ATTACHMENT 1

Conformance of Oconee RCP Trip Setpoint with NRC Guidance

The proposed Oconee RCP trip setpoint, manual pump trip on loss of primary system subcooling, is in accord with the guidance provided by the NRC as an enclosure to Generic Letter 83-10 of February 8, 1983. The applicable points in the guidance are discussed below.

1. Setpoints for RCP Trip

a) The setpoint ensures that the RCP's will be tripped for all loss of primary coolant events for which RCP trip is necessary. As shown in the response to Item 2 of IE Bulletin 79-05C [Parker, W. O. Jr. (Duke), letter to O'Reilly, J. P. (NRC-Region II), August 29, 1979], SBLOCA's which could result in violation of the requirements of 10 CFR 50.46 involve a limited range of break sizes and subsequent RCP trip with a highly voided system. Loss of subcooled margin must occur before significant primary system voiding develops. Therefore, a RCP trip on loss of subcooling will preclude exceeding a peak cladding temperature of 2200°F.

Use of the loss of subcooled margin criterion for reactor coolant pump trip will ensure continuous forced RCS flow during steam generator tube ruptures up to and including the design basis tube rupture. An analysis of a design basis tube rupture event at Oconee, described in Attachment 2, showed that automatic plant actions will

prevent a loss of subcooled margin, so that a manual RCP trip would not be required.

The symptom used to alert the operators of the need to manually trip the reactor coolant pumps is, to a large extent, uniquely attributable to LOCA's. Apart from LOCA's, the only credible scenario which would lead to a loss of subcooled margin is an overcooling event. Overcooling transients can be caused by secondary side failures such as a steam generator overfeed or a loss of steam generator pressure control. Mitigation of these events depends on action to isolate the affected steam generator. Continued forced flow in the RCS is not necessary to ensure core cooling or to limit offsite radiation doses. Forced circulation will be promptly restored when the RCP restart criteria are met. For these reasons a RCP trip on loss of subcooled margin caused by an overcooling transient does not significantly affect plant safety. Furthermore, a simple and unambiguous RCP trip criterion such as a loss of subcooled margin is preferable to a complicated setpoint methodology that would attempt to differentiate between small break LOCA's and overcooling transients.

b) The RCP trip setpoint precludes the possibility of extended RCP operation in a voided system. A loss of subcooled margin must occur before significant voids develop in the RCS. Therefore, extended pump operation in a two-phase system will not occur.

c) RCP trip on a loss of subcooled margin will not lead to challenges to the pressurizer PORV. Overcooling events initially lead to a large

RCS pressure decrease. Following high pressure injection (HPI) initiation on Engineered Safeguards activation, and after operator action to stop excessive primary to secondary heat removal, RCS pressure will turn around and increase as injection flow exceeds the contraction of the reactor coolant. Once the subcooled margin is recovered the operators have clear and unambiguous instructions to throttle HPI and prevent a primary system repressurization. This will prevent a challenge to the automatic pressure relief function of the PORV. Furthermore, forced circulation will be promptly restored when the RCP restart criteria are met, thus allowing normal pressurizer pressure control. Moreover, there is no need for manual PORV operation to depressurize the RCS during an overcooling event.

d) It is possible, although unlikely, that a RCP trip on loss of subcooled margin will lead to the establishment of a hot, stagnant fluid region in the high points of the primary system. The operations staff has been trained to detect, manage, and remove coolant voids that result from flashing. Furthermore, extensive guidance for dealing with RCS voids will be provided in our upgraded, symptom oriented emergency procedures.

e) Extended RCP operation at Oconee requires cooling water to the pump motor and either seal injection flow or component cooling to the pump thermal barrier. Containment isolation does not isolate seal injection, so it will not lead to pump seal damage. Containment isolation will interrupt component cooling and RCP motor cooling; however, these services can be restored quickly when a pump restart

is desired. Furthermore, containment isolation results from a high Reactor Building pressure (4 psig) signal. The only overcooling event which could lead to high building pressure is a steam line break inside containment. It is much more likely that an overcooling transient would not degrade Reactor Building conditions and therefore not result in containment isolation of RCP cooling water systems.

f) Loss of RCS subcooled margin is considered to be the optimal indicator of when the RCP's should be tripped. This setpoint will ensure that the pumps will be tripped for LOCA's which could exceed the limits of 10 CFR 50.46 if the pumps were left on and then tripped at a later time. The indication is not a completely unambiguous indication of a LOCA since some moderately severe overcooling transients could also result in a loss of subcooled margin. However, a RCP trip during such an overcooling event will not affect plant safety since a pump can be promptly restarted when the restart criteria are met. Furthermore, a simple and unambiguous trip setpoint is preferable to a complicated set of parameters which would differentiate between LOCA's and overcooling events.

2. Guidance for Justification of Manual RCP Trip

a) Analyses demonstrate that operator action to trip the RCP's two minutes following a loss of subcooled margin will not result in exceeding the limits set forth in 10 CFR 50.46 (see Attachment 3).

b) An analysis utilizing some best estimate values and some conservative assumptions has shown that at least 10 minutes are available for the operators to trip the reactor coolant pumps following a loss of sub-cooled margin (see Attachment 3). This is considered to be more than adequate time considering the high level of emphasis on tripping the RCP's that will be maintained in the training program, and the clear and unambiguous setpoint for RCP trip. In the event that the RCP's are not tripped in the preferred time frame the operators are trained to leave the pumps running, thus ensuring adequate core cooling.

It should be noted that the generic analysis performed to determine the best estimate time available to trip the reactor coolant pumps was conservative relative to Oconee in several respects. First, the initial core power level assumed was 2772 MW, which is greater than the Oconee full power rating of 2568 MW. The lower steady state power at Oconee results in lower decay heat power levels and less severe consequences from a loss of primary coolant. In addition, Oconee's HPI capacity is significantly greater than the flow assumed in the generic analysis. The additional injection flow would provide quicker recovering of the core following a delayed RCP trip. Furthermore, the radial and axial power peaking assumed in the generic analysis is much worse than the peaking seen in normal operation at Oconee. Less severe peaking would result in lower cladding temperatures in the hottest core channel and thus longer times available for a manual reactor coolant pump trip.

3. Other Considerations

a) The parameter employed in the RCP trip setpoint, subcooled margin, is determined by the RCS pressure and the temperature of the hot legs (hot leg subcooled margin) and by the RCS pressure and the temperature of the core exit thermocouples (core subcooled margin).

A loss of subcooled margin from any indication will require RCP trip. The instrumentation used to detect the RCS pressure and hot leg temperatures is redundant and safety related, since these instruments also feed the Reactor Protective System. The core exit thermocouples will be improved to safety grade as a part of the Oconee inadequate core cooling upgrades. Normally the pressure and temperature are converted to a subcooled margin indication by the plant computer. In this determination of subcooled margin an allowance will be made for instrument error, so that a loss of subcooled margin cannot occur without the appropriate indication. The subcooled margin indications themselves are a prominent LED's on the primary control console. In the unlikely event that a unit is being operated while the plant computer is down, the operators will use the RCS pressure and temperature indication to determine the subcooled margin. In any event, they will be alerted to a potential loss of subcooled margin by a low RCS pressure Engineered Safeguards alarm and activation at 1500 psig.

b) The upgraded, symptom oriented emergency procedures provide extensive guidance for the timely restart of the reactor coolant pumps when conditions which will support safe pump operation are established.

c) The operator training program instructs operators in their responsibility for RCP trip in the event of a SBLOCA. In the emergency procedures, prompt RCP trip is required upon loss of subcooled margin regardless of its time of occurance in the emergency procedures.

ATTACHMENT 2

Oconee Design Basis Steam Generator Tube Rupture Analysis

The analysis of a design basis steam generator tube rupture (SGTR) event at an Oconee unit was performed using the RETRAN02-MOD002 transient analysis computer code. The analysis showed that the minimum subcooled margin to be expected during such an event is 26°F. Therefore there is ample margin to the manual reactor coolant pump trip setpoint during a design basis tube rupture event.

The transient analysis was characterized by the following assumptions:

1) 1.0 x ANS Standard decay heat
2) 435 gpm initial tube leak rate
3) No primary makeup prior to Engineered Safeguards (ES) injection
4) ES actuation signal on low RCS pressure (1500 psig)
5) One train of high pressure injection (HPI) flow delivered to the RCS 35 seconds after the ES actuation signal was received
6) No operator action

The choice of assumptions led to significant conservatisms in the analysis. The 435 gpm initial leak rate was the FSAR value, but the actual initial leak rate from a double-ended rupture is calculated to be less than 325 gpm. As the pressurizer level dropped the automatic makeup system would provide at least 160 gpm to mitigate the loss of primary coolant, but no credit was taken for makeup flow. The actual plant ES setpoint is 1550 psig, which would result in a quicker system refill. Realistically, flow from three HPI pumps rather

than one would be available to compensate for the tube leakage and refill the RCS. The actual delay time for HPI flow following the receipt of the low pressure ES signal would be less than ten seconds rather than 35 seconds (which is applicable to a loss of offsite power event). Most importantly, the assumption of no operator action is highly pessimistic. The symptoms of a tube rupture are clear and unambiguous -- high secondary side radiation alarms, decreasing RCS pressure, decreasing pressurizer level, and high makeup flow. The operators would certainly take prompt action to increase injection and shutdown the reactor, thus avoiding any possiblity of a reactor coolant pump trip. However, even with very conservative assumptions the reactor coolant pump trip setpoint (loss of subcooled margin) was not exceeded.

The transient was characterized by the following sequence of events:

Time(set)	Event
0	Steam generator tube rupture
13	Pressurizer heaters on due to low RCS pressure
326	Reactor trip on variable low pressure-temperature
	Subcooled margin = 36°F
328	MSSV's begin to lift
346	All MSSV's reseated
	Pressurizer level offscale low
364	ES actuation signal on low RCS pressure
399	One train of HPI begins injection to RCS

400	Minimum subcooled margin = 26°F
	Minimum RCS pressure = 1335 psig
600	End of analysis

As soon as HPI flow begins the transient is turned around, because the injection flow from one pump is greater than the tube leakage at low RCS pressures. A sensitivity study was performed with the more appropriate 10 second delay between the ES actuation signal and the beginning of HPI flow. For that case the minimum subcooled margin was 29°F.

Plots of RCS pressure, temperature, subcooled margin, and pressurizer level are shown on Figures 1-4.

Figu

OCONEE DESIGN BASIS TUBE RUPTURE
RCS PRESSURE vs. TIME

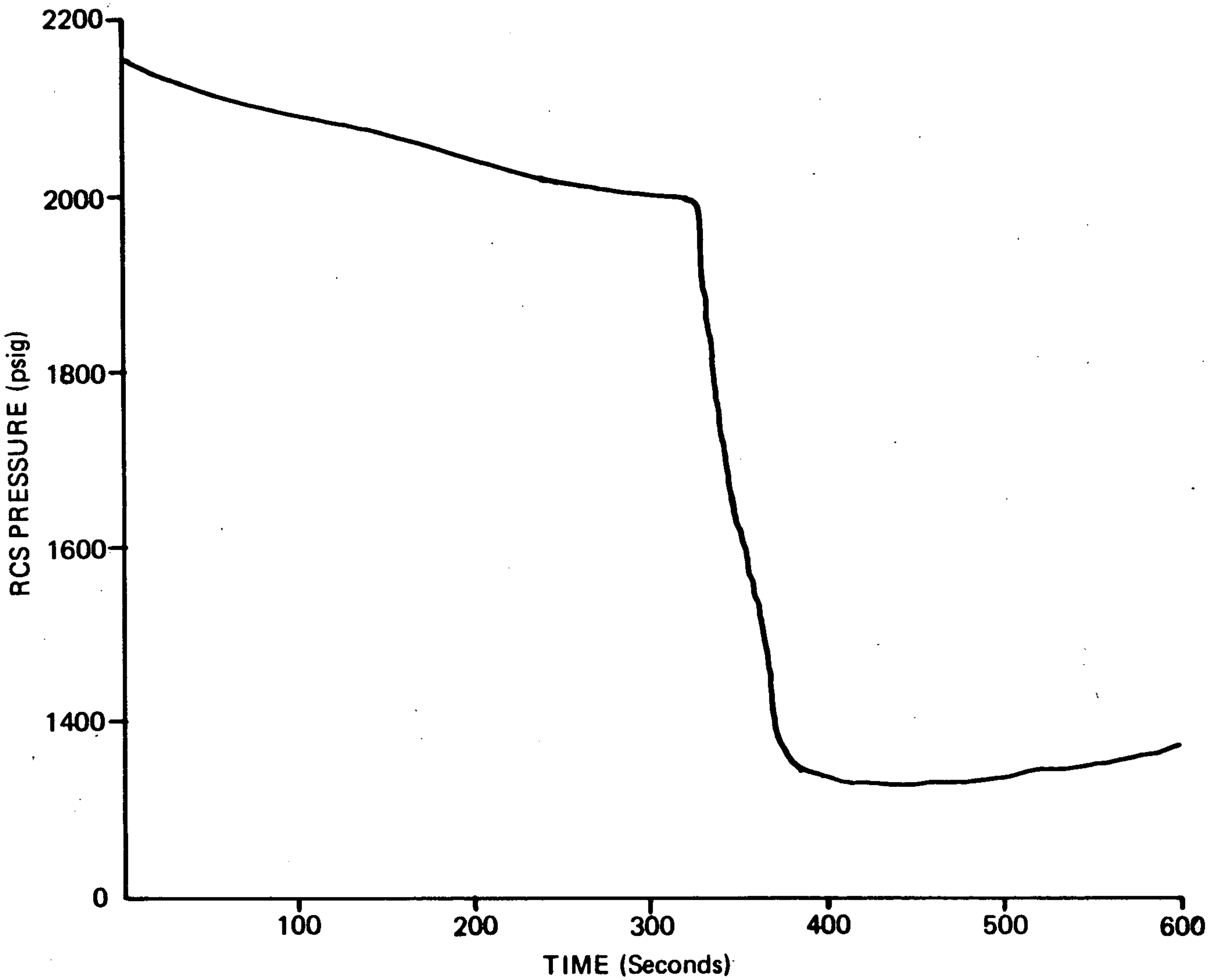
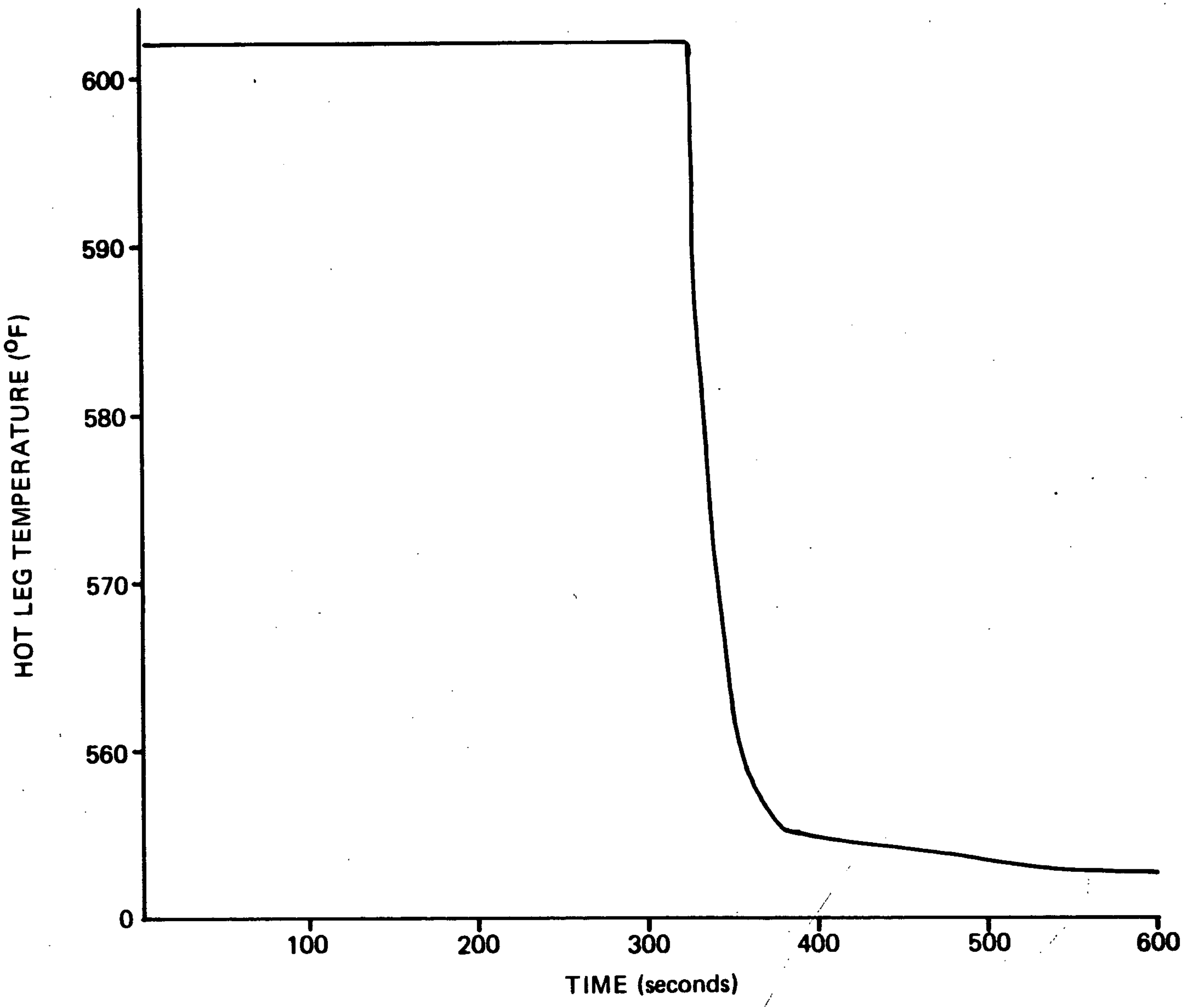


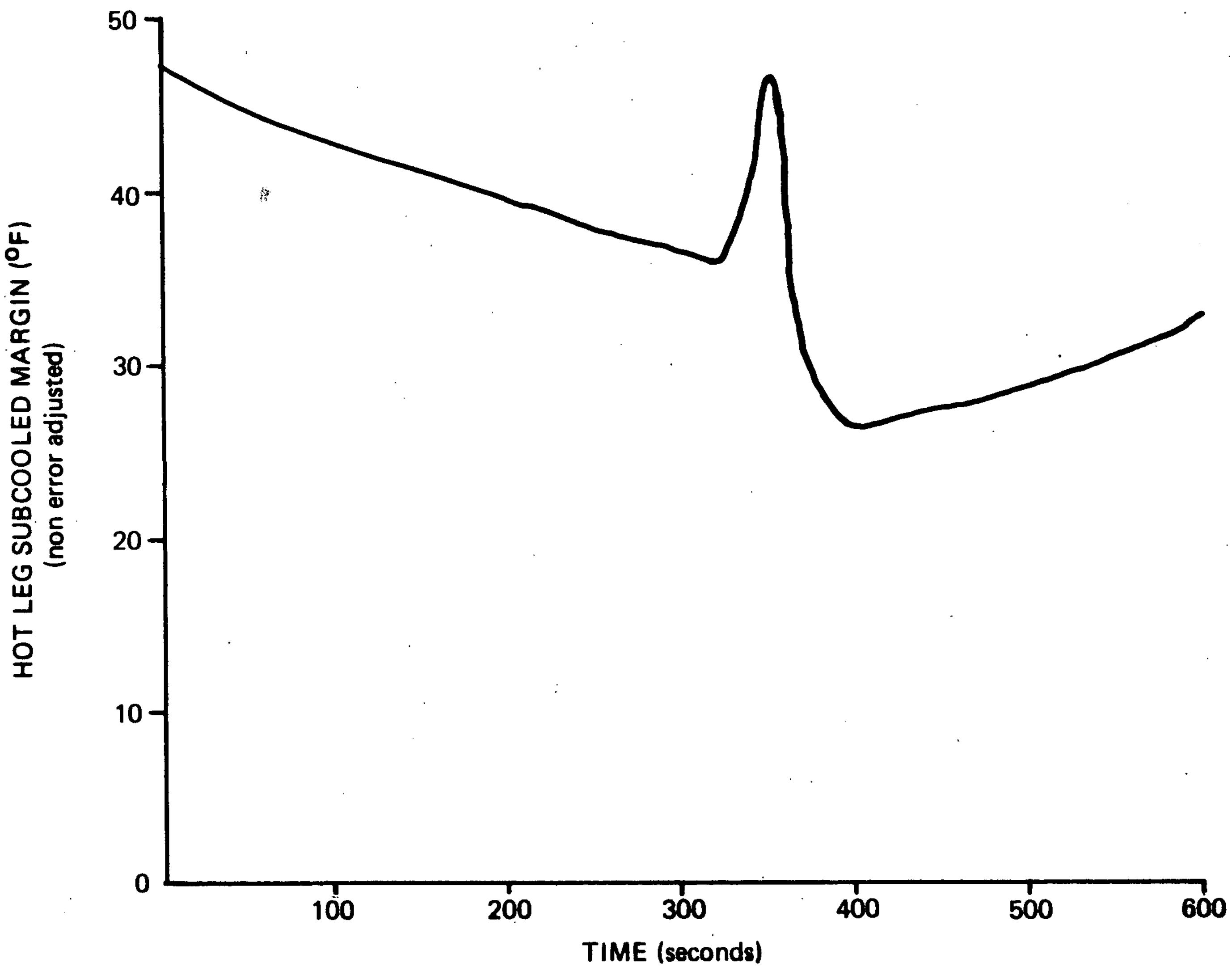
Figure 2

OCONEE DESIGN BASIS TUBE RUPTURE
HOT LEG TEMPERATURE vs. TIME



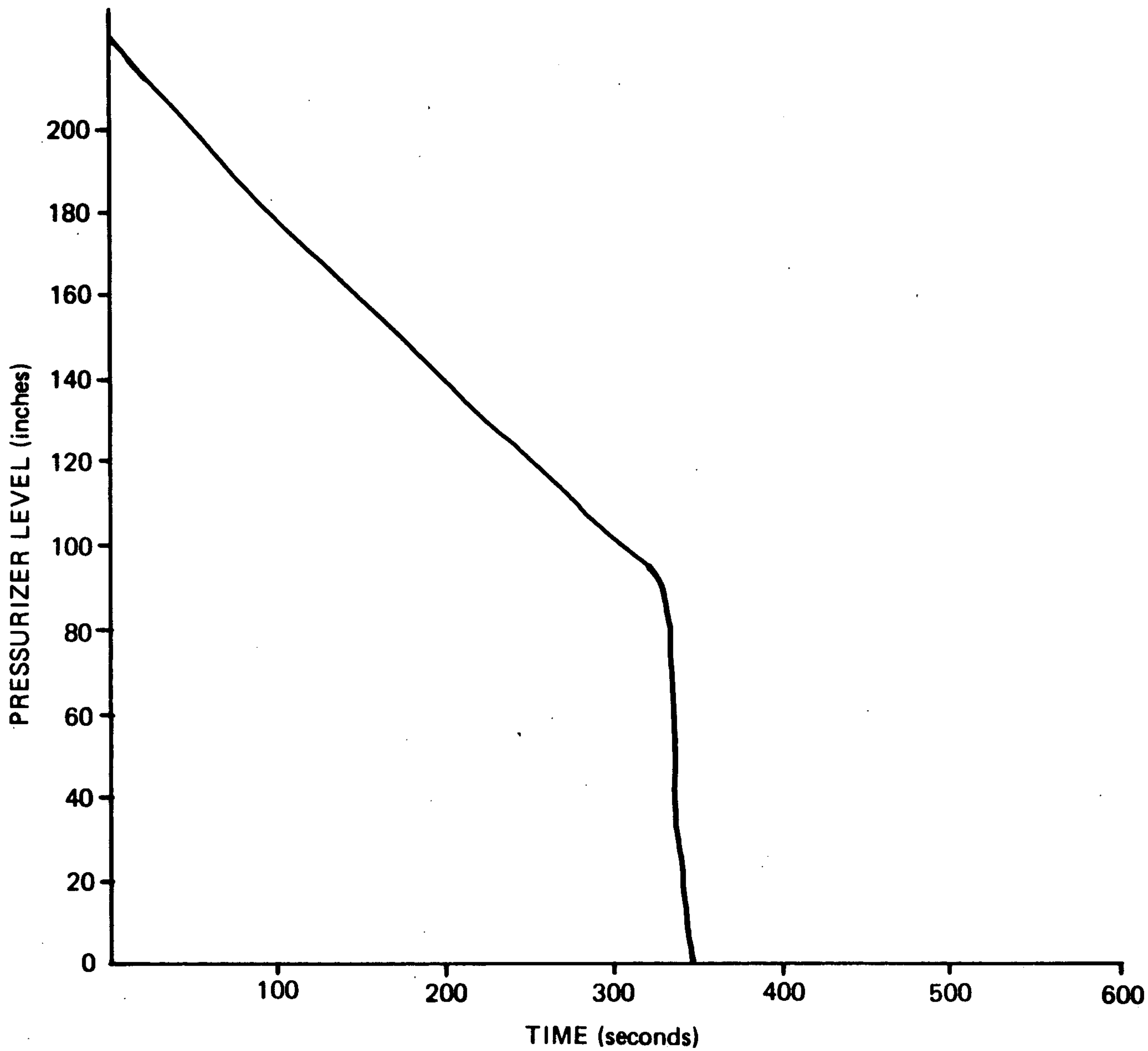
ure 3

OCONEE DESIGN BASIS TUBE RUPTURE
SUBCOOLED MARGIN vs. TIME



e 4

OCONEE DESIGN BASIS TUBE RUPTURE
PRESSURIZER LEVEL vs. TIME



ATTACHMENT 3

Analytical Justification for Manual Reactor Coolant Pump Trip on Loss of Subcooled Margin

Note: Sections of this report pertaining to steam generator tube rupture have been deleted. The analysis of an FSAR-type steam generator tube rupture is included in Attachment 2.

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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 heat-up 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 coast-down. 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

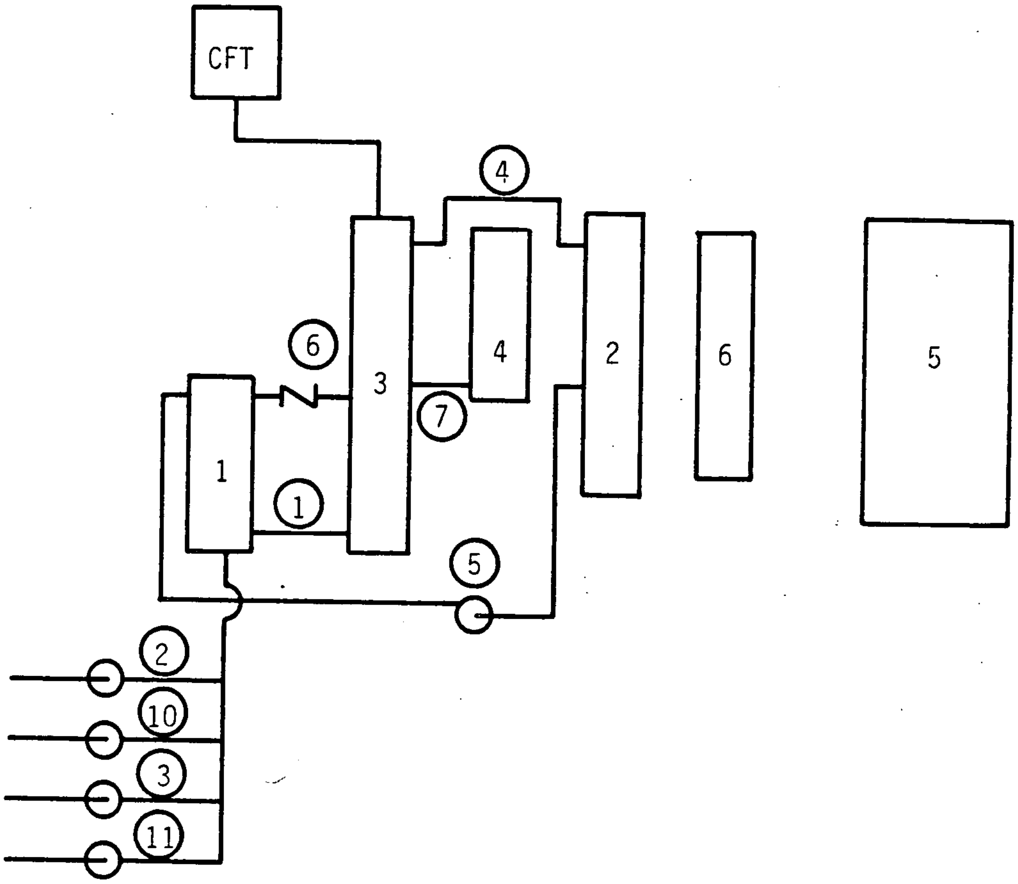
The RCP trip criteria, based on loss of subcooling margin, was developed with the intent of assuring that an indication for RC pump trip would occur

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.

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



Node No.	Identification
1	PD piping, DC, LP
2	Primary SG
3	Core, UP, hot legs
4	Pressurizer
5	Containment
6	Secondary SG

Path No.	Identification
1	Core
2	LPI
3,10,11	HPI
4	Hot legs
5	Pumps
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

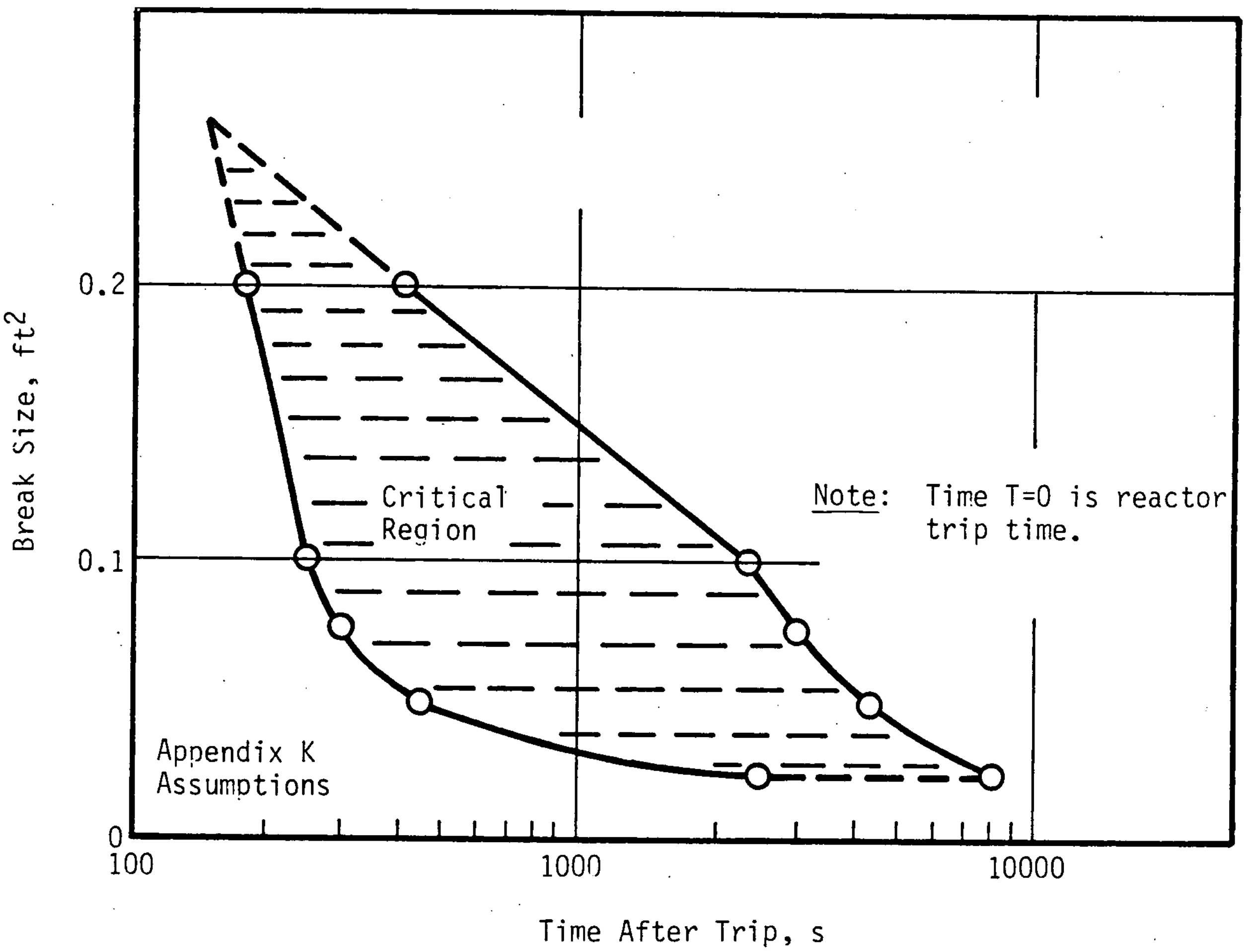


Figure 5-3. "Realistic" Core Axial Peaking Distribution

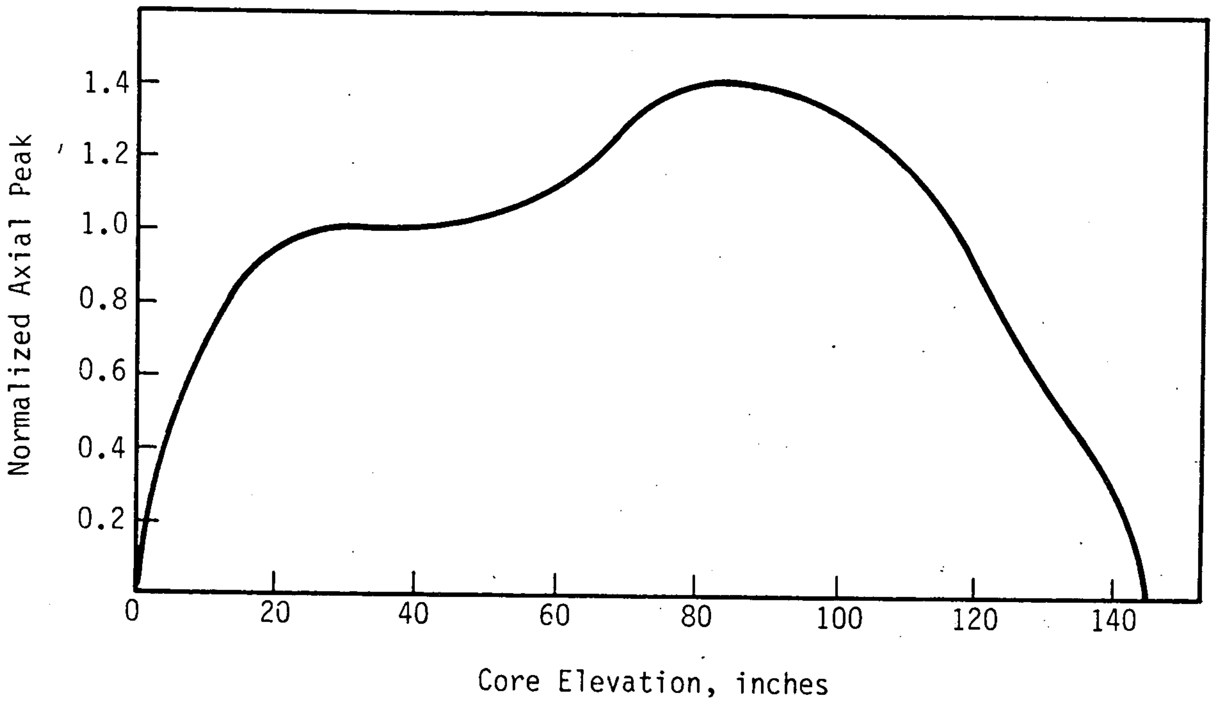


Figure 5-4. Average Primary System Void Fraction
Excluding Pressurizer

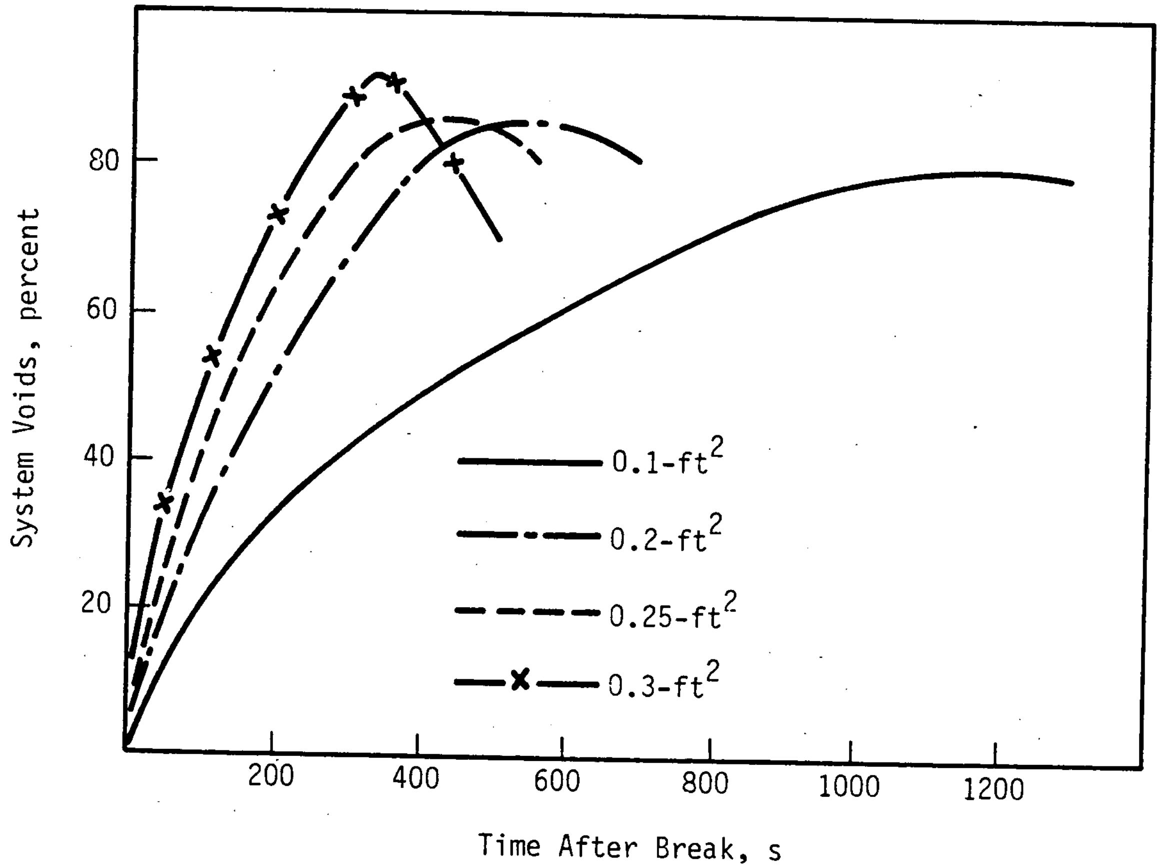


Figure 5-5. RC System Pressure Vs Time (0.1-ft^2 CLD Break -- RCP Trip at 1000 s)

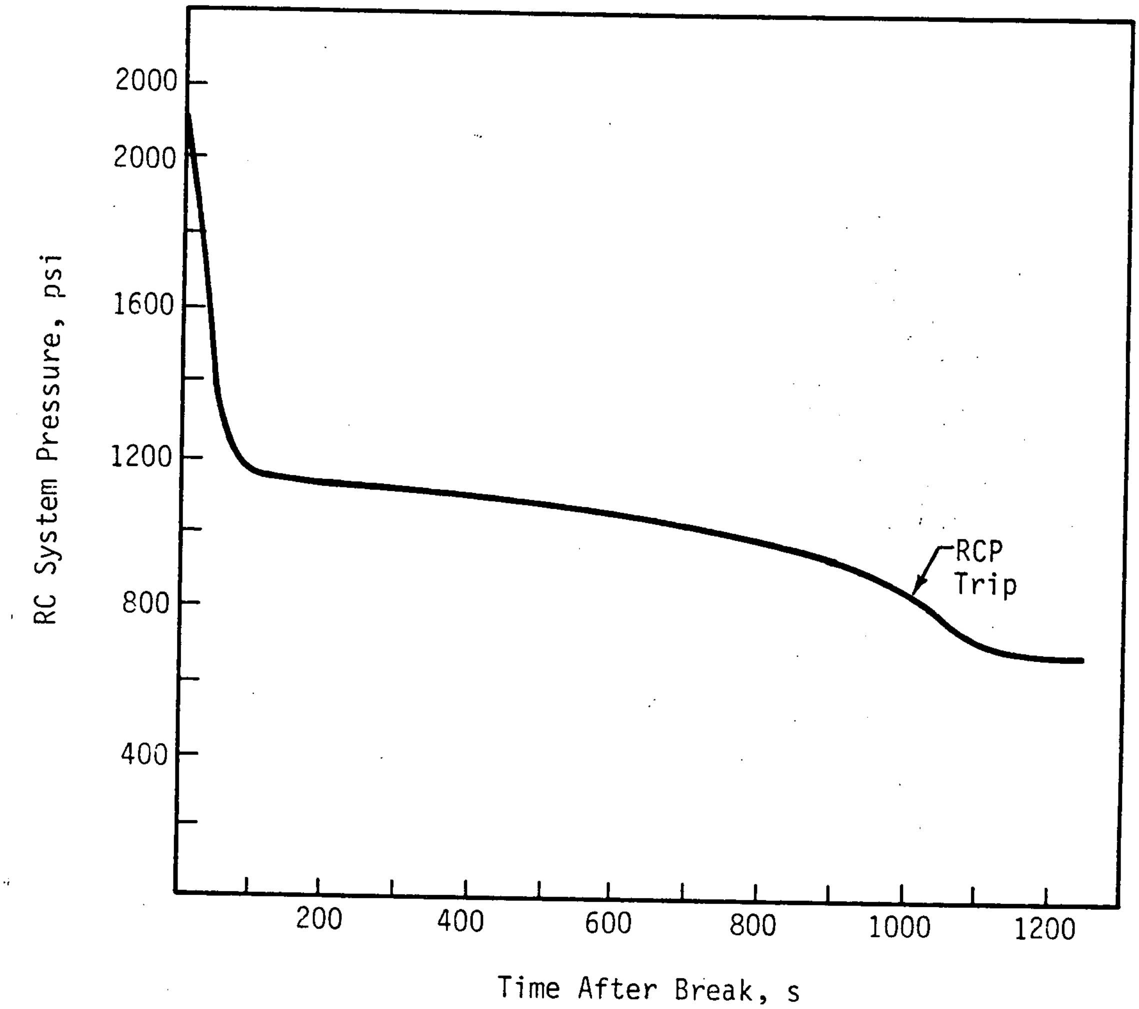


Figure 5-6. Liquid Volume in Reactor Vessel for 0.1-ft^2 CLD Break -- RCP Trip @ 1000 s

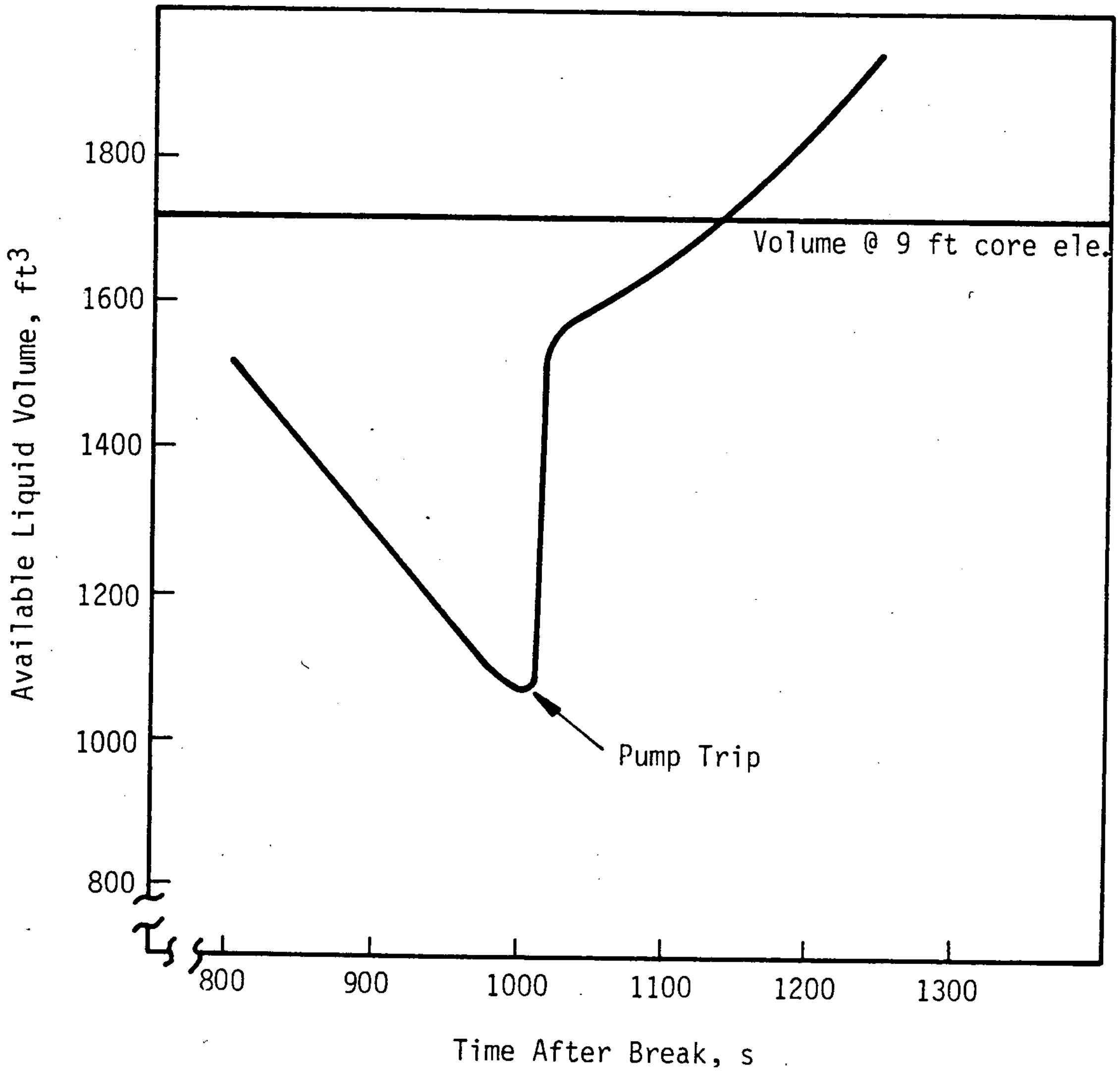


Figure 5-7. RC System Pressure Vs Time (0.2-ft^2 CLD Break -- RCP Trip @ 400 s)

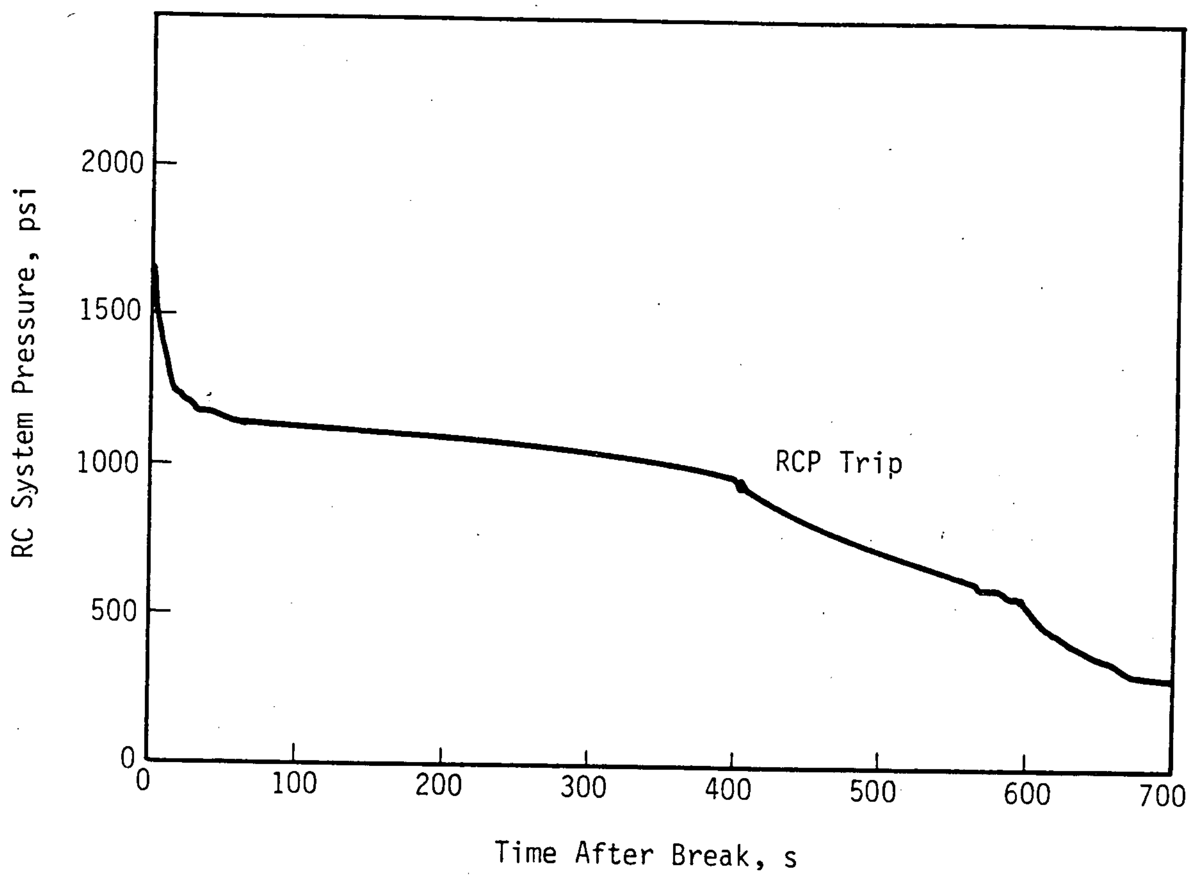


Figure 5-8. Liquid Volume in Reactor Vessel for 0.2-ft^2 CLD Break -- RCP Trip @ 400 s

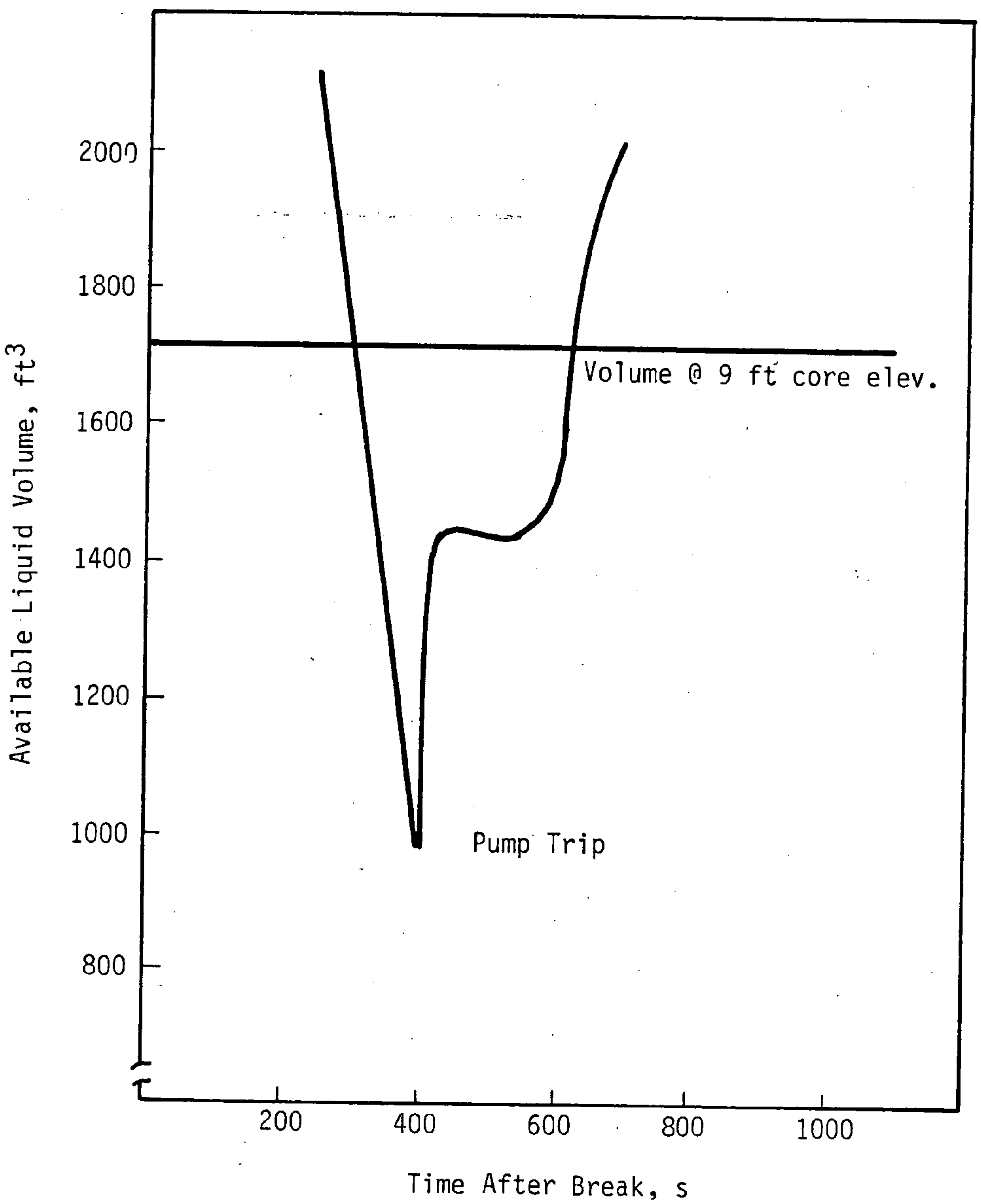


Figure 5-9. RC System Pressure Vs Time (0.25-ft^2 CLD Break -- RCP Trip @ 300 s)

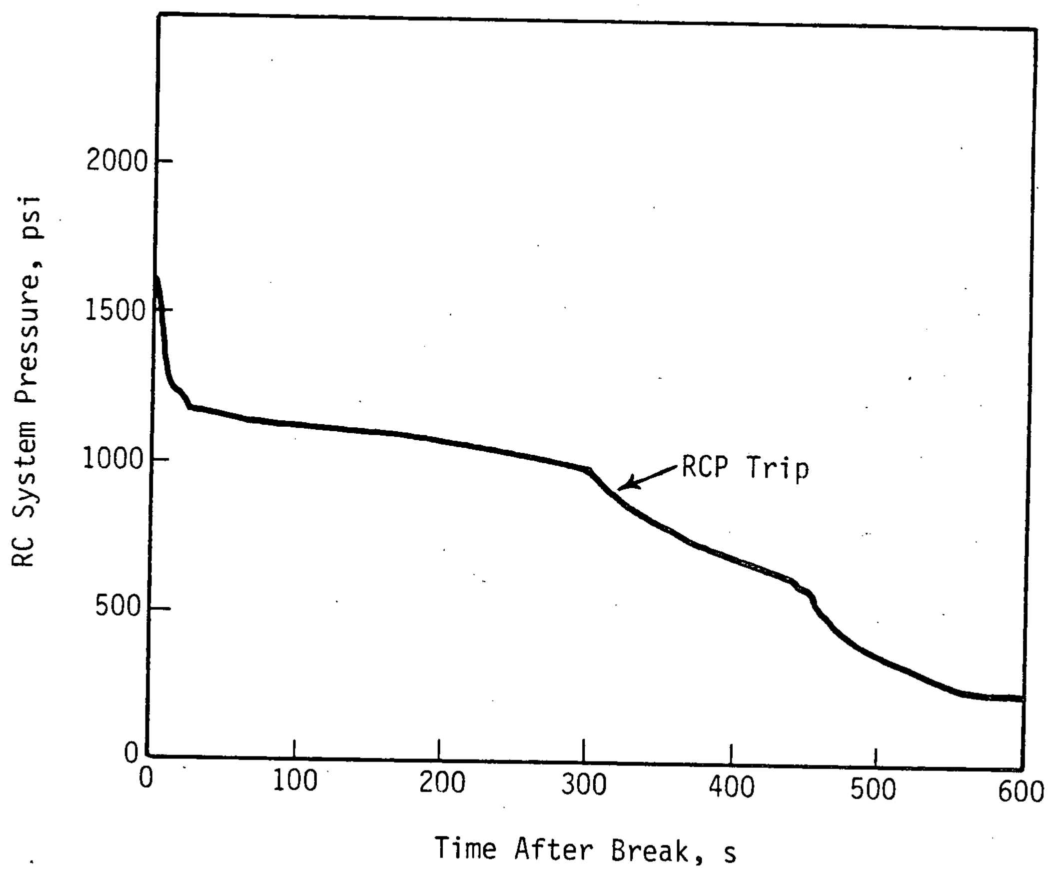


Figure 5-10. Liquid Volume in Reactor Vessel for 0.25-ft^2 CLD Break -- RCP Trip @ 300 s

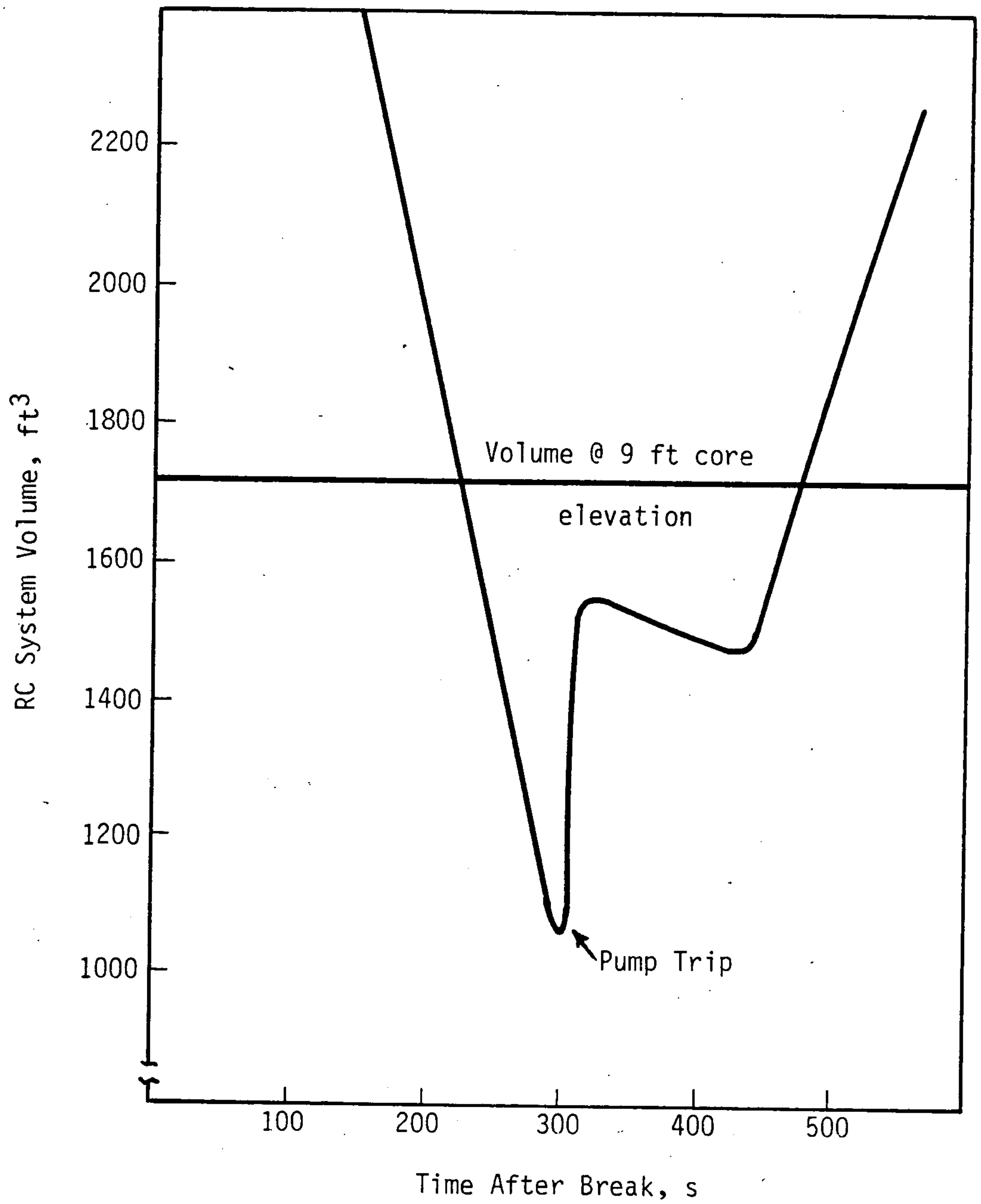


Figure 5-11. RC System Pressure Vs Time (0.3-ft^2 CLD Break -- RCP Trip @ 300 s)

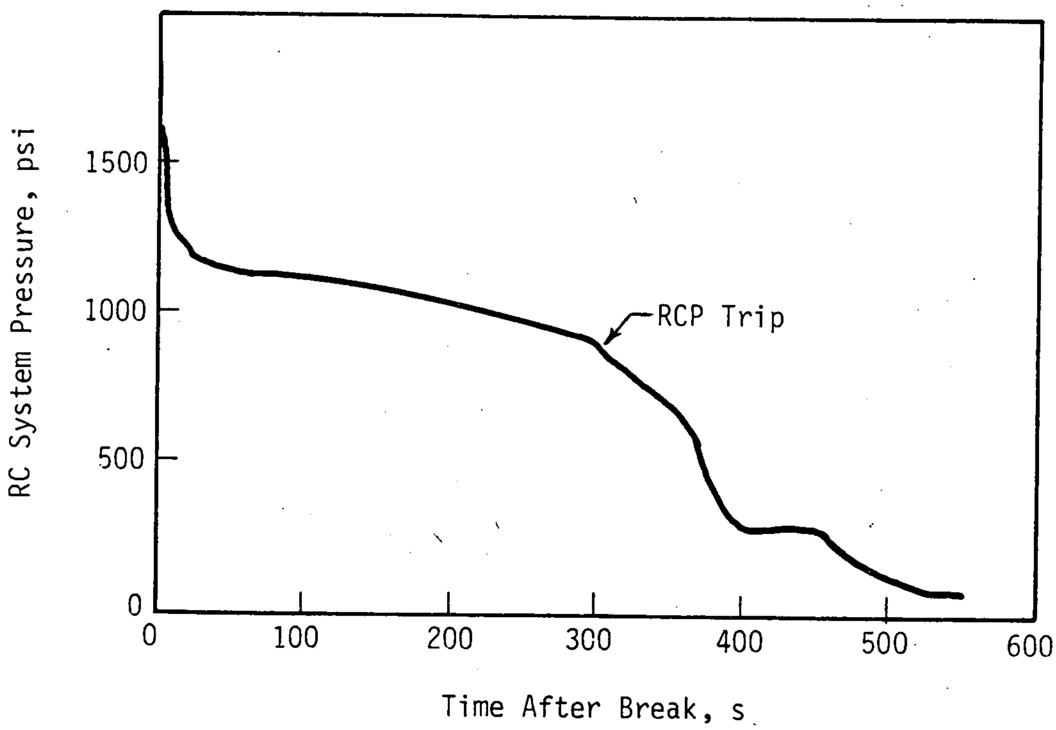


Figure 5-12. Liquid Volume in Reactor Vessel for 0.3-ft^2 CLD Break -- RCP Trip @ 300 s

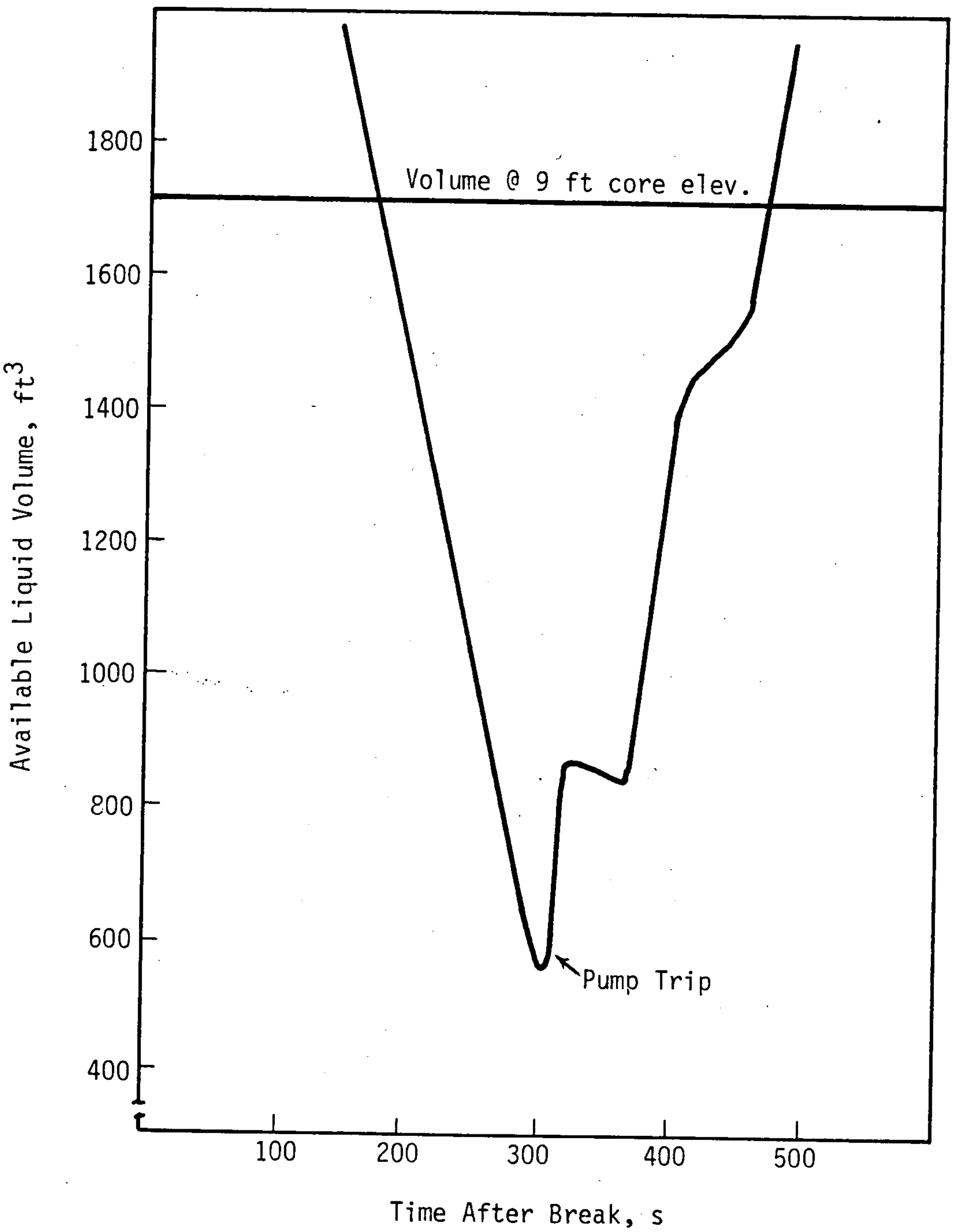


Figure 5-13. SBLOCA Spectrum System Pressure Vs Time

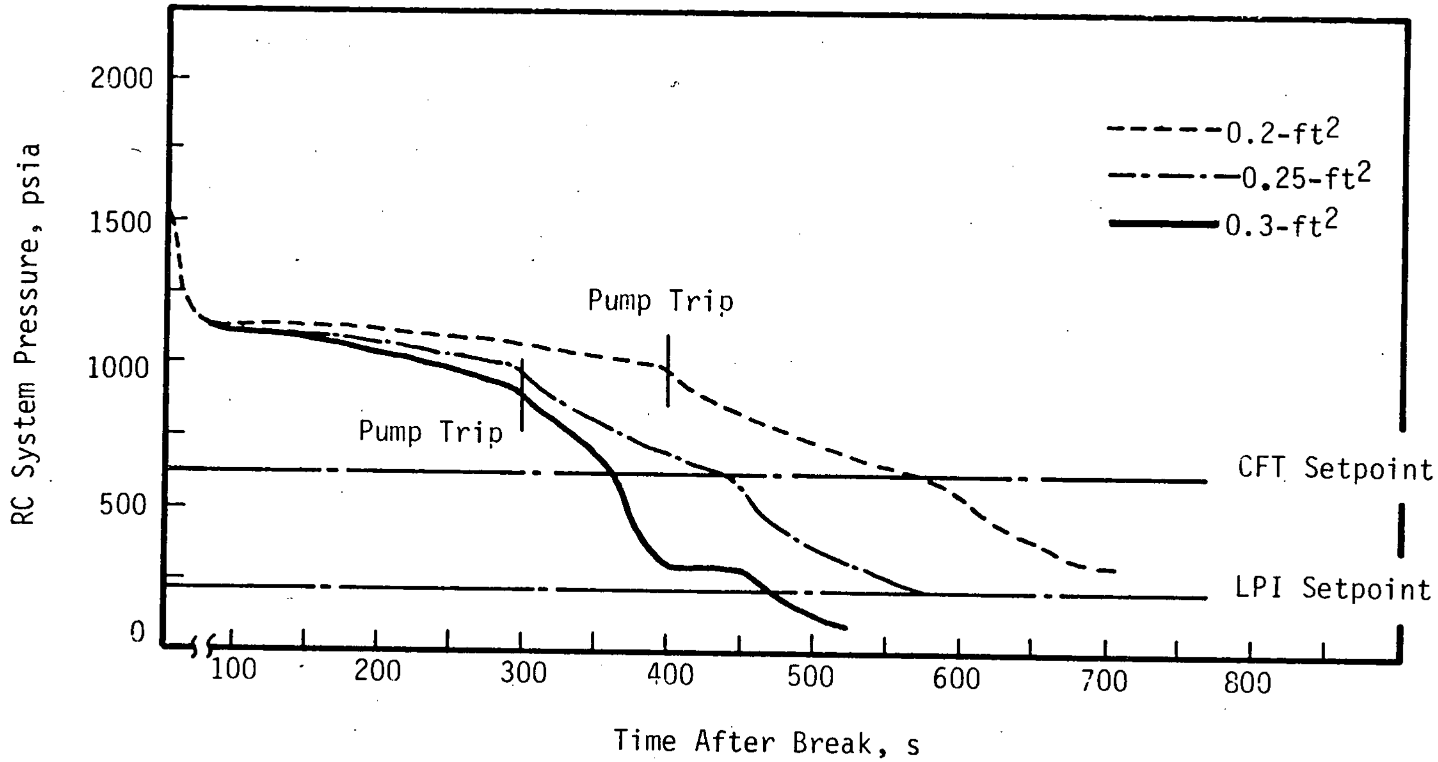


Figure 5-14. Liquid Volume in Reactor Vessel Following RC Pump Trip for a 0.2-ft^2 PD Break

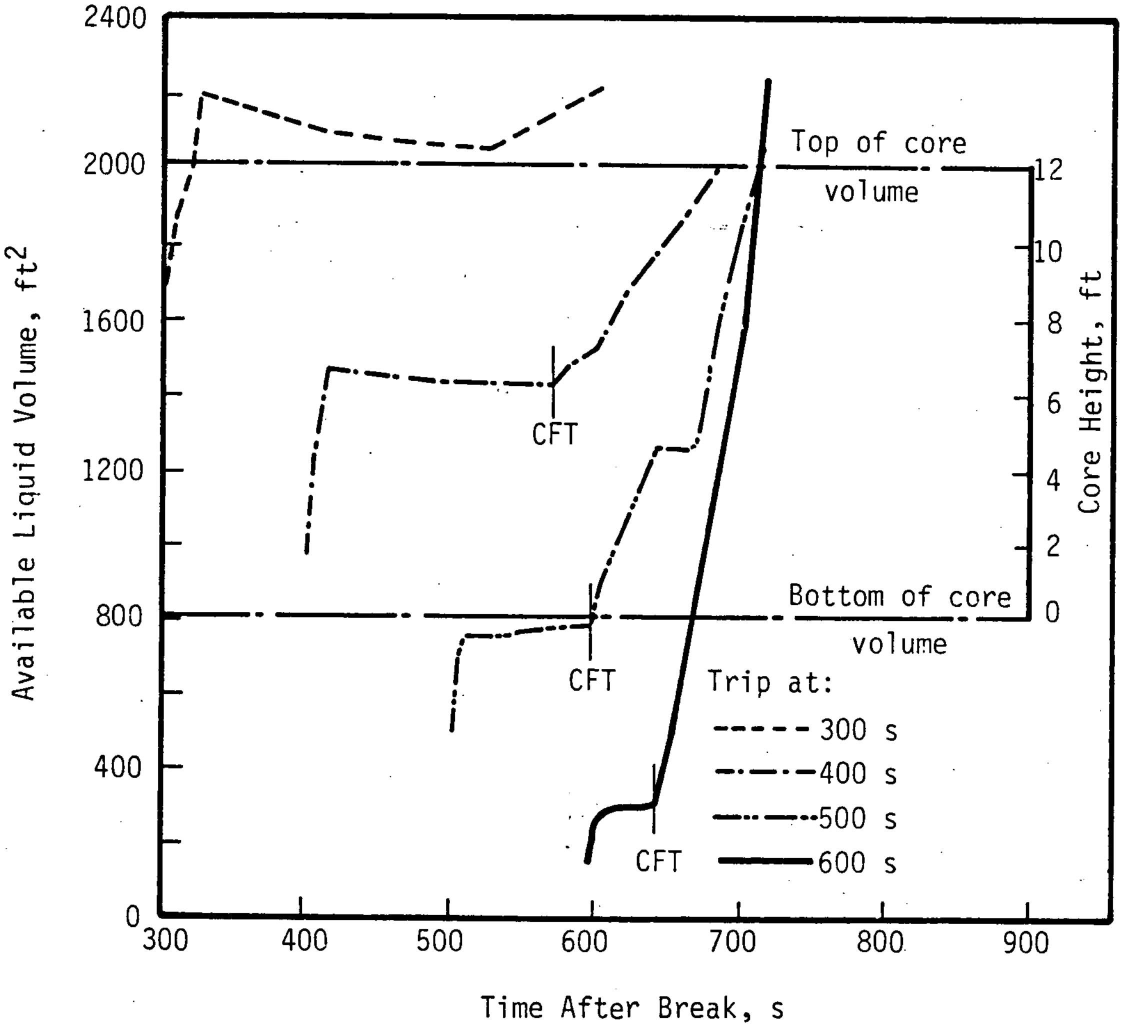


Figure 5-15. System Pressure Vs Time for a 0.2-ft^2 Break at the Pump Discharge

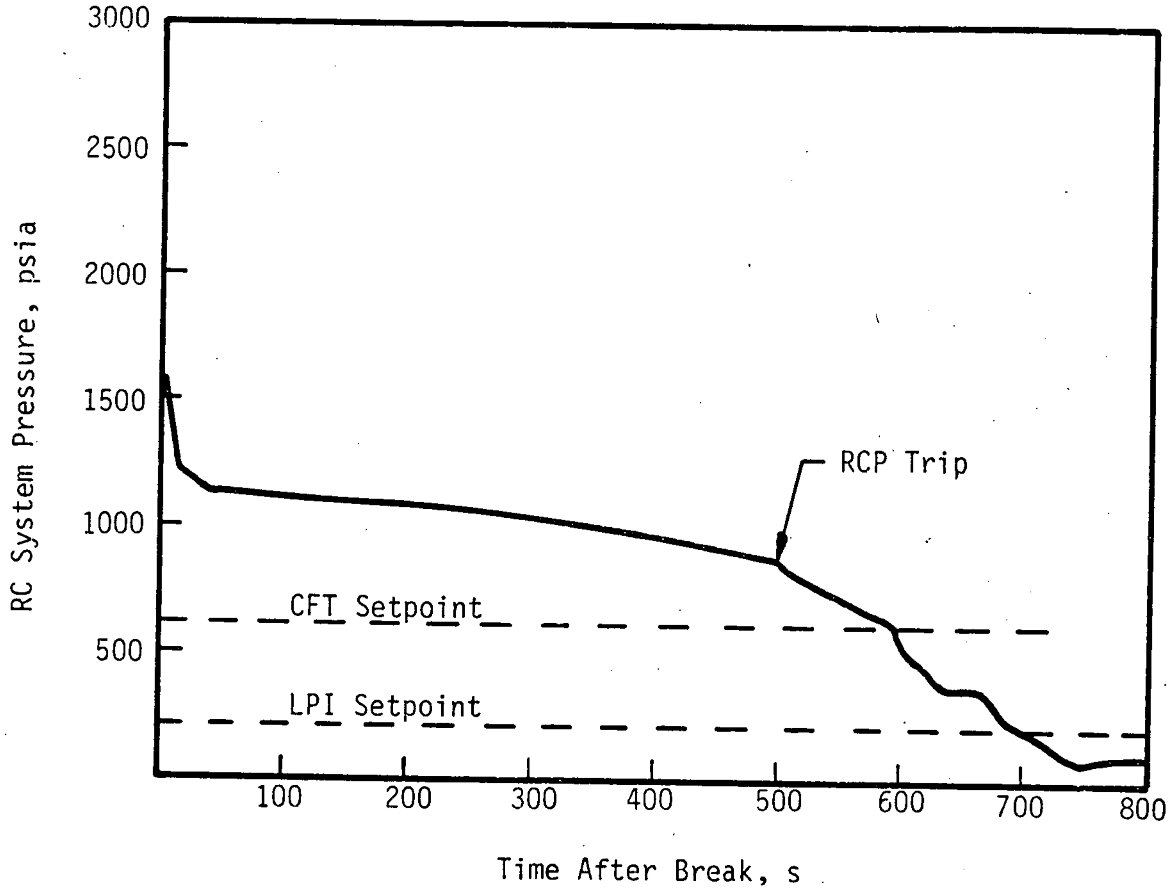


Figure 5-16. Maximum Hot Spot Cladding Volume-Average Temperature Vs Time for a 0.2-ft^2 Break at Pump Discharge

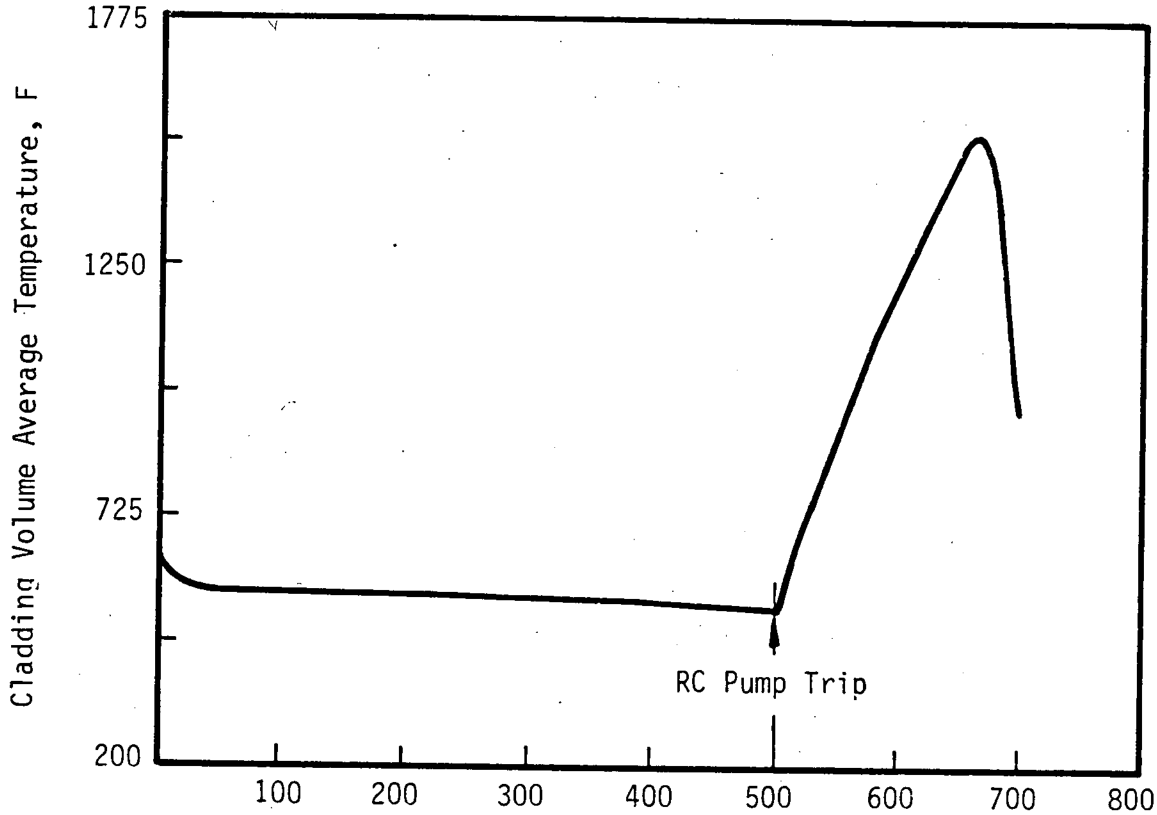
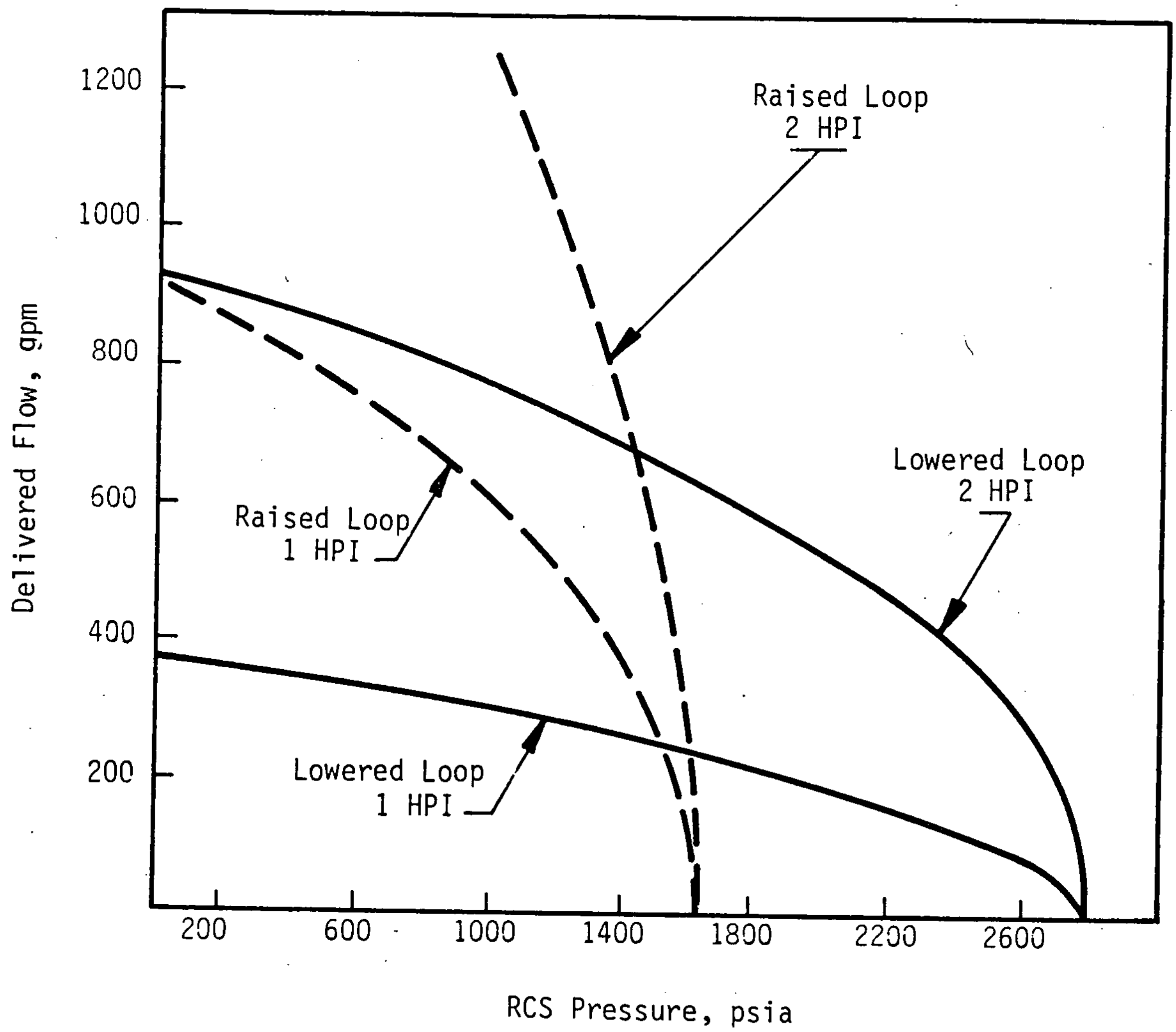


Figure 5-17. Comparison of High Pressure Injection System Capacities for Raised Loop and Lowered Loop Design Plants



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