump trip is assumed at 300 seconds (90% void fraction). The primary system depressurizes rapidly following the pump trip, and the CFTs are actuated at 360 seconds causing a rapid recovery of the core at about 470 seconds. Due to this fast recovery, this break size is not considered limiting for this analysis as discussed in section 5.2.3.

5.2.3. Break Size -- Trip Time Sensitivity

The combined results of this analysis and those from the analysis with the Appendix K assumptions, described in section 5.1, demonstrated that, for breaks 0.1-ft^2 and smaller, more than 10 minutes time is available for manual RC pump trip following an indication of the loss of subcooling margin. For breaks larger than 0.3-ft^2, the rapid system depressurization to LPI setpoint assures quick core recovery. In the range between 0.1-ft^2 and 0.3-ft^2 extended core uncovering resulted in a limited time available for manual RC pump trip when evaluated using Appendix K assumptions and an adiabatic heatup estimation of clad temperature. The detailed method used in the best estimate analysis, taking credit for steam cooling, realistically improves the previous results. The most limiting break size is selected by comparing the results of the analyses described in section 5.2.2. Figure 5-13 shows the RCS pressure response for different break sizes. For break sizes of 0.3- and 0.25-ft^2 the system pressure drops rapidly to the CFT setpoint. The pressure drop is even faster following the pump trip as the system collapses. Thus, a faster recovery is ensured. The 0.2-ft^2 break results in CFT actuation at a later time, and the core recovery is slower since the CFT flow is a function of backpressure. For breaks between 0.1- and 0.2-ft^2, the system inventory is lost at an even slower rate. An early trip will not result in complete core uncovering. Therefore, steam cooling and a froth level provide significant core cooling. A worst case would be a trip at the later time, at very high void fractions, where the core becomes completely uncovered. Figure 5-14 shows the available liquid volume in the vessel for a 0.2-ft^2 break with the RC pump trip at different times. The RC pump trip at 500 seconds represents the worst case for which the complete core uncovery period is the longest ($\sim$95 seconds). As shown, the system collapses and the volume remains just below the core bottom elevation. An earlier trip (400 second trip) results in only partial core uncovering and thus higher heat transfer. Trips at later times (600-second trip) results in shorter complete core uncovering time due to increased ECCS at lower pressure and thus higher recovery rates.

For breaks smaller than 0.2-ft^2, similar trends as those shown in Figure 5-14 will result, however, shifted to the right on the time axis. The heatup rates for these breaks will be lower than that for the 0.2-ft^2 break due

to lower decay heat rates. Therefore, the available trip time for these breaks would be longer than that for a 0.2-ft^2 break.

The range of interest, therefore, is narrowed to break areas of approximately 0.2-ft^2. The RCP trip for breaks larger than 0.2-ft^2 will result in faster core recovery relative to the 0.2-ft^2 results shown in Figure 5-14. Furthermore, as shown in Figure 5-13, the LPI will actuate earlier, providing even faster recovery. The limiting case, therefore, was selected as a 0.2-ft^2 break at the pump discharge with the RC pumps tripped at 500 seconds. This case, as shown in Figure 5-14, has the longest period of complete core uncovering and is considered representative of the most limiting condition. Therefore, this break size was chosen to evaluate the cladding temperature response for an extended core uncovery period to demonstrate acceptability of the peak cladding temperature.

5.2.4. Analysis Results

The limiting break configuration analyzed was a 0.2-ft^2 break at the pump discharge with the RC pumps tripped at 500 seconds. As discussed in section 5.2.3, this case resulted in prolonged complete core uncovering. Figure 5-15 shows the primary system pressure response. Following the initiation of the break and reactor trip on low pressure, the system depressurized very rapidly to the secondary pressure. The depressurization continued at a slower rate as the two-phase mass and energy was being released through the break. The RC pump trip at 500 seconds resulted in phase separation and higher quality discharge from the break, thus, a faster depressurization of the system. The available liquid volume drained to the bottom of the vessel and refilled the vessel to just below the bottom of the core, as shown in Figure 5-14. At this point the core became completely uncovered. However, the steam flashing in the lower plenum provided partial core cooling. The system continued to depressurize until the CFTs were actuated at 595 seconds at which time the core refill started. The core was rapdily recovered and quenched by a mixture level at about 680 seconds. This break size/RCP trip time combination resulted in a core uncovery/recovery transient that did not achieve LPI for the period of interest.

The heatup calculations were performed using the THETA code in the manner described in section 5.2.1. Figure 5-16 shows the hot spot cladding volume

average temperature response. The cladding temperature dropped from its initial value following the reactor trip and remained close to the saturation temperature until the pumps were tripped at 500 seconds. The cladding temperature started to rise at about 505 seconds during the RC pump coastdown. The steam production due to flashing in the lower plenum provided steam cooling of the core. However, the hot spot cladding temperature continued to rise and reached a maximum of 1532F at 662 seconds and began a rapid downturn as the hot spot quenched. The core was covered by a mixture level at 680 seconds and continuous core cooling was established thereafter.

5.2.5. Analysis Applicability to Raised Loop Design

The significant parametric differences between the raised-loop design plant and the preceding generic lowered-loop analysis are in the high pressure injection (HPI) delivery rate and the amount of liquid volume which can effectively be used to cool the core.

The liquid volume differential is due to the basic design difference; raised versus lowered loops. Because of the raised-loop design, system water available after the RC pumps trip will drain into the reactor vessel. For the lowered loop designs, the available water is split between the reactor vessel and the pump suction piping. Thus, for the same average system void fraction, the collapsed core liquid level following a RC pump trip is higher for the raised loop design than for the lowered loop design.

Figure 5-17 shows a comparison of HPI system capacities for the raised-loop design plant and the lowered-loop plants. As shown, the HPI pumps utilized in the raised loop design will deliver more flow versus the RCS pressure below 1300 psia. The RCS pressure generally falls below this pressure when the RCS evolves to a high void fraction that may result in an extended period of core uncovering. When the RC pumps are tripped under such condition, the HPI pumps will deliver more flow into the system and shorten the core recovery time. Therefore, the results and conclusions provided in sections 5.2.2, 5.2.3, 5.2.4, 5.2.5, and 5.2.6 are applicable to the raised-loop plant design.

5.2.6. Conclusions

The best estimate analysis of RC pump trip following a SBLOCA was performed with realistic assumptions described in section 5.2.1. The FOAM2 and THETA codes were used to calculate steaming rates, core mixture level, and cladding temperatures during the most limiting transient. The results of this analysis, described in previous sections, can be summarized as follows:

1. Following a SBLOCA, if the RC pumps remain operative, core cooling is assured regardless of system void evolution. However, continuous RC pump operation in a highly voided system is not desirable for pump integrity reasons.

2. Prompt tripping of the RC pumps upon receipt of indication of loss of subcooling margin will maintain the PCT well below the limits of 10 CFR 50.46.

3. Based on the results of the analysis under realistic assumptions, an RC pump trip at any time following a SBLOCA for break sizes 0.05 and smaller will not result in PCT exceeding the 10 CFR 50.46 limit.

4. For breaks 0.2-ft^2 and smaller, more than 10 minutes time is available for manual RC pump trip following indication of a loss of subcooling margin.

5. Small breaks larger than 0.2-ft^2 cause a rapid depressurization of the RCS and early actuation of the CFTS and LPI system. Therefore, a delayed RC pump trip for break sizes larger than 0.2-ft^2 will not result in PCT exceeding the 10 CFR 50.46 limit.

6. In summary, as a result of a realistic analysis, at least 10 minutes time is available for a manual RC pump trip following a small break LOCA without exceeding the limits of 10 CFR 50.46. This conclusion is applicable to all B&W 177-FA lowered-loop and raised-loop plants.

5.3. Setpoint and Signal Selection for Non-LOCA Events

The RCP trip criteria, based on loss of subcooling margin, was developed with the intent of assuring that an indication for RC pump trip would occur

for those SBLOCAs where pump trip was required to meet the criteria of 10 CFR 50.46. The spectrum of SBLOCA analyses discussed in the previous sections demonstrate that a loss of subcooling will always occur for small breaks that have the potential to uncover the core if the RCPs are tripped under certain two-phase conditions. The actual value of the setpoint is determined on a plant-specific basis to ensure that this indication will allow continued forced RCS flow during realistic SGTRs up to and including the design basis SGTR -- a single double-ended rupture. The setpoint also includes consideration for minimizing the indication for need to trip RC pumps for more likely non-LOCA events such as a mild overcooling transient due to excessive steam or feedwater flow.

While optimum selection of the loss of subcooling setpoint is expected to discriminate against the more likely overcooling, severe overcooling events such as a large steam line break can result in a loss of subcooling and an indication for manual tripping of the RC pumps. The FSAR safety analyses aee performed assuming the most limiting case of pumps on or pumps off, therefore tripping of the pumps will not result in consequences more severe than previously analyzed.

A best estimate analysis of the steam generator tube rupture event has been performed to demonstrate that steam generator tube leaks up to a single double-ended rupture will not result in sufficient loss of subcooling if ATOG procedures are followed, to produce an indication for the need to trip the RC pumps.

5.3.1. Best Estimate Analysis of a Single Double-Ended Steam Generator Tube Rupture

A single double-ended steam generator tube rupture (SGTR) was simulated for a generic 177-FA lowered-loop plant design. The analysis is intended to demonstrate that sufficient loss of subcooling margin will not occur resulting in a need to trip the RC pumps if an operator follows the ATOG for the event. The results of this analysis will support the bases for utilizing manual tripping of the RC pumps on the criteria of loss of subcooling margin.

5.3.1.1 SGTR General Characteristics

A SGTR is a loss-of-coolant accident (LOCA) that allows reactor coolant to leak into the secondary side of the steam generator where it is released into the steam plant and can lead to significant offsite doses if this steam is released to the environment. For a complete severance of one SG tube, a leak rate of approximately 400 gpm at normal system pressure and temperature would be expected.

The leak from a failed tube cannot be isolated and reactor coolant will continue to be lost until the plant is completely cooled and depressurized and the primary loops have been drained.

Since a tube rupture can exhibit the same general characteristics as a small break LOCA, the general procedures for LOCA mitigation must be followed. A continuous cooldown and depressurization of the RCS is essential to avoid opening of the SG safety valves minimizing the risk of releasing radiation. Forced circulation by the RC pumps will provide a continuous uninterrupted cooldown and depressurization of the RCS.

The following best estimate analysis of a single double-ended SGTR for a generic 177-FA lowered-loop plant was performed to demonstrate that operator actions as per ATOG to control RCS inventory, perform plant runback, and initiate low power reactor trip, preclude loss of subcooling margin and RCP trip during a single double-ended SGTR event.

5.3.1.2 Method of Analysis (Best Estimate)

The analysis was performed with the TRAP2[10] computer code. A description of the TRAP2 model simulating the RCS and the steam generator (SG) secondary system is shown in Figures 5-18 and 5-19 and Tables 5-2 and 5-3. The model has been developed to predict the system behavior during a SGTR event and simulate important operator actions as described in ATOG. In addition, the model utilizes a non-equilibrium pressurizer model capable of predicting two-phase pressurizer inventory and mixture level.

Operator actions leading to a cold shutdown, systems initial conditions, and operation and other input assumptions are described in the following subsection.

5.3.1.2.1. General Operator Actions

Although the ATOG guidelines may differ slightly among 177-FA lowered-loop plants, the most important operator actions and stages in treatment of the SGTR are similar. The operator actions for a SGTR event are summarized in Figure 5-16.

Event Identification

The first stage for mitigation of SGTR is prompt recognition of the event and determination of the affected steam generator. The occurrence of secondary radiation alarms (steamline monitor or condenser air ejector) almost simultaneous with SGTR and RCS pressure and pressurizer level drops are unique indicators that a SGTR has occurred. This analysis assumed the operator diagnoses the SGTR following the radiation alarm and pressurizer low level alarm and begins to take prescribed action.

Plant Control at Power

In the second stage of the event, the RCS pressure and pressurizer level must be stabilized so that the plant may be run back without tripping. A trip at high core power may result in venting radioactive steam to the environment through the secondary safety valves. Normally, the makeup system (MU) will automatically increase MU flow to stabilize pressurizer level. For a double-ended rupture (DER) of a single tube the leak flow (400 gpm) is greater than the MU flow. The operator is instructed to take action to increase makeup flow or HPI and terminate letdown in order to stabilize the RCS. Once RCS pressure and pressurizer level are stable, plant runback to hot zero power (HZP) should begin.

This analysis assumes that the operator starts a second makeup pump and isolates letdown in order to stabilize RCS pressure. If the output of two makeup pumps does not match the leak flow, the operator is assumed to shift suction to the BWST and initiate full HPI flow. Analysis[8] has shown that for a DER of a single tube, the operator has approximately 11 minutes to stabilize RCS pressure before the reactor trips automatically. Once the system is stabilized, the operator is assumed to runback the power.

Plant Runback to Hot Zero Power

Operator action should be initiated to stabilize RCS pressure and pressurizer level while conducting a plant runback to low power without tripping. RCS inventory should be monitored during the plant runback. This analysis assumed operator action to reduce the MFW demand to match a 10% per minute runback in core power level.

Upon reaching a low power level where the available turbine bypass (TB) capacity is sufficient to avoid lifting the steam safeties, the plant is tripped. This action was assumed at 15% power. Although the turbine bypass capacity may vary from plant to plant, reactor trip near these power levels will exhibit a negligible difference in plant response.

Cooldown and Depressurization

The initial objective of the cooldown is to bring the RCS hot leg temperature to a value (540F) that corresponds to a saturation pressure which is below the steam safety valve setpoint. This action will limit atmosphere radiation releases. Below 540F the SG with the tube rupture can be isolated. The cooldown should be continued to cold shutdown conditions at a rate of 100F/hour using the good generator and DHRS while maintaining the RCS subcooled.

5.3.1.2.2. Assumptions and Initial Conditions

The following assumptions and initial conditions were used in this analysis.

- Core power is 2568 MWt. (The transient is not expected to be sensitive to initial power.)
- Offsite power is available throughout the transient.
- Reactor coolant pumps operate throughout the analysis.
- Initial pressurizer level is 180 inches (indicated). This is conservative with respect to B&W 177-FA plants because this corresponds to an initial liquid volume of 820 ft^3. No B&W 177-FA plants operate with a lower nominal pressurizer level.

- HPI and makeup flow characteristics are as shown in Table 5-1. The results are not sensitive to flow capacity as long as the flow from two HPI pumps can match the leak flow rate at rated power.
- The primary to secondary leak flow is conservatively modeled as subcooled discharge with a discharge coefficient of 1.0.
- Initial plant condition --

Power level	2568 MWt
Hot leg temp	605F
Hot leg pressure	2160.2 psia
T_{avg}	582.7F
RC system flow	36,597 lbm/s
Pressurizer level (as measured from the lower tap)	180 inches
Total steam flow	3258 lbm/s
Subcooling margin	42F

5.3.1.3. <u>Results of Analysis</u>

The primary-to-secondary leak flow rate which resulted from the double-ended rupture was approximately 50 lbm/s (see Figure 5-21). Operator action was modeled to manually start a second makeup pump and isolate letdown 1 minute after indication of rupture. The charging flow of two makeup pumps was not sufficient to match the leak flow. Subsequently, pressurizer level continued to decrease and HPI injection was initiated manually at t=4 min. to stabilize RCS inventory. This is shown in Figure 5-28.

A runback in reactor power of 10% per minute was initiated when the plant was considered stable (t=7 min., see Figure 5-22). Main feedwater was manually ramped to match core power (Figure 5-23). The ICS would normally ramp power and feedwater flow to maintain a constant T_{avg}, however, modeling difficulties caused T_{cold} to remain constant while T_{hot} decreased. This forced T_{avg} to decrease causing the system to contract more during the runback than would actually have occurred. However, this had little effect on the analysis since the total system contraction is from full power to zero power, and thus the pressurizer outsurge remains approximately the same.

When reactor power and steam load were within the turbine bypass capacity, 15%, the reactor and turbine were tripped. The turbine bypass system controlled secondary pressure while the loss of heat source on the primary side caused the RCS to contract. The RCS contraction caused a pressurizer outsurge and the pressurizer level (Figure 5-24) decreased. The pressurizer level continued to decrease until the energy removed by the turbine bypass system was equal to the energy generated in the core at which time pressurizer level began to increase due to high pressure injection.

The minimum indicated post trip pressurizer level was 13 inches. Once pressurizer level was increasing, the analysis was terminated because it was assumed the operator would cooldown and depressurize the RCS while maintaining minimum subcooled margin. The subcooled margin at termination of the analysis was 91F. Figures 5-21 through 5-30 show the transient responses of pertinent parameters.

Plant Condition at Termination of Analysis --Table 5-2 lists the sequence of events for this analysis.

Power level	68 MWt
Hot leg temp	545F
Hot leg pressure	2000 psia
T_{avg}	544F
System flow	37,546 lbm/s
Pressurizer level (indicated)	21 inches
Bypass steam flow	53 lbm/s
Subcooling margin	91F

5.3.1.4. Conclusions

Since the unique indication of a SGTR lends itself to quick diagnosis, the operator has time to increase makeup or initiate HPI to stabilize the RCS inventory. If no action is taken to increase makeup, the reactor will eventually trip on the variable low pressure trip. The pressurizer may subsequently drain because of insufficient pressurizer inventory to accommodate the RCS contraction.

Operator action, as per ATOG, to stabilize RCS pressure and pressurizer level, runback the plant, and trip the reactor does preclude sufficient loss-of-subcooling margin to result in the indication for a need to trip the RC pumps.

5.4. Other Considerations

5.4.1. Potential Containment Isolation of RC Pump Services

A primary objective of the parameter and setpoint selection is the avoidance of reactor coolant pump trip for the more probable of the non-LOCA events. Loss of subcooling margin has been shown to occur for all small break LOCAs of concern and may also occur for design basis overcooling type transients which may exhibit a similar primary system response as a LOCA.

Item I.1.e of the enclosure to Generic Letter No. 83-10 expresses the concern that "Transients and accidents which produce the same initial symptoms as a LOCA (i.e., depressurization of the reactor and actuation of engineered safety features) and result in the termination of systems essential for continued operation of the reactor coolant pumps (i.e., component cooling water and/or seal injection water)." "...In particular, if a facility design terminates water services essential for RCP operation, then it should be assured that these water services can be restored in a timely manner once a non-LOCA situation is confirmed, and prevent seal damage or failure."

The following contains some of the generic guidance provided in the Abnormal Transient Operating Guidelines (ATOG) to assure water services when RCP operation is desired. The potential for containment isolation signals resulting from various transient types is discussed as well as guidance for the restart of RC pumps under degraded service conditions.

5.4.1.1. ATOG Guidance

ATOG emphasizes protection of the RCP seals and motor to prevent damage. It is desirable to trip the pumps to prevent mechanical damage in case they must be restarted at a later time. Preserving the pumps for long-term

cooling or cooldown is desirable, and it is recommended that they be shut down if high vibration or loss of auxiliary cooling water services occurs. (See a discussion of cooling water services and the effect on pump operation under Containment Isolation Signal Guidance below.) Limits on continued pump operation are given in "Plant Limits and Precautions." The rules for RCP trip to prevent mechanical damage are applicable in all cases <u>except</u> (1) when the pumps were not tripped immediately (i.e., within two minutes after the subcooling margin was lost) or (2) when severe inadequate core cooling (ICC) exists. In these cases the operator should try to restore the RCP service which is lost.

Operators are instructed to perform the remedial action of verifying or reestablishing RCP seal injection and service water flow if potential isolation signals have been verified (i.e., ESF signals). The sections of ATOG which permit continued RCP operation or anticipate RCP restart give instructions for the verification or reestablishment of seal injection and cooling water.

ATOG provides specific instructions for RC pump restart. The Equipment Operation chapter of ATOG shows the conditions under which the RC pumps can be restarted. For situations where inadequate core cooling is not a concern, confirmation that no RC pump damage will occur is among the requirements for pump restart. (If inadequate core cooling is a concern, RC pump restart is required even if damage can occur). Some specific actions which must be taken to verify pump integrity before restart include:

- If component cooling water is isolated and pumps are allowed to run, bearings and stator will rise in temperature. If pumps are then tripped, they should not be restarted until component cooling water is restored and bearings have cooled down to normal operating temperature.
- If pumps are tripped before bearings get hot (exceed alarm setpoint), pumps can be restarted and run once component cooling water is restored.
- Unless bypassed, interlocks will prevent pump restart until component cooling water and seal injection are available to the pump.
- Controlled bleed-off flow should be reestablished prior to pump restart.

Other considerations for RCP restart are verified on a plant specific basis. The Plant Limits and Precautions Documents, Plant Setpoints Documents, and RCP vendor instruction manuals provide further instructions.

5.4.1.2. Generic Containment Isolation Signal Philosophy

In considering all B&W Owner's Group plants generically, it must be recognized that some differences exist. For example, (1) the plants utilize three different RCP vendors (Byron-Jackson, Bingham, and Westinghouse), (2) the plants utilize various motor suppliers (Siemans-Allis, GE, Westinghouse, and AEG), (3) the 205-FA plants operate with a lesser subcooling margin than the 177-FA plants, (4) system configuration and design differ slightly and, (5) as a result of recommendations for increased RCP reliability and response to NUREG-0737, Item II.E.4.2, each utility has opted for various isolation signals which affect RCP cooling water services availability.

The containment isolation signals presently being used, depending on plant and function, are:

- Low RC pressure (1500-1700 psig).
- Low RC pressure or high containment pressure (3-4 psig).
- Low-low RC pressure (400 psig).
- ESF interlocked with RCPs not running.
- High-high containment pressure (25-38 psig).
- ESF with seal bleedoff redirected to quench tank.
- High radiation.
- Low CCW surge tank level.

Non-LOCA transients such as a severe overcooling resulting from a steam line break inside containment have the potential to produce the same initial symptoms as a LOCA. If it were desired to continue with RCP operation for these transients, low RC pressure (1500-1700 psig) and high containment pressure (3-4 psig) actuation have the greater potential to cause unwanted isolation of RCP services and require timely reestablishment of these services. Using high-high containment pressure as an isolation signal would prevent isolation of the service functions for the majority of such transients and accidents and yet maintain isolation capability for design basis

accident (large LOCA and SLB). The other isolation signals mentioned above are used to preclude RCP or equipment damage.

The cooling water services supporting the RCP with the potential of being isolated are:

- Seal injection.
- Seal bleedoff (resulting from seal injection provided to pump).
- Component cooling water to seal area coolers.
- Component cooling water to RCP motors and oil coolers.

For a running RCP, seal integrity can be maintained with either seal injection or component cooling, within operating parameters, to the seal area coolers. More than ten years ago, B&W removed the seal injection isolation signal from the makeup and purification system design and recommended that the signal be removed from the operating plants. Presently there is only one plant with seal injection isolation which is signaled to close on 400 psig RC pressure. Four hundred psig RC pressure is an indication of a large break LOCA where RCP operation is not anticipated.

The seal bleedoff is important to seal integrity in that it provides pressure staging across the seals and removal of the heat generated by the rotating seals. Closure of the seal bleedoff line with the RC pumps running can cause significant seal damage. In this situation with a low seal face leakage rate, the heat generation rate at the upper seal will triple due to tripling of the seal P and there is not enough flow to remove this heat. ATOG instructions call for quickly reestablishing seal bleedoff flow or tripping the pumps. Some plants have initiated a change to redirect the seal bleedoff flow to the quench tank upon receipt of a containment isolation signal. Another alternative to minimize the occurrence of ESF closure is to actuate close only on hi-hi containment pressure (the same signal that actuates containment spray flow). For plants utilizing ESF signals (1500-1700 psig RC pressure, 3-4 psig containment pressure) for isolation, timely reestablishment of seal bleedoff flow per the ATOG instructions is required for continued RC pump operation.

With proper seal injection and seal return, integrity of the seals can be maintained indefinitely on loss of component cooling water to a running RC

pump. However, the RCP motors cannot be run indefinitely on a loss of cooling water. If it is desired to continue RCP operation, cooling water must be reestablished within a given time frame as indicated in ATOG to preclude damage. The time frame varies depending on the type of motor.

Table 5-1. HPI and Makeup System Capacities

1. HPI Flow Vs Pressure

Flow, gpm	Pressure, psig 0.0	300	600	900	1200	1500	1800	2100	2300
1 HPI	550	525	500	470	440	405	365	320	280
2 HPI	1070	1025	970	915	850	780	705	615	535

2. Makeup (MU) Flow Vs Pressure (One Pump)

Pressure, psig	0	1000	1600	2100	2205	2265	2300	3000
Flow, gpm	317	250	200	150	53	53	32	32

Table 5-2. MINITRAP2 Node Description

Node number	
1	Reactor vessel, lower plenum
2	Reactor vessel, core
3	Reactor vessel, upper plenum
4,10	Hot leg piping (including "candy cane")
32,33	"Candy cane" and upper SG shroud
5-7,11-13	Primary, steam generator tube region
8,14	Cold leg piping
9	Reactor vessel downcomer
80	Pressurizer
16,24	Steam generator downcomer
17,25	Steam generator lower plenum
18-20,26-28	Secondary, steam generator tube region
21,29	Steam risers
22,30	Main steam piping
23	Turbine
31	Containment
34	OTSG Tube

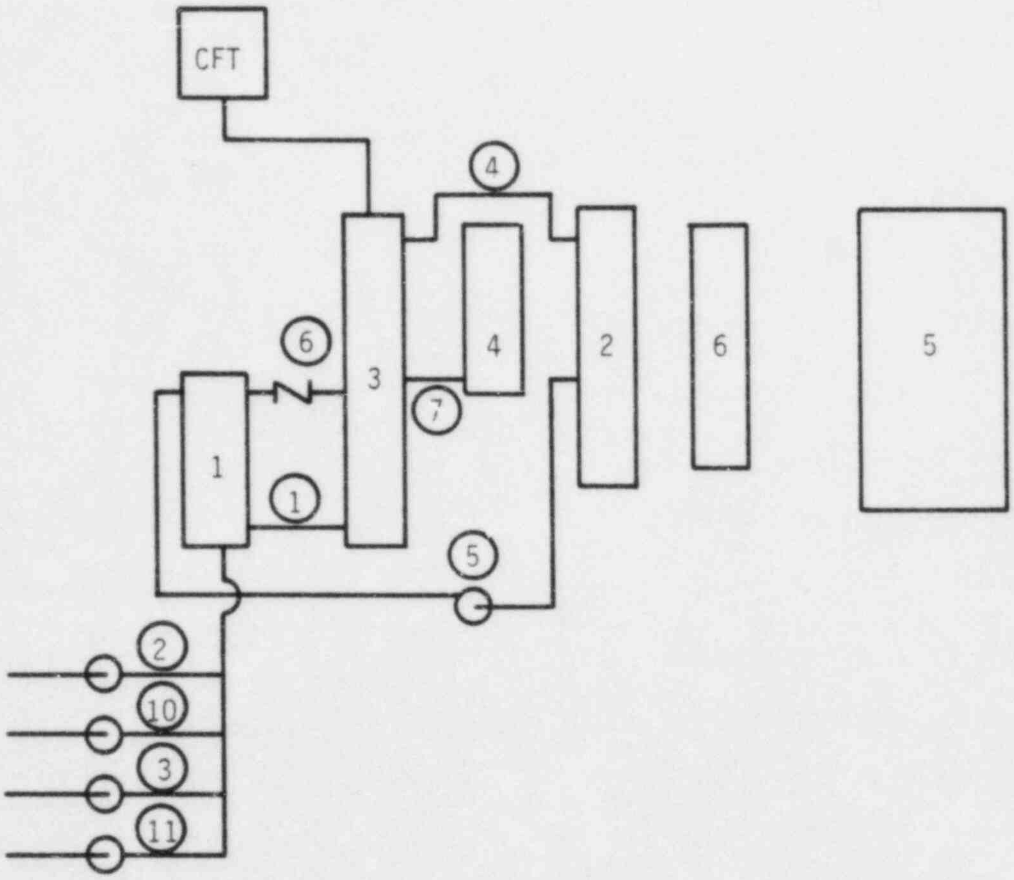
Table 5-3. MINITRAP2 Path Description

Path number	Description
1	Core
2	Core bypass
3	Upper plenum, reactor vessel
4,11	Hot leg piping
5,12	Upper steam generator shroud
45,46,47,48	Top of hot leg "candy cane"
6,7,13,14	Primary heat transfer region, SG
8,15	RC pumps
9,16	Cold leg piping
10	Downcomer, reactor vessel
17	Pressurizer surge line
18,19,26,27	Steam generator downcomer and plenum
20,21,28,29	Secondary heat transfer region, SC
22,30	Aspirator
23,31	Steam riser, steam generator
24,32	Main steam piping
25,33	Turbine piping
34,35	Break (or leak) path
36,37	HPI
38,39,43,44	AFW
40,41	Main feed pumps
42	LPI
49	MSL crossover
50,51	Turbine bypass valves
52	Pressurizer spray
53	Letdown
54	Makeup
55	Upper OTSG tube (steady state only)
56	Lower OTSG tube
57	Lower end of ruptured OTSG tube
58	Upper end of ruptured OTSG tube
Q1	Pressurizer heaters
Q2	RC pump heat

Table 5-4. SGTR Sequence of Events

Event	Time after rupture, min:sec
SGTR at full power	0:00
MSL radiation monitors and/or condenser air ejector radiation monitors trip high secondary activity alarms	0:00
Low pressurizer level alarm	0.20
Operator starts second makeup pump; letdown isolated	1:00
Operator switches pumps to HPI mode and begins HPI	4:00
Reactor runback begins	7:00
Reactor and turbine trip	15:30
Minimum indicated pressurizer level of 13 inches	16:35
Analysis terminated	17:40

Figure 5-1. CRAFT2 Noding Diagram for Small Breaks
(Six-Node Model)



Node No.	Identification	Path No.	Identification
1	PD piping, DC, LP	1	Core
2	Primary SG	2	LPI
3	Core, UP, hot legs	3,10,11	HPI
4	Pressurizer	4	Hot legs
5	Containment	5	Pumps
6	Secondary SG	6	Vent valve
		7	Pzr surge line
		8,9	Leak & return path

Figure 5-2. Critical Region for RC Pumps Trip, Break Size Vs Time After Trip

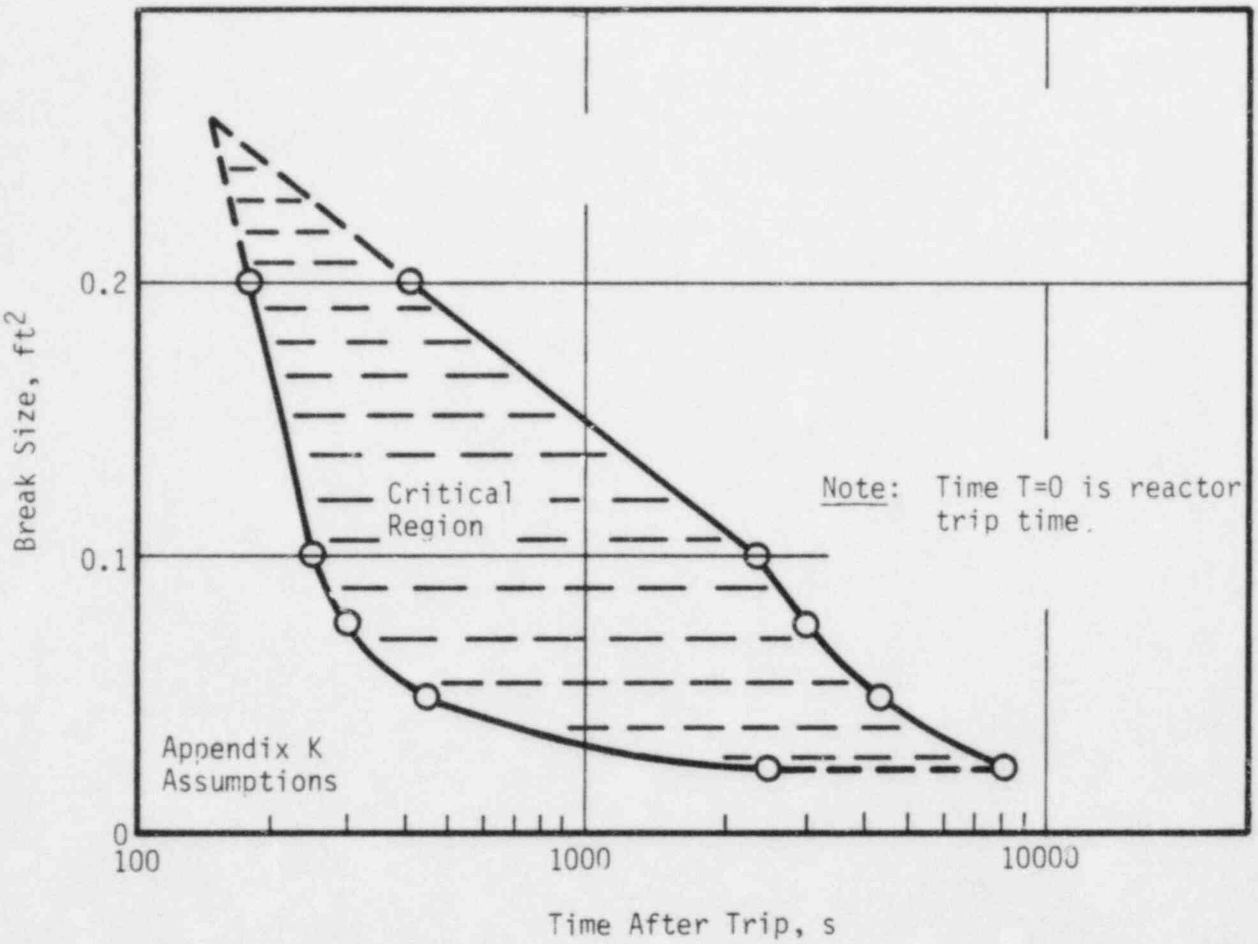


Figure 5-3. "Realistic" Core Axial Peaking Distribution

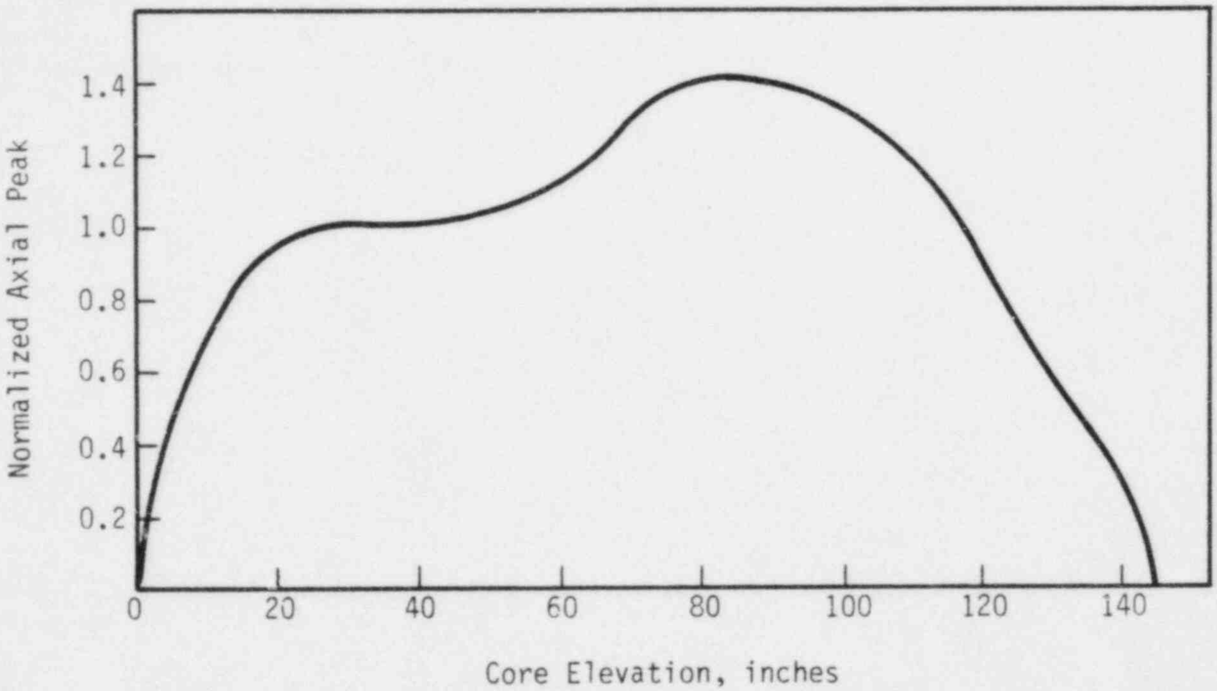


Figure 5-4. Average Primary System Void Fraction Excluding Pressurizer

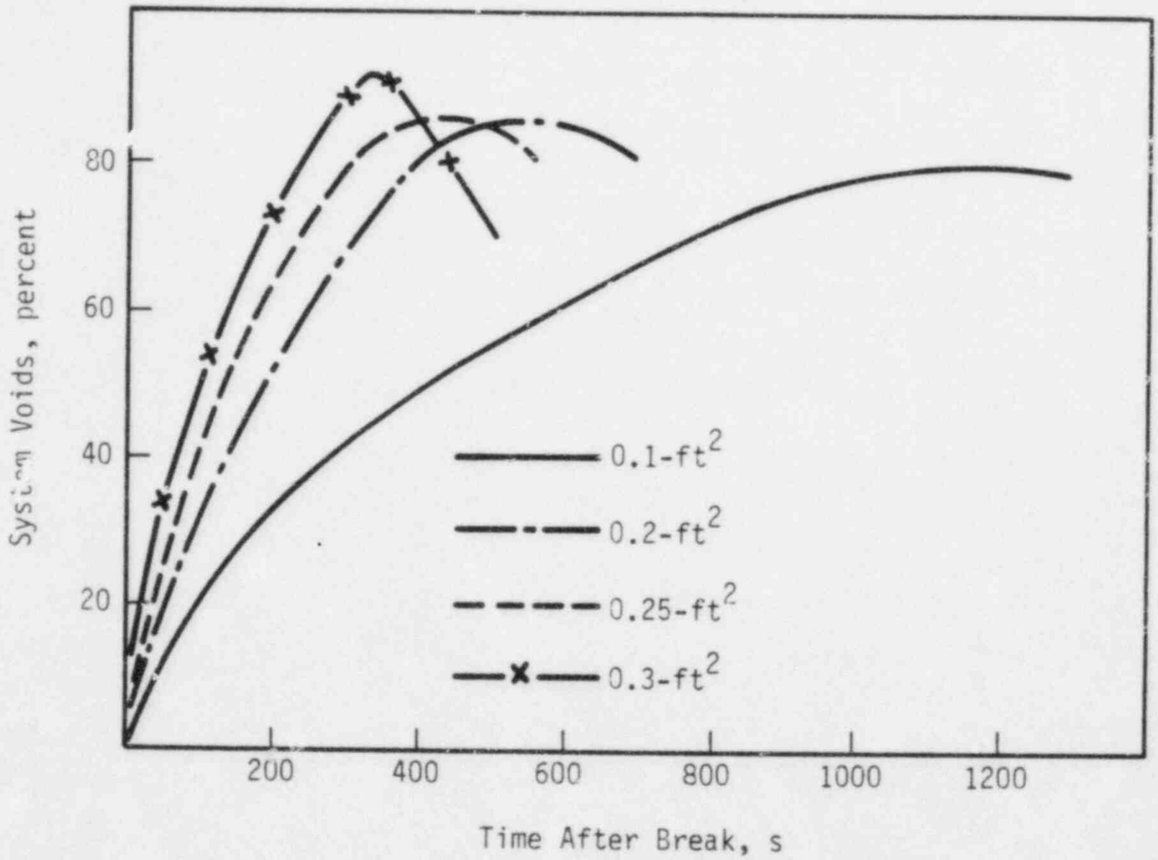


Figure 5-5. RC System Pressure Vs Time (0.1-ft^2 CLD Break -- RCP Trip at 1000 s)

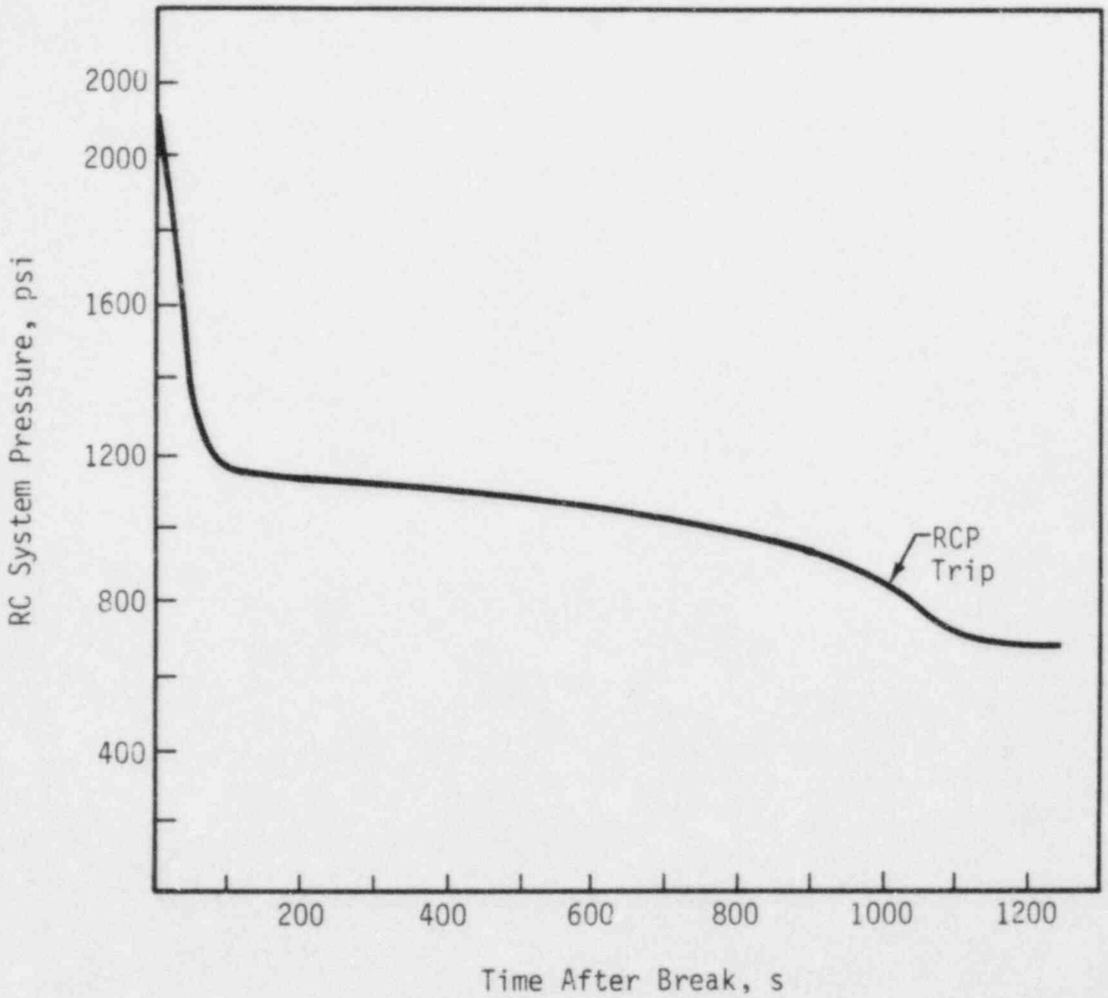


Figure 5-6. Liquid Volume in Reactor Vessel for 0.1-ft^2 CLD Break -- RCP Trip @ 1000 s

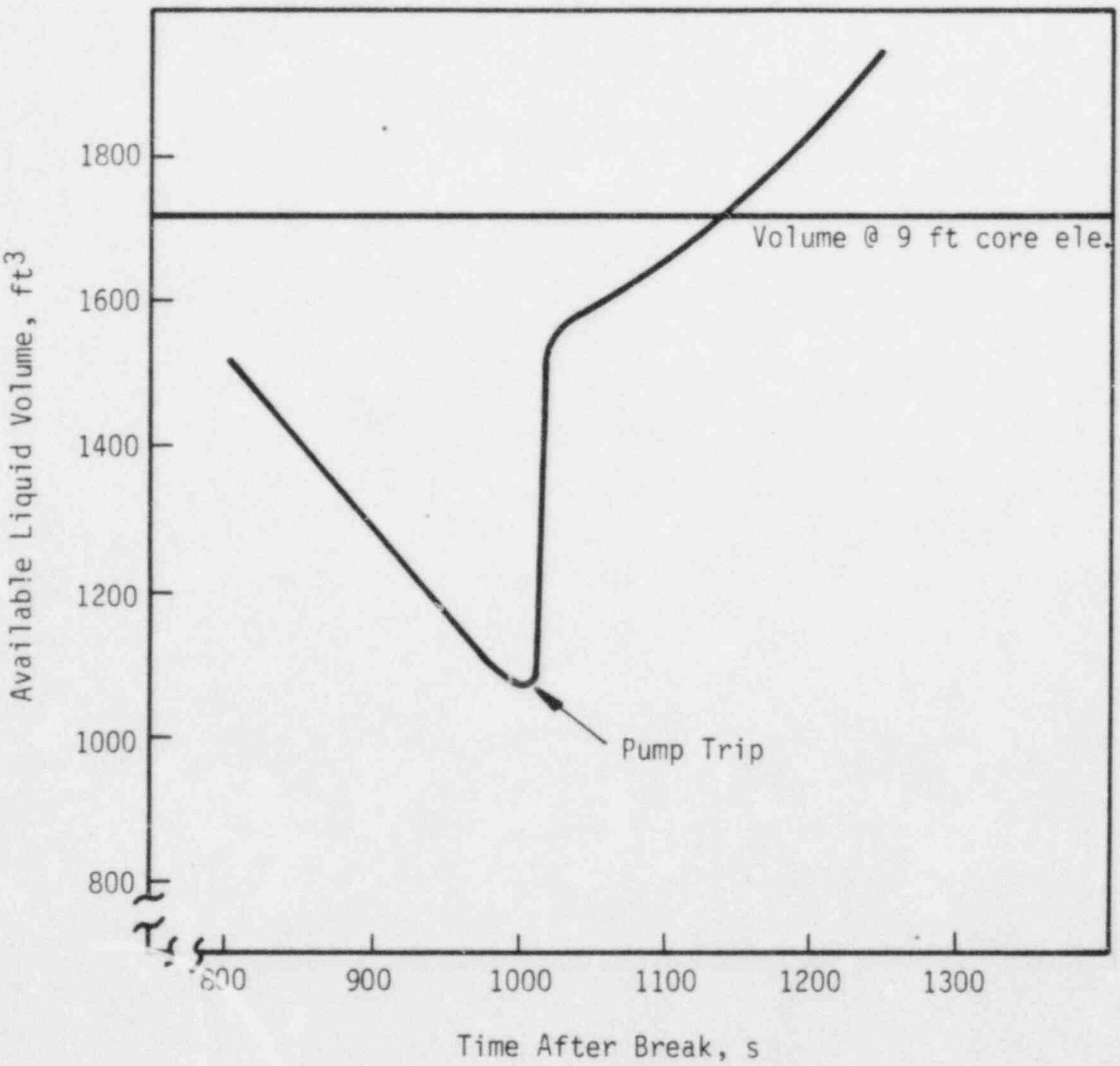


Figure 5-7. RC System Pressure Vs Time (0.2-ft^2 CLD Break -- RCP Trip @ 400 s)

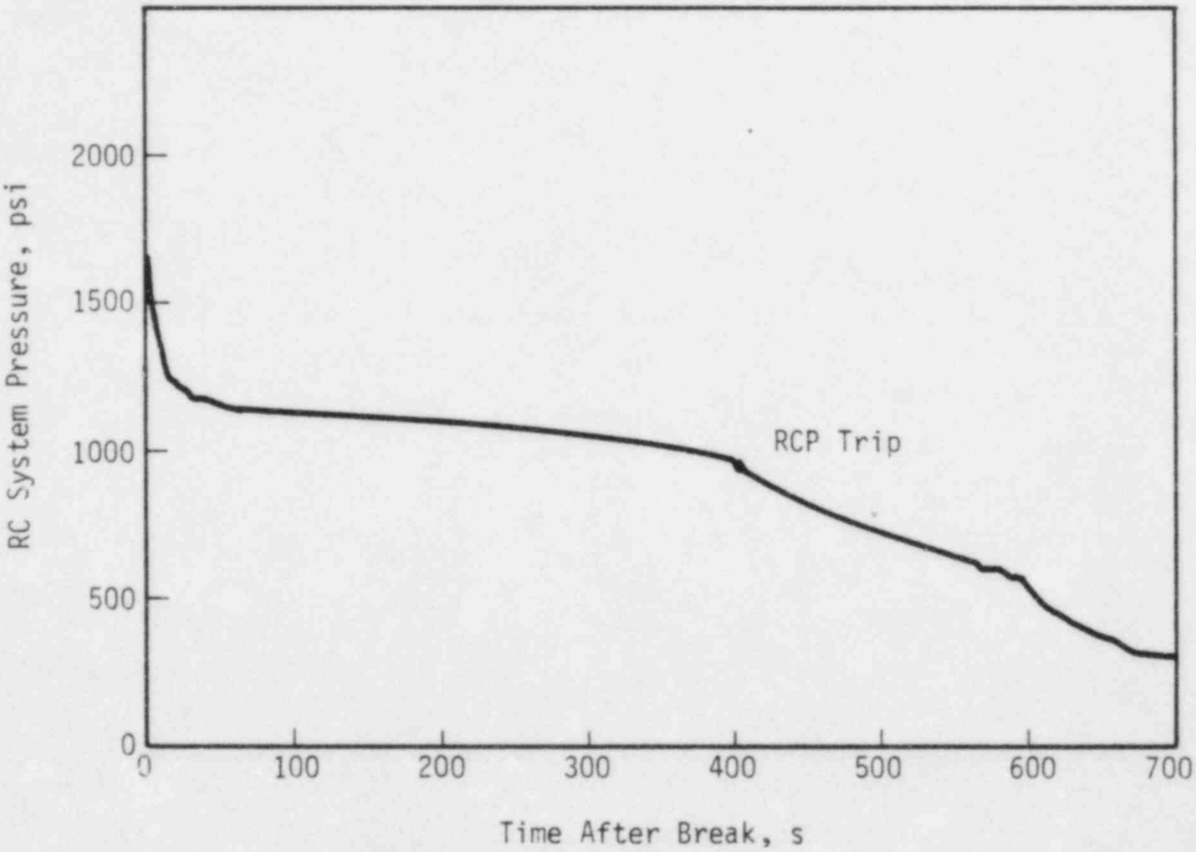


Figure 5-8. Liquid Volume in Reactor Vessel for 0.2-ft^2 CLD Break -- RCP Trip @ 400 s

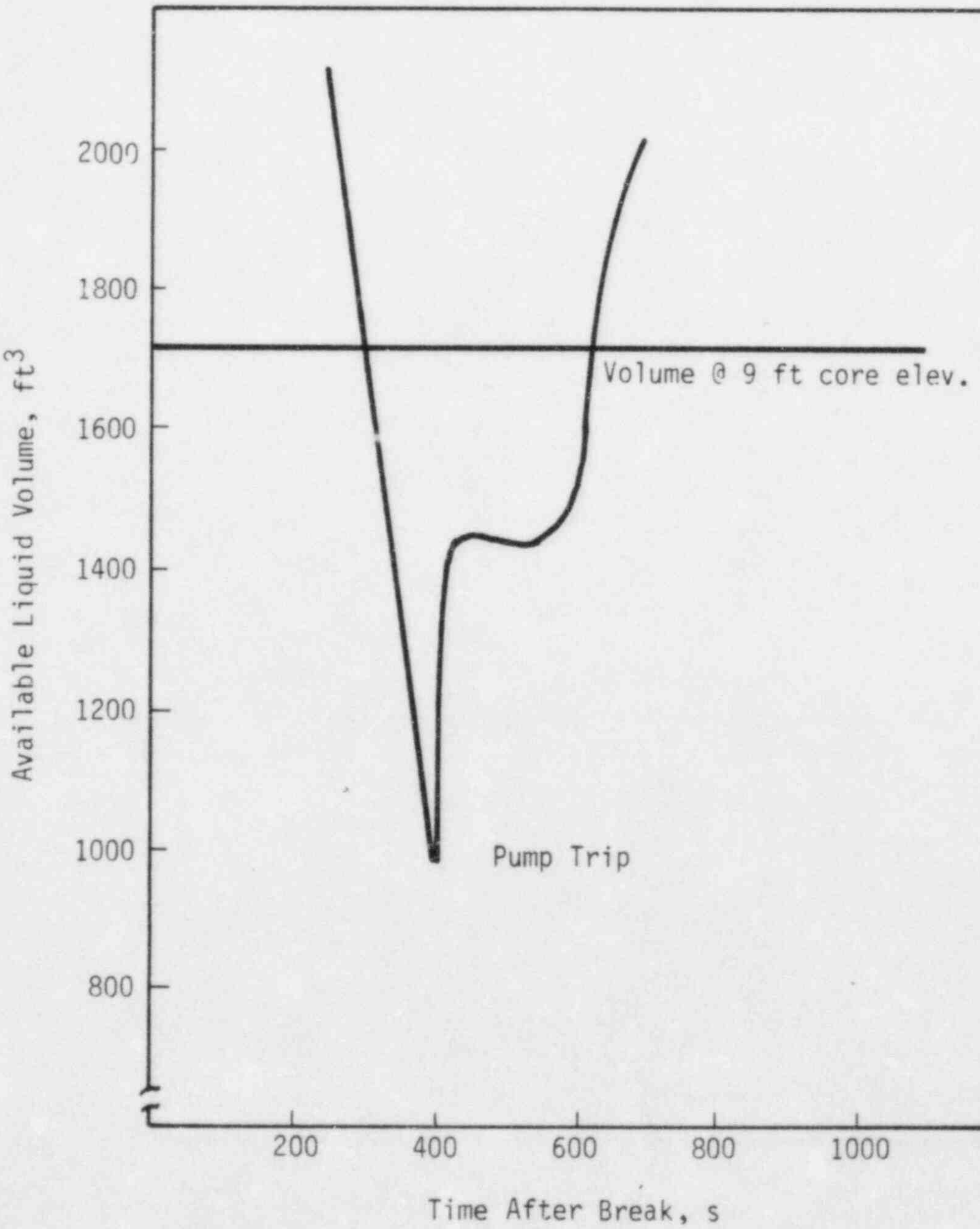


Figure 5-9. RC System Pressure Vs Time (0.25-ft^2 CLD Break -- RCP Trip @ 300 s)

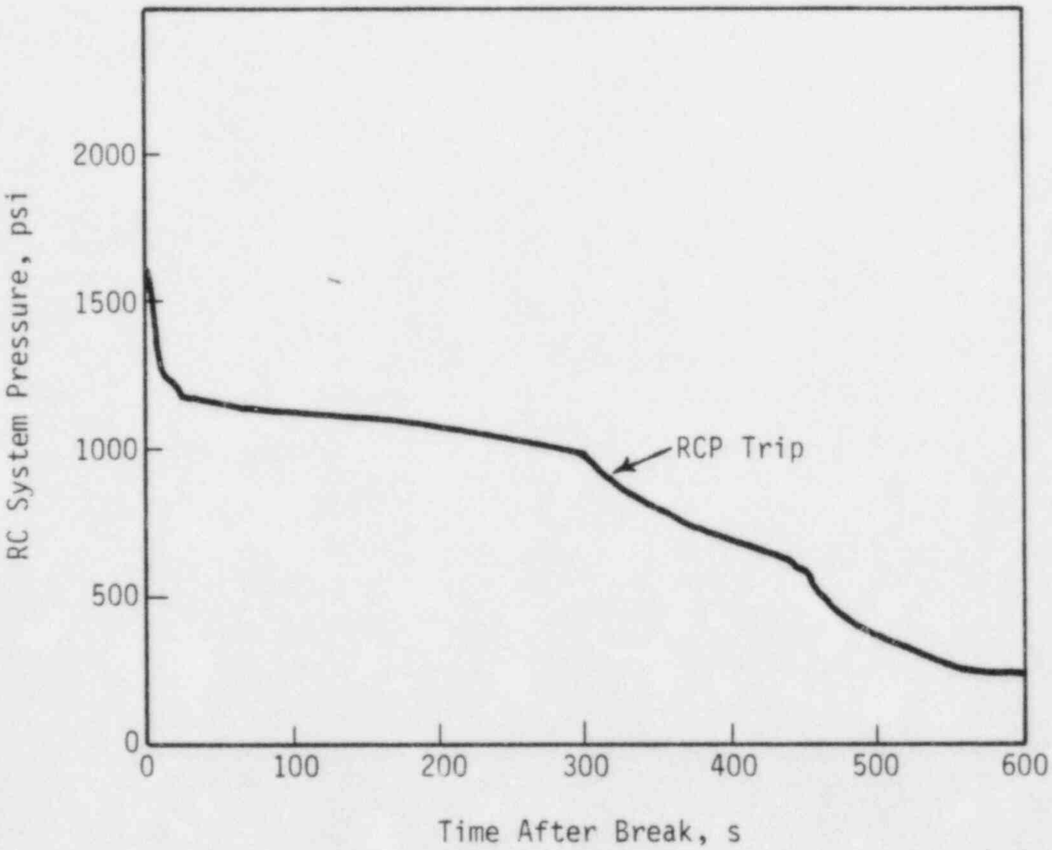


Figure 5-10. Liquid Volume in Reactor Vessel for 0.25-ft^2 CLD Break -- RCP Trip @ 300 s

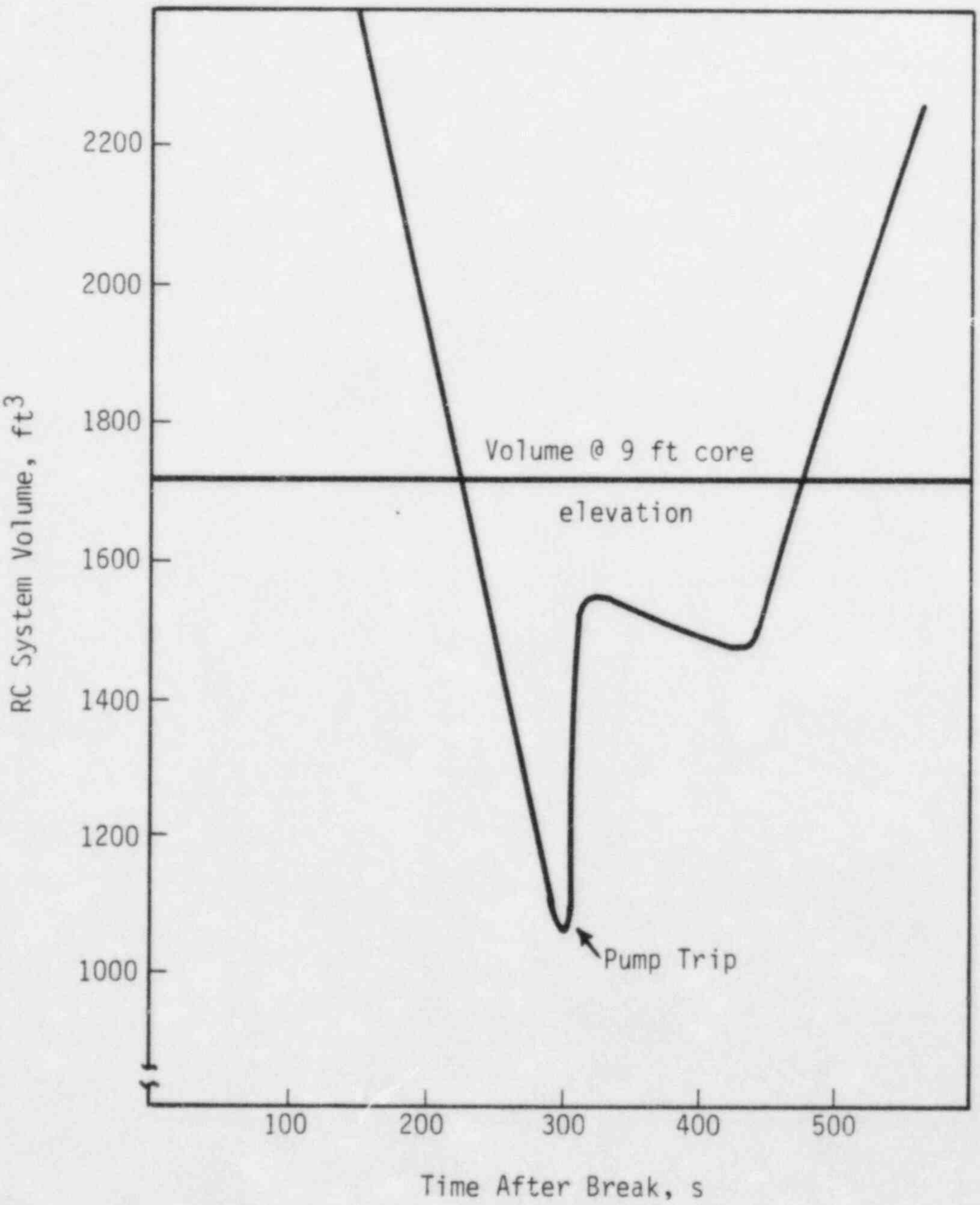


Figure 5-11. RC System Pressure Vs Time (0.3-ft^2 CLD Break -- RCP Trip @ 300 s)

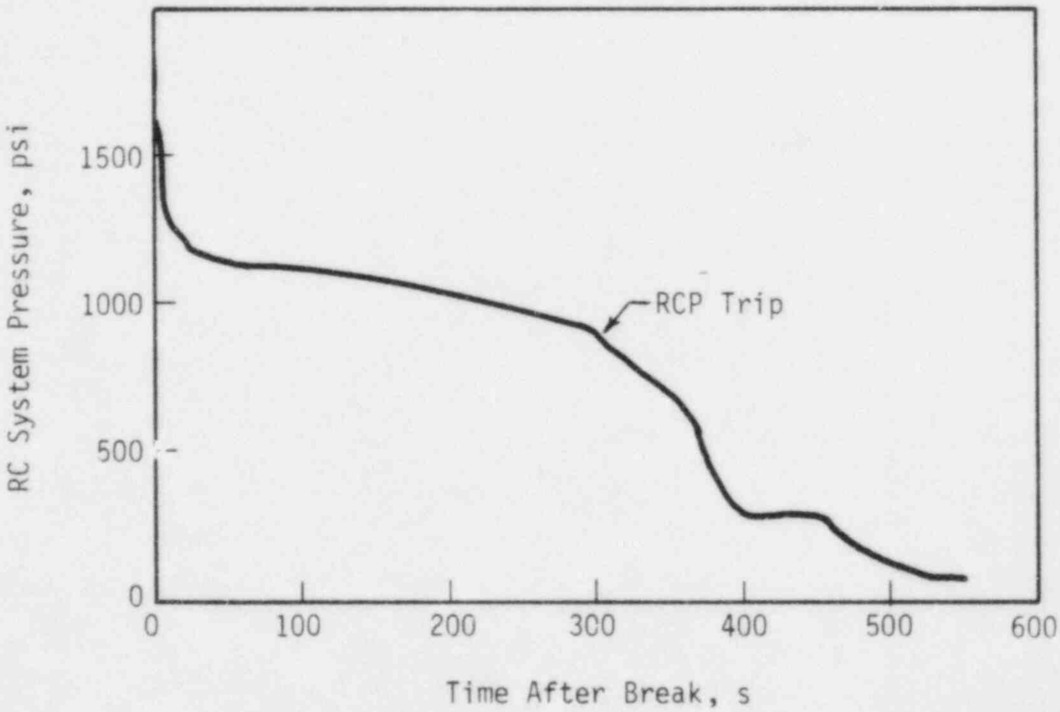


Figure 5-12. Liquid Volume in Reactor Vessel for 0.3-ft^2 CLD Break -- RCP Trip @ 300 s

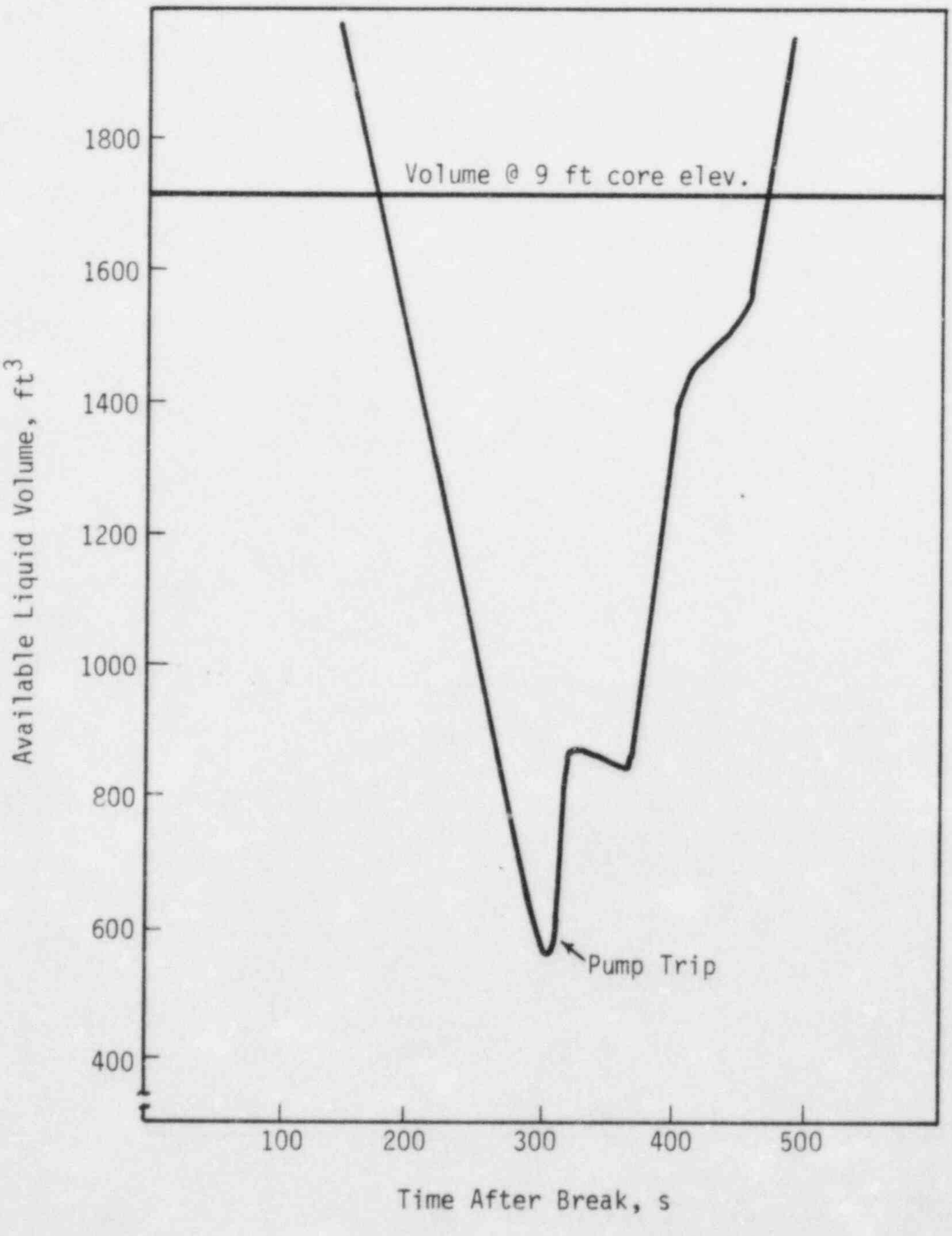


Figure 5-13. SBLOCA Spectrum System Pressure Vs Time

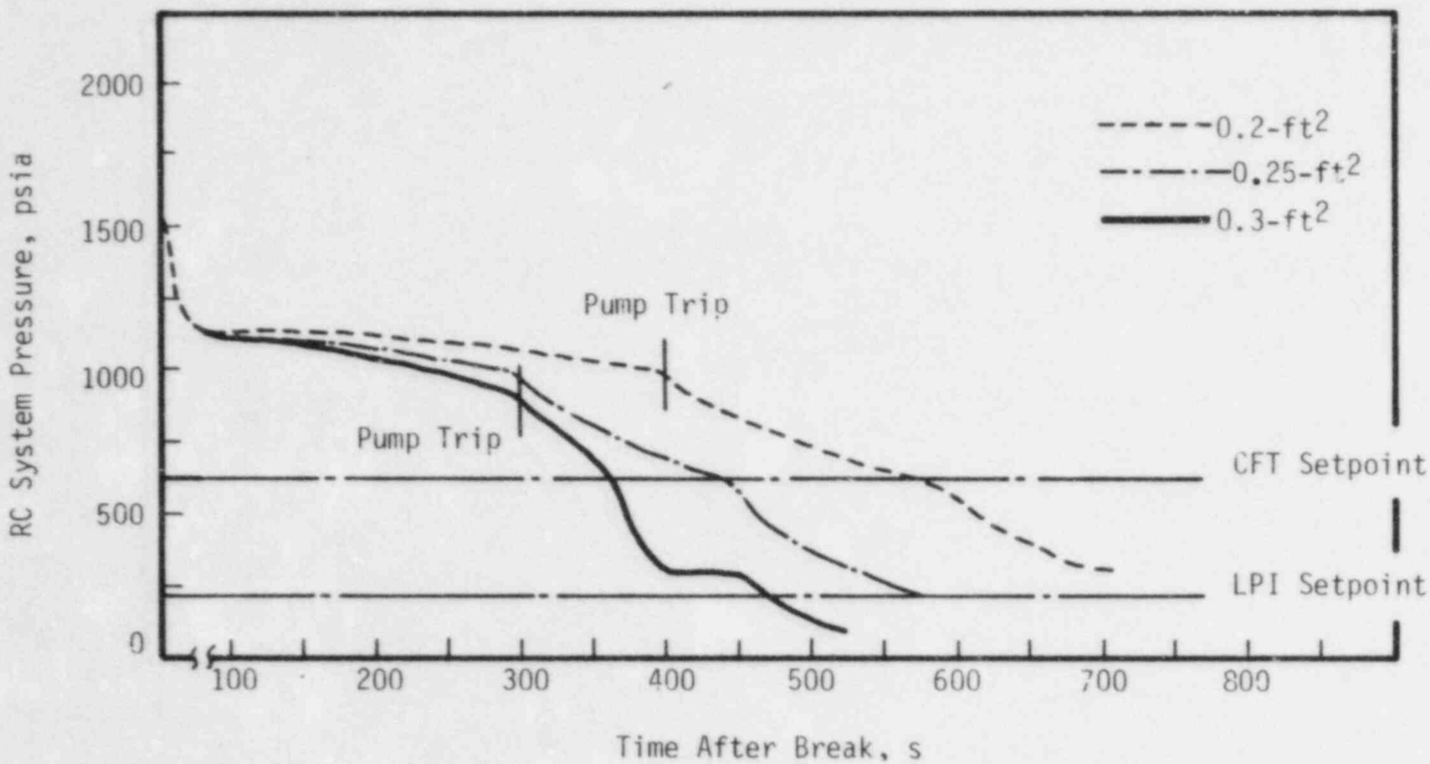


Figure 5-14. Liquid Volume in Reactor Vessel Following RC Pump Trip for a 0.2-ft^2 PD Break

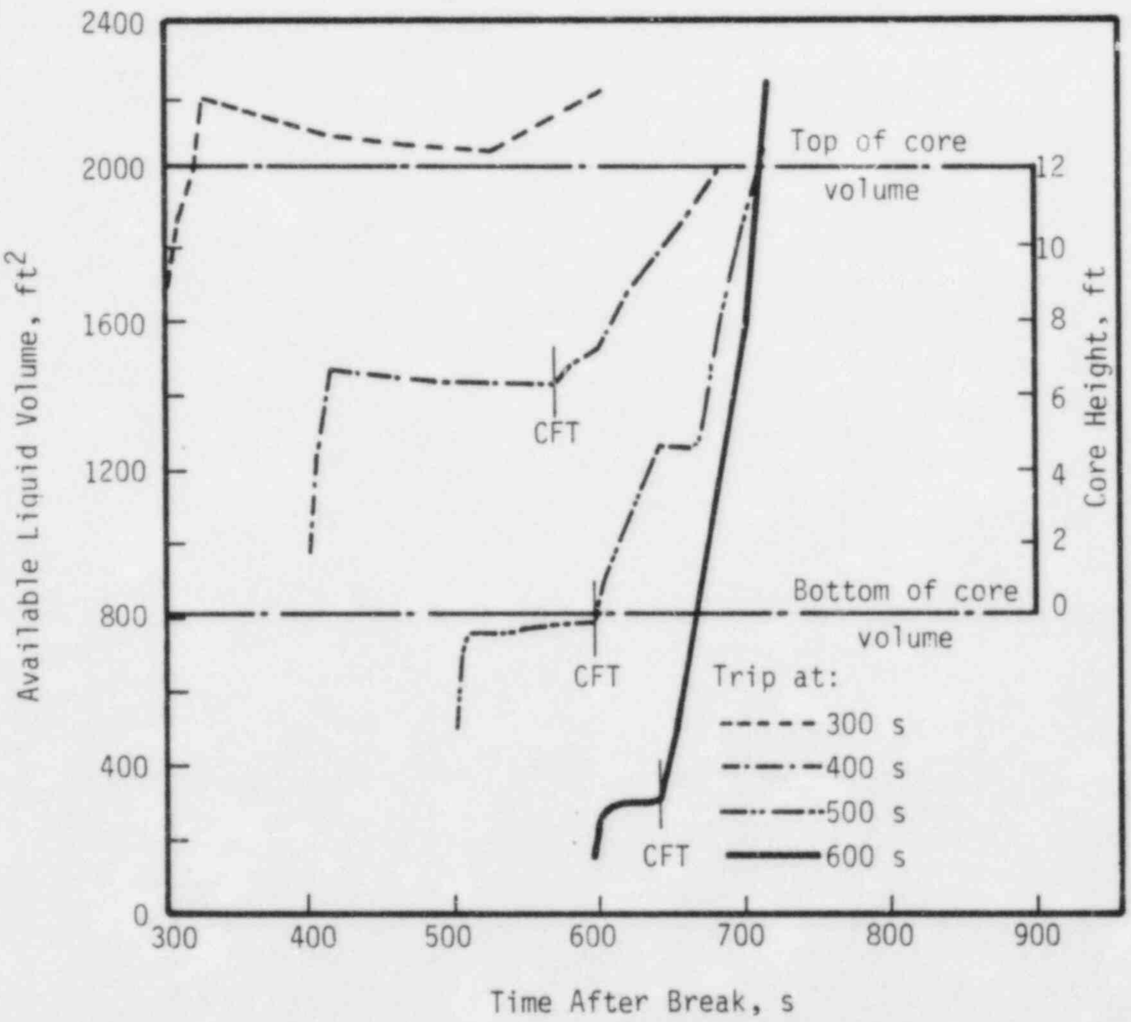


Figure 5-15. System Pressure Vs Time for a 0.2-ft^2 Break at the Pump Discharge

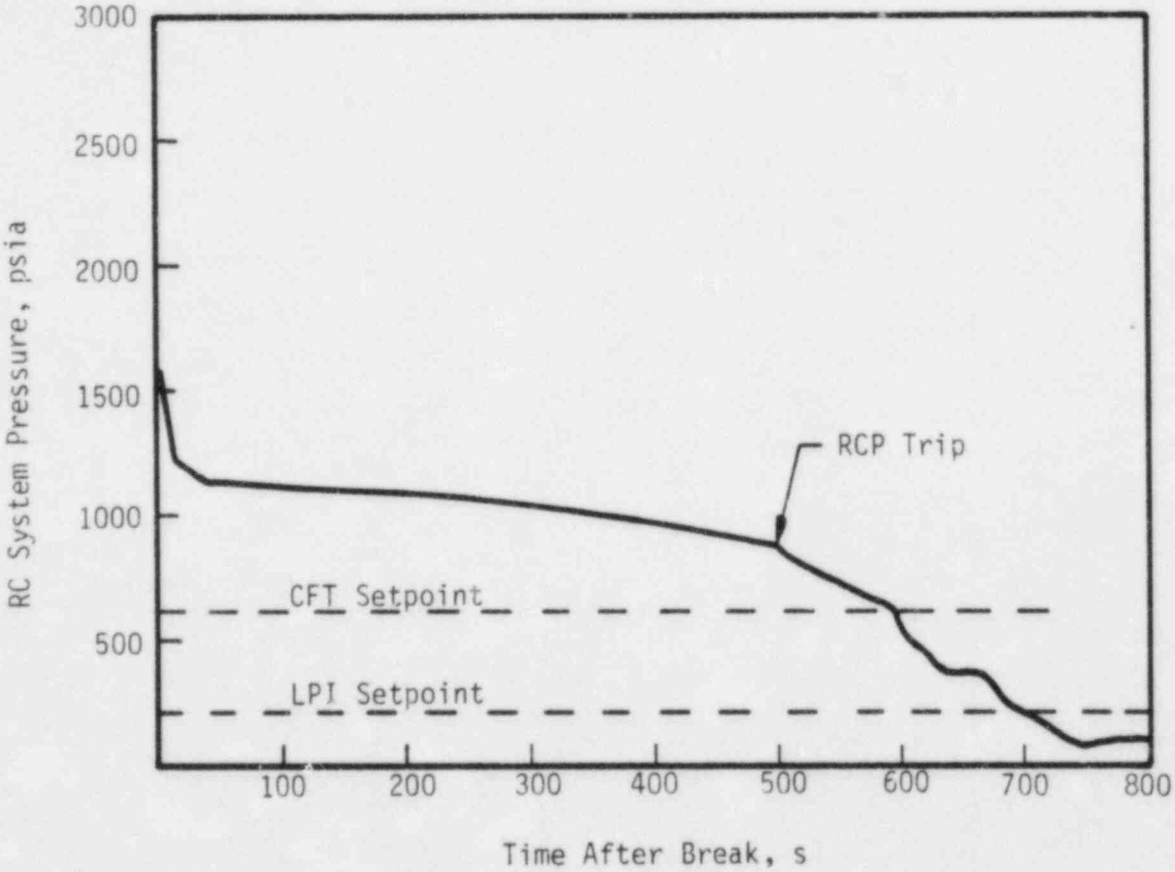


Figure 5-16. Maximum Hot Spot Cladding Volume-Average Temperature Vs Time for a 0.2-ft^2 Break at Pump Discharge

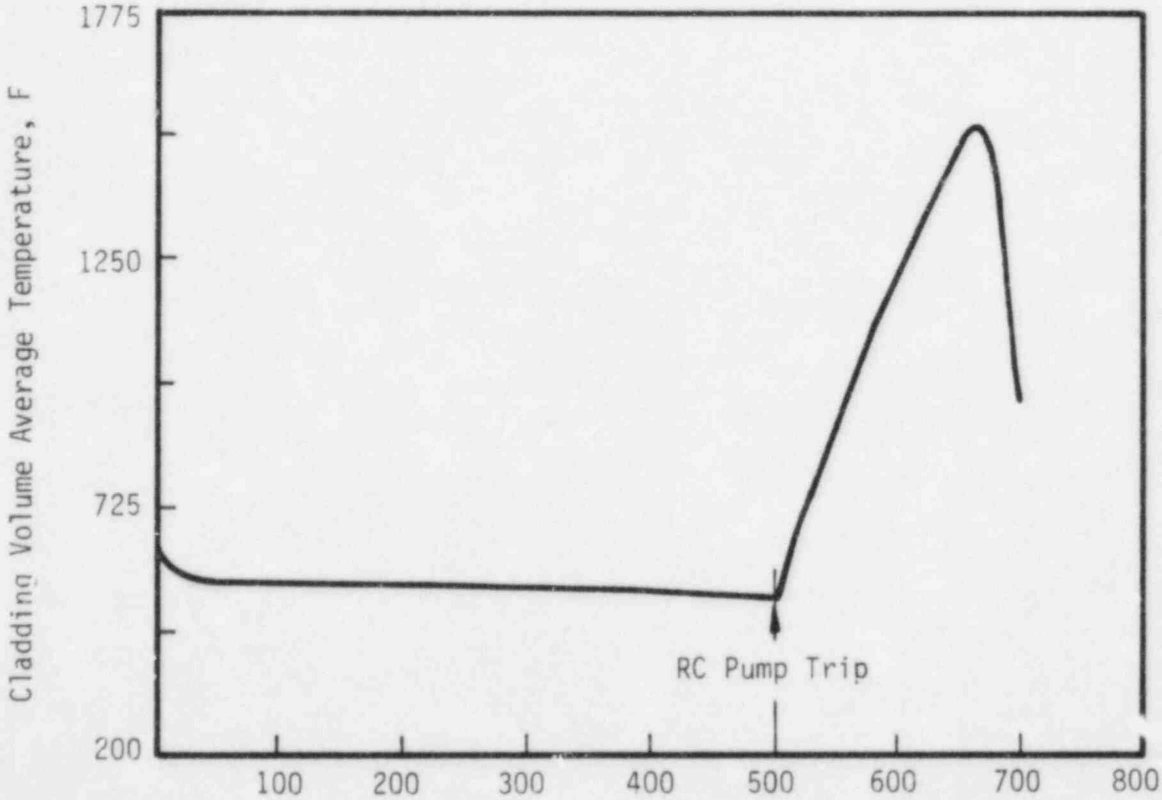


Figure 5-17. Comparison of High Pressure Injection System Capacities for Raised Loop and Lowered Loop Design Plants

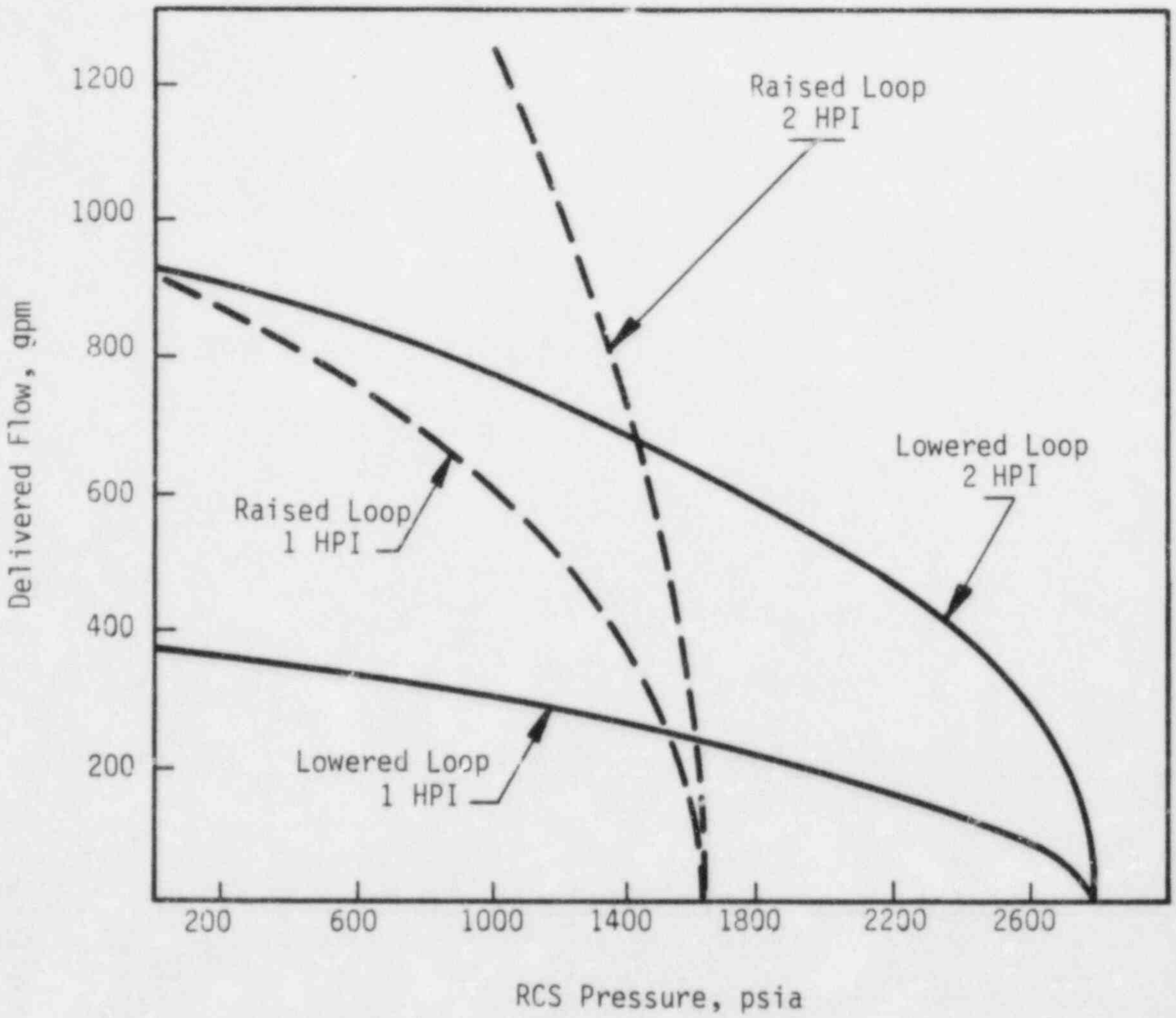


Figure 5-18. Mini-TRAP2 Noding and Flow Path Schematic

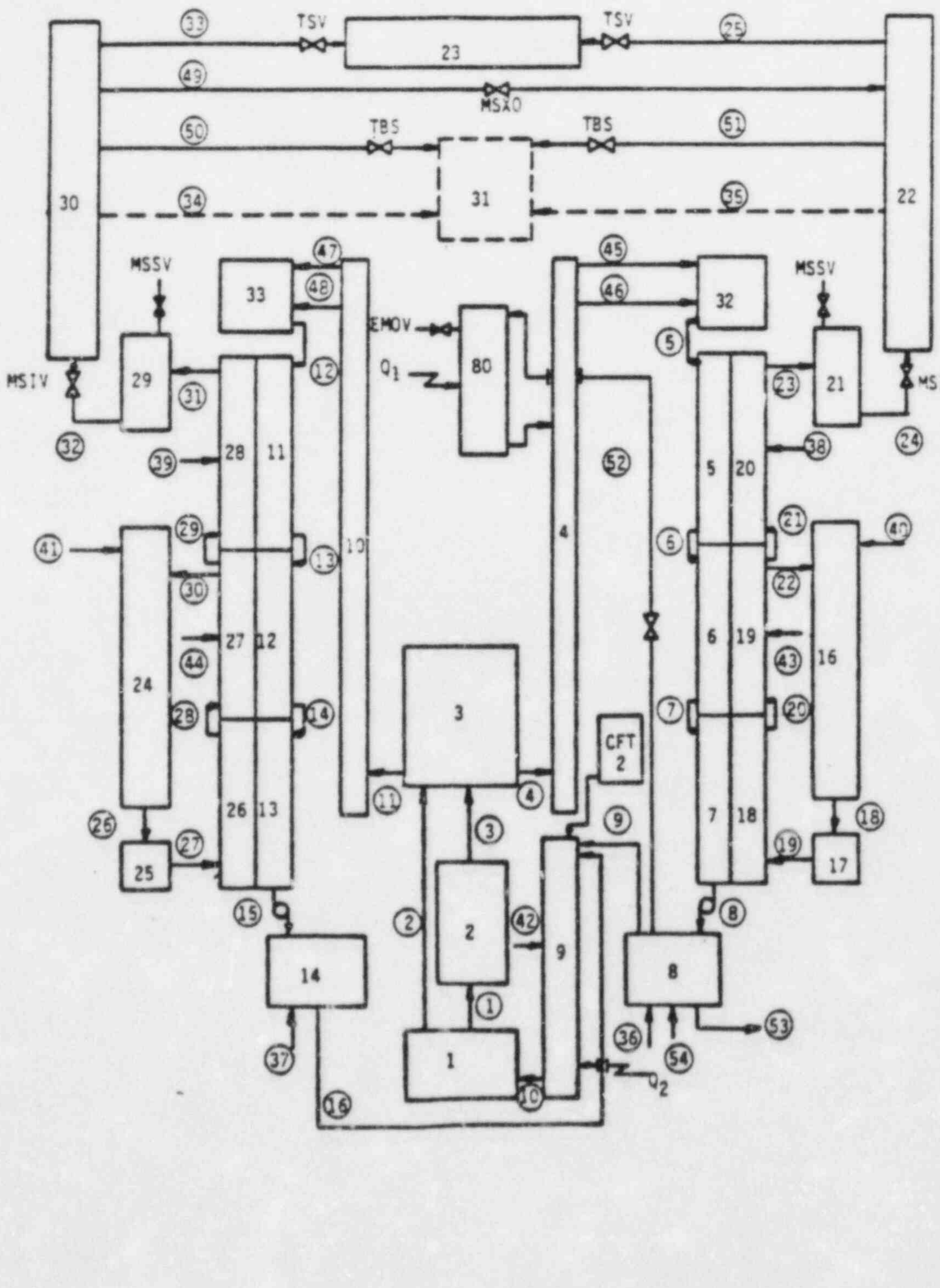
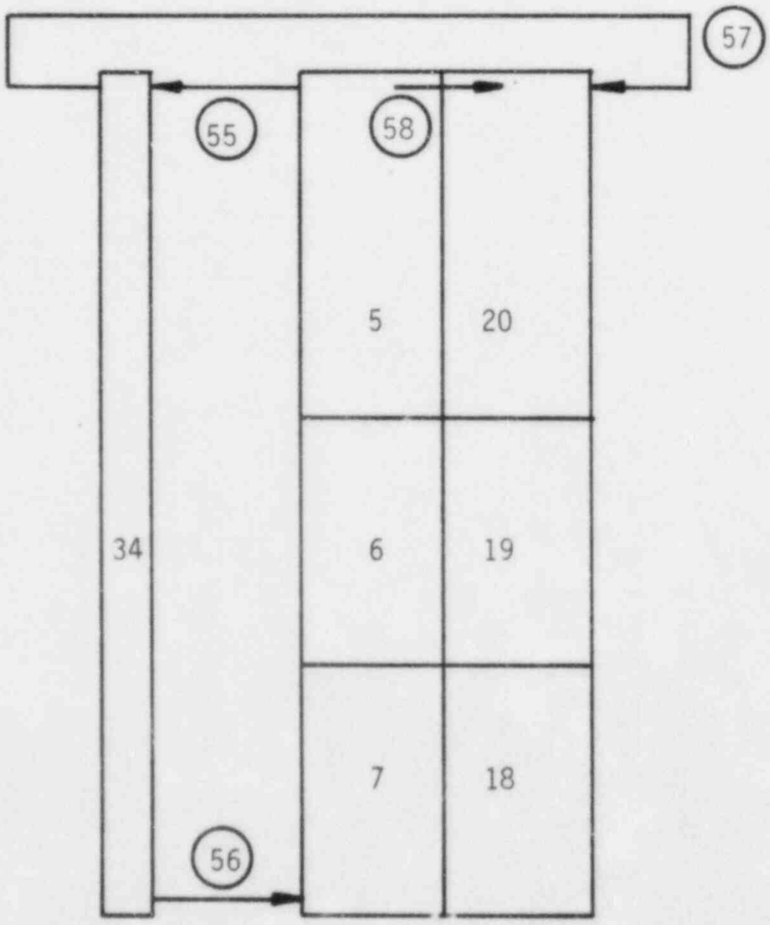


Figure 5-19. Detailed Mini-TRAP2 Noding/Flowpath Diagram for SGTR Model



Node Description

5,6,7	Primary OTSG Tube Region
18,19,20	Secondary OTSG Tube Region
34	OTSG Tube

Flowpath Description

55	OTSG Tube (steady-state only)
56	OTSG Tube
57	Lower End of Ruptured OTSG Tube
58	Upper End of Ruptured OTSG Tube

Figure 5-20. Steam Generator Tube Leaks -- Operator Action Outline

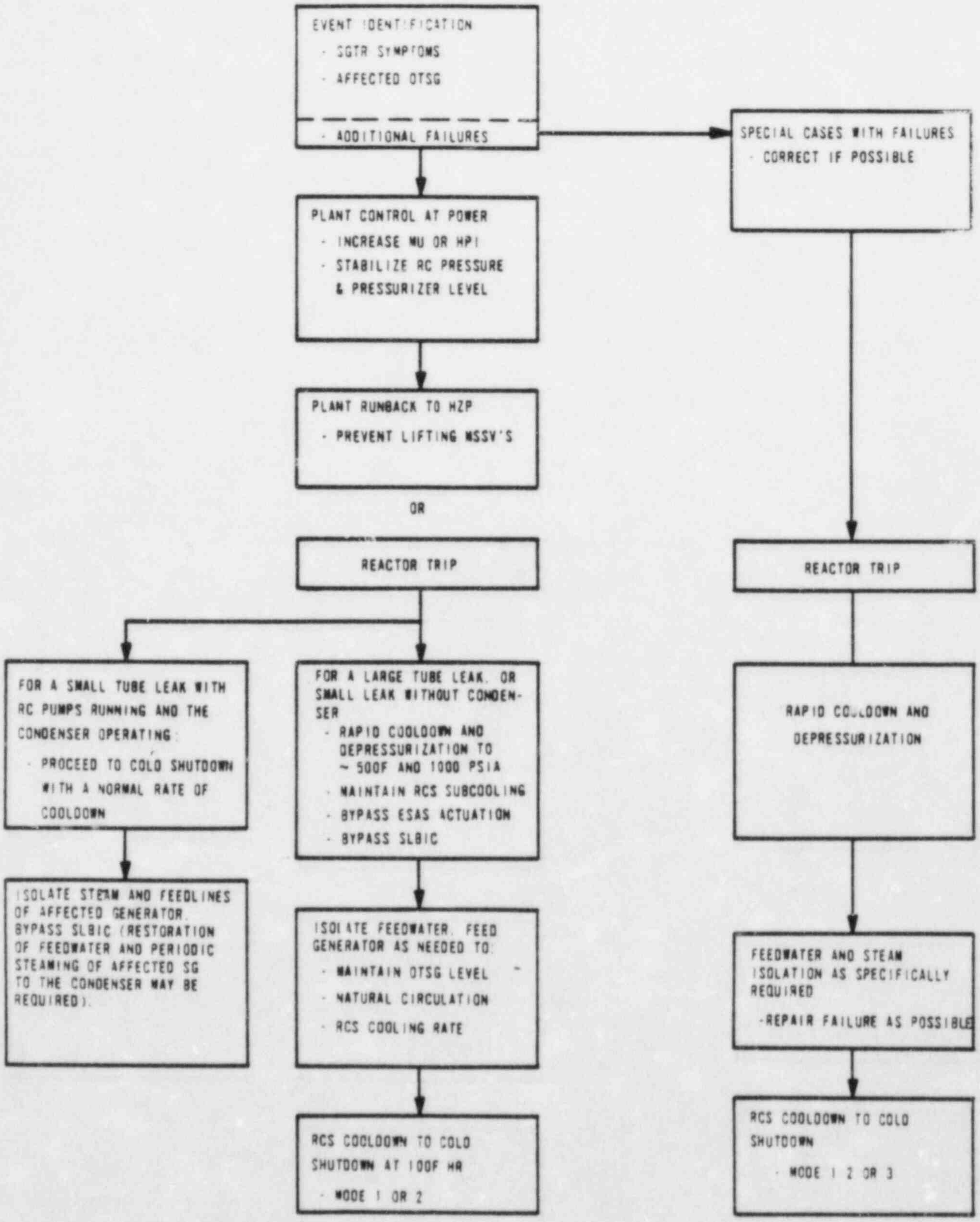


Figure 5-21. Leak Flow Vs Time After Rupture

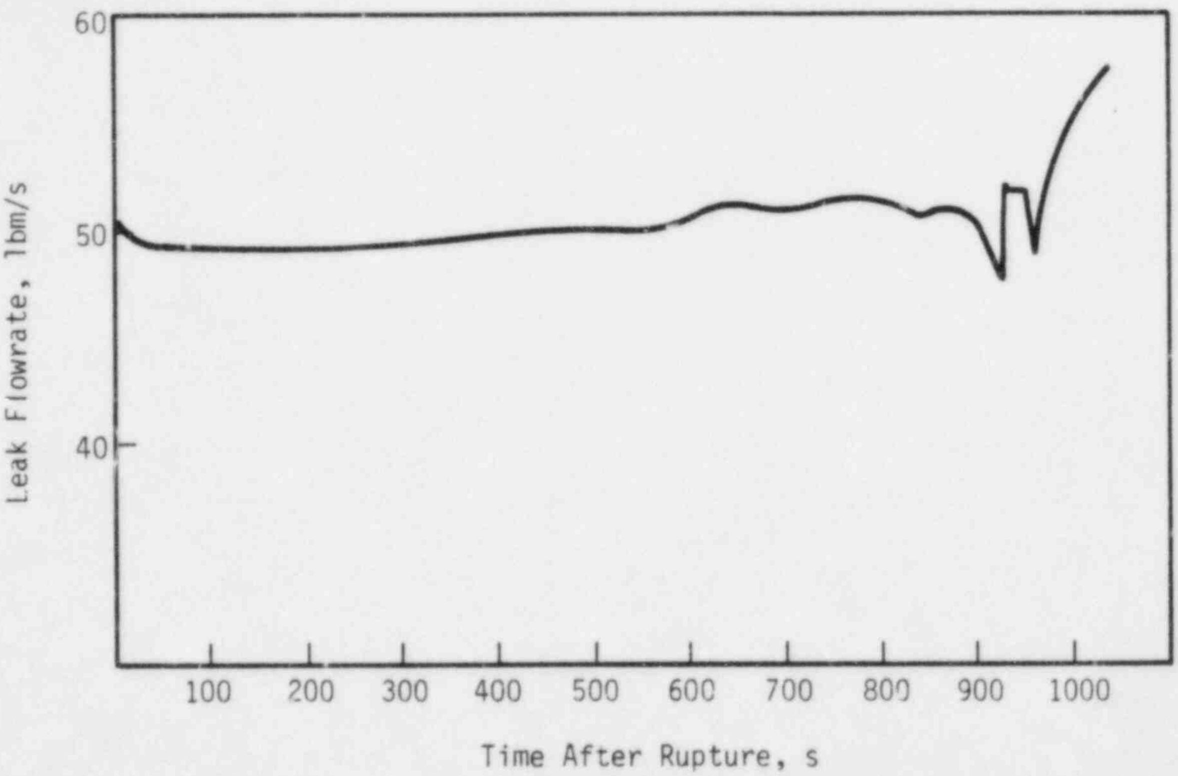


Figure 5-22. Total Reactor Power Vs Time After Rupture of a Single Double-Ended SG Tube

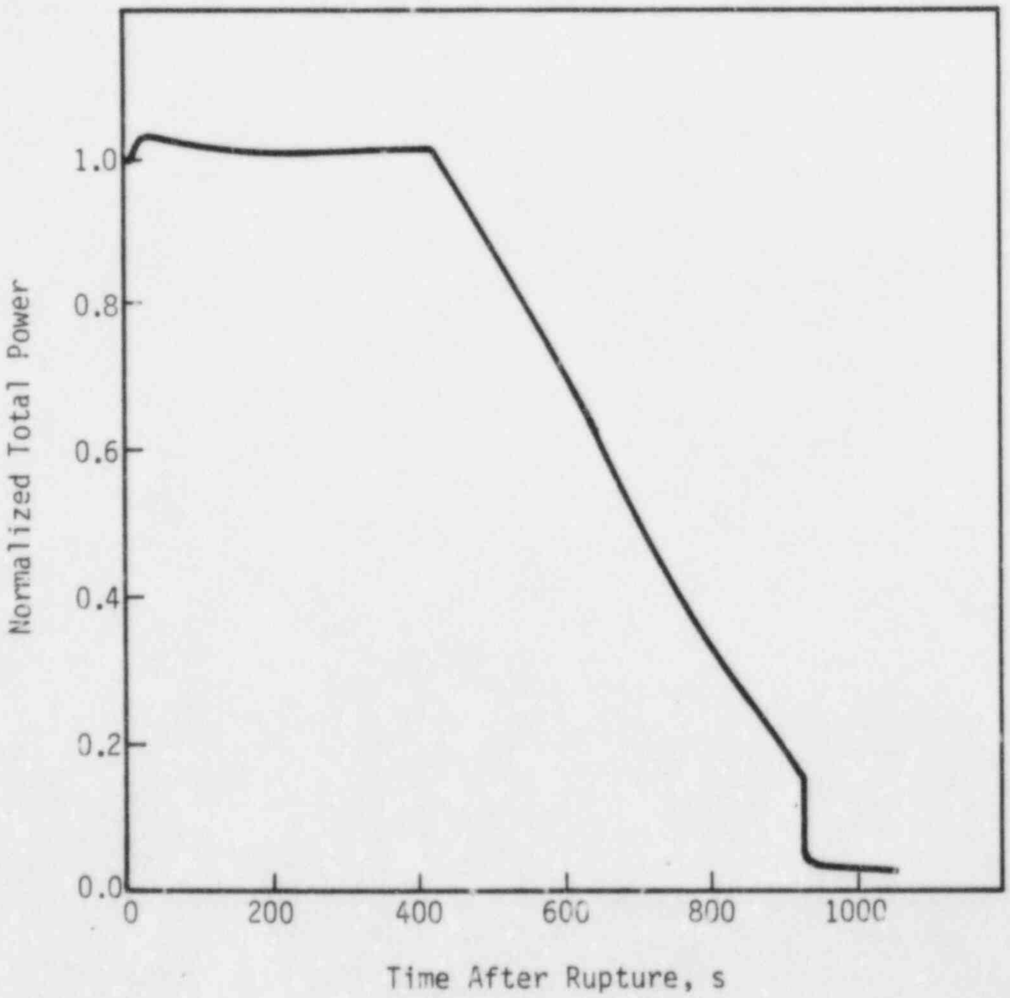


Figure 5-23. Power Runback to 15%

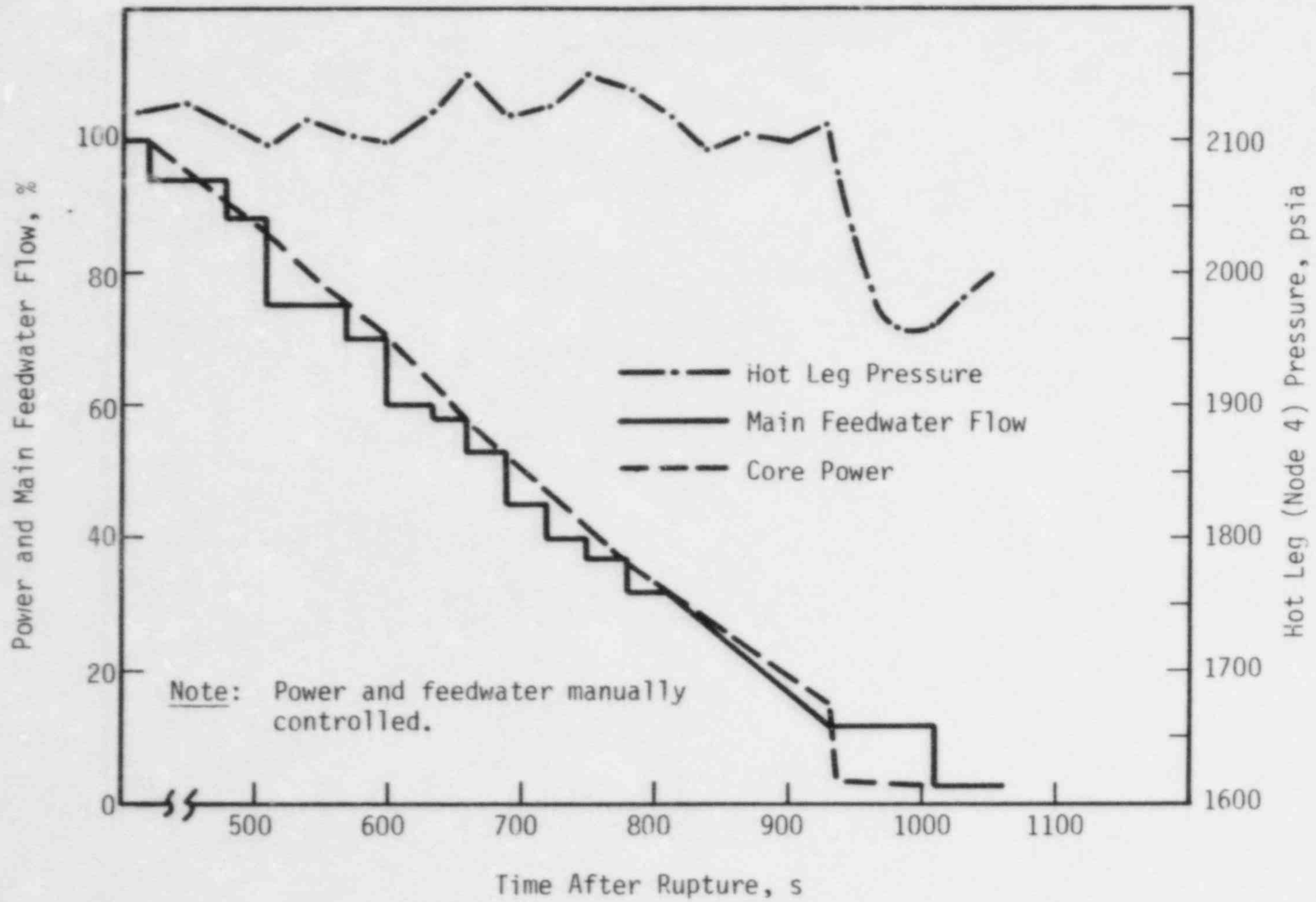


Figure 5-24. Pressurizer Inventory Vs Time After Rupture

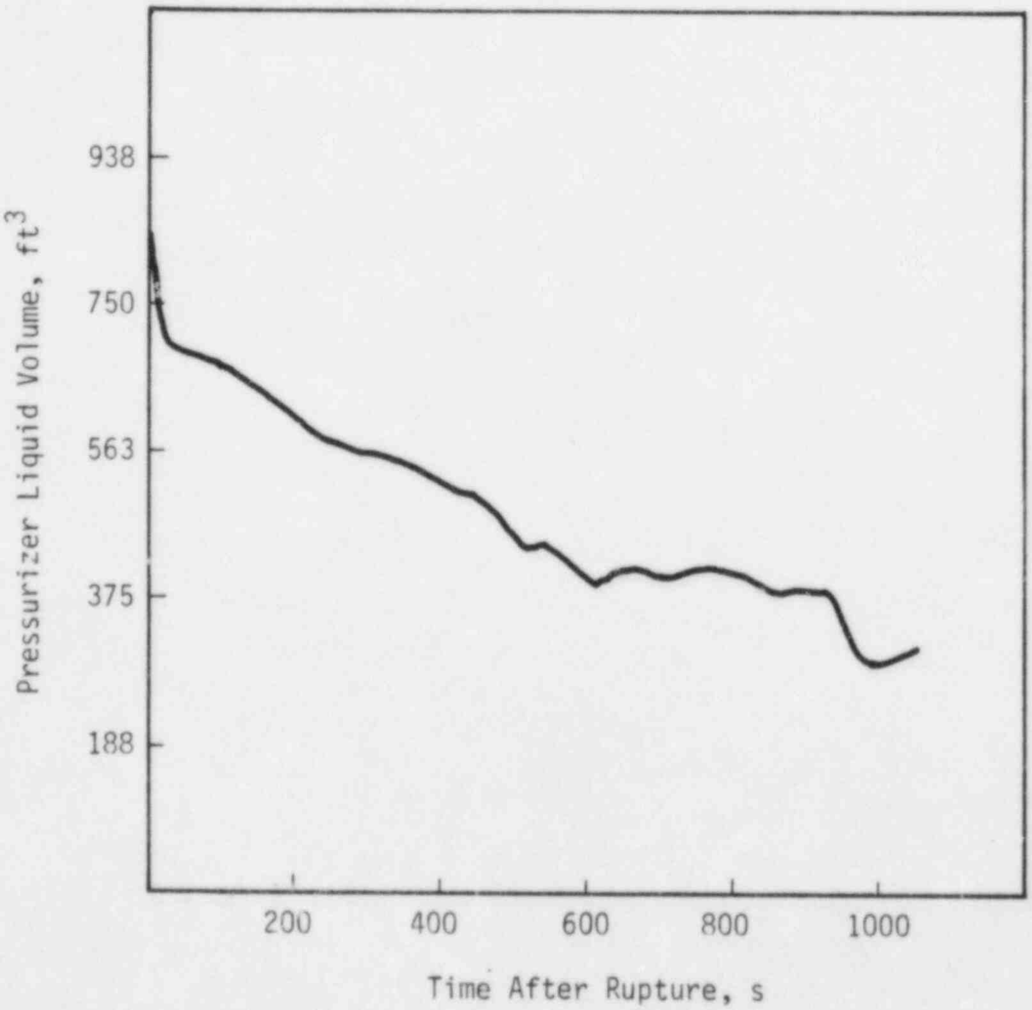


Figure 5-25. Hot and Cold Leg Temperature
Vs Time After Rupture

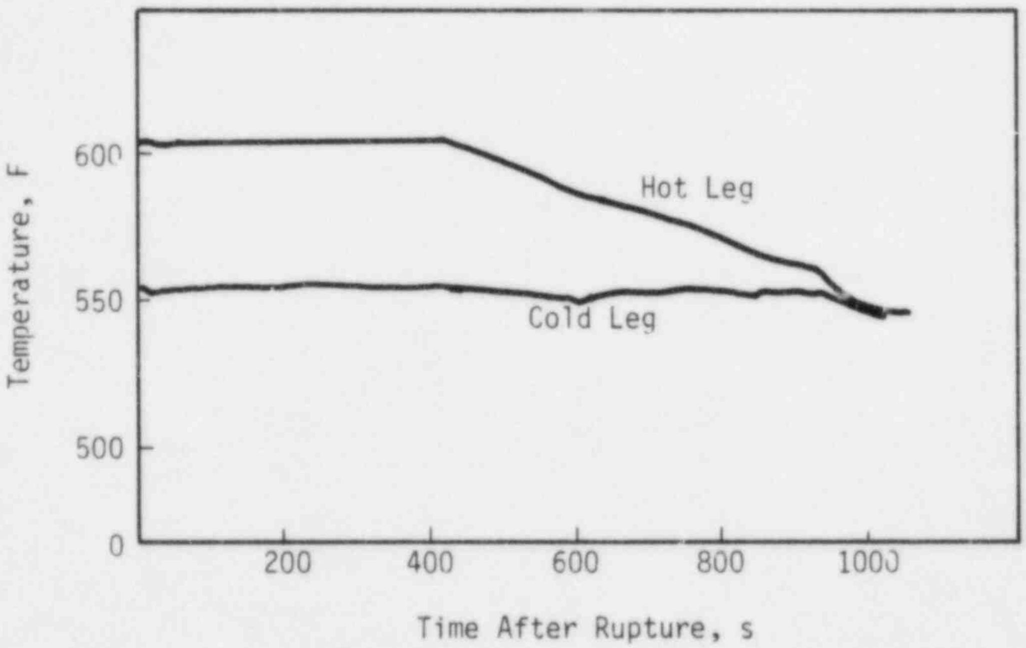


Figure 5-26. Core Outlet Pressure Vs
Time After Rupture

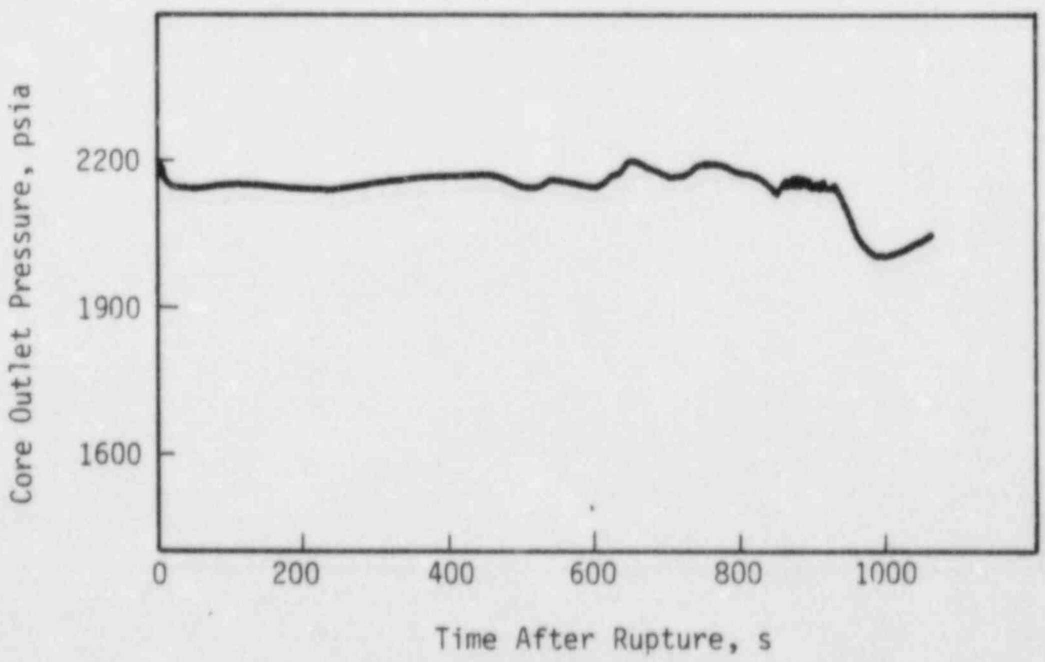


Figure 5-27. Surge Line Flow Vs Time After Rupture

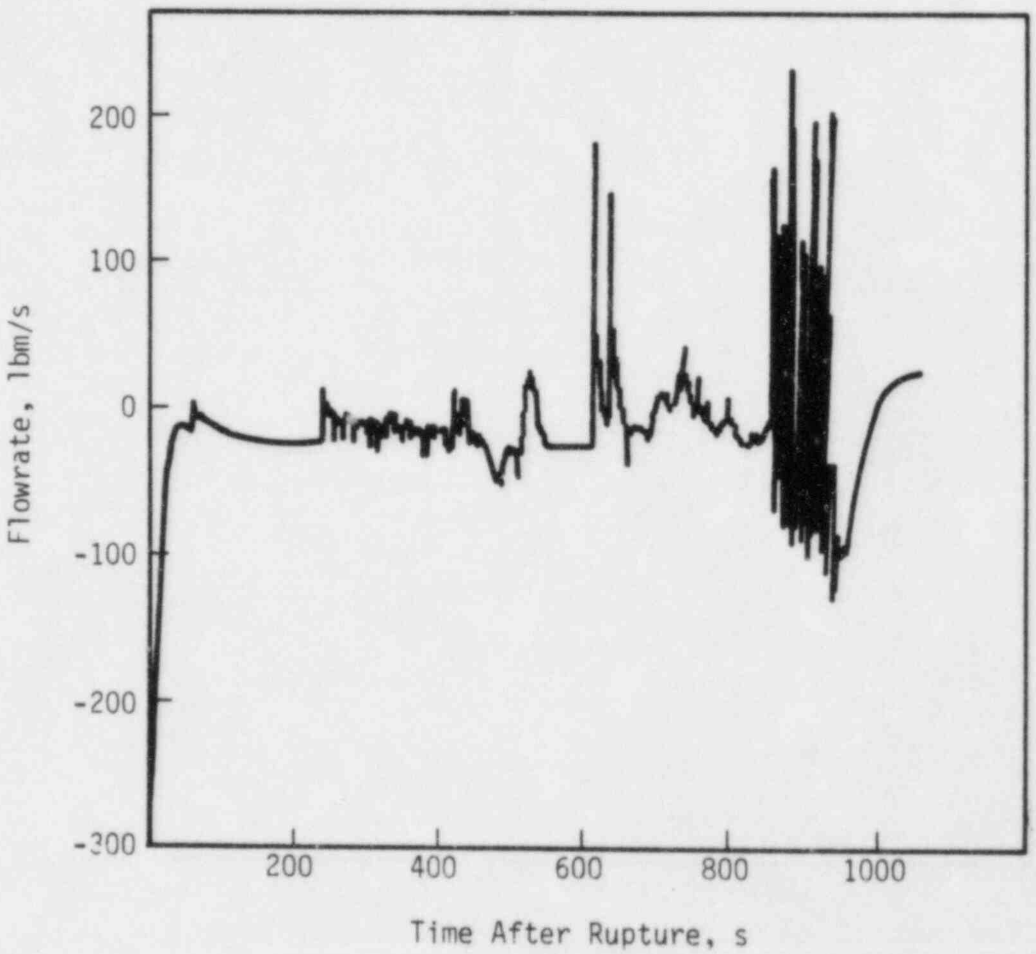


Figure 5-28. Total RCS Charging Flow Vs Time After Tube Rupture

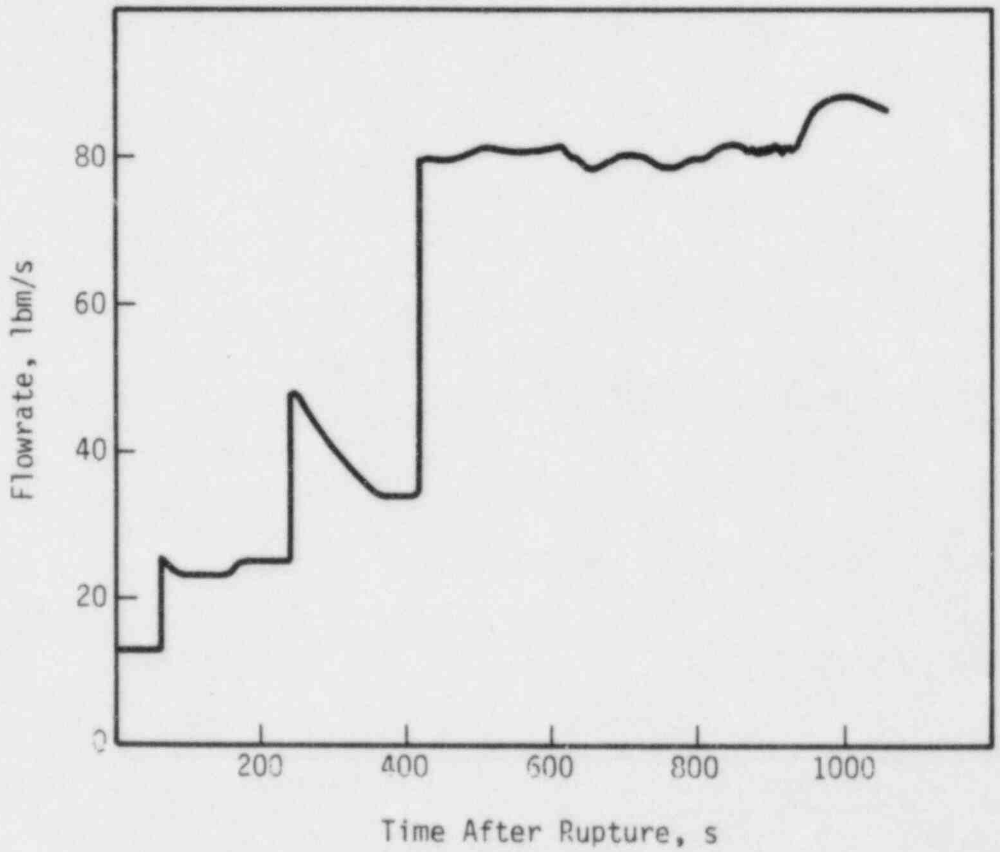


Figure 5-29. SG Tube Flow Vs Time After Rupture

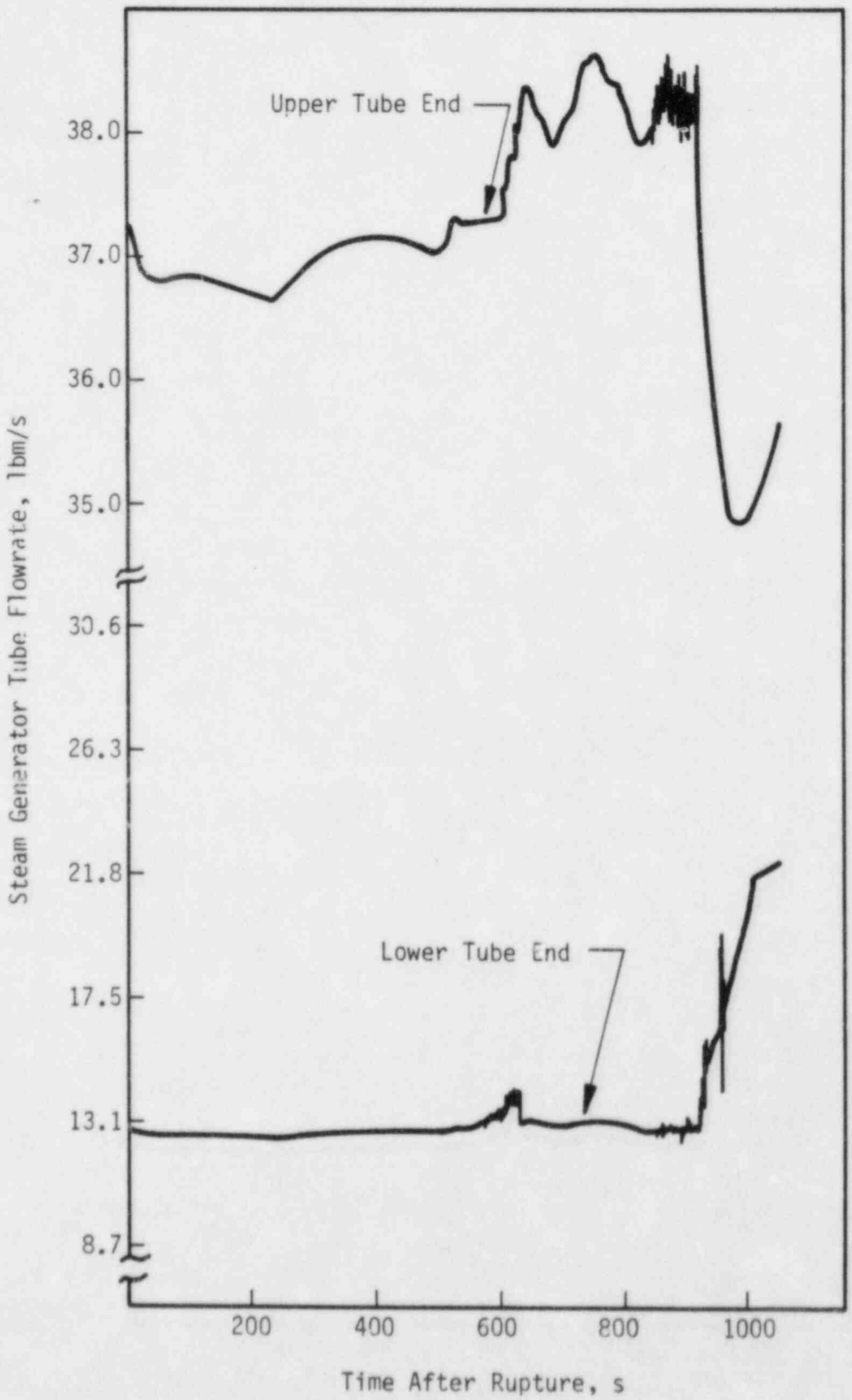
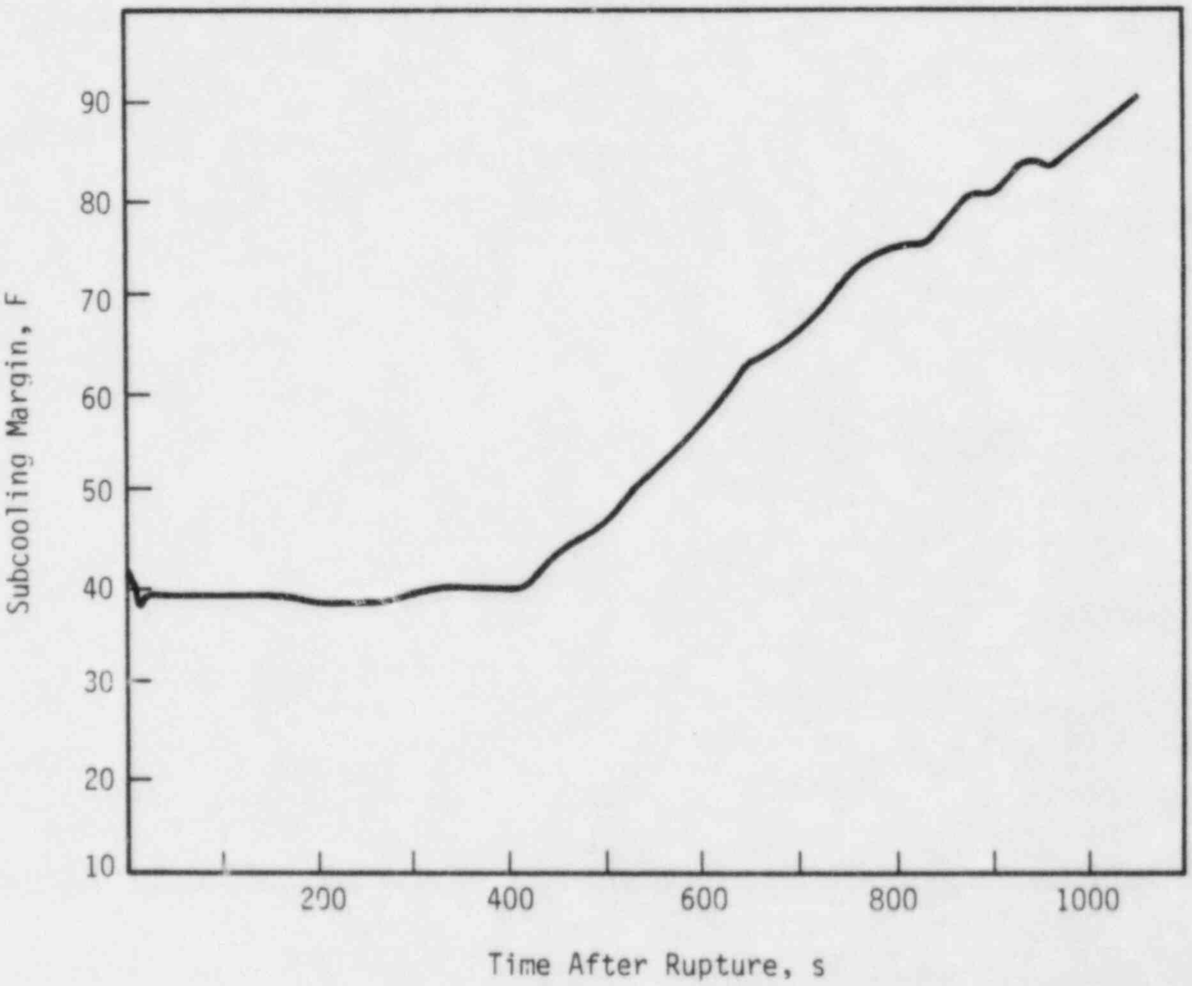


Figure 5-30. Subcooling Margin Vs Time After Rupture



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