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Florida Power
CORPORATION

Sseptember 14, 1979

File: 3-0-3-a-3

Mr. J. P. O'Reilly, Director
U.S. Nuclear Regulatory Commission
Office of Inspection and Enforcement, Suite 3100
101 Mariett Street
Atlanta, GA 30303

SUBJECT: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
I.E. Bulletin 79-05C

Dear Mr. O'Reilly:

Enclosed is our response to short term action Items 2 and 4 on the subject Bulletin. The response to Item 2 is the additional information we committed to provide you by our earlier response dated August 24, 1979.

If you have any questions concerning these responses, please contact this office.

Very truly yours,

FLORIDA POWER CORPORATION

W. P. Stewart

W. P. Stewart
Manager, Nuclear Operations

Enclosure

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D6

cc: Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
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Item 2 Perform and submit a report of LOCA analyses for your plants for a range of small break sizes and a range of time lapses between reactor trip and pump trip. For each pair of values of the parameters, determine the peak cladding temperature (PCT) which results. The range of values for each parameter must be wide enough to assure that the maximum PCT or, if appropriate, the region containing PCTs greater than 220 degrees F is identified.

Response:

On August 24, 1979, Florida Power Corporation submitted the B & W report entitled "Analysis Summary In Support of an Early RC Pump Trip" as our response to Item 2 above. At that time we indicated that additional work related to Item 2 was underway at B & W and that FPC would submit this information on September 14, 1979. In that regard, enclosed is a copy of the report "Supplemental Small Break Analysis" which is being submitted in response to Item 2.

Also enclosed is a revised copy of Section III, Impact Assessment of a RC Pump Trip on Non-LOCA Events, of the B & W report "Analysis Summary In Support of an Early RC Pump Trip". This section has been revised to correct errors discovered by B & W in its calculational techniques which affected the results provided in our August 24, 1979 submittal. The NRC staff was notified by B & W on September 7, 1979 about this problem and additional discussions of the problem were held with the NRC staff on September 11, 1979 and September 13, 1979. As a result of these discussions with the NRC staff it was agreed that the revised Section III would be submitted on September 14, 1979 and B & W would perform additional analyses to insure that the 12.2 sq. ft. steam line break case is the worst case event for the non-LOCA Analysis. The results of this additional work will be submitted by Florida Power Corporation upon its completion.

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SUPPLEMENTAL SMALL BREAK ANALYSIS

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1. Introduction

Babcock & Wilcox has evaluated the effect of a delayed reactor coolant (RC) pump trip during the course of a small loss-of-coolant accident. The results of this evaluation are contained in Section II of the report entitled "Analysis Summary in Support of an Early RC Pump Trip."[1] (Letter R.B. Davis to B&W 177 Owner's Group, "Responses to IE Bulletin 79-05C Action Items," dated August 21, 1979.) The above letter demonstrated the following:

a. A delayed RC pump trip at the time that the reactor coolant system is at high void fractions will result in unacceptable consequences when Appendix K evaluation techniques are used. Therefore, the RC pumps must be tripped before the RC system evolves to high void fractions.

b. A prompt reactor coolant pump trip upon receipt of the low pressure ESFAS signal provides acceptable LOCA consequences.

The following sections in this report are provided to supplement the information contained in reference 1. Specifically discussed in this report are:

a. The analyses to determine the time available for the operator to trip the reactor coolant pumps such that, under Appendix K assumptions, the criteria of 10 CFR 50.46 would not be violated.

b. The RC pump trip times for a spectrum of breaks for which the peak cladding temperature, evaluated with Appendix K assumptions, will exceed 10 CFR 50.46 limits.

c. A realistic analysis of a typical worst case to demonstrate that the consequences of a RC pump trip at any time will not exceed the 10 CFR 50.46 limits.

2. Time Available for RC Pump Trip Under Appendix K Assumptions

A spectrum of breaks was analyzed to determine the time available for RC pump trip under Appendix K assumptions. The breaks analyzed ranged from 0.025 to 0.3 ft^2. As was demonstrated in reference 1, the system evolves to high void fractions early in time for the larger sized breaks. Values in excess of 90% void fraction were predicted as early as 300 seconds for the 0.2 ft^2 break. For the smaller breaks it takes much longer (hours) before the system evolves to high void fraction. Therefore, the time available to trip the RC pump is minimum for the larger breaks. However, as will be shown later, for the larger small breaks (>0.3 ft^2), a very rapid depressurization is achieved upon the trip of RC pumps at high system void fraction. This results in early CFT and LPI actuation, and

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a subsequent rapid core refill. Thus, only a small core uncovery time will ensue. The following paragraphs will discuss the available time to trip the RC pumps for different break sizes. In performing this evaluation, only one HPI system was assumed available rather than the two HPI systems assumed in the reference 1 analyses.

a. 0.3 ft^2 Break - Figures 1 and 2 show the system void fraction and available liquid volume in the vessel versus time for RC pump trips at 95, 83, and 63% void fractions for a 0.3 ft^2 break at the RC pump discharge. For the pump trip at 95% void the system void fraction slowly decreases and then it drops faster following the CFT and LPI actuations. Following the RCP trip, the pressure drops rapidly and CFT is actuated at 250 seconds. The core begins to refill at this time and, with LPI actuation at 300 seconds, the core is flooded faster and is filled to a liquid level of 9 feet (equivalent to approximately 12 feet swelled mixture) at 370 seconds. The total core uncovery time is 170 seconds. Assuming an adiabatic heatup of 6.5°F/sec, as explained in reference 1, the consequences of a RC pump trip at 95% void will not exceed the 220F limit.

   As seen in Figure 2 for the RC pump trip at 63% or lower void fractions, the available liquid in the core will keep the core covered above the 11 feet elevation for about 350 seconds, and above 12 feet elevation at all other times. Therefore, tripping the RC pumps at void fractions ≤ 63% will not result in unacceptable consequences as the core will never uncover.

   A RC pump trip at 83% void fraction demonstrates an uncovery time of 350 seconds. However, previous detailed small break analysis (reference 2) have shown that a 10 ft of mixture height in the core will provide sufficient core cooling to assure that the criteria of 10 CFR 50.46 is satisfied. For this case, the 10 feet of mixture height is provided by approximately 1600 ft^3 liquid in the vessel. At this level in Figure 2, the core uncovery time is 220 seconds. Again, even with the assumption of adiabatic heatup over this period, the consequences are acceptable. It should be pointed out that if credit is taken for steam cooling of the upper portion of the fuel pin, the resulting PCT will be significantly lower then that obtained from the adiabatic heatup assumption.

   From Figure 2, it can be concluded that a RC pump trip at 120 seconds will result in little core uncovery. For RC pumps trip at system void fractions

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higher than 95% (at 200 seconds), the system will be at a lower pressure and with the CFT and LPI actuation there will be little or no core uncovery. Although core uncoveries are predicted for trips at 83% and 95% system void fractions, as shown earlier, the consequences are acceptable. Thus, a delayed RC pump trip at anytime for this break will provide acceptable consequences even if Appendix K evaluation techniques are used.

For breaks larger than 0.3 ft^2, a delayed RC pump trip at any time during the transient is also acceptable as the faster depressurization for these breaks will result in smaller delays between the pump trip and CFT and LPI actuation. Therefore, core uncovery times will be smaller than that shown for the 0.3 ft^2 break.

b. 0.2 ft^2 Break - Figures 3 through 5 show the system void fraction and available liquid volume in the vessel versus time for RC pump trips at 98, 73, 60 and 45% void fraction for a 0.2 ft^2 break at the RC pump discharge. As seen in Figure 5, the RC pump trip at 45 and 60% void fraction does not result in core uncovery. The available liquid volume is sufficient to keep the core covered above the 10 ft elevation at all times. For the trip at 98% void fraction in Figure 4, the core is refilled very rapidly with the actuation of CFT and LPI at approximately 420 and 450 seconds, respectively. The core is refilled to an elevation of 9 feet at 460 seconds. The core uncovery time is in the order of 60 seconds, and the consequences are not significant. The RC pump trip at 73% void fraction as seen in Figure 4, results in a 450 seconds core uncovery time. Although a 450 seconds uncovery time seems to result in unacceptable consequences, if credit is taken for steam cooling and using the same rationale as that given for the RC pump trip at 83% system void in section 1.a, it is believed that the consequences will not be significant. Should the RC pumps be tripped at system voids less than 70%, there will be little or no core uncovery. However, for void fractions between 73% and 98%, there is a potential for a core uncovery depth and time which might be unacceptable. Thus, a time region can be defined in which a RC pump trip, evaluated under Appendix K assumptions, could result in peak cladding temperatures exceeding the 10 CFR 50.46 criteria. This window is narrow and extends from 180 seconds (73% void) to 400 seconds (98% void) after ESFAS. A RC pump trip at any other time will not result in unacceptable consequences.

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c. 0.1 ft^2 Break - Figures 6 and 7 shows system void fractions and available liquid volume for trips at 90, 60, and 40% system void fractions for a 0.1 ft^2 break at the RC pump discharge. The same discussions as those presented in sections 2.a and 2.b can be applied here. However, due to slower depressurization of the system for this break, complete core cooling is not provided until the actuation of LPI's. As seen in Figure 7, the time to trip the RC pumps without any core uncovery is approximately 250 seconds. In Figure 6, with the RC pumps operating the LPI's are actuated at approximately 2350 seconds. Tripping the RC pumps at any time before 2350 seconds will actuate the LPIs earlier in time. Therefore, unacceptable consequences are predicted for a delayed RC pump trip in a time range of 250 seconds to 2350 seconds. For any other time, all the consequences are acceptable.

d. 0.075, 0.05 and 0.025 ft^2 Breaks - Figures 8 and 9 show a comparison of system void fractions for pumps running and pumps tripped[3] conditions. As seen in Figure 8, with the RC pumps tripped coincident with the reactor trip, in the short term, the evolved system void fraction is greater than that with the RC pumps operative. The two curves cross at about 300 seconds. Before this time, a RC pump trip will not result in unacceptable consequences since the system is at a lower void fraction than RC pumps trip case. Therefore, the time available for RC pumps trip with acceptable results is estimated at 300 seconds. As the system depressurizes and LPI's are actuated, the core will be flooded, and a RC pump trip after this time will have acceptable consequences. From the analyses performed, the LPI actuation time is estimated at approximately 3000 seconds. Therefore, the region between 300 and 3000 seconds defines the time region in which a RC pump trip could result in unacceptable consequences.

For a 0.05 ft^2 break, the same argument can be made using Figure 9. As seen from this figure, the time available to trip the RC pumps is approximately 450 seconds. The LPI actuation time for this break size is estimated at approximately 4350 seconds. Therefore, the unacceptable times for RC pump trip is defined between 450 and 4350 seconds.

As discussed in reference 1, the system evolves to high void fractions very slowly for 0.025 ft^2 or smaller breaks. The system depressurization is very slow and it takes on the order of hours before the LPI's are actuated. A RC pump trip at 2400 seconds for the 0.025 ft^2 break results in a system

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void fraction below 50% and the core remains completely covered. A study of the 0.025 ft^2 break with 2 HPI's available shows with the RC pumps operative the system void fraction never exceeds 61%. The CFT is actuated at approximately 4800 seconds and the system void starts to decrease and available liquid volume in the RV starts to increase. Thus, the core will remain completely covered for any RC pump trip time and, thus, will result in acceptable consequences. With one HPI available, a slower depressurization is expected but the system evolution to high void fraction will still be very slow. Thus, the conclusion that a RC pump trip at any time yields acceptable consequences for the 0.025 ft^2 break holds whether one or two HPI's are assumed available.

The LPI actuation time for the 0.025 ft^2 break can be extrapolated using the available data of the other breaks. Figure 10 shows the extrapolated LPI actuation time at approximately 8000 seconds. Thus, a conservative unacceptable time region for pump trip can be defined between 2500 and 8000 seconds for the 0.025 ft^2 break under Appendix K assumptions.

3. Critical Time Window for RC Pumps Trip

As discussed in section 2, there is a time region for each break size in which the consequences of the RC pump trip could exceed the 10 CFR 50.46 LOCA limit. These critical time windows were defined in section 2. Figure 11 shows a plot of the break size versus trip time RC pump which results in unacceptable consequences. The region indicated by dashed lines represent a boundary in which unacceptable consequences may occur if the RC pumps are tripped. However, this region is defined using Appendix K assumptions. It should be recognized that this region, even under Appendix K assumptions, is smaller than what is shown in Figure 11 as the 0.2 and 0.025 ft^2 breaks may not even have an unacceptable region. The time available to trip the RC pumps can be obtained from the lower bound of this region and is on the order of two to three minutes after ESFAS.

4. "Realistic" Evaluation of Impact of Delayed RC Pump Trip for a Small LOCA

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a. Introduction

As discussed in the previous sections, there exists a combination of break sizes and RC pump trip times which will result in peak cladding temperatures in excess of 2200F if the conservative requirements of Appendix K are utilized in the analysis. The analysis discussed in this section was performed utilizing "realistic" assumptions and demonstrates that a RC pump trip at any time will not result in peak cladding temperatures in excess of the 10 CFR 50.46 criteria.

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b. Method of Analysis

There are three overriding conservatisms in an Appendix K small break evaluation which maximizes cladding temperatures. These are:

(1) Decay heat must be based on 1.2 times the 1971 ANS decay heat curve for infinite operation.

(2) Only one HPI pump and one LPI pump are assumed operable (single failure criterion).

(3) The axial peaking distribution is skewed towards the core outlet. The local heating rate for this power shape is assumed to be at the LOCA limit value.

In performing a realistic evaluation of the effect of a delayed RC pump trip following a small LOCA, the conservative assumptions described above were modified. The evaluation described in this section utilized a decay heat based on 1.0 times the 1971 ANS standard and also assumed that both HPI and LPI systems were available. The axial peaking distribution was chosen to be representative of normal steady-state power operation.

Figures 12 and 13 show the axial peaking distributions utilized in this evaluation. These axial distributions were obtained from a review of available core follow data and the results of manuvering analyses which have been performed for the operating plants. A radial peaking factor of 1.651, which is the maximum calculated radial (without uncertainty) pin peak during normal operation, was utilized with these axial shapes. As such, the combination of radial and worst axial peaking are expected to provide the maximum expected kw/ft values for the top half of the core for at least 90% of the core life. Since the worst case effect of a delayed RC pump trip is to result in total core uncovery with a subsequent bottom reflooding, maximum pin peaking towards the upper half of the core will produce the highest peak cladding temperatures. Thus, this evaluation is expected to bound all axial peaks encountered during steady-state, power operation for at least 90% of core life.

The actual case evaluated in this section is a 0.05 ft^2 break in the pump discharge piping with the RC pump trip at the time the RC system average void fraction reaches 90%. As discussed in reference 1, RC pump trips at 90% system void fraction are expected to result in approximately the highest peak cladding temperatures. The CRAFT2 results for this case and the evaluation techniques utilized are discussed in section II.B.5 of reference 1. A realistic peak

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cladding temperature evaluation of this case, which is discussed below, is expected to yield roughly the highest peak cladding temperature for any break size and RC pump trip time. As shown in reference 1, maximum core uncovery times of approximately 600 seconds occur over the break size range of 0.05 ft^2 through 0.1 ft^2 using 1.2 times the ANS curve. Break sizes smaller than 0.05 ft^2 and larger than 0.1 ft^2 will yield smaller core uncovery times as demonstrated in reference 1 and the preceeding sections. Use of 1.0 times the ANS decay heat curve would result in a similar reduction in core uncovery time, approximately 200 seconds, for breaks in the 0.05 through 0.1 ft^2 range. Thus, the core refill rate, uncovery time, and peak cladding temperatures for the 0.05 ft^2 case is typical of the worst case values for the break spectrum.

c. Results of Analysis

Figure 4 shows the liquid volume in the reactor vessel for the 0.05 ft^2 break with a RC pump trip at the time the system average void fraction reaches 90%. The core initially uncovers and recovers approximately 375 seconds later. Using the previously discussed realistic assumptions the peak cladding temperature for this case is below 1900F. Therefore, the criteria of 10 CFR 50.46 is met.

The temperature response given above was developed in a conservative manner by comparing adiabatic heat up rates to maximum possible steady-state cladding temperatures. First, a temperature plot versus time is made up for each location on the hottest fuel assembly assuming that the assembly heats up adiabatically. Second, a series of FOAM[4] runs are made to produce the maximum steady-state pin temperatures at each location as a function of core liquid volume. FOAM calculates the mixture level in the core and the steaming rate from the portion of the core which is covered. Both the mixture height and steaming rate calculations are based on average core power. Fluid temperatures in the uncovered portion of the fuel rod are obtained by using the calculated average core steaming rate and by assuming all energy generated in the uncovered portion of the hot rod is transferred to the fluid. The surface heat transfer coefficient is calculated, based on the Dittus-Boelter correlation[5], from the fluid temperature and steaming rate and the steady-state clad temperature is obtained. The FOAM data are then combined with the core liquid inventory history (derived from Figure 14) to produce a maximum possible cladding temperature as a function of time. This graph might be termed maximum steady-state cladding temperature as a function of time and decreases in value with time because the core liquid

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inventory is increasing. By cross plotting the adiabatic heat up curve with the maximum steady-state curve a conservative peak cladding temperature prediction is obtained.

5. Conclusions

From this analysis, and the results in reference 1, the following conclusions have been drawn:

a. Using Appendix K evaluation techniques, there exists a combination of break size and RC pump trip times which result in a violation of 10 CFR 50.46 limits.

b. Prompt tripping of the RC pumps upon receipt of a low pressure ESFAS signal will result in cladding temperatures which meet the criteria of 10 CFR 50.4y. The minimum time available for the operator to perform this function is 2 to 3 minutes.

c. Under realistic assumptions, a delayed RC pump trip following a small break will result in cladding temperatures in compliance with 10 CFR 50.46.

[redacted]

REFERENCES

POOR ORIGINAL

1 "Analysis Summary in Support of an Early RC Pump Trip," Section II of letter R.B. Davis to B&W 177 Owner's Group, Responses to IE Bulletin 79-05C Action Items, dated August 21, 1979.

2 Letter J.H. Taylor (B&W) to Robert L. Baer, dated April 25, 1978.

3 Letter J.H. Taylor to S.A. Varga, dated July 18, 1978.

4 B.M. Dunn, C.D. Morgan, and L.R. Cartin, Multinode Analysis of Core Flooding Line Break for B&W's 2568 MWt Internals Vent Valve Plants, BAW-10064, Babcock & Wilcox, April 1978.

5 R.H. Stoudt and K.C. Heck, THETA1-B - Computer Code for Nuclear Reactor Core Thermal Analysis - B&W Revisions to IN-1445, (Idaho Nuclear, C.J. Hocevar and T.W. Wineinger), BAW-10094, Rev. 1, Babcock & Wilcox, April 1975.

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Figure 1 : 0.30 FT^2 BREAK @ P.D., SYSTEM

VOID FRACTION VS TIME

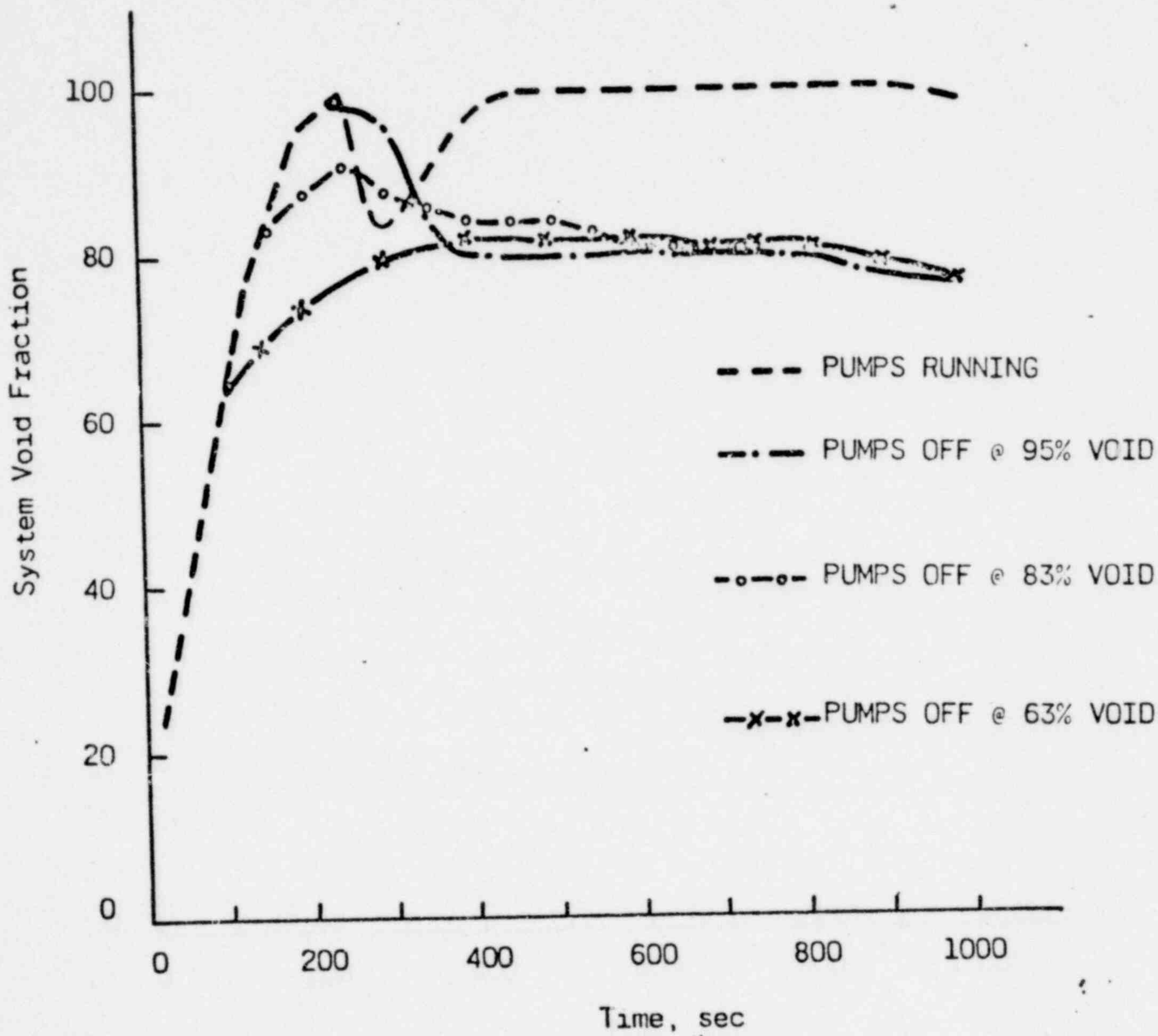

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Figure 2: 0.30 FT^2 BREAK @ P.D., AVAILABLE LIQUID VOLUME IN RV VS TIME, 1 HPI AVAILABLE

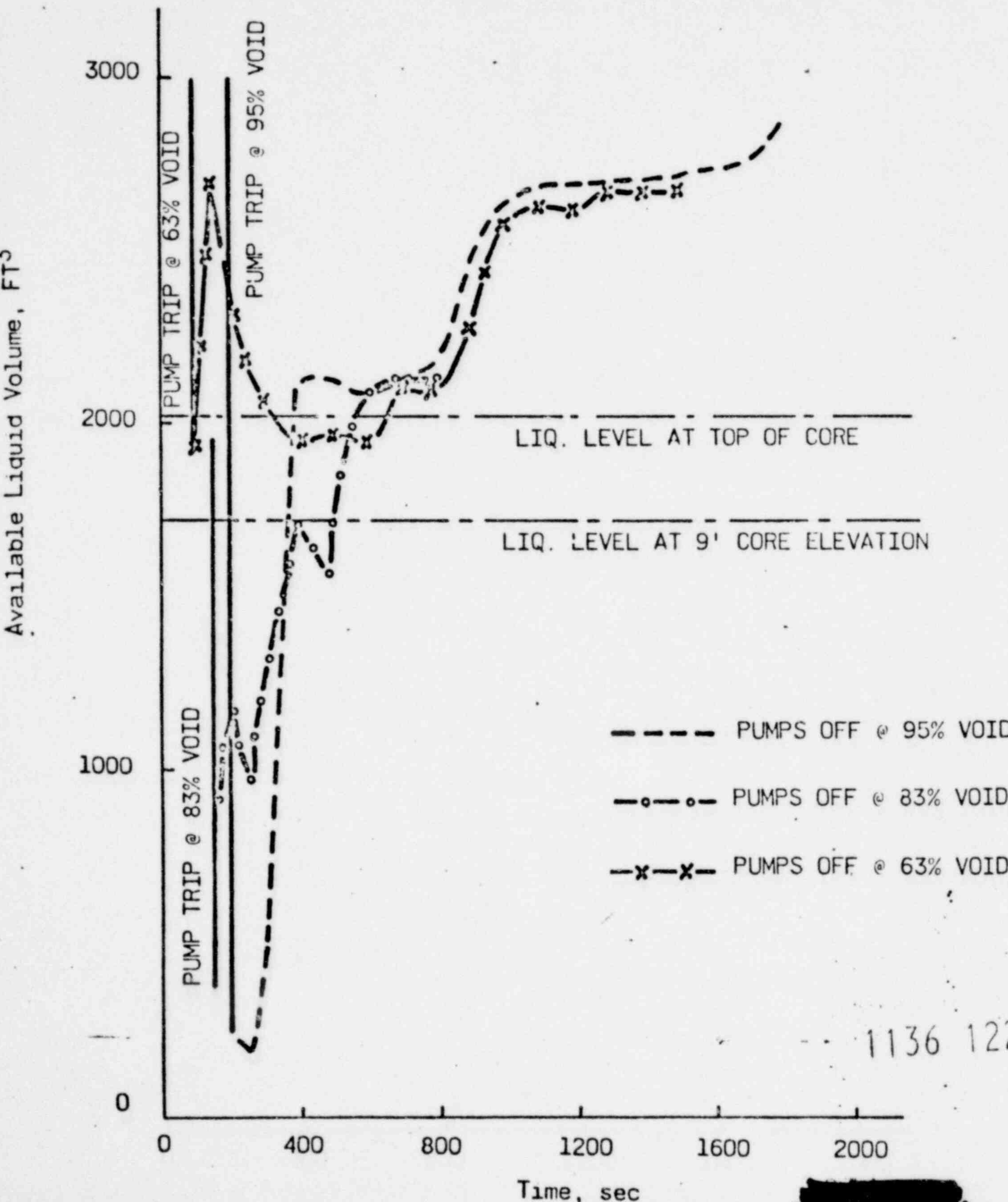

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Figure 3: 0.20 FT^2 BREAK @ P.D., SYSTEM VOID FRACTION VS TIME

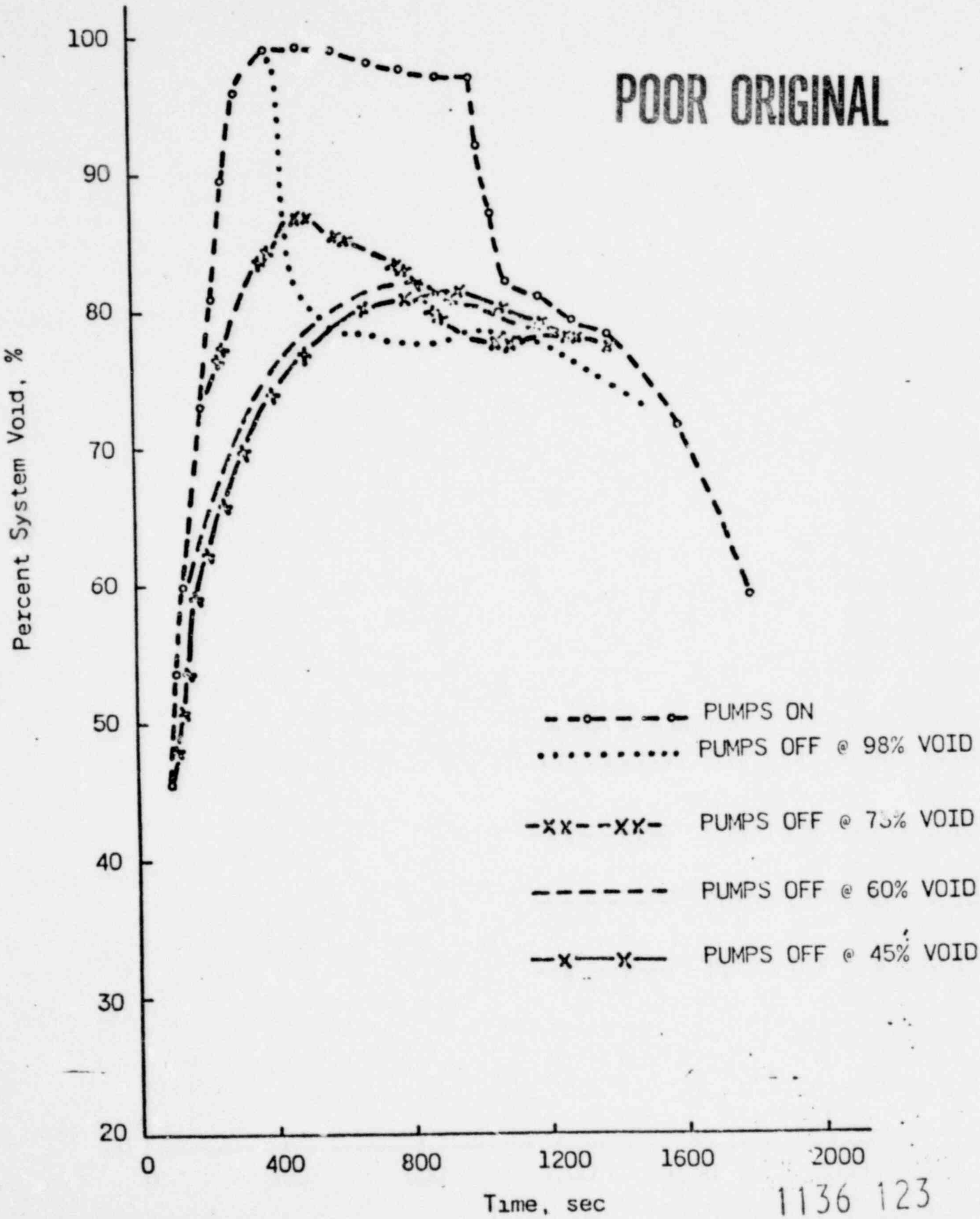

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Figure 4 : 0.20 FT^2 BREAK @ P.D., AVAILABLE LIQ. VOL. IN RV VS TIME 1 HPI AVAILABLE

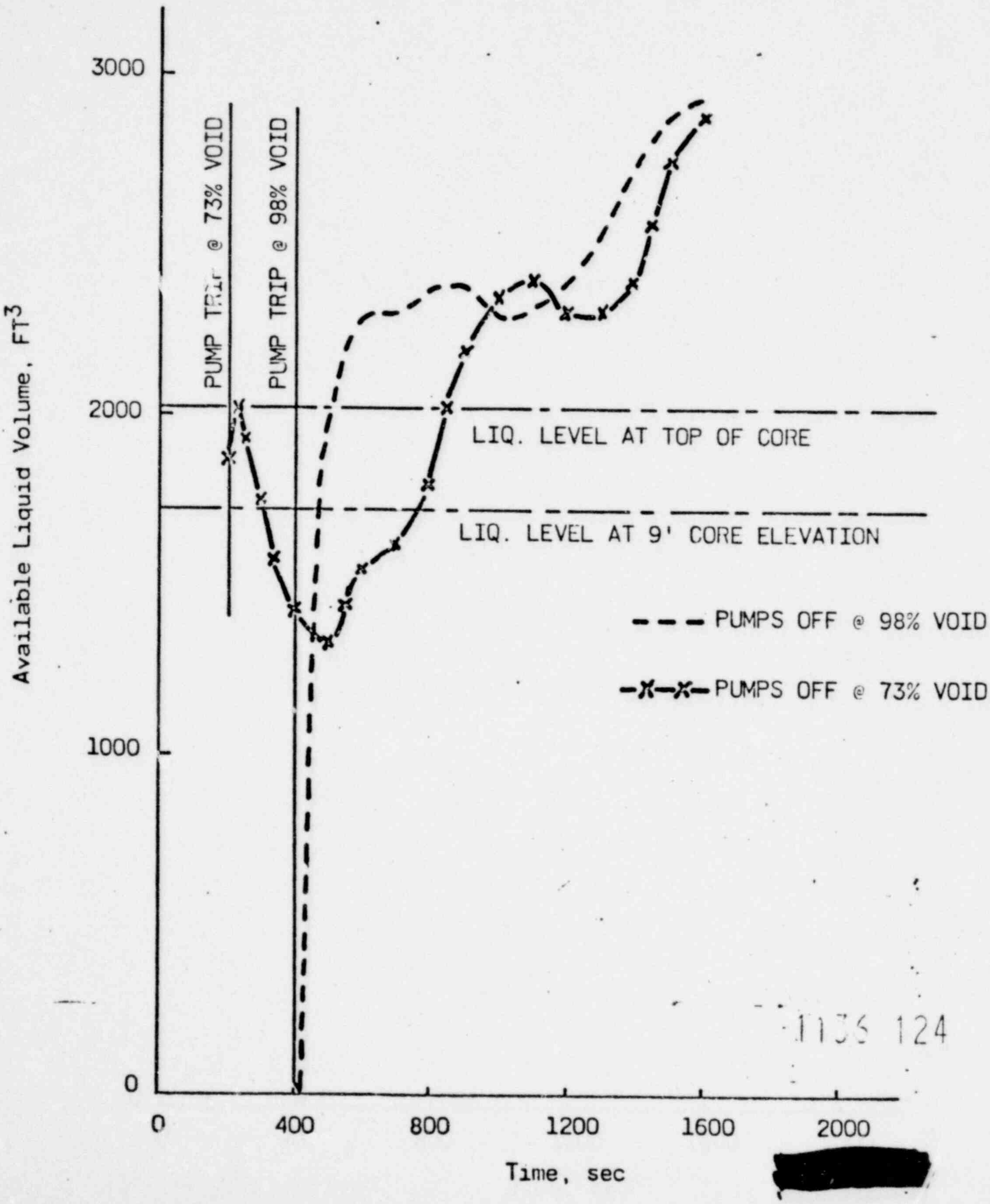

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Figure 5 : 0.20 FT^2 BREAK P.D. AVAILABLE LIQ. VOL. IN RV VS TIME 1 HPI AVAILABLE

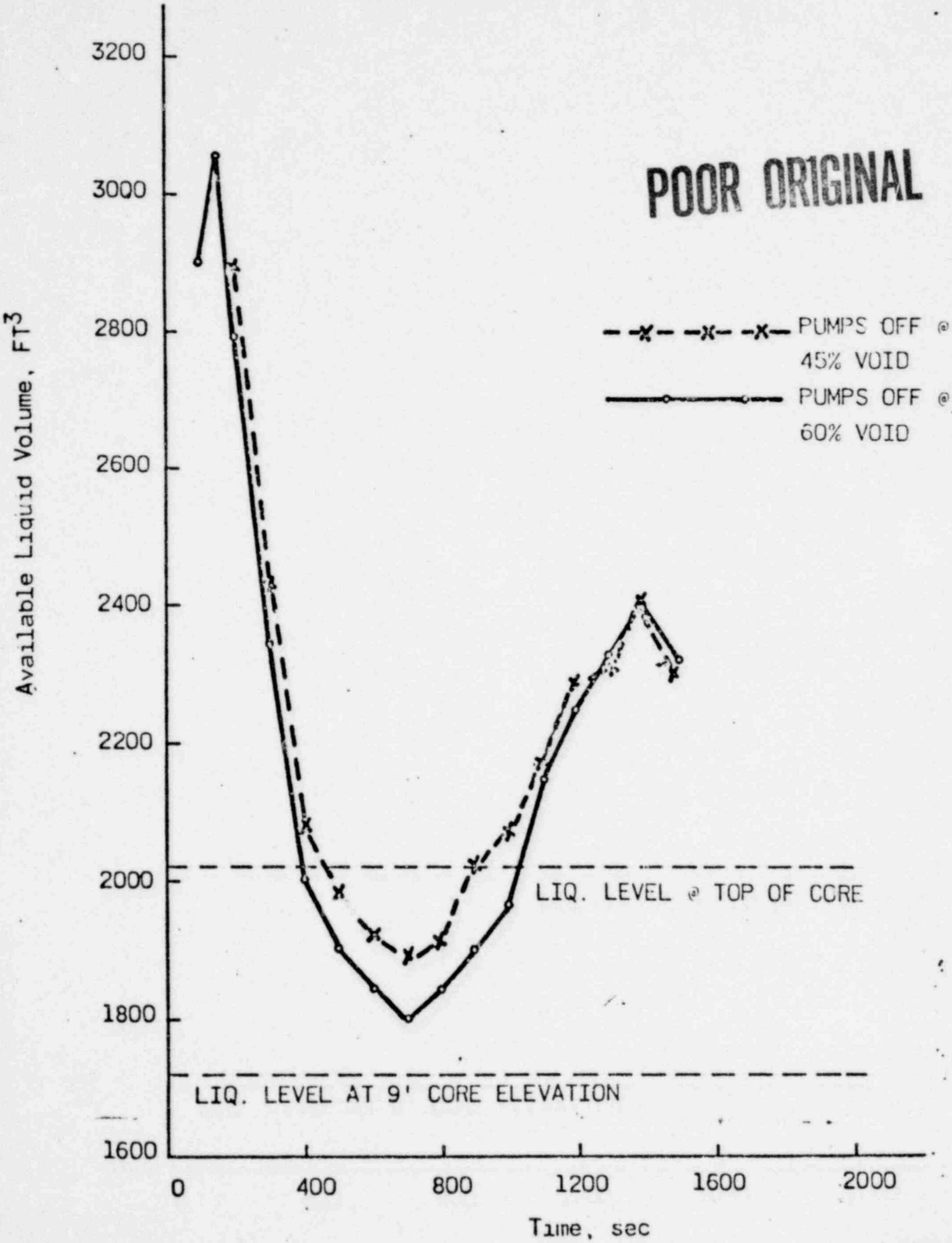

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Figure 6 : 0.1 FT^2 BREAK @ P.D., AVAILABLE LIQUID VOLUME : IN RV VS TIME, 1 HPI AVAILABLE

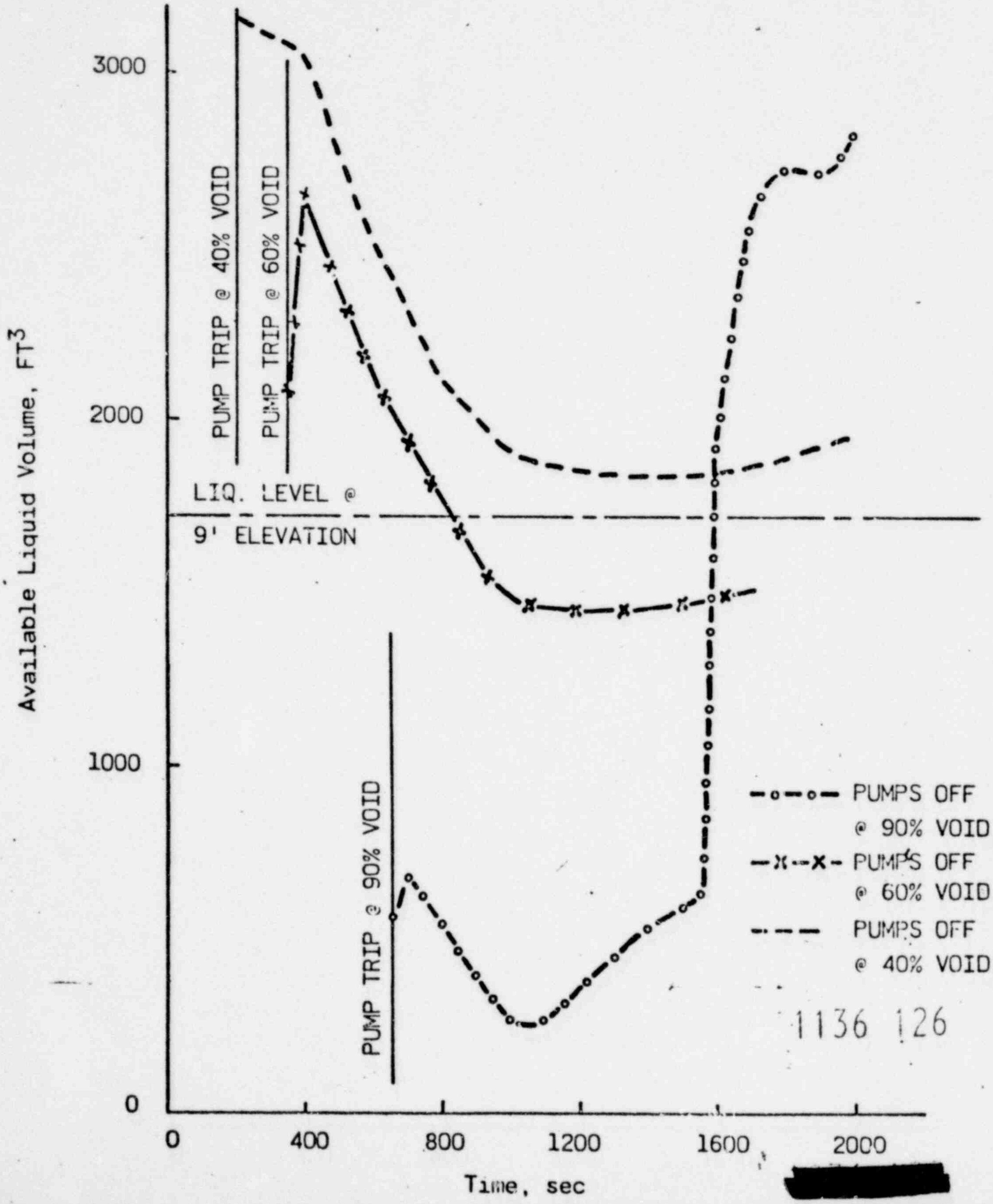

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Figure 7: 0.1 FT^2 BREAK @ P.D., SYSTEM VOID FRACTION VS TIME

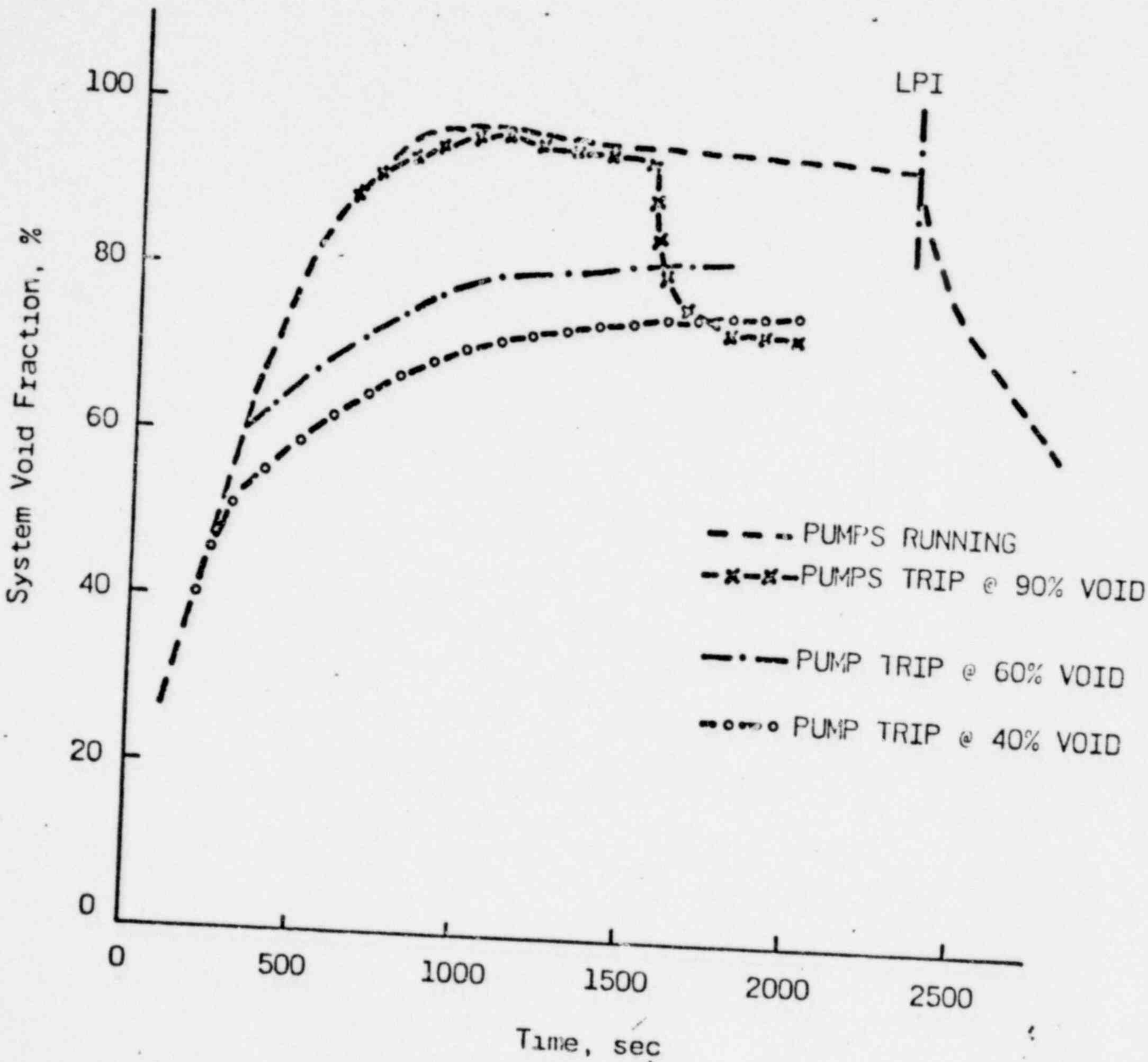

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Figure 8: SYSTEM VOID FRACTION VERSUS TIME

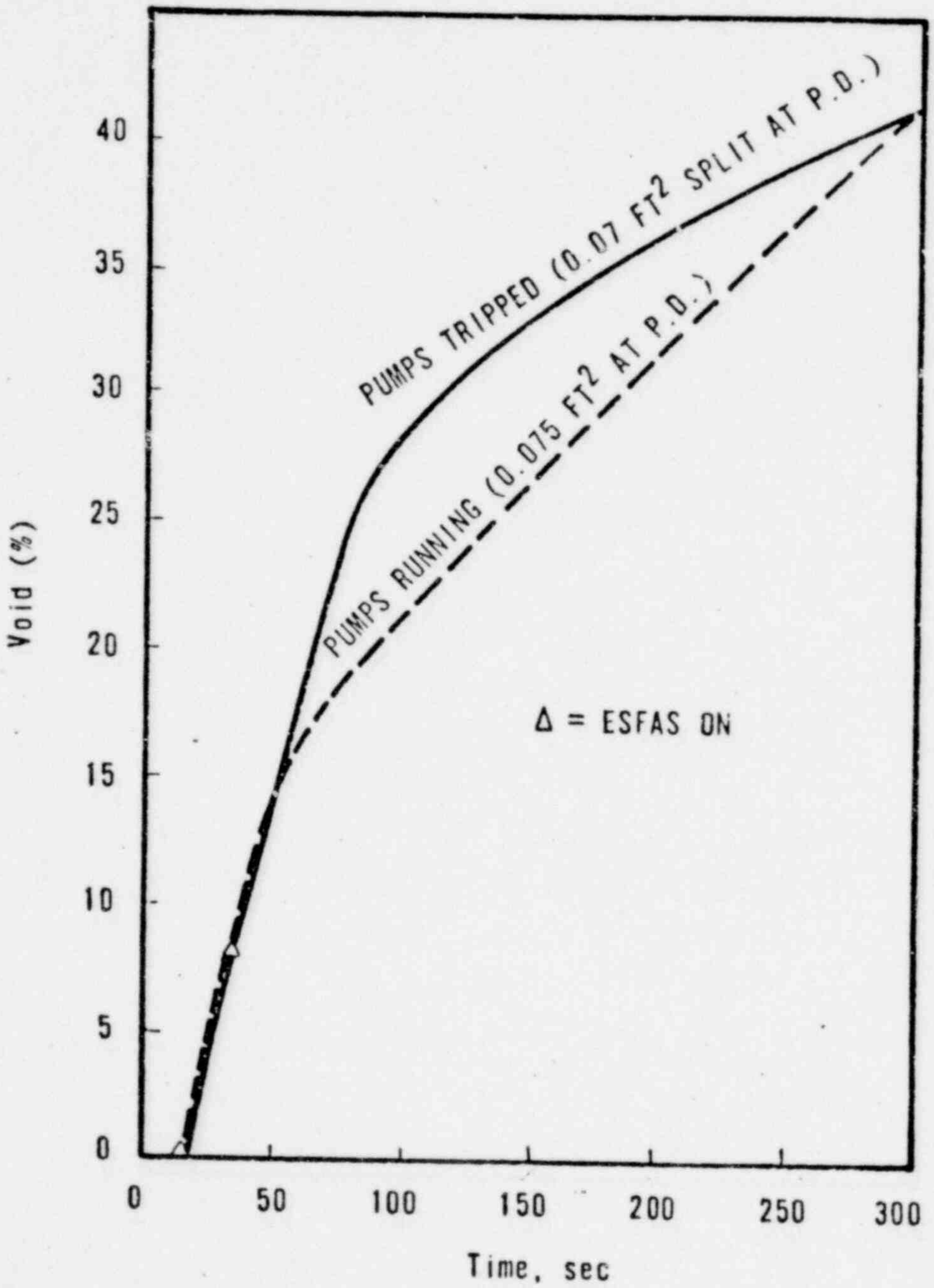

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Figure 9 : SYSTEM VOID FRACTION VERSUS TIME
PUMPS RUNNING AND PUMPS TRIPPED MODEL

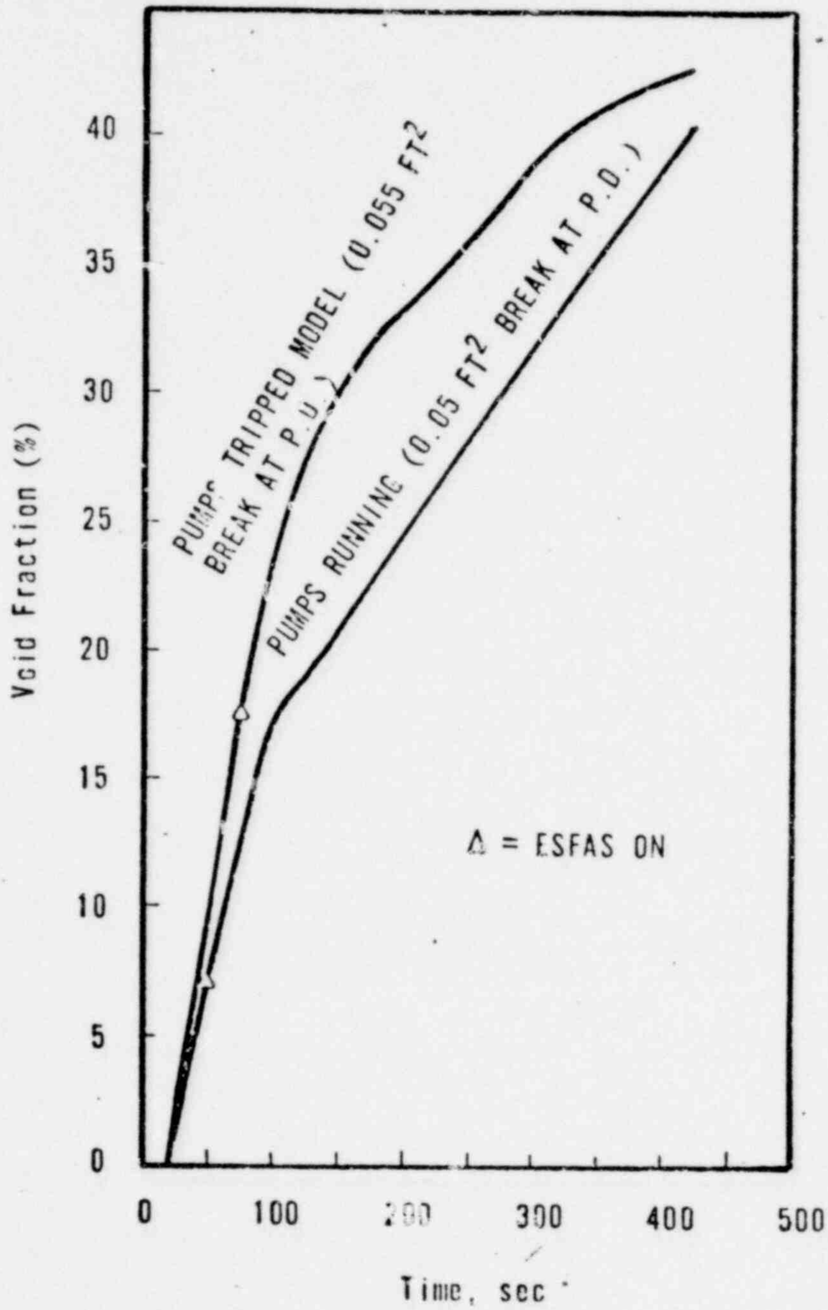

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Figure 10 : BREAK SIZE VS LPI ACTUATION TIME

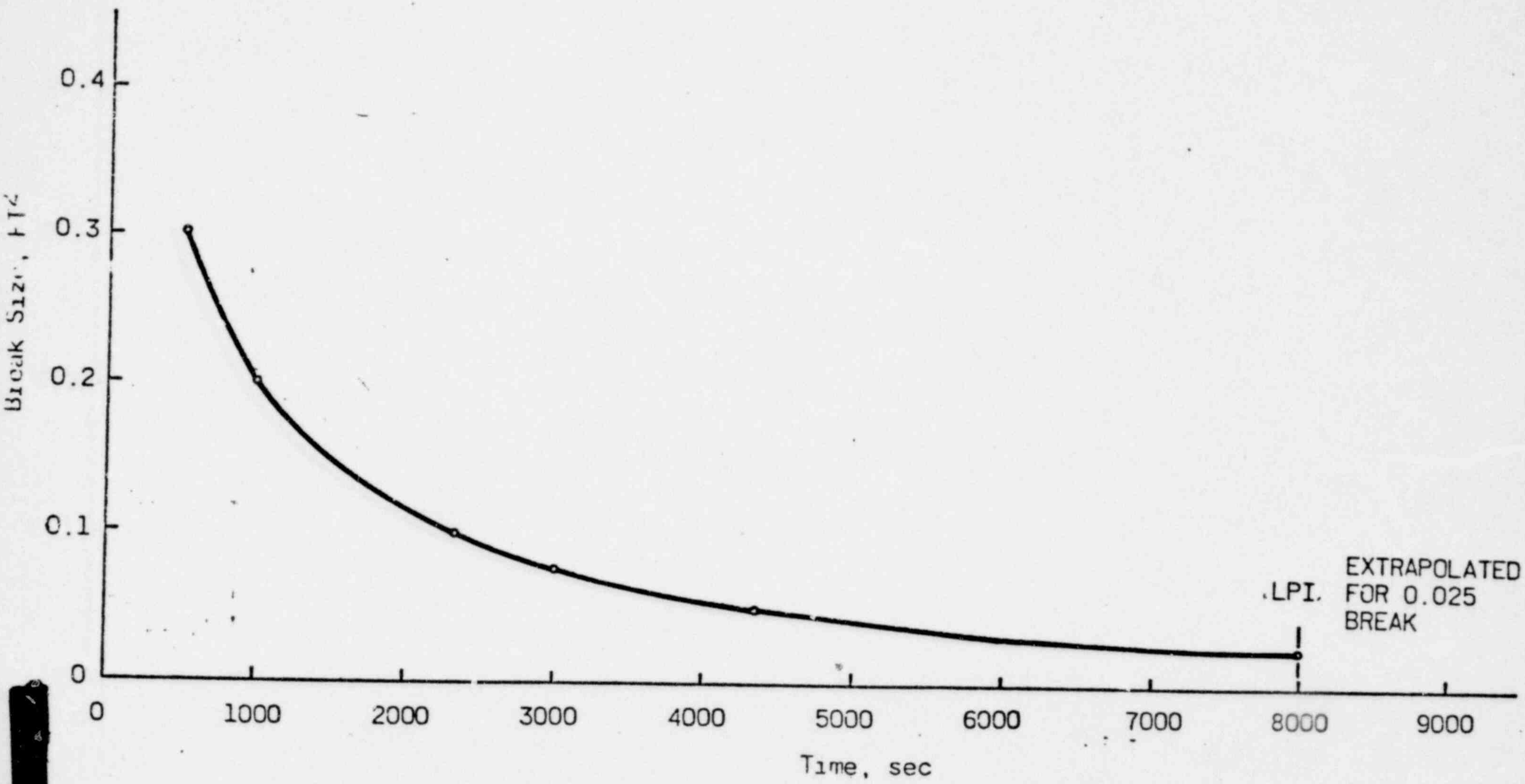
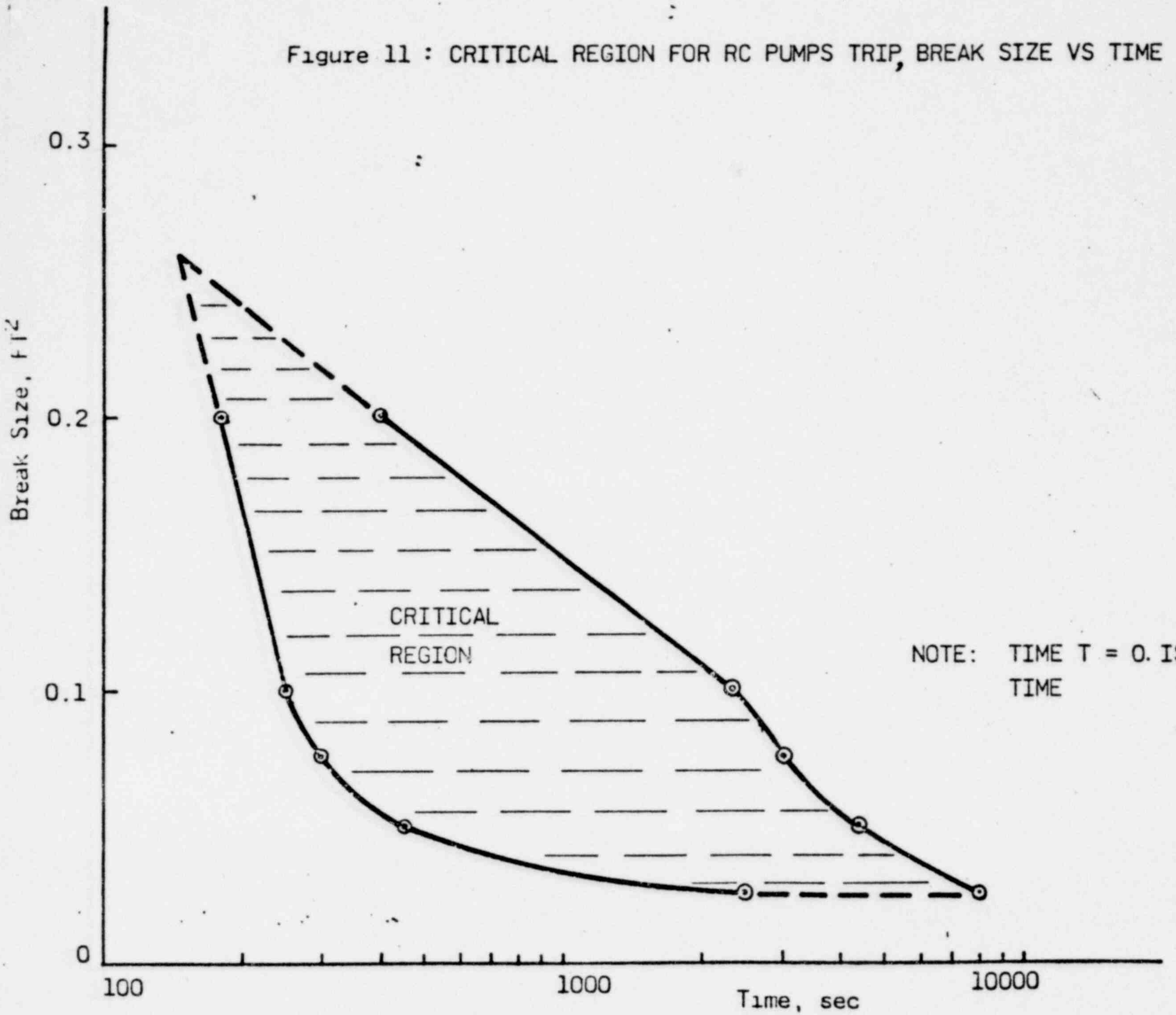


Figure 11 : CRITICAL REGION FOR RC PUMPS TRIP, BREAK SIZE VS TIME



NOTE: TIME T = 0. IS REACTOR TRIP TIME

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Figure 12: "REALISTIC" CORE AXIAL PEAKING DISTRIBUTION-CASE 1

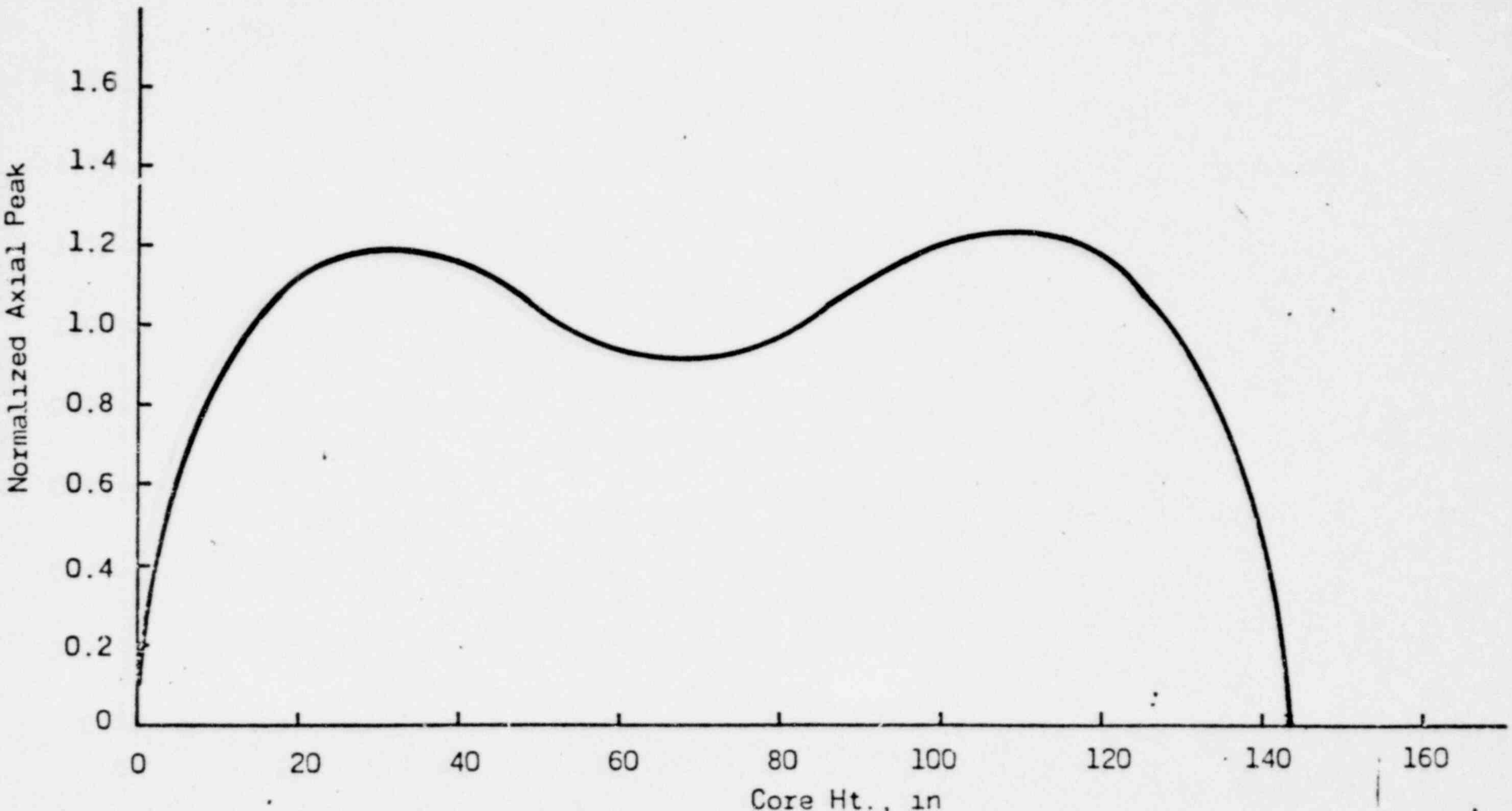

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Figure 13: "REALISTIC" CORE AXIAL PEAKING DISTRIBUTION-CASE 2

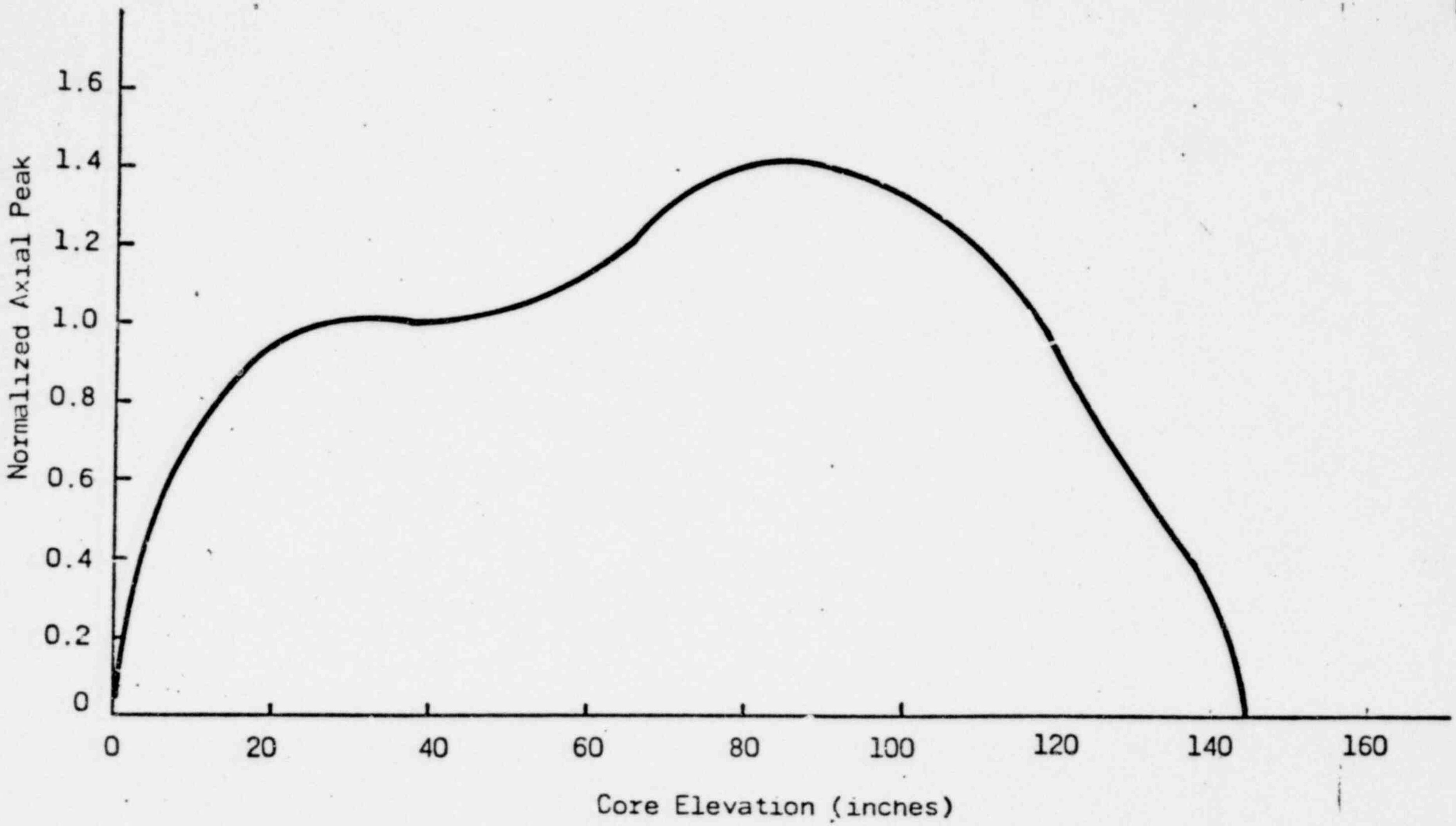
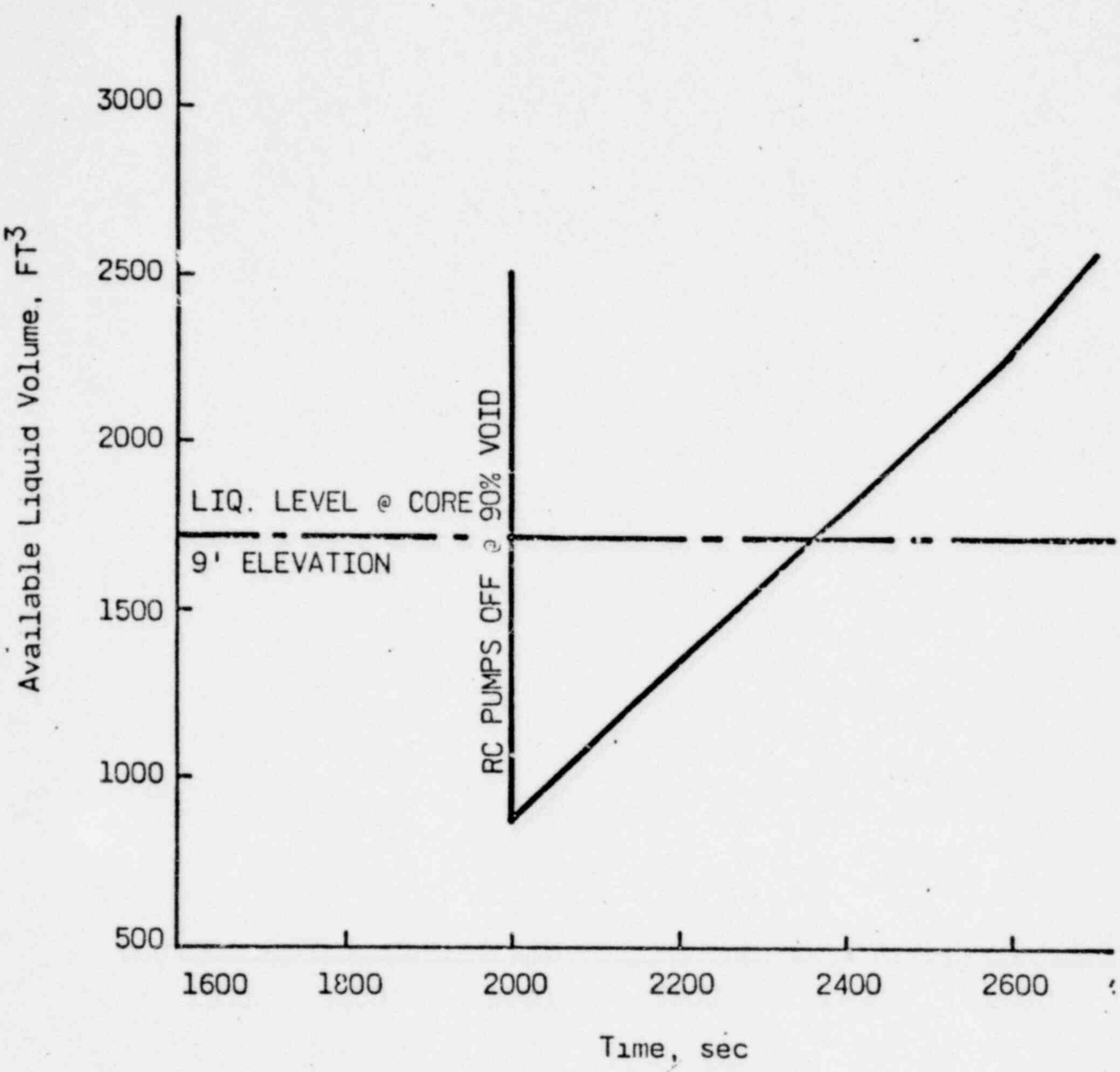


Figure 14 : AVAILABLE LIQUID VOLUME VS TIME FOR 0.05 FT^2 BREAK WITH 1.0 ANS DECAY CURVE



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III. IMPACT ASSESSMENT OF A RC PUMP TRIP ON NON-LOCA EVENTS

A. Introduction

Some Chapter 15 events are characterized by a primary system response similar to the one following a LOCA. The Section 15.1 events that result in an increase in heat removal by the secondary system cause ~~a~~ primary system cooldown and depressurization, much like a small break LOCA. Therefore, an assessment of the consequences of an imposed RC pump trip, upon initiation of the low RC pressure ESFAS, was made for these events.

B. General Assessment of Pump Trip in Non-LOCA Events

Several concerns have been raised with regard to the effect that an early pump trip would have on non-LOCA events that exhibit LOCA characteristics. Plant recovery would be more difficult, dependence on natural circulation mode while achieving cold shutdown would be highlighted, manual fill of the steam generators would be required, and so on. However, all of these drawbacks can be accommodated since none of them will on its own lead to unacceptable consequences. Also, restart of the pumps is recommended for plant control and cooldown once controlled operator action is assumed. Out of this search, three major concerns have surfaced which have appeared to be substantial enough as to require analysis:

1. A pump trip could reduce the time to system fill/repressurization or safety valve opening following an overcooling transient. If the time available to the operator for controlling HPI flow and the margin of subcooling were substantially reduced by the pump trip to where timely and effective operator action could be questionable, the pump trip would become less desirable.
2. In the event of a large steam line break (maximum overcooling), the blowdown may induce a steam bubble in the RCS which could impair natural circulation, with severe consequences on the core, especially if any degree of return to power is experienced.
3. A more general concern exists with a large steam line break at EOL conditions and whether or not a return to power is experienced following the RC pump trip. If a return to critical is experienced, natural circulation flow may not be sufficient to remove heat and to avoid core damage.

Overheating events were not considered in the impact of the RC pump trip since they do not initiate the low RC pressure ESFAS, and therefore, there would be no coincident pump trip. In addition, these events typically do not result in an empty pressurizer or the formation of a steam bubble in the primary system. Reactivity transients were also not considered for the same reasons. In addition, for overpressurization, previous analyses have shown that for the worst case conditions, an RC pump trip will mitigate the pressure rise. This results from the greater than 100 psi reduction in pressure at the RC pump exit which occurs after trip.

C. Analysis of Concerns and Results

1. System Repressurization

In order to resolve this concern, an analysis was performed for a 177 FA plant using a MINITRAP model based on the case set up for TMI-2. Figure 3.1 shows the noding/flow path scheme used and Table 3.1 provides s description of the nodes and flow paths. This case assumed that, as the result of a small steam line break (0.6 ft.^{2} split) or of some combination of secondary side valve failure, secondary side heat demand was increased from 100% to 138% at time zero. This increase in secondary side heat demand is the smallest which results in a (high flux) reactor trip and is very similar to the worst moderate frequency overcooling event, a failure of the steam pressure regulator. In the analysis, it was assumed that following HPI actuation on low RC pressure ESFAS, main feedwater is ramped down, MSIV's shut, and the auxiliary feedwater initiated with a 40-second delay. This action was taken to stop the cooldown and the depressurization of the system as soon as possible after HPI actuation, in order to minimize the time of refill and repressurization of the system. Both HPI pumps were assumed to function.

The calculation was performed twice, once assuming two of the four RC pumps running (one loop), and once assuming RC pump trip right after HPI initiation. The analysis shows that the system behaves very similarly with and without pumps. In both cases, the pressurizer refills in about 14 to 16 minutes from initiation of the transients, with the natural circula-

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tion case refilling about one minute before the case with two of four pumps running (See Figures 3.2, 3.3). In both cases, the system is highly subcooled, from a minimum of 30°F to 120°F and increasing at the end of 14 minutes (refer to Figure 3.4). It is concluded that an RC pump trip following HPI actuation will not increase the probability of causing a LOCA through the pressurizer code safeties, and that the operator will have the same lead time, as well as a large margin of subcooling, to control HPI prior to safety valve opening. Although no case with all RC pumps was made, it can be inferred from the one loop case (with pumps running) that the subcooled margin will be slightly larger for the all pumps running case. The pressurizer will take longer to fill but should do so by 16 minutes into the transient. Figure 3.4 shows the coolant temperatures (hot leg, cold leg, and core) as a function of time for the no RC pumps case.

2. Effect of Steam Bubble on Natural Circulation Cooling

For this concern, an analysis was performed for the same generic 177 FA plant as outlined in Part 1, but assuming that as a result of an unmitigated large SLB (12.2 $ft.^2$ DER), the excessive cooldown would produce void formation in the primary system. The intent of the analysis was to also show the extent of the void formation and where it occurred. As in the case analyzed in Part 1, the break was symmetric to both generators such that both would blow down equally, maximizing the cooldown (in this case there was a 6.1 $ft.^2$ break on each loop). There was no MSIV closure during the transient on either steam generator to maximize cooldown. Also, the turbine bypass system was assumed to operate, upon rupture, until isolation on ESFAS. ESFAS was initiated on low RC pressure and also actuated HPI (both pumps), tripped RC pumps (when applicable) and isolated the MFWIV's. The AFW was initiated to both generators on the low SG pressure signal, with minimum delay time (both pumps operating).

This analysis was performed twice, once assuming all RC pumps running, once with all pumps being tripped on the HPI actuation (after ESFAS), with a short (~5 second) delay. In both cases, voids were formed in the hot legs, but the dura-

tion and size were smaller for the case with no RC pump trip (refer to Figure 3.7). Although the RC pump operating case had a higher cooldown rate, there was less void formation, resulting from the additional system mixing. The coolant temperatures in the pressurizer loop hot and cold legs, and the core, are shown for both cases in Figures 3.5, 3.6. The core outlet pressure and SG and pressurizer levels versus time are given for both cases in Figures 3.8, 3.9. This analysis shows that the system behaves similarly with and without pumps, although maintaining RC pump flow does seem to help mitigate void formation. The pump flow case shows a shorter time to the start of pressurizer refill than the natural circulation case (Figure 3.9), although the time difference does not seem to be very large.

Since the volume of the hot leg loop above the lowest point in the candy cane portion is about 63 cubic feet, these steam formations have the potential for blocking natural circulation in the hot leg loops. As a result of these findings and since TRAP had not been programmed to closely follow this specific condition, an additional TRAP case was run. It is based on the unmitigated 12.2 ft^2 steam line break with RC pump trip, since this case represented the bounding event for steam formation. This case included a more detailed noding scheme and conservative bubble rise velocities (5.0 ft/sec) to the upper regions of the hot legs such that the effect of steam formation on natural circulation in the loops could be observed.

The noding and flow path scheme used in this model is shown in Figure 3.10. Table 3.2 provides a description of these nodes and flow paths. Figure 3.11 details the hot leg - candy cane - upper steam generator shroud noding and flow path model superimposed over a scaled figure of those regions. The flow path positions and sizes were carefully chosen to allow for countercurrent steam and liquid flow at the top of the candy cane. This model is consistent with that used for the small break LOCA analyses described in Section 6.2.4.2 of Ref. 5.

The results of this analysis showed steam formation only in the pressurizer loop (refer to Figure 3.12). These steam volumes are conservative since they include all of the steam that was calculated as being entrained as bubbles in the liquid. The additional steam volumes calculated for this loop, compared with those shown in Figure 3.7, are due to the additional boiling and steam separation

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that occurs in the candy cane as the liquid flow rates are reduced by steam formation and aided by metal heating. The lack of steam formation in the non-pressurizer loop 'B' is attributed to a correction in the metal heat transfer and metal heat capacities calculated for the hot legs. The previous analysis erroneously included half of the steam generator tubes, based on the calculations from the ECCS CRAFT model. Since the TRAP code already accounts for the tube metal in its steam generator model, this represented an unnecessary conservatism and it was deleted from the model for this case.

This case showed that the natural circulation flow was temporarily reduced. This flow reduced in the pressurizer loop to 45 to 100 lb/sec from 250 to 360 seconds (refer to Figure 3.13), with flow steadily increasing after this time period. The flow in the non-pressurizer loop remained relatively unchanged at about 1000 lb/sec (refer to Figure 3.14). Core flow was maintained from 1000 to 2000 lb/sec and no void formation occurred (refer to Figures 3.15 and 3.16). The steam bubble was collapsed, natural circulation fully restored, and a greater than 50°F subcooled margin achieved in the pressurizer loop (refer to Figure 3.16). Both steam generators and the pressurizer established level and the system pressure was turned around from the HPI flow by 14 minutes into the transient (refer to Figures 3.17 and 3.18).

3. Effect of Return to Power

There was no return to power exhibited by any of the BOL cases analyzed above. Previous analysis experience (ref. Midland FSAR, Section 15D) has shown that a RC pump trip will mitigate the consequences of an EOL return to power condition by reducing the cooldown of the primary system. The reduced cooldown substantially increases the subcritical margin which, in turn, reduces or eliminates return to power.

D. Conclusions and Summary

A general assessment of Chapter 15 non-LOCA events identified three areas that warranted further investigation for impact of a RC pump trip on ESFAS low RC pressure signal. [redacted]

1. It was found that a pump trip does not significantly shorten the time to filling of the pressurizer and approximately the same time interval for operator action exists.

2. For the maximum overcooling case analyzed, the RC pump trip increased the amount of void formation in the hot leg 'candy cane' of the pressurizer loop; however, natural circulation was not completely blocked. The steam bubble was collapsed and full natural circulation was restored. Core cooling was maintained throughout the transient and no void formation occurred in the core.
3. The subcritical return-to-power condition is alleviated by the RC pump trip case due to the reduced overcooling effect.

Based upon the above assessment and analysis, it is concluded that the consequences of Chapter 15 non-LOCA events are not increased due to the addition of a RC pump trip on ESFAS low RC pressure signal, for all 177 FA lowered loop plants. Although there were no specific analyses performed for TECO, the conclusions drawn from the analyses for the lowered loop plants are applicable.

[redacted]

MINITRAP2 NODE DESCRIPTION

NODE NUMBER	DESCRIPTION
1	Reactor Vessel, Lower Plenum
2	Reactor Vessel, Core
3	Reactor Vessel, Upper Plenum
4,10	Hot Leg Piping and Upper S, G. Shroud
5-7,11-13	Primary, Steam Generator Tube Region
8,14	Cold Leg Piping
9	Reactor Vessel Downcomer
15	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

MINITRAP2 PATH DESCRIPTION

PATH NUMBER	DESCRIPTION
1	Core
2	Core Bypass
3	Upper Plenum, Reactor Vessel
4,11	Hot Leg Piping
5,12	Hot Leg Piping and Upper S. G. Shroud
6,7,13,14	Primary, Steam Generator
8,15	RC Pumps
9,16	Cold Leg Piping
10	Downcomer, Reactor Vessel
17	Pressurizer Surge Line
18,19,26,27	Steam Generator Downcomer
20,21,28,29	Secondary, Steam Generator
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

Table 3.1

[redacted]

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MINITRAP2 NODE DESCRIPTION

NODE NUMBER	DESCRIPTION
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 S. G. Shroud
5-7,11-13	Primary, Steam Generator Tube Region
8,14	Cold Leg Piping
9	Reactor Vessel Downcomer
15	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

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, S. G.
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, S. G.
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

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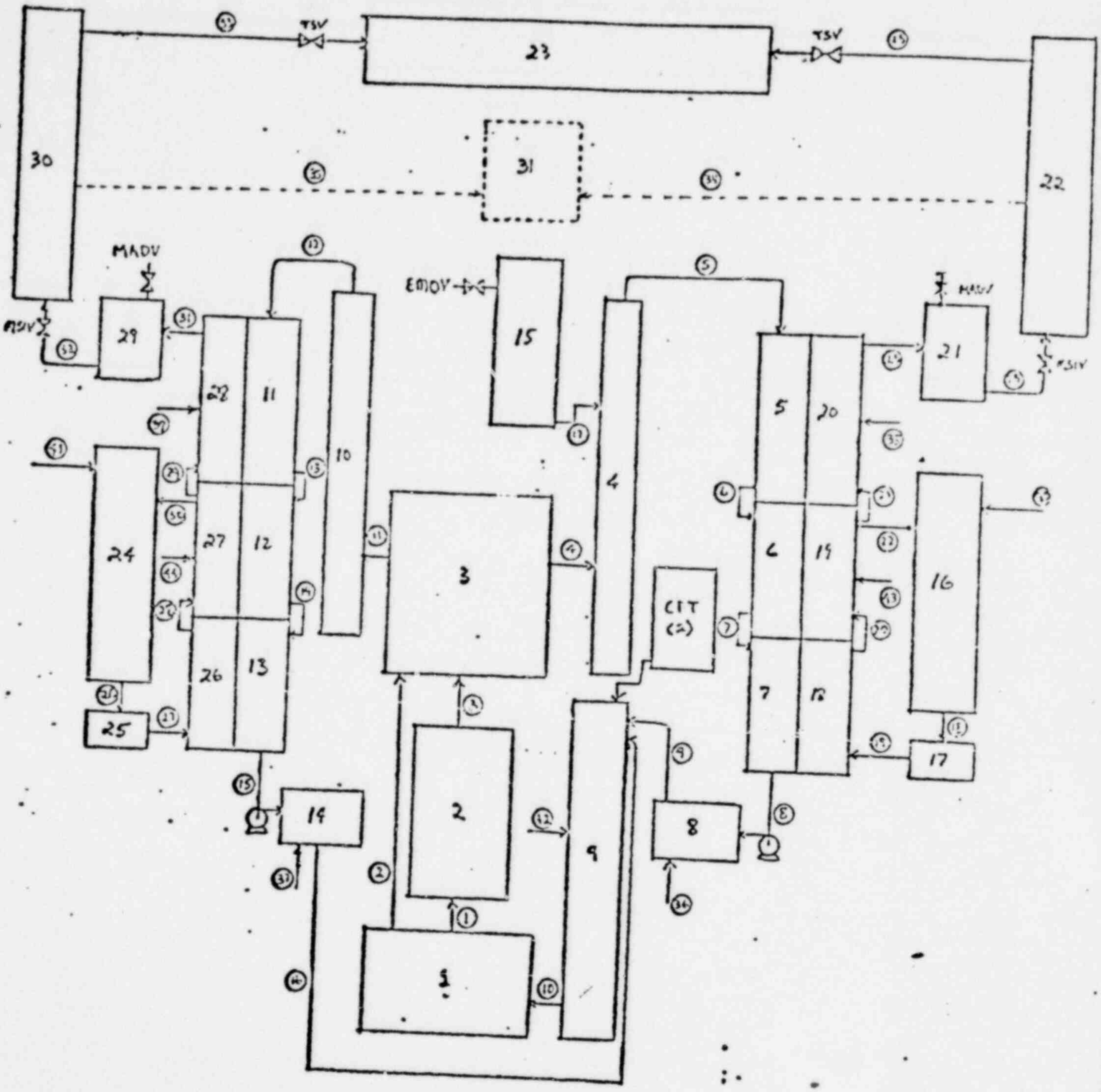


Figure
3.1
MINITRAP2 Noding and
Flow Path Scheme

[redacted]

1136 143

POOR ORIGINAL

PRESSURIZER AND STEAM GENERATOR LIQUID LEVEL VERSUS TRANSIENT TIME (102% FP, END OF LIFE, 0.6 FT^2 STEAMLINE BREAK (BOUNDING MODERATE FREQ.). (RC PUMP TRIP)

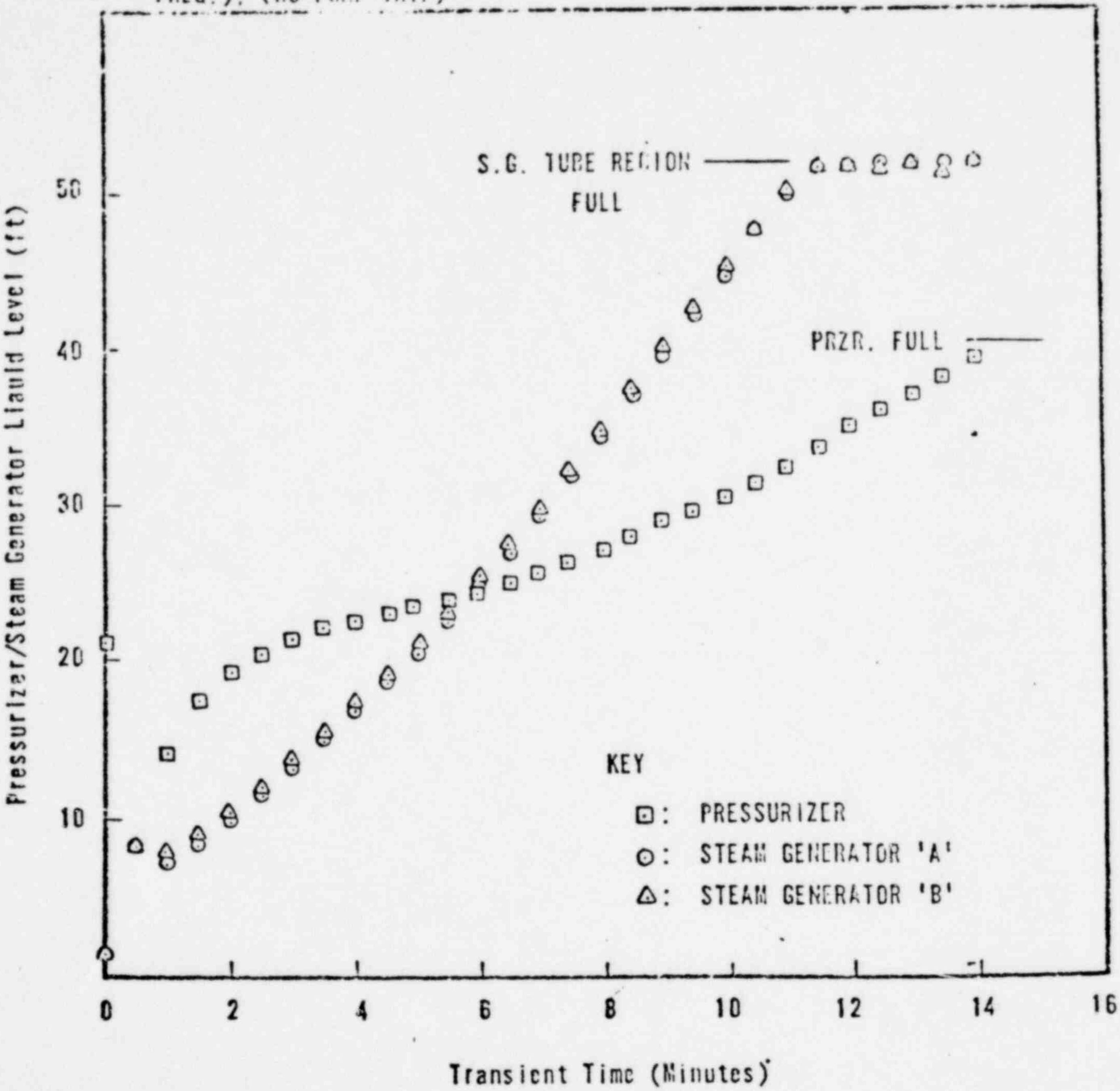


Figure 3.2

[redacted]

1136 144

PRESSURIZER AND STEAM GENERATOR LIQUID LEVEL VERSUS TRANSIENT TIME (102% FP, BEGINNING OF LIFE, 0.6 FT^2 STEAMLINE BREAK (BOUNDING MODERATE FREQ.), 1 LOOP ('B') RC PUMP TRIP).

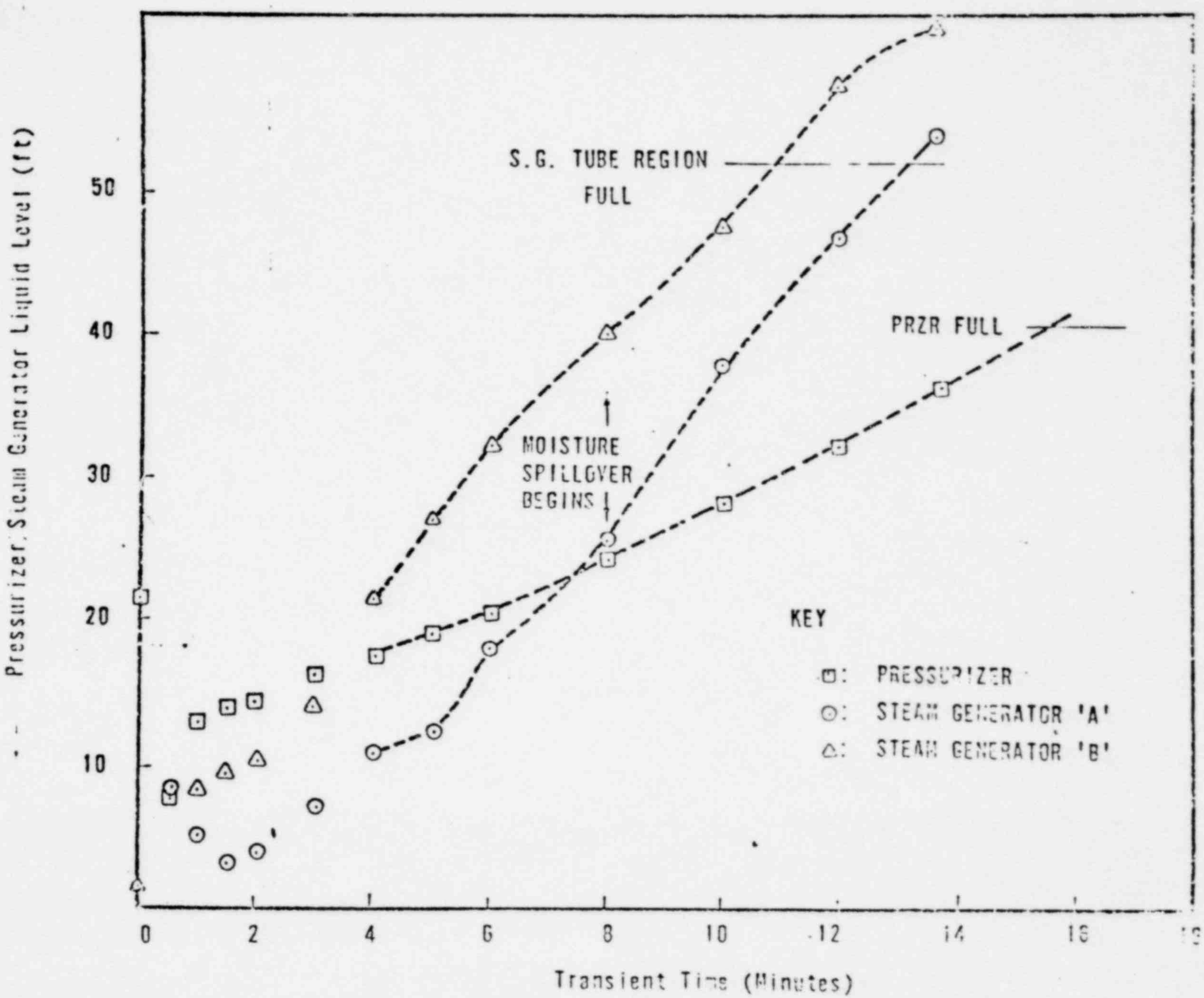


Figure 3.3

1136 145

COOLANT TEMPERATURES VERSUS TRANSIENT TIME
(102% FP, 0.6 FT^2 STEAMLINE BREAK, RC PUMP TRIP
(WORST MOD. FREQ).)

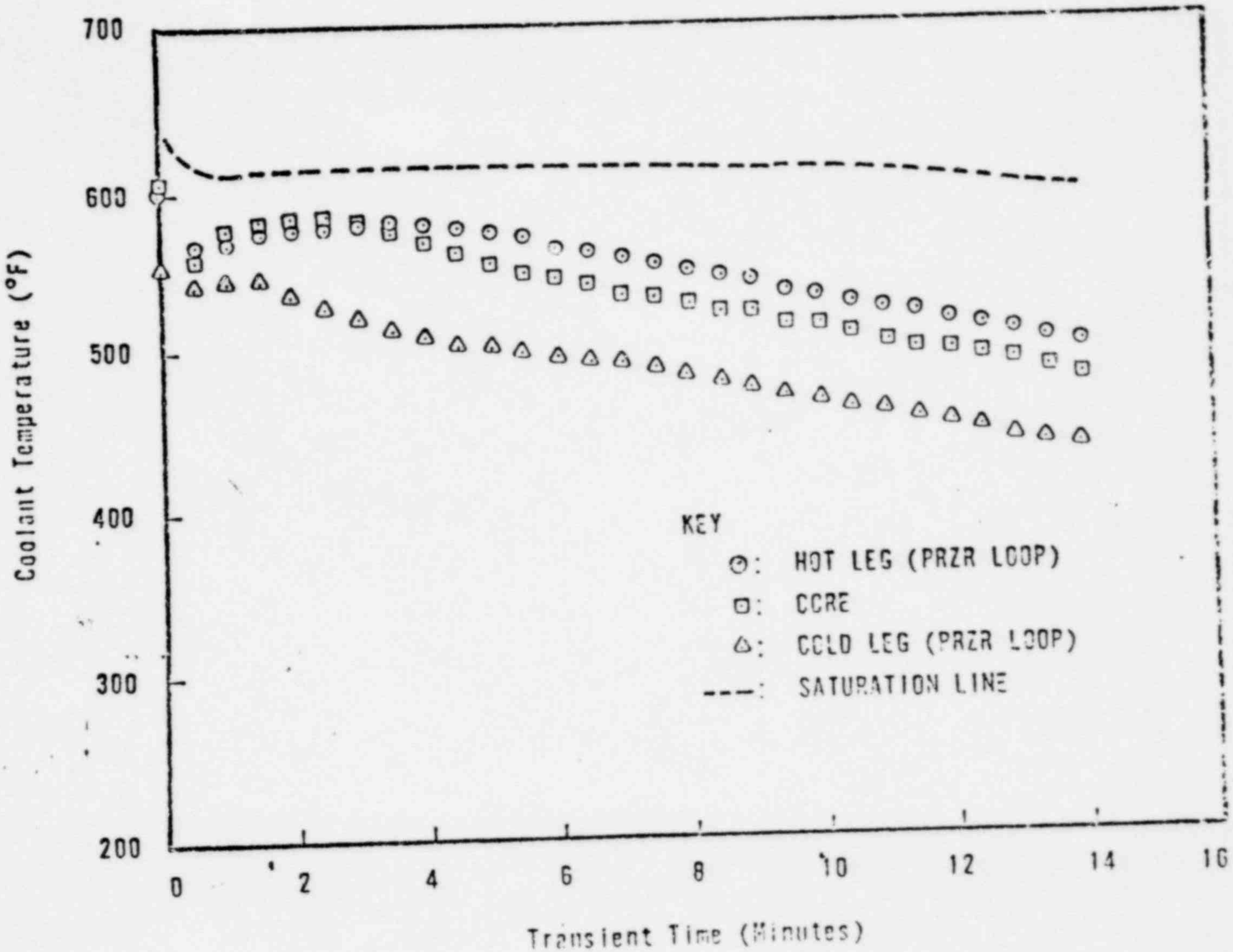


Figure 3.4

POOR ORIGINAL

[redacted]

1136 146

COOLANT TEMPERATURES VERSUS TRANSIENT TIME
(102% FP, BEGINNING OF LIFE, 12.2 FT^2 DOUBLE
END RUPTURE, STEAMLINE BREAK (UNMITIGATED)
NO RC PUMP TRIP)

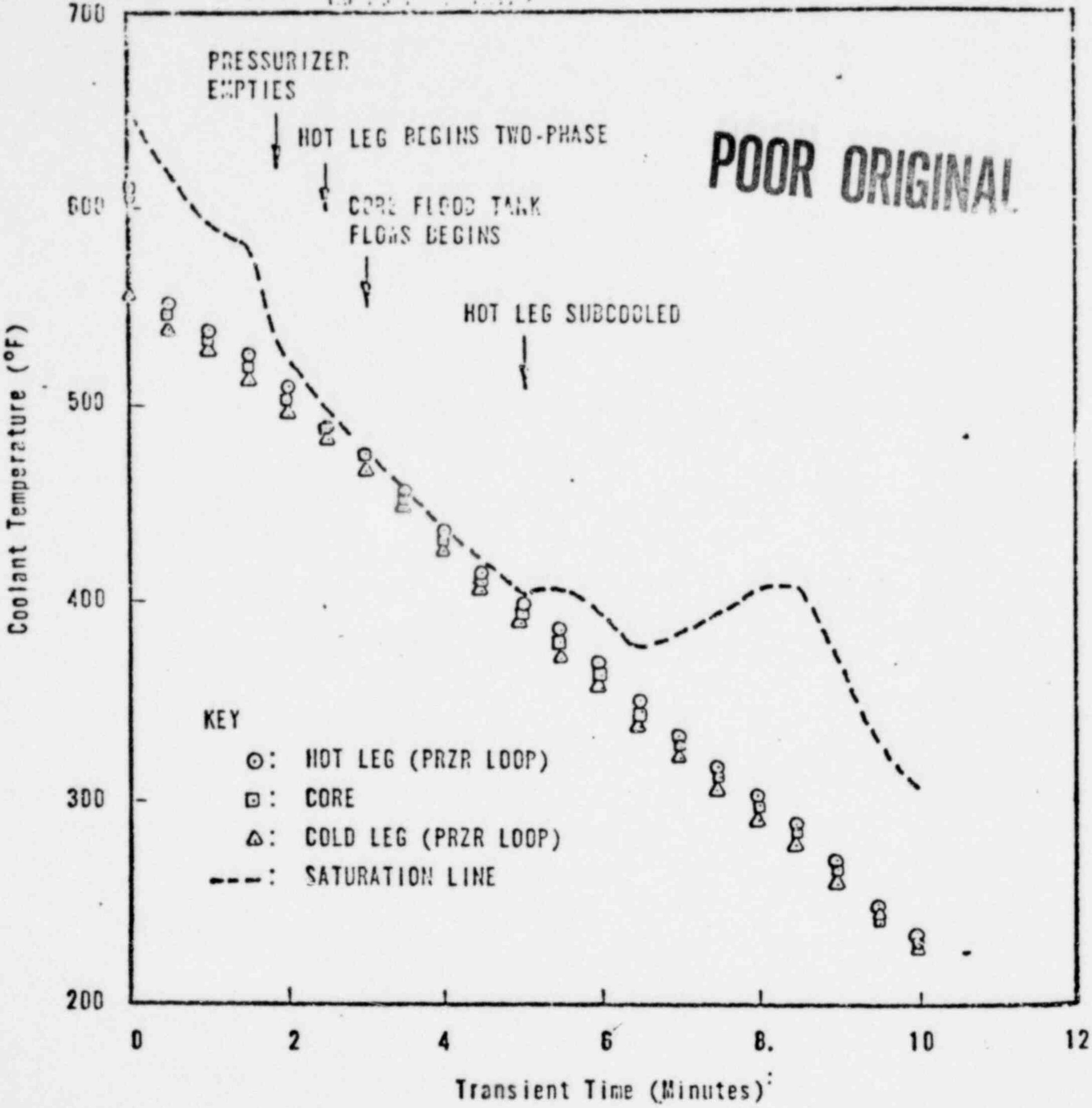


Figure 3.5

POOR ORIGINAL

[redacted]

1136 147

POOR ORIGINAL

COOLANT TEMPERATURES VERSUS TRANSIENT TIME (102% FP, BEGINNING OF LIFE, 12.2 FT² DOUBLE END RUPTURE, UNMITIGATED STEAMLINE BREAK, RC PUMP TRIP)

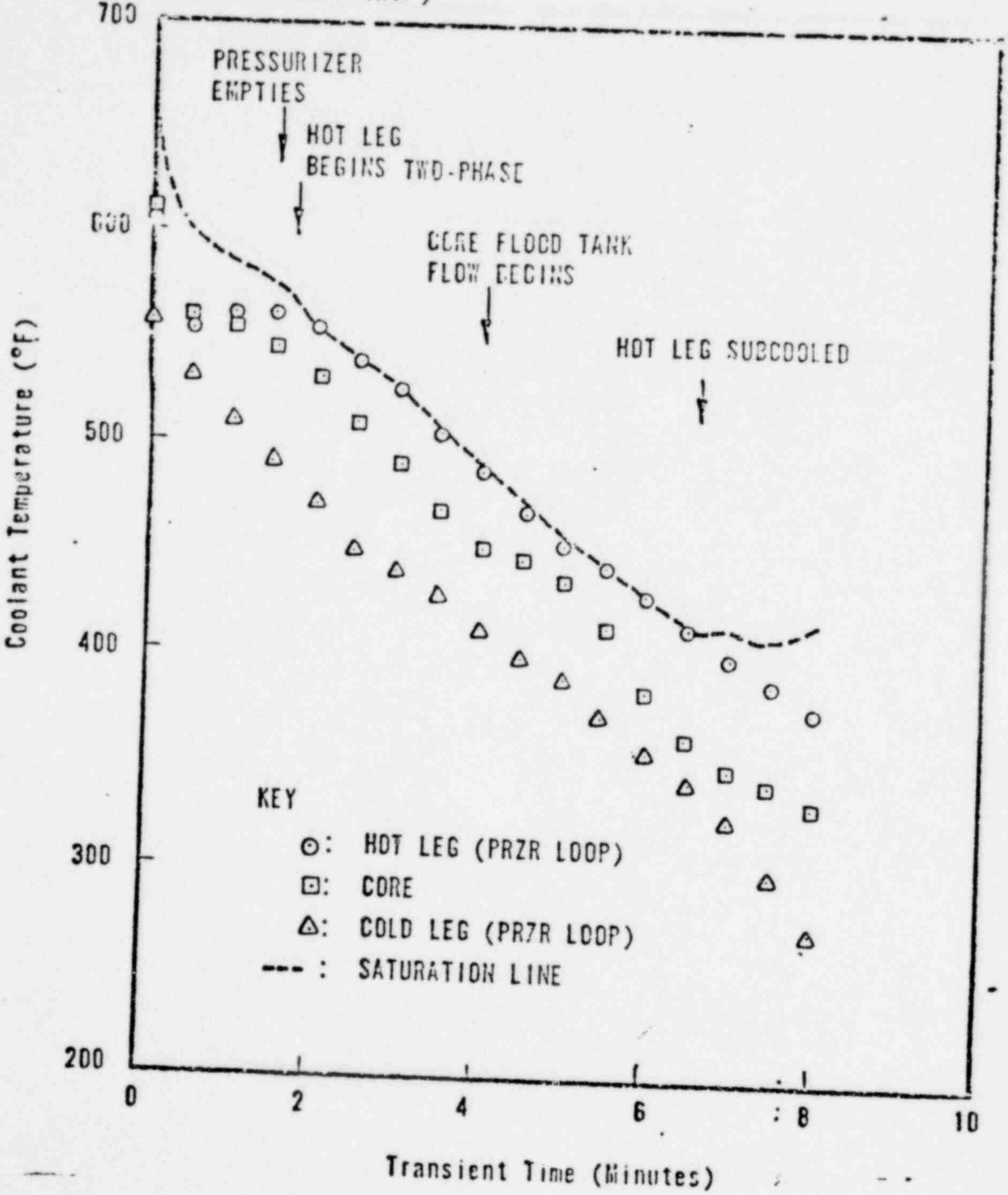


Figure 3.6

[redacted]

1136 148

POOR ORIGINAL

TOTAL STEAM BUBBLE VOLUME VERSUS TRANSIENT TIME
(102% FP, 12.2 FT^2 UNMITIGATED DOUBLE-ENDED
STEAMLINE BREAK)

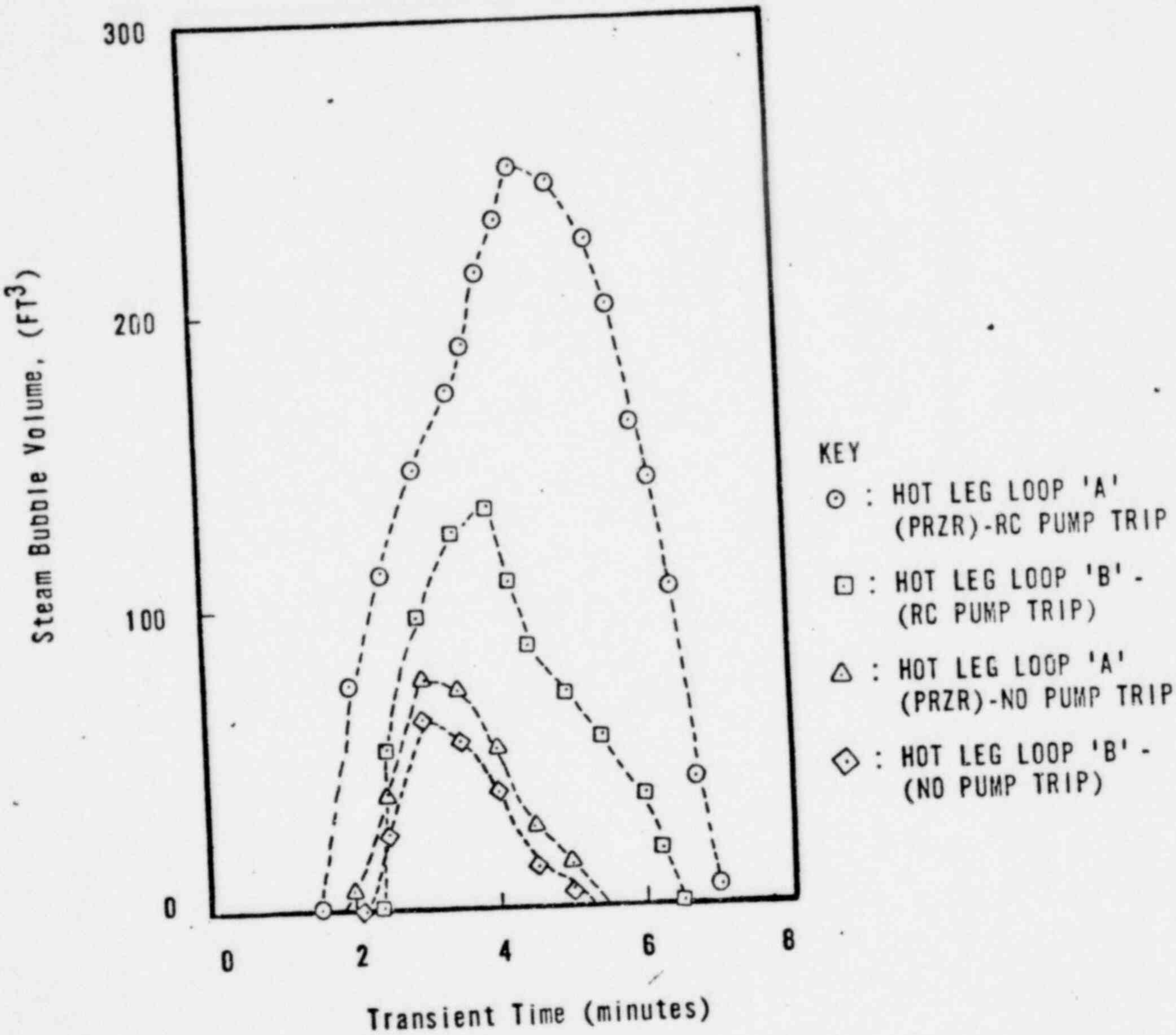
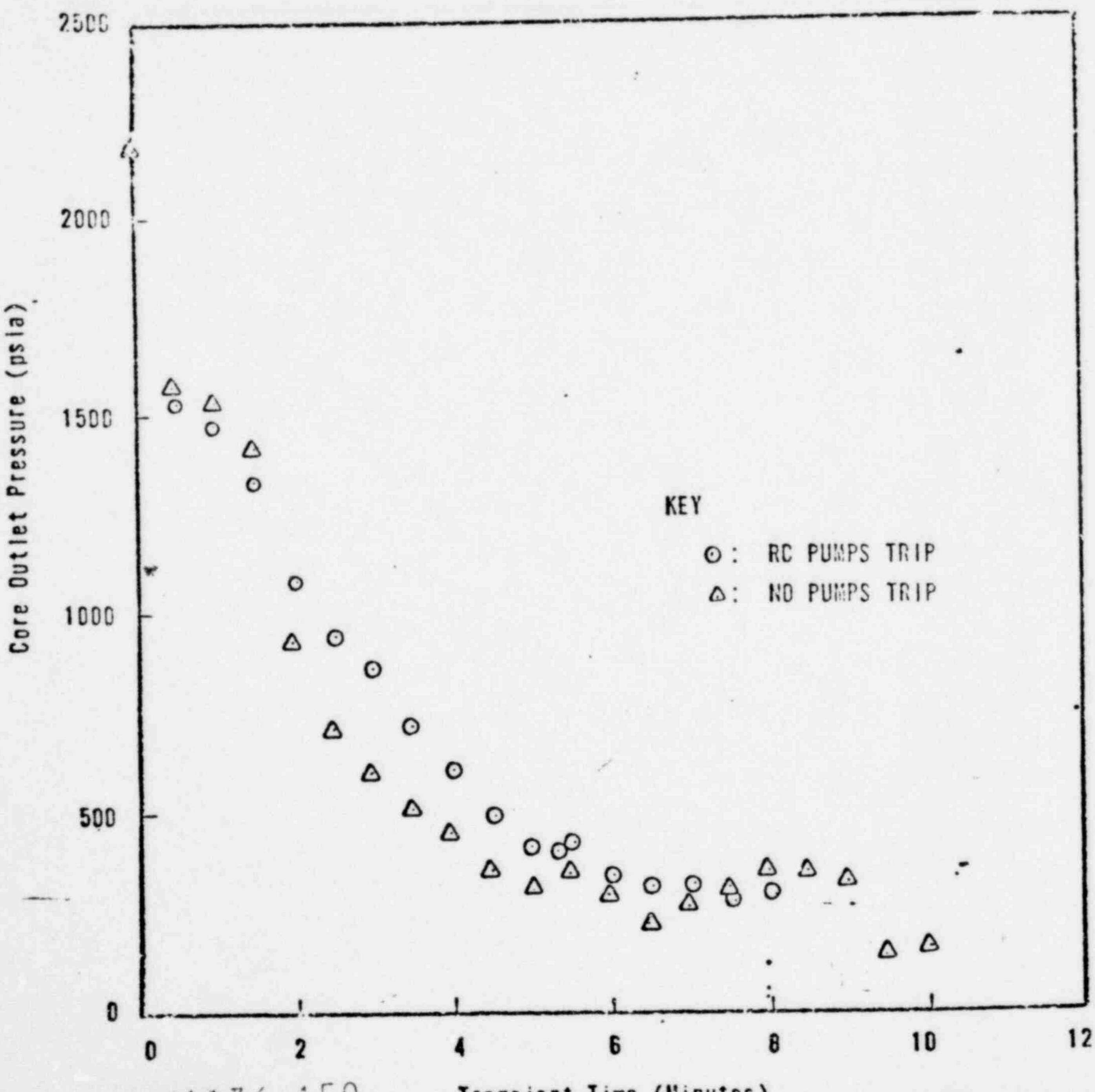


Figure 3.7

[redacted]

POOR ORIGINAL

CORE OUTLET PRESSURE VERSUS TRANSIENT TIME
(102% FP, BEGINNING OF LIFE, 12.2 FT^2 DOUBLE
END RUPTURE, UNMITIGATED STEAMLINE BREAK)



1136 150

Figure 3.8

[redacted]

POOR ORIGINAL

STEAM GENERATOR AND PRESSURIZER LIQUID LEVEL VERSUS TRANSIENT TIME
(102% FP, BEGINNING OF LIFE, 12.2 FT^2 DOUBLE END RUPTURE-UNMITIGATED
STEAMLINE BREAK)

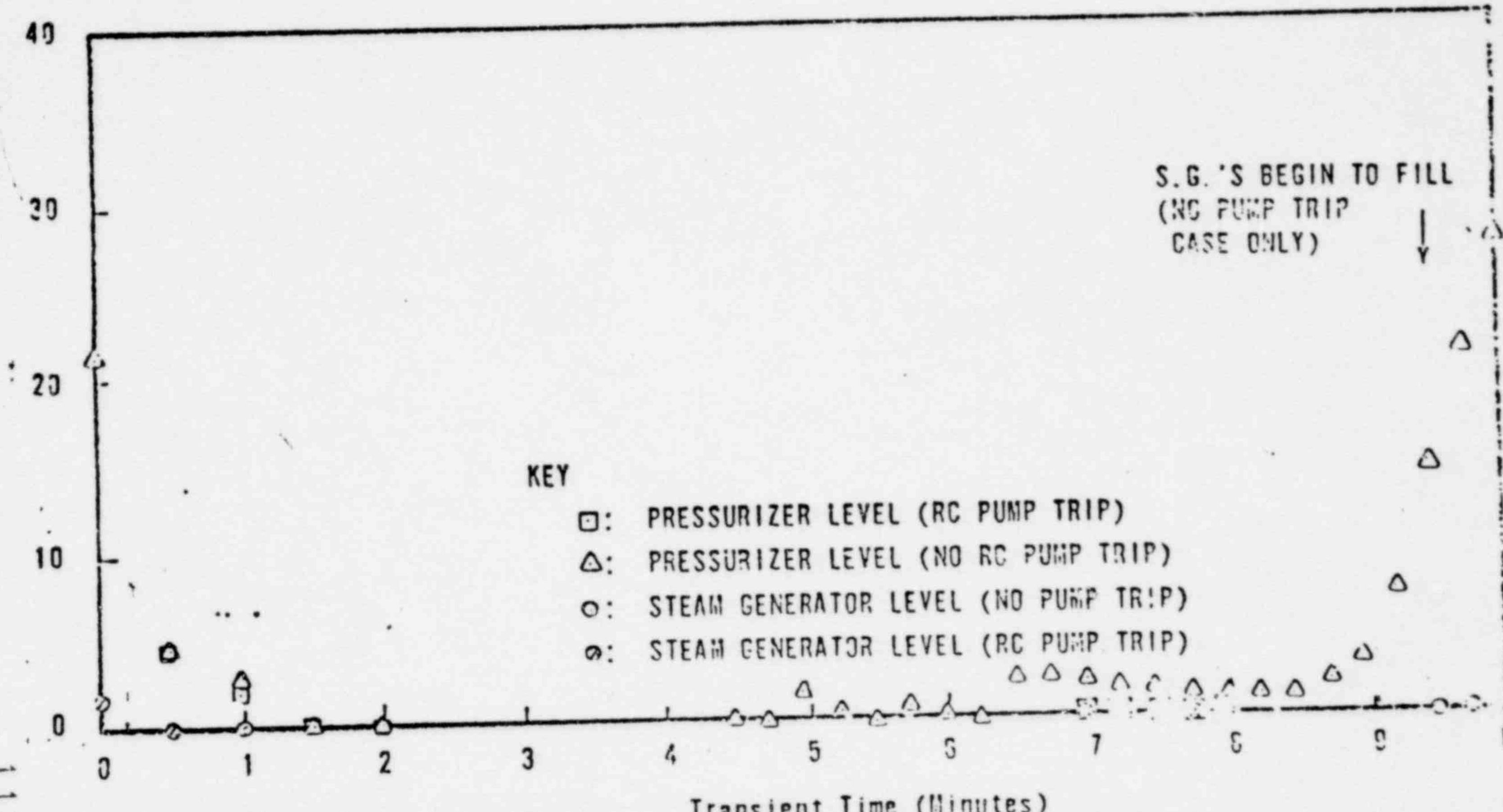


Figure 3.9

1136 151

[redacted]

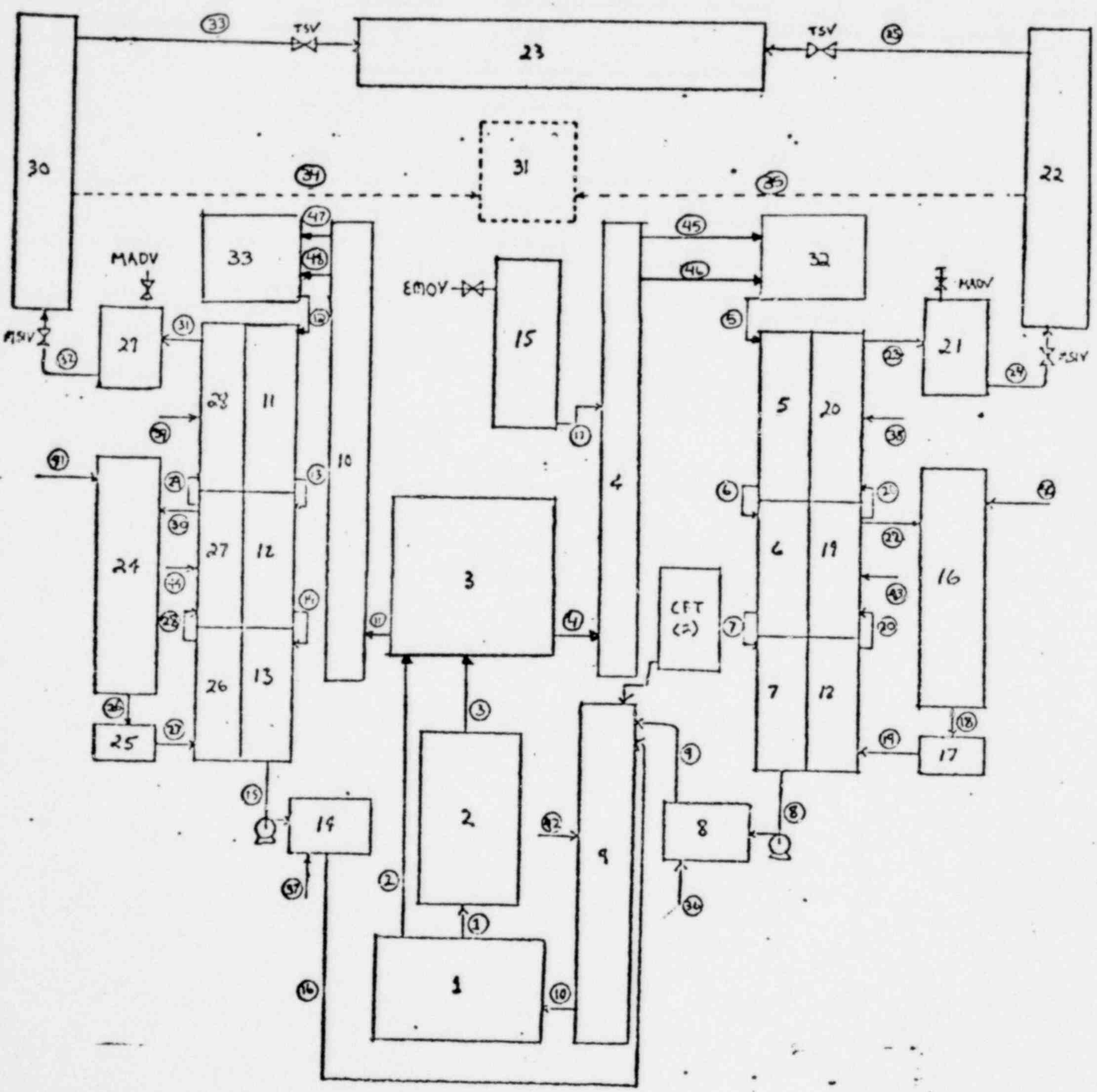
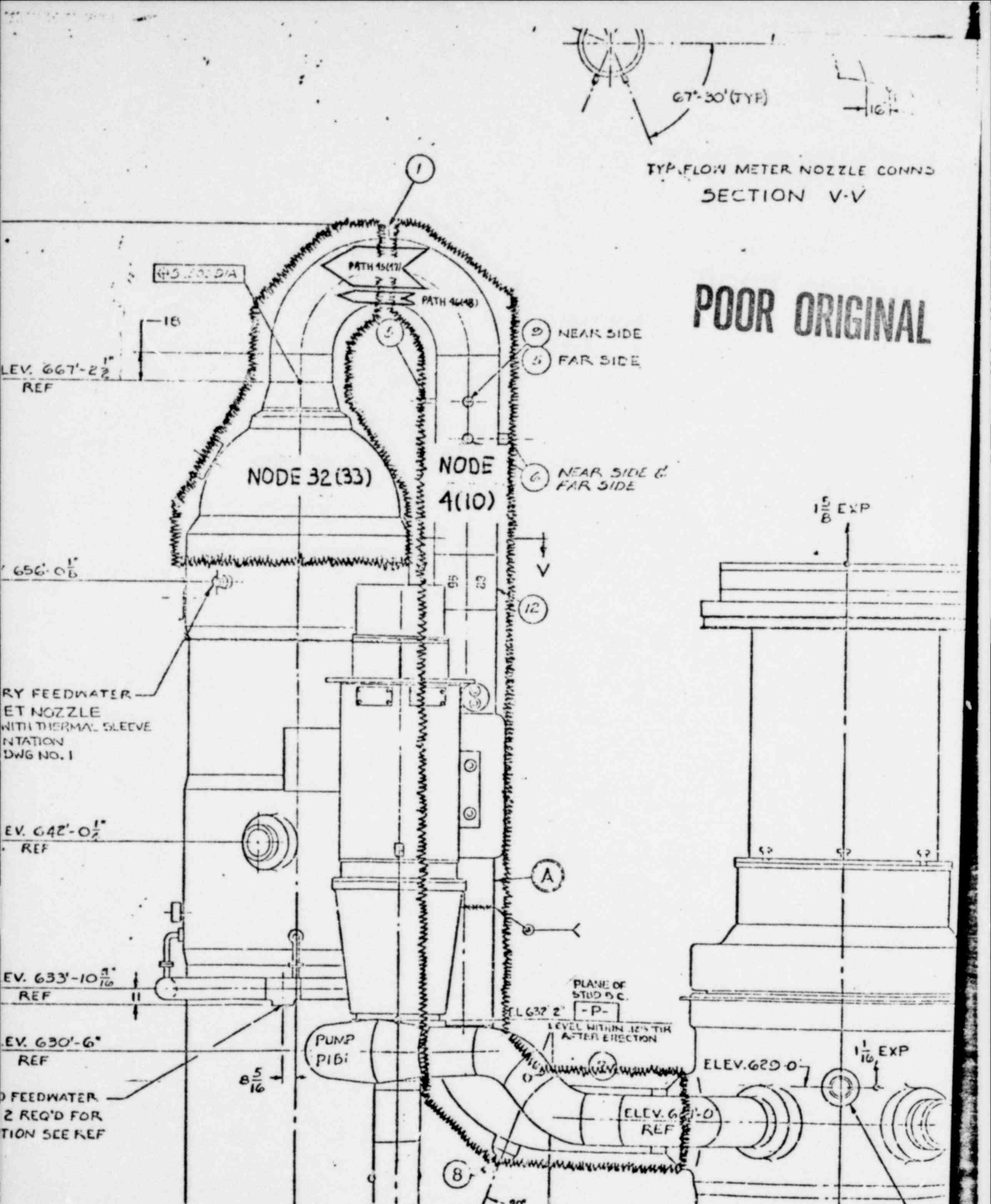


Figure 3.10

MINITRAP2 Noding and

Flow Path Scheme

[redacted]



Hot Leg Noding

Figure 3.11

1136 153 [redacted]

TOTAL STEAM BUBBLE VOLUME VS TRANSIENT TIME
(102% FP, 12.2 FT^2 DOUBLE-ENDED UNMITIGATED
STEAMLINE BREAK, RC PUMP TRIP)

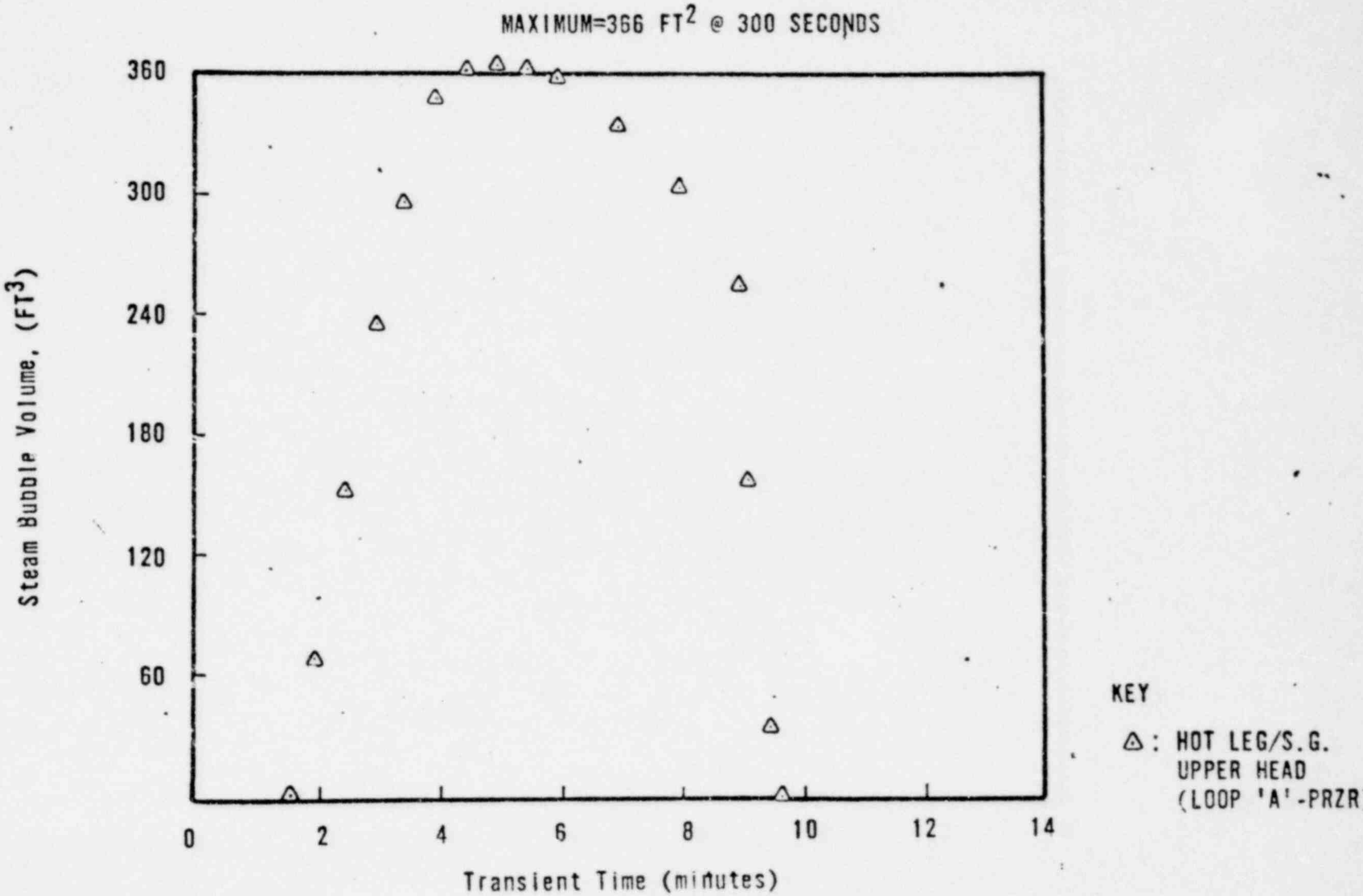


Figure 3.12

1136 154 [redacted]

LOOP 'A' CANDY CANE FLOW VERSUS TRANSIENT TIME
(102% FP, 12.2 FT^2 DOUBLE-ENDED UNMITIGATED
STEAMLINE BREAK, RC PUMP TRIP)

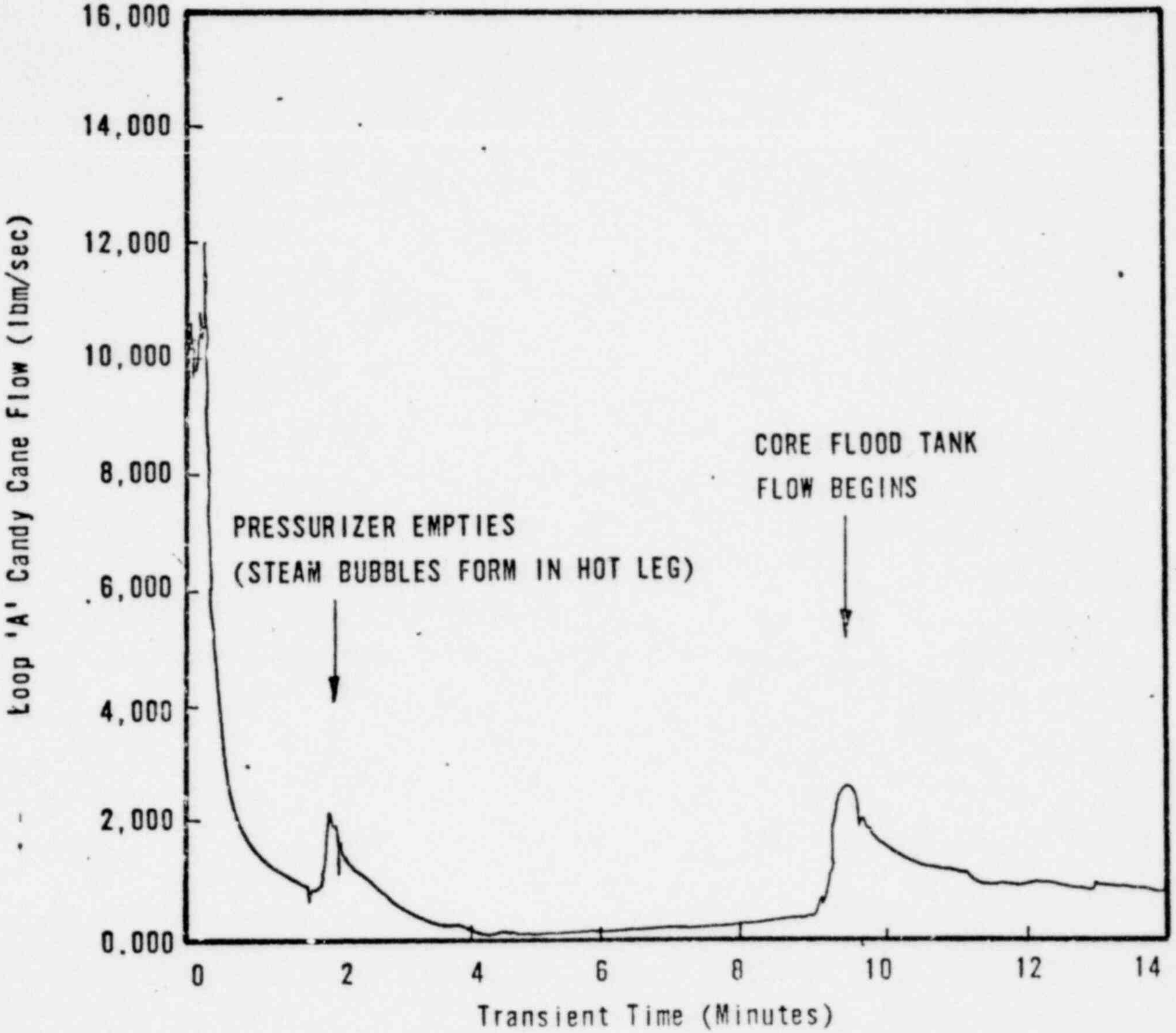


Figure 3.13

[redacted]

1136 155

Loop 'B' CANDY CANE FLOW VERSUS TRANSIENT TIME
(102% FP, 12.2 FT^2 DOUBLE-ENDED UNMITIGATED
STEAMLINE BREAK, RC PUMP TRIP)

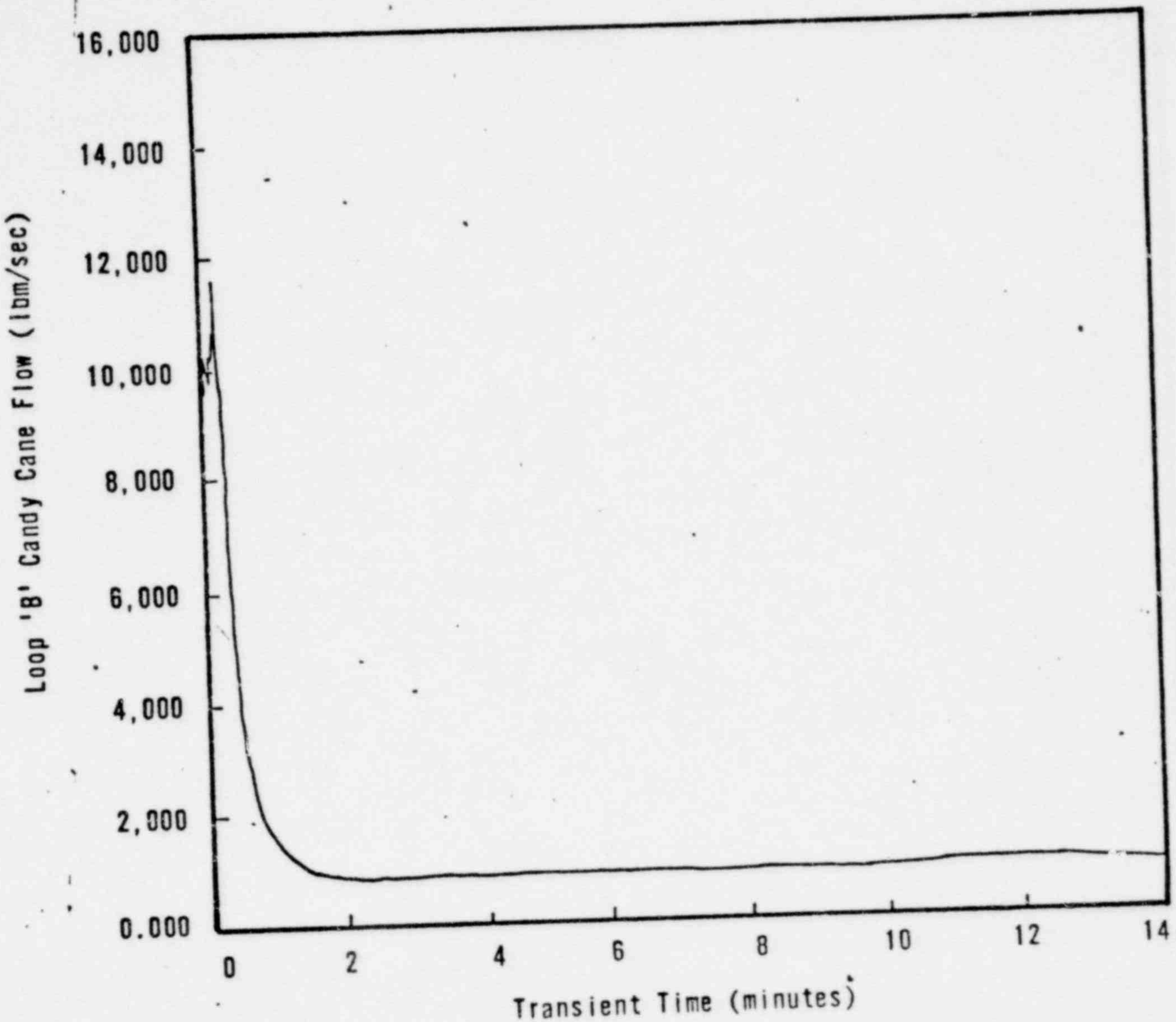


Figure 3.14

[redacted]

CORE FLOW VERSUS TRANSIENT TIME
(102% FP, 12.2 FT^2 DOUBLE-ENDED
UNMITIGATED STEAMLINE BREAK, RC
PUMP TRIP)

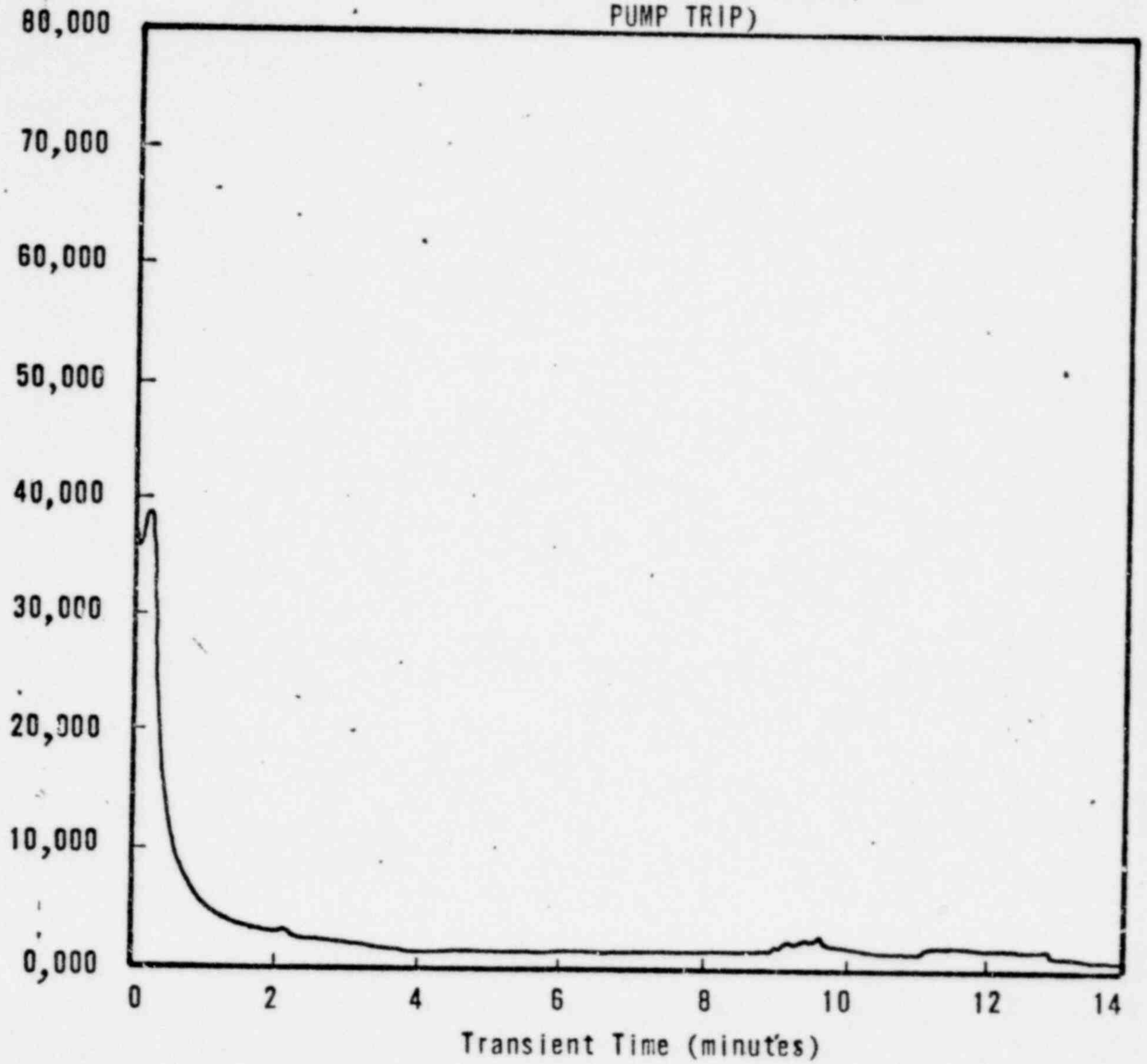


Figure 3.15

[redacted]

1136 157

COOLANT TEMPERATURES VERSUS TRANSIENT TIME
(102% FP, 12.2 FT^2 DOUBLE-ENDED UNMITIGATED
STEAMLINE BREAK, RC PUMP TRIP)

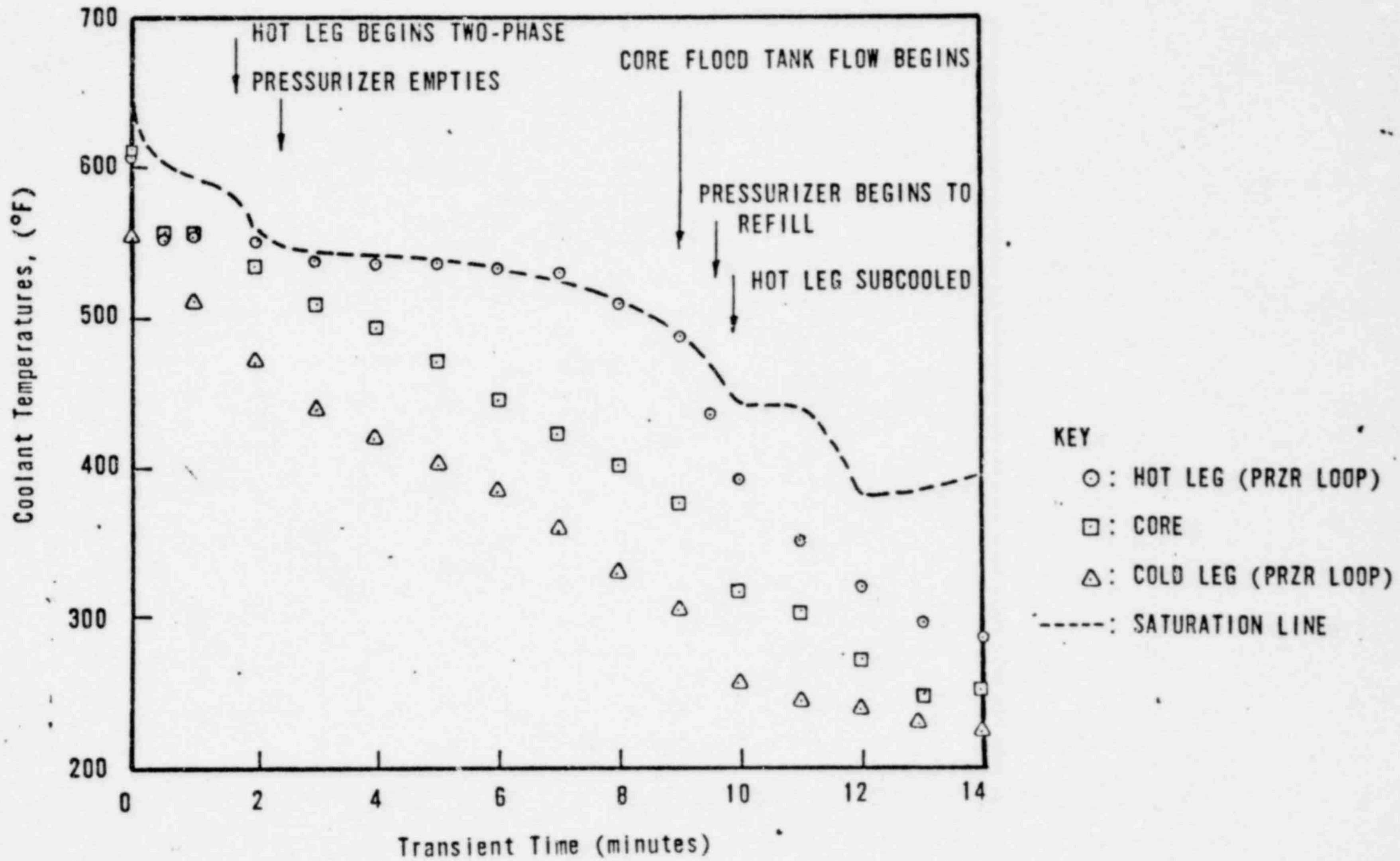


Figure 3.16

[redacted]

1136 158

PRESSURIZER AND STEAM GENERATOR LIQUID LEVEL VERSUS TRANSIENT TIME (102% FP, 12.2 FT^2 UNMITIGATED DOUBLE-ENDED STEAMLINE BREAK, RC PUMP TRIP)

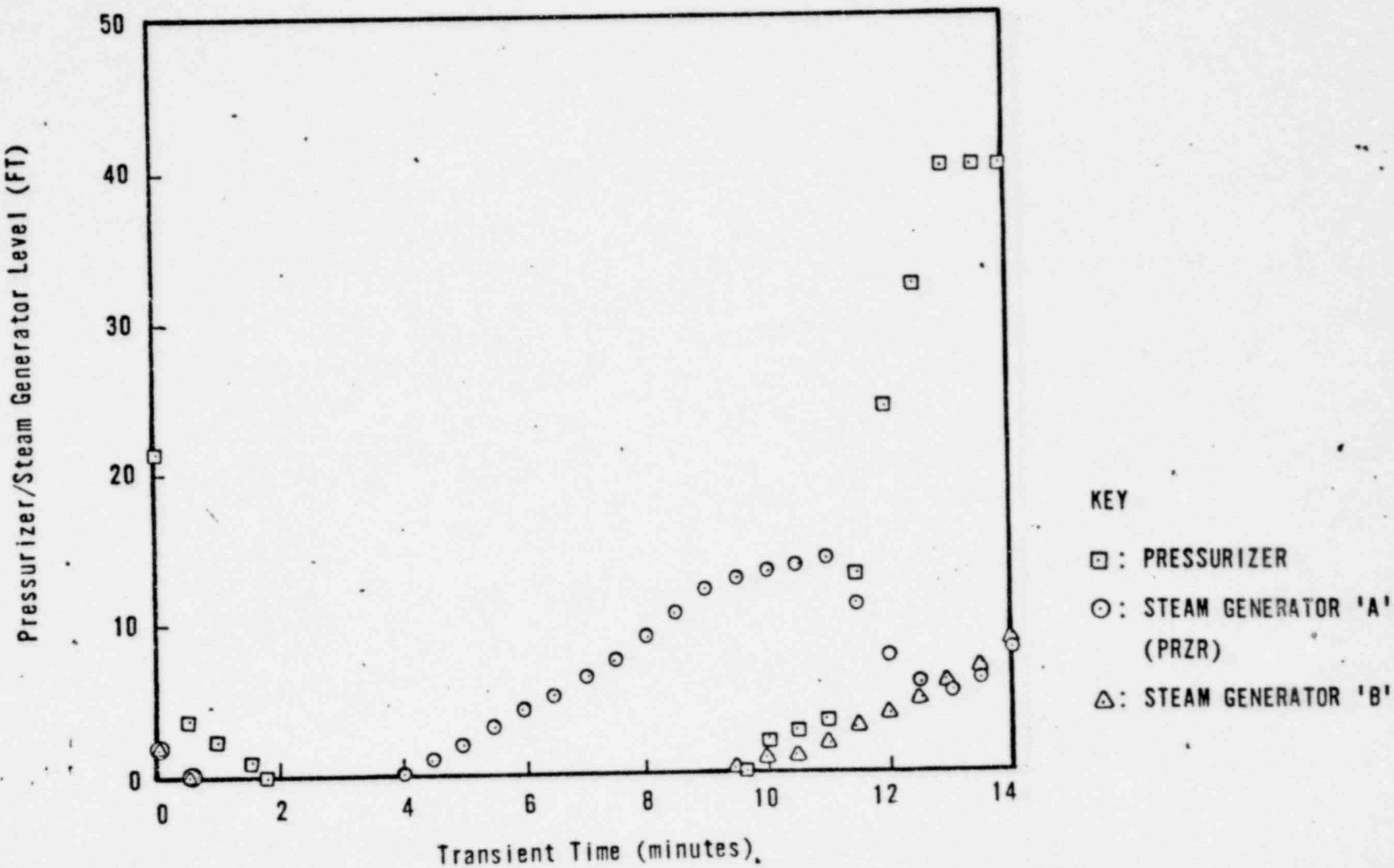


Figure 3.17

[redacted]

CORE OUTLET PRESSURE VERSUS TRANSIENT TIME
(102% FP, 12.2 FT^2 DOUBLE-ENDED UNMITIGATED
STEAMLINE BREAK, RC PUMP TRIP)

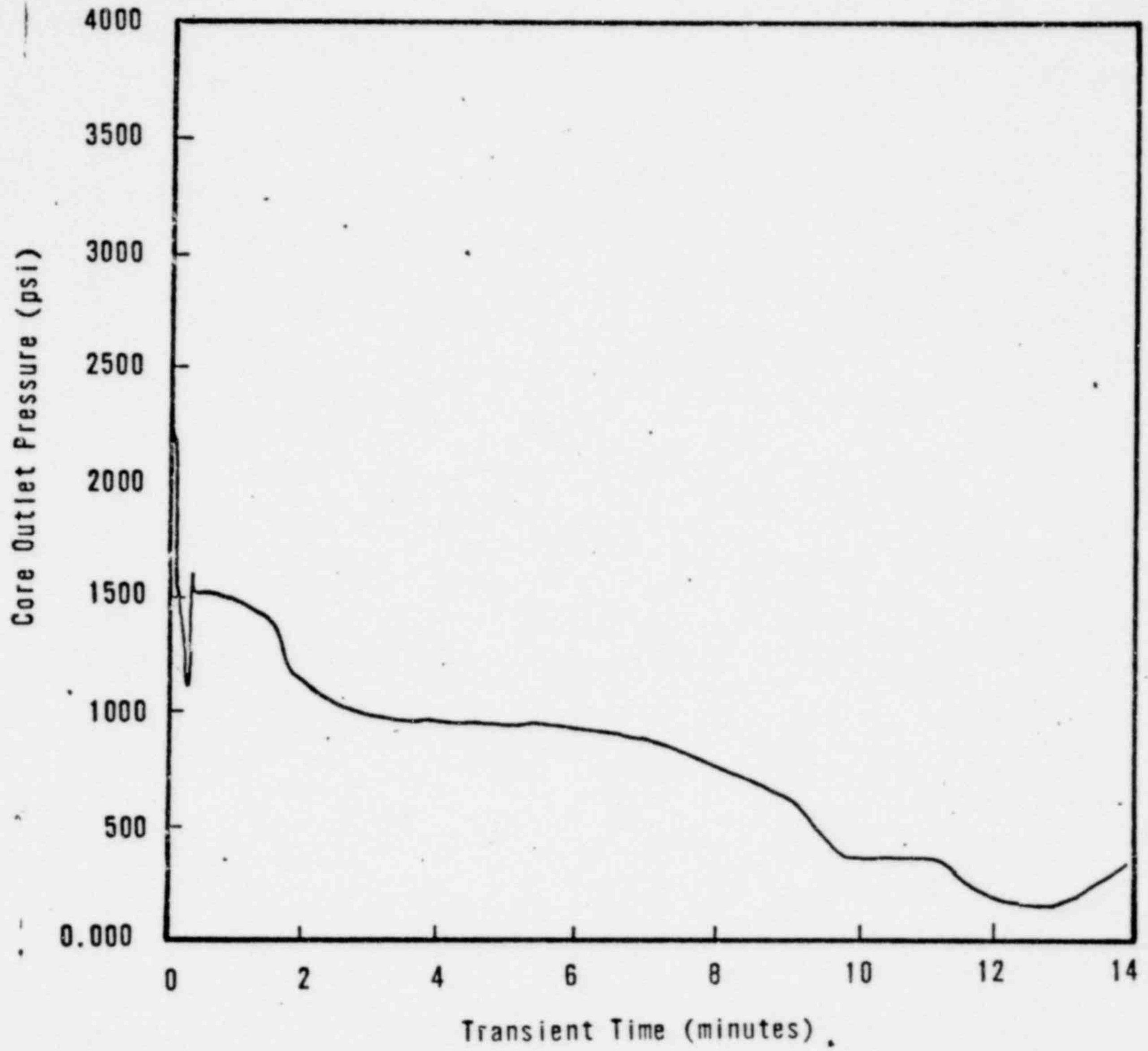


Figure 3.18

[redacted]

1136 160

Response to I.E. Bulletin 79-05C

Short-Term Actions

Item 4: Revise emergency procedures and train all licensed reactor operators and senior operators based on the guidelines developed under Item 3 above.

Response 4: The emergency procedures have been revised and all licensed reactor operators and senior operators have been trained based on the guidelines developed under I.E. Bulletin 79-05C, Item 3.

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