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August 29, 1979

1-089-16

Mr. K. V. Seyfrit, Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Comm.
Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

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Subject: Arkansas Nuclear One-Unit 1
Docket No. 50-313
License No. DPR-51
IE Bulletin 79-05C
(File: 1510.1)

Gentlemen:

Pursuant to the requests of the subject bulletin, the following is provided in response to the short-term actions.

Item 1

In the interim, until the design change required by the long-term action of this Bulletin has been incorporated, institute the following actions at your facilities:

a. Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs.

b. Provide two licensed operators in the control room at all times during operation to accomplish this action and other immediate and follow-up actions required during such an occurrence. For facilities with dual control rooms, a total of three licensed operators in the dual control room at all times meets the requirements of this Bulletin.

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Response

Upon receipt of IE Bulletin 79-05 C, Arkansas Nuclear One-Unit 1 (ANO-1) was shut down. Prior to heatup, the operators were instructed to immediately trip all operating RCPs upon reactor trip and initiation of HPI caused by low RCS pressure. Two licensed operators are in the control room at all times during operation to accomplish this action and other immediate and followup actions required during such an occurrence.

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 2200 degrees F is identified.

Response

Attachment 1, "Analysis Summary in Support of an Early RC Pump Trip", Section II provides the results and conclusions of the requested LOCA analyses. Section III of Attachment 1 is included to allow development of non-LOCA guidelines as required in Item 3.

Item 3

Based on the analyses done under Item 2 above, develop new guidelines for operator action, for both LOCA and non-LOCA transients, that take into account the impact of RCP trip requirements. For Babcock & Wilcox designed reactors, such guidelines should include appropriate requirements to fill the steam generators to a higher level, following RCP trip, to promote natural circulation flow.

Response

New guidelines for operator action during LOCA and non-LOCA transients have been developed based on the results of analysis discussed in response to Item 2. These guidelines will be issued via site instructions and incorporated into the appropriate procedures.

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Item 4

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

Response

No later than September 13, 1979, emergency procedures will be revised based on the guidelines discussed in response to Item 3. Reactor Operators (ROs) and Senior Reactor Operators (SROs) will receive training on the revised procedures prior to being utilized on shift.

Item 5

Provide analyses and develop guidelines and procedures related to inadequate core cooling (as discussed Section 2.1.9 of NUREG-0578, "TMI 2 Lessons Learned Task Force Status Report and Short-Term Recommendations") and define the conditions under which a restart of the RCPs should be attempted.

Response

A response to Item 5 will be provided by October 31, 1979.

The long-term action requires us to propose and submit a design which will assure automatic tripping of the operating RCPs under certain conditions. A conceptual design for automatic RCP tripping will be provided to you by October 31, 1979, for your review. This schedule will allow sufficient time to evaluate the alternatives available and choose the best means for providing this automatic trip. Upon notification of your approval of our concept, engineering and procurement will begin. The modification will be installed at the first available outage of sufficient duration following completion of engineering and procurement.

Very truly yours,

David C. Trimble

David C. Trimble
Manager, Licensing

DCT/DGM/ew

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Attachment

cc: Mr. Victor Stello, Jr., Director
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ATTACHMENT 1

ANALYSIS SUMMARY IN SUPPORT OF
AN EARLY RC PUMP TRIP

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ANALYSIS SUMMARY IN SUPPORT OF AN EARLY RC PUMP TRIP

I. INTRODUCTION

B&W has evaluated the effect of a delayed RC pump trip during the course of small loss-of-coolant accidents and has found that an early trip of the RC pumps is required to show conformance to 10CFR50.46. A summary of the LOCA analyses performed to date is provided in Section II. This discussion includes:

1. A description of the models utilized.
2. Break spectrum results with continuous RC Pump Operation.
3. Break spectrum results with delayed RC pump trips including estimates of peak cladding temperatures.
4. Justification that a prompt pump trip following ESFAS actuation on low RC pressure provides LOCA mitigation.

An impact assessment of the required pump trip on non-LOCA events has also been completed and is presented in Section III. This evaluation supports the use of a pump trip following ESFAS actuation for LOCA mitigation since no detrimental consequences on non-LOCA events were identified.

II. SMALL BREAK ANALYSES

A. Introduction

Previous small break analyses have been performed assuming a loss-of-offsite power (reactor coolant pump coastdown) coincident with reactor trip. These analyses support the conclusion that an early RC pump trip for a LOCA is a safe condition. However, a concern has been identified regarding the consequences of a small break transient in which the RC pumps remain operative for some time period and then are lost by some means (operator action, loss-of-offsite power, equipment failure, etc.). This section contains the results of a study to further understand how the small break LOCA transient evolves with the RC pumps operative. Specifically, section B. describes the system response with the RC pumps running for B&W's 177-FA lowered-loop plants. Included in this section is the development of the model used for the analysis, a break spectrum sensitivity study, and peak cladding temperature assessments for cases where the RC pumps trip at the worst time.

Section C demonstrates the applicability of the conclusions drawn in section B to a 177-FA raised-loop plant (Davis-Besse 1). The effect of a prompt tripping of the RC pumps upon receipt of a low pressure ESFAS signal is discussed in section D . Finally, section E summarizes the conclusions of this analysis.

B. System Response With RC Pumps Running

1. Introduction

Recent evaluations have been performed to examine the primary system response during small breaks with the RC pumps operative. During the transient with the RC pumps available, the forced circulation of reactor coolant will maintain the core at or near the saturated fluid temperature. However, for a range of break sizes, the reactor coolant system (RCS) will evolve to high void fractions due to the slow system depressurization and the high liquid (low quality fluid) discharge through the break as a result of the forced circulation. In fact, the RCS void fraction will increase to a value in excess of 90% in the short term. In

the long term, the system void fraction will decrease as the RCS RCS depressurizes, HPI flow increases, and decay heat diminishes.

With the RCS at a high void fraction, if all RC pumps are postulated to trip, the forced circulation will no longer be available and the residual liquid would not be sufficient to keep the core covered. A cladding temperature excursion would ensue until core cooling is reestablished by the ECC systems. The following paragraphs summarizes the results of the analyses which were performed for the 177-FA lowered-loop plants, to develop the consequences of this transient.

2. Method of Analysis

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"[1] and the letter J.H. Taylor (B&W) to S.A. Varga (NRC), dated July 18, 1978[2], which is applicable to the 177-FA lowered-loop plants for power levels up to 2772 MWt. The analysis uses the CRAFT2[3] code to develop the history of the RCS hydrodynamics. However, the CRAFT2 model used for this study is a modification of the small break evaluation model described in the above references. Figure 2-1 shows the CRAFT2 noding diagram for small breaks from the above referenced letter. The modified CRAFT2 model consists of 4 nodes to simulate the primary side, 1 node for the secondary side of the steam generator, and 1 node representing the reactor building. Figure 2-2 shows a schematic diagram of this model. Node 1 contains the cold leg pump discharge piping, downcomer, and lower plenum. Node 2 is the primary side of the SG and the pump suction piping. Node 3 contains the core, upper plenum, and the hot legs. Node 4 is the pressurizer and nodes 5 and 6 represent the reactor building and the SG secondary side, respectively. This 6 node model is highly simplified compared to those utilized in past ECCS analyses. It does, however, maintain RCS volume and elevation relationships which are important to properly evaluate the system response during a small break with the RC pumps running.

The breaks analyzed in this section are assumed to be located in the cold leg piping between the reactor coolant pump discharge and the reactor vessel. Section B.7 demonstrates that this is the worst break location. Key assumptions which differ from those described in the July 18, 1978, letter are those concerning the equipment availability and phase separation. These are discussed below.

a. Equipment Availability

The analyses which were performed assumed that the RC pumps remain operative after the reactor trips. For select cases, after the system has evolved to high void fractions (approximately 90%) the RC pumps were assumed to trip. Also, the impact of 1 versus 2 HPI systems for pump injection were examined. The majority of the analyses performed assumed 2 HPI pumps. However, as is demonstrated later, even with 2 HPI pumps available, cladding temperatures will exceed the criteria of 10 CFR 50.46 using Appendix K evaluation techniques. Therefore, further analysis with only 1 HPI pump would only be academic.

b. Phase Separation

The present ECCS evaluation model created to evaluate small breaks without RC pumps operative, (quiescent RCS) utilizes the Wilson[4] bubble rise correlation for all primary system control volumes in the CRAFT evaluation. In this analysis, for the time period that the RC pumps are operative, the primary system coolant is assumed to be homogeneous, i.e., no phase separation in the system. In reality, the flow rates in the core and hot legs are low enough that slip will occur. This will cause an increased liquid inventory in the reactor vessel compared to that calculated with the homogeneous model. With the homogeneous assumption, core fluid is continuously circulated throughout the primary system and a portion of that fluid is lost via the break. During the later stages of the transient, a slip model will result in fluid being trapped in the reactor vessel and the hot legs. The only method of losing liquid during this period will be by boiling caused by the core decay heat. Thus, the assumption of homogeniety for the period with the RC pumps operative is conservative.

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Following tripping of the RC pumps and the subsequent loss-of-forced circulation, the system will collapse and separate. The residual liquid will then collect in the reactor vessel and the loop seal in the cold leg suction piping. For this period of the transient, the Wilson bubble rise model is utilized.

The homogeneous assumption for the period with the RC pumps operating applies to nodes 1, 2, and 3 in the CRAFT model. Node 4, the pressurizer, and node 6, the secondary side of the steam generators, utilize the Wilson bubble rise model throughout the transient as these nodes are not in the direct path of the forced circulation.

3. Benchmarking of the 6 Node CRAFT Model

Studies were performed to compare the results of the 6 node model to the more extensive evaluation model for B&W's 177-FA lowered-loop plants as described in the letter J.H. Taylor (B&W) to S.A. Varga (NRC), dated July 18, 1978. The break size selected for this comparison is a 0.025 ft^2 break at pump discharge. This break represents the largest single-ended rupture of a high energy line (2-1/2 inch sch 160 pipe) on the operating plants. The break can be viewed as "realistic" or the worst that would be expected on a real plant. Figures 2-3 and 2-4 are the results of this comparison. System pressure and percent void fraction shown in Figures 2-3 and 2-4, respectively, compare very well with those from the more extensive (23 nodes) CRAFT2 small break model. As seen in these figures, the difference is not significant and is less than a few percent. The computer time for this 6 node model is, however, significantly decreased. The model utilized for this study is thus justified based on comparison of results to the more extensive small break model and desirable because of its economical run time.

4. Analysis Results

The break sizes examined for this analysis ranged from 0.025 ft^2 to 0.2 ft^2 in area and are located in the pump discharge piping. Breaks of this size do not result in a rapid system depressurization and rely predominantly upon the HPIs for mitigation.

Table 2-1 summarizes the analyses performed for this evaluation. The majority of the analyses performed utilized 2 HPI pumps throughout the transient. The effect of utilizing 1 HPI pump is discussed in this section.

Figures 2-5 and 2-6 show the system pressure and average system void fraction transients for the break spectrum analyzed assuming continuous RC pump operation and 2 HPI's available. In Figure 2-6, the average system void fraction is defined as

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

where
V_1 = total primary liquid volume excluding the pressurizer at time = 0,
V_2 = total primary liquid volume excluding the pressurizer at time = t.

This parameter was utilized in place of the mixture height in that the coolant will tend to be homogeneously mixed with the RC pumps operative. Under these assumptions, the core is cooled by forced circulation of two-phase fluid and not by pool boiling as in the case where the RC pumps are not running and separation of steam and water occurs. As shown in Figure 2-5, the system pressure response is basically independent of break size during the first several hundred seconds into the transient. This occurs because the forced circulation of reactor coolant maintains adequate heat transfer in the steam generators; the primary system thus depressurizes to a pressure (about 1100 psia) corresponding to the secondary control pressure (i.e., set pressure of SG safety relief valves). After some time (250 seconds for the 0.1 ft^2 break), the system pressure will decrease as the break alone relieves the core energy.

Figure 2-6 shows the evolution of the system void fraction; values in excess of 90% are predicted very early (300 seconds) into the transient. For the larger breaks the system high void fractions occur early in time. For the smaller breaks it takes in the order of hours before the system evolves to high void fraction. Core cooling is maintained during a small break with continuous RC pump

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operation regardless of void fraction. In the long term, the system will depressurize and the enhanced performance of the ECCS (HPI and LPI) will result in reduced system void fraction.

Figure 2-7 illustrates this long term system behavior for a 0.10 ft^2 break. For this case, the LPIS are operative at approximately 2300 seconds, and a substantial decrease in system void fraction results. An arbitrary pump trip after approximately 2700 seconds would not result in core uncovery. The potential for core uncovery due to an RC pump trip is thus limited to a discrete time period during which the natural evolution of the system produces high void fractions and prior to LPI actuation. For a 0.1 ft^2 break, this time period is on the order of 2000 seconds. For smaller breaks, this critical time could be a few hours even if the operator initiated a controlled cooldown and system depressurization as recommended in the small break guidelines.

40 min

Must trip within 2-3 min

Although the analyses described above used 2 HPI pumps, the effect of only 1 HPI pump available on the system void fraction evolution while the RC pumps are operating is not significant. Figures 2-8 and 2-9 show the impact of one versus two HPI pumps on system pressure and average void fraction transients for a 0.05 ft^2 break with the RC pumps operative. As seen from these figures, the results with one HPI pump are not significantly different to the two HPI pump case and are bounded by the spectrum approach utilized. With one HPI pump, the system does depressurize more slowly (less steam condensation) and a higher short term equilibrium void fraction is achieved. Also, recovery of the core following a loss of the RC pumps would be significantly longer with only 1 HPI pump available.

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The majority of the analyses provided in this report uses two HPI pumps and demonstrates a core cooling problem with worst time pump trip given that assumption. As analysis of one HPI available cases would only show a larger problem, such cases have not been extensively considered. As demonstrated in section 3.4, the resolution of this problem, forced early pump trip, provides assurance of core cooling for both one or two HPIs available cases. Therefore,

there is no need for further pursuit of the single HPI available case.

The effect of the RCP tripping during the transient was studied by assuming that the pumps are lost when the system reaches 90% void fraction. Loss of the RC pumps at this void fraction is expected to produce essentially the highest peak cladding temperature. After the RC pumps are tripped, the fluid in the RCS separates and liquid falls to the lowest regions, i.e., the lower plenum of the RV and the pump suction piping. At 90% void fraction, the core will be totally uncovered following the RC pump trip. Thus, the time required to recover the core is longer than that for RC pump trips initiated at lower system void fractions. System void fractions in excess of 90% can possibly result in slightly higher temperatures due to the longer core refill times that may occur. However, the peak cladding temperature results are not expected to be significantly different as the system pressure and core decay heat, at the time that a higher void fraction is reached, will be lower.

Table 2-2 shows the core uncovery time for the cases analyzed with the RC pumps tripping at 90% void fraction with 2 HPI pumps available for core recovery. As shown, the core will be uncovered for approximately 600 seconds for the breaks analyzed. Figures 2-10 and 2-11 show the system pressure and void fraction response for the 0.075 ft^2 break with a RC pump trip at 90% void fraction. As seen in these figures, the system depressurizes faster after the RC pump trip, due to the change in leak quality, and the void fraction decreases indicating that the core is being refilled. Figure 2-12 shows the core liquid level response following the RC pump trip. The core is refilled to the 9 foot level with collapsed liquid approximately 625 seconds after the assumed pump trip. Once the core liquid level reaches the 9 foot elevation, the core is expected to be covered by a two-phase mixture and the cladding temperature excursion would be terminated.

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5. Effect of 1.0 ANS versus 1.2 ANS Decay Curve

An analysis was performed using the more realistic 1.0 ANS decay curve instead of 1.2 ANS decay curve. The study was done for a 0.05 ft^2 break with 2 HPI;s available and pumps tripped at 90% system void fraction. Figures 2-13 and 2-14 show a comparison of system pressure and average system void fraction for 1.0 and 1.2 ANS decay curves. As seen in Figure 2-13, the system pressure for 1.0 ANS case begins to drop from saturation pressure (~1100 psia) about 200 seconds earlier than the case with 1.2 ANS as a result of reduced decay heat. Also, the system will evolve to a lower average void fraction as shown in Figure 2-14. After the pumps trip at 90% system void fraction, the case with 1.0 ANS decay curve has a shorter core uncovery time by approximately 200 seconds compared to 1.2 ANS case. This case demonstrates that the effect of a delayed RC pump trip may be acceptable when viewed realistically. A peak cladding temperature assessment for this case will be provided in a supplementary response planned for September 15th, to the I&E Bulletin 7905-C.

6. Effect of No Auxiliary Feedwater

Analyses have also been performed with the RC pumps available and no auxiliary feedwater. These analyses all assumed 2 HPI pumps were available. The system void fraction evolutions for these calculations were not significantly different from those discussed with auxiliary feedwater. Thus the conclusions of the cases with auxiliary feedwater apply.

.7 Break Location Sensitivity Study

A study was conducted to demonstrate that the break location utilized for the preceeding analyses is indeed the worst break location. As stated previously, the analyses were performed assuming that the break was located in the bottom of the pump discharge piping. A 0.075 ft^2 hot leg break was analyzed to provide a direct comparison to a similar case in the cold leg. For this evaluation, the RC pumps were assumed to trip after the RCS void fraction reaches 90%. Figure 2.15 shows the average system void fraction transient and the core uncovery times for both the 0.075 ft^2 hot and cold leg breaks. As shown, the cold leg break reaches 90% void fraction approximately 150 seconds earlier than the hot leg break. Also, the cold leg break yields a core uncovery time of 175 seconds longer than the hot leg break. The quicker core recovery time for the hot leg break is caused by the greater penetration of the HPI fluid for this break. For a cold leg break in the pump discharge piping, a portion of the HPI fluid is lost directly out the break and is not available for core refill. For a hot leg break, the full HPI flow is available for core refill. Thus, as shown by direct comparison and for the reasons given above, hot leg breaks are less severe than breaks in the pump discharge piping.

8 Peak Cladding Temperature Assessment

As described previously, a RC pump trip, at the time the RCS void fraction is 90%, will result in core uncovery times of approximately 600 seconds. The peak cladding temperatures for these cases were evaluated using the small break evaluation model core power shape used to demonstrate compliance with Appendix K and 10CFR50.46. Also, an adiabatic heatup assumption during the time of core uncovery was utilized. This approach is extremely conservative in that the power shape and

local power rate (kw/ft) analyzed is not expected to occur during normal plant operation. Furthermore, use of an adiabatic heatup assumption neglects any credit for the steam cooling that will occur during the core refill phase and also neglects the effect of any radiation heat transfer. Using a decay heat power level based on 1.2 ANS at 1500 seconds, the cladding will heatup at a rate will be 6.5 F/S under the adiabatic assumption. With a core uncovery period of 600 seconds and the adiabatic heatup assumption, cladding temperatures will exceed the criteria of 10CFR50.46. Use of a more realistic heat transfer approach with the extreme power shape utilized for this evaluation is also expected to result in cladding temperature in excess of the criteria. In order to ensure compliance of the 177 FA lowered loop plants to the criteria of 10CFR50.46 a prompt tripp~~ing of~~ the RC pumps is required. Section B. demonstrates that a prompt trip of the RC pumps upon receipt of a low pressure ESFAS signal will result in compliance to the criteria.

An evaluation of the peak cladding temperature using a power shape encountered during normal operation for a realistic transient response with delayed RC pump trip will be provided by September 15, 1979.

C. Analysis Applicability to Davis-Besse I

The significant parametric differences between the raised-loop Davis-Besse I plant and the preceeding 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 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 an RC pump trip is higher for the raised loop design than for the lowered loop design.

Figure 2-16 shows a comparison of the delivered HPI flow for the Davis-Besse I plant and the lowered loop plants. As shown, for a similar number of HPI pumps available, the Davis-Besse I pumps will deliver more flow. For the delayed pump trip cases presented in section B.4 of this report, the Davis-Besse I plant will take approximately 450 seconds to recover the core as opposed to ≈600 seconds for the lowered-loop plants. However, it is noted that the core recovery time is based on using two HPI's rather than one, as required by Appendix K. Use of only one HPI pump for Davis-Besse I will result in core uncovery times in excess of 600 seconds. The Davis-Besse I plant cannot be shown to be in compliance with 10CFR50.46 for a delayed RC pump trip.

Prompt reactor coolant pump trip is, therefore, necessary to ensure compliance of the Davis-Besse I plant with 10CFR50.46.

D. Effect of Prompt RC Pump Trip on Low Pressure ESFAS Signal

As demonstrated by the previous sections, the ECC system can not be demonstrated to comply with 10CFR50.46 using present evaluation techniques and Appendix K assumptions under the assumption of a delayed RC pump trip. Thus, prompt tripping of the RC pumps is necessary to ensure conformance. Operating guidelines for both LOCA and non-LOCA events have been developed which require prompt tripping of the RC pumps upon receipt of a low pressure ESFAS signal. Because no diagnosis of the event is required by the operator and ESFAS initiation is alarmed in the control room, prompt tripping of the RC pumps can be assumed.

The effect of a prompt reactor coolant pump trip on an ESFAS signal has been examined to ensure that the consequences of a small LOCA are bounded by previous small break analyses[2] which assume RC pump trip on reactor trip. As shown by Table 2-3 at the time of low pressure ESFAS initiation, keeping the RC pumps running results in a lower average system void fraction. This occurs because the availability of the RC pumps results in lower hot leg temperatures and thus less flashing in the RCS at a given pressure. Thus, a prompt trip upon receipt of an ESFAS signal will result in a less severe system void fraction evolution than cases previously analyzed assuming RC pump on reactor trip.

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E. Conclusions

The results of the analyzes described in this section can be summarized as follows:

1) If the RC pumps remain operative, core cooling is assured regardless of system void fraction.

2) For breaks greater than 0.025 ft^2, the RCS may evolve to system void fractions in excess of 90%.

3) 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.

4) 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.

5) Even with 2 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 10CFR50.46 if Appendix K assumptions are utilized.

6) A prompt RC pump trip upon receipt of a low pressure ESFAS signal will provide compliance to 10CFR50.46.

7) The above conclusions are applicable to both the B&W 177 FA lowered and raised loop NSS designs.

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 not precluded 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 unacceptable.
2. In the event of a large steam line break (maximum overcooling), th 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 EOI conditions and whether or not a return to power is experienced following the RC pump trip. If a return to critical is experienc natural circulation flow may not be sufficient to remove heat and to avoid core damage.

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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 combinatic 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 th four RC pumps running (one loop), and once assuming RC pump trip right after HPI initiation. The analysis shows that th system behaves very similarly with and without pumps. In both cases, the pressurizer refills in about 14 to 16 minute 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 tapping. 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-

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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 very similarly with and without pumps, although maintaining RC pump flow does seem to help mitigate void formation. Th pump flow case shows a shorter time to the start of pressurizer refill than the natural circulation case (Figure 3. although the time difference does not seem to be very large

3. <u>Effect of Return to Power</u>

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 w: mitigate the consequences of an EOL return to power condit: by reducing the cooldown of the primary system. The reduce cooldown substantially increases the subcritical margin which, in turn, reduces or eliminates return to power.

D. <u>Conclusions and Summary</u>

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

1. It was found that a pump trip does not significantly short the time to filling of the pressurizer and approximately t! same time interval for operator action exists.
2. For the maximum overcooling case analyzed, the RC pump tri increased the amount of two-phase in the primary loop; however, the percent void formation is still too small to affect the ability to cool on natural circulation.
3. The subcritical return-to-power condition is alleviated by the RC pump trip case due to the reduced overcooling effec

Based upon the above assessment and analysis, it is concluded that the consequences of Chapter 15 non-LOCA events are

POOR ORIGINAL

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.

Table 2-1. Analysis Scope With AFW Available

Break size, (ft^2)	Break location		Continuous RC pump operation		RC pump trip @ 90% void
	Cold leg	Hot leg	2 HPI	1 HPI	2 HPI
0.025	X		X		
0.05	X		X*	X	X*
0.075	X	X	X		X
0.10	X		X		X
0.20	X		X		

- - - - -

*Analyzed with both 1.0 and 1.2 ANS decay curves.

Table 2-2. Impact Assessment of Break Spectrum With RC Pump Trip at 90% Void

Break size (ft^2)	Core uncovery time (sec)
0.10	550
0.075	625
0.05	575

- - - - -

Notes: 1. Two HPIs available during the transient.

2. Core uncovery time is the time period following pump trip required to fill the inner RV with water to an elevation of 9. ft in the core which is approximately 12.ft when swelled.

Table 2-3. Comparison of System Void Fractions at ESFAS Signal

Break size, (ft^2)	System void fraction at ESFAS	
	Pumps on	Pumps tripped
0.02463	0.0	
0.04		4.47
0.05	0.04	
0.055		6.74
0.07		8.06
0.075	0.90	
0.085		8.45
0.10	2.17	7.97
0.15		10.70
0.20	6.78	

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Takes 1700 ft^3 to fill to 9 ft elev.

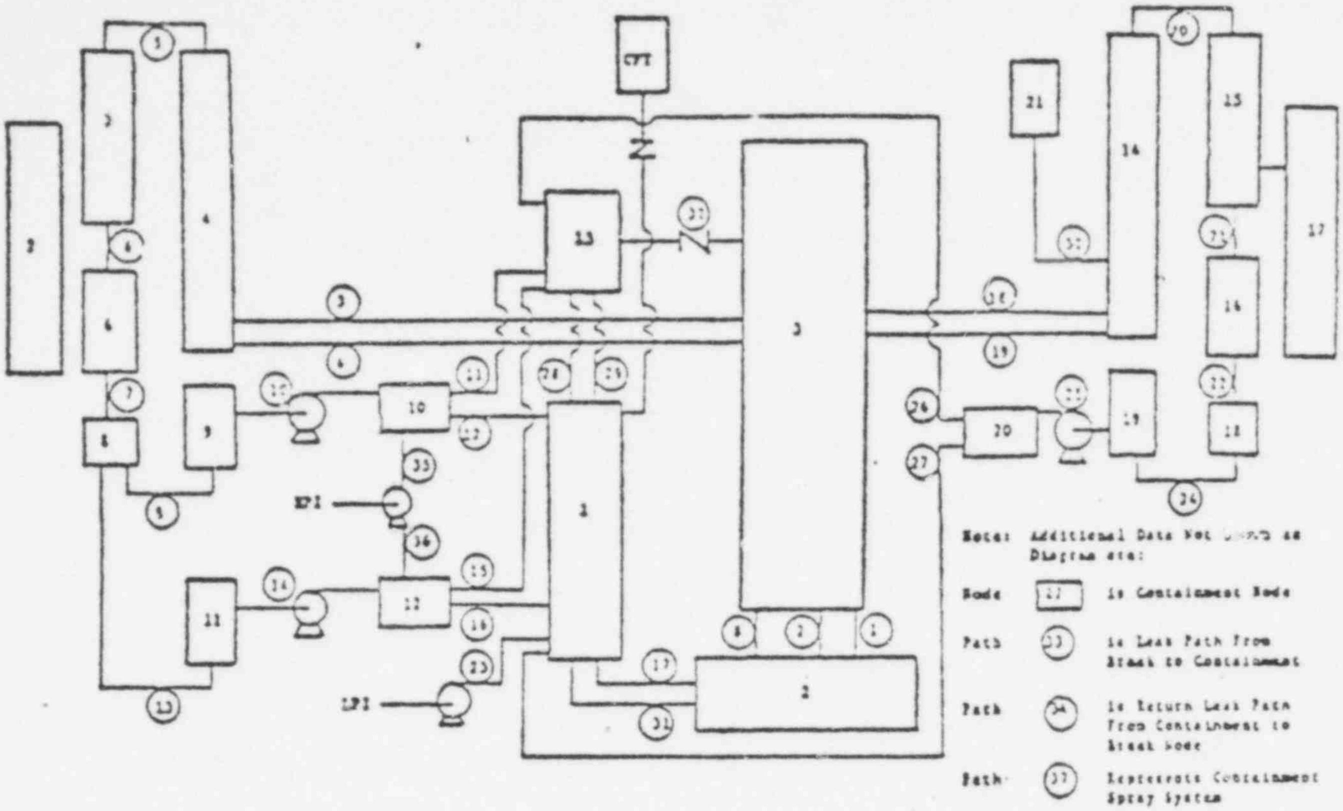
MINITRAP2 NODE DESCRIPTION

NODE NUMBER	DESCRIPTION
1,33	Reactor Vessel, Lower Plenum
2,34	Reactor Vessel, Core
3,35	Reactor Vessel, Upper Plenum
4,10	Hot Leg Piping
5-7,11-13	Primary, Steam Generator
8,14	Cold Leg Piping
9,32	Reactor Vessel Downcomer
15	Pressurizer
16,24	Steam Generator Downcomer
17,25	Steam Generator Lower Plenum
18-20,26-28	Secondary, Steam Generator
21,29	Steam Risers
22,30	Main Steam Piping
23	Turbine
31	Containment

MINITRAP2 PATH DESCRIPTION

PATH NUMBER	DESCRIPTION
1,2	Core
45,46	Core Bypass
3,5,5,11,12,44	Hot Leg Piping
6,7,13,14	Primary, steam Generator
8,15	RC Pumps
9,16	Cold Leg Piping
10,43	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
24,32	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

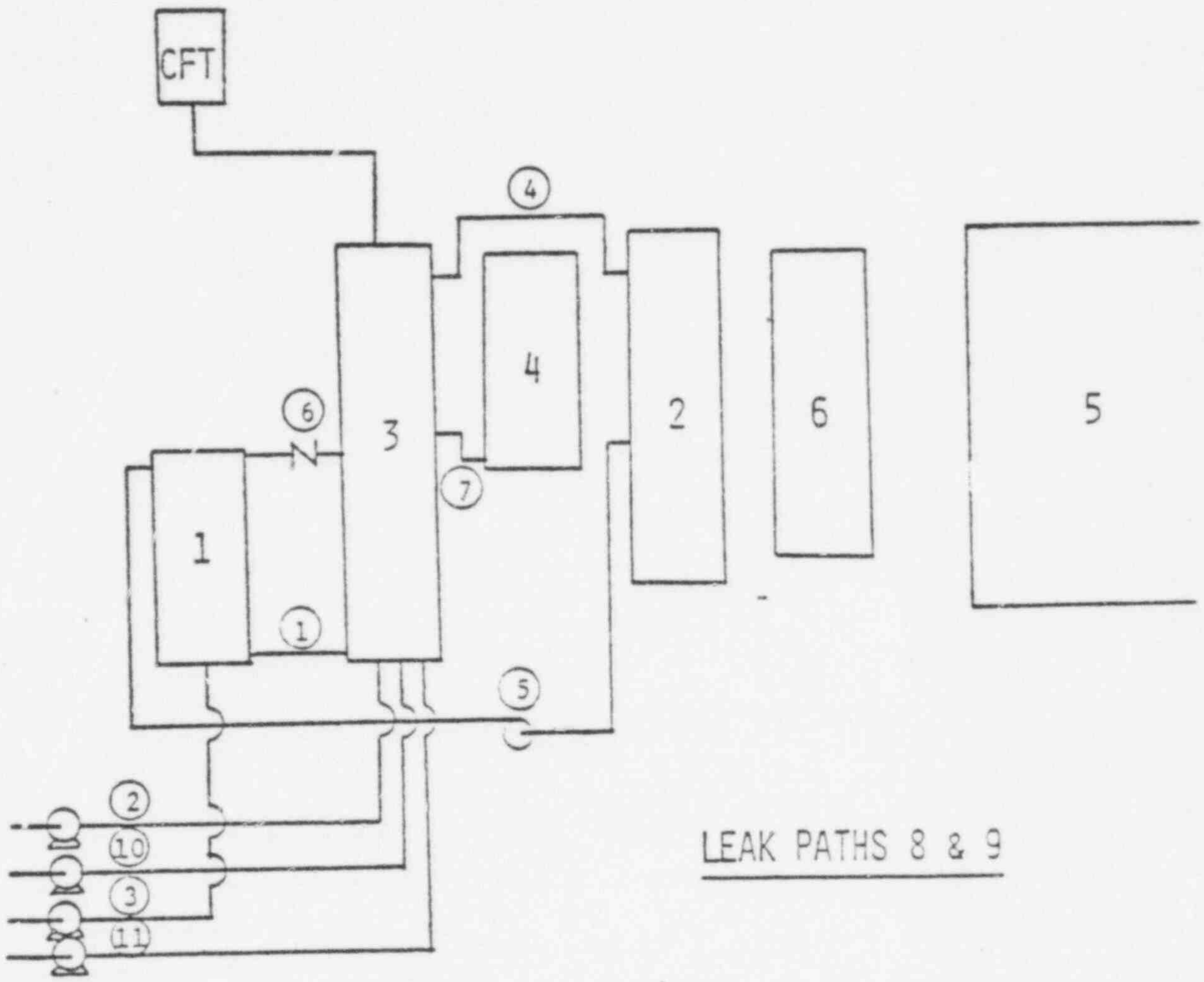
Table 3.1



Node No.	Identification
1	Downcomer
2	Lower Plenum
3	Core, Core Bypass, Upper Plenum, Upper Head
4,14	Hot Leg Piping
5,15	Steam Generator Upper Head, SG Tubes (Upper Half)
6,16	SG Tubes (Lower Half)
8,18	SG Lower Head
9,11,19	Cold Leg Piping (Pump Suction)
10,12,20	Cold Leg Piping (Pump Discharge)
13	Upper Downcomer (Above the ℄ of Nozzle Belt)
21	Pressurizer
22	Containment

Path No.	Identification
1,2	Core
3,4,18,19	Hot Leg Piping
5,20	Hot Leg, Upper
6,21	SG Tubes
7,22	SG Lower Head
8	Core Bypass
9,13,24	Cold Leg Piping
10,14,25	Pumps
11,12,15,16,26,27	Cold Leg Piping
17,31	Downcomer
23	LPI
28,29	Upper Downcomer
30	Pressurizer
32	Vent Valve
33,34	Leak & Return Path
35,36	HPI
37	Containment Sprays

Figure 2-2. CRAFT2 NODING DIAGRAM FOR SMALL BREAKS
(6 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	Pressurizer
8,9	Leak & Return Path

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946084

CORE PRESSURE VS TIME, 177-LL, 2772 MWt, PUMPS ON

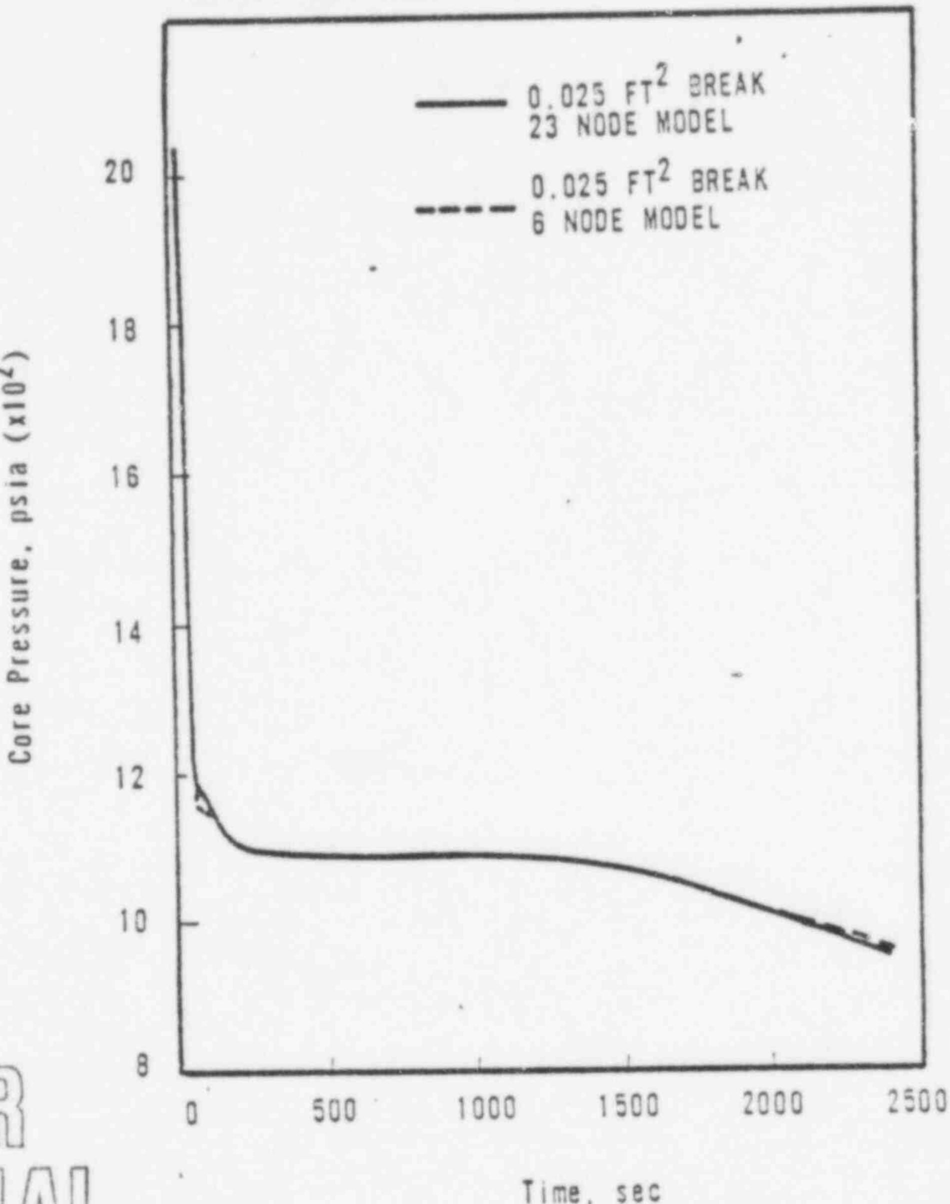

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Figure 2-3

946085

PERCENT SYSTEM VOIDS VS TIME, PUMPS ON

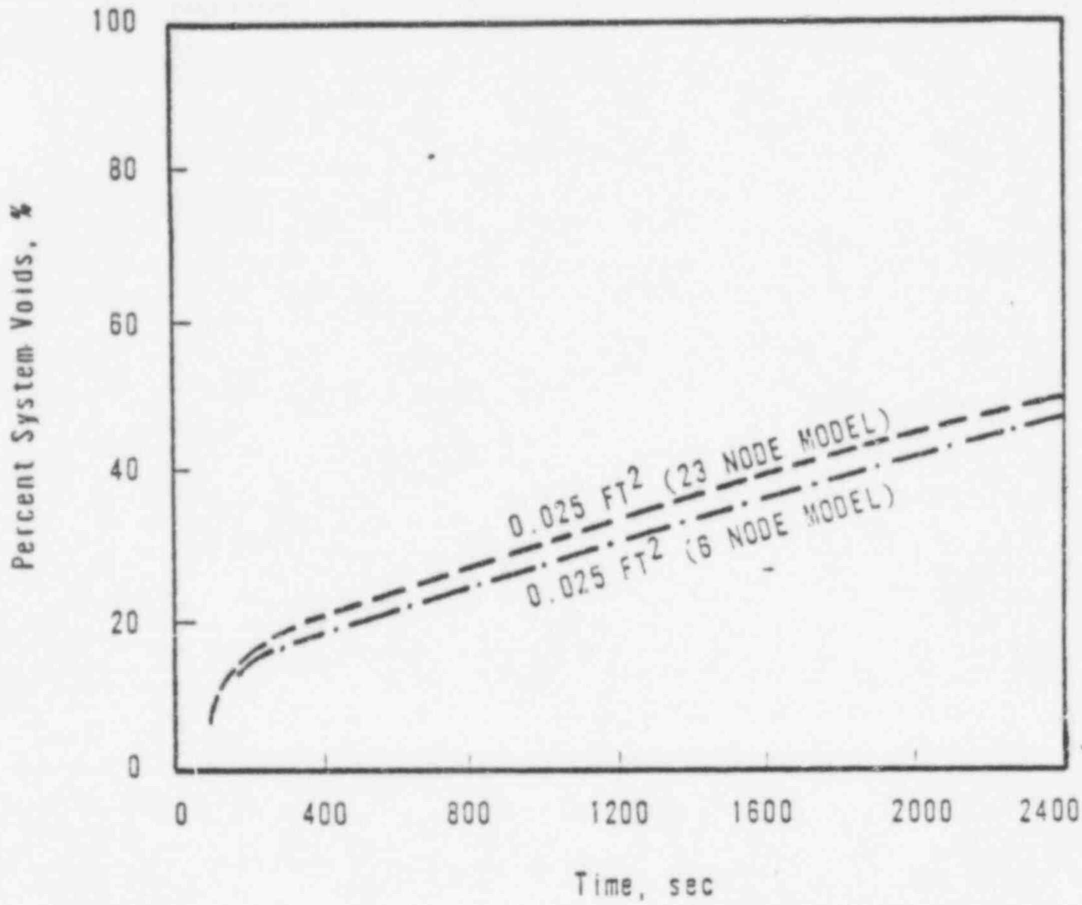

Figure 2-4

BREAK SPECTRUM-RC PRESSURE WITH
THE RC PUMPS OPERATIVE AND 2 HPI PUMPS

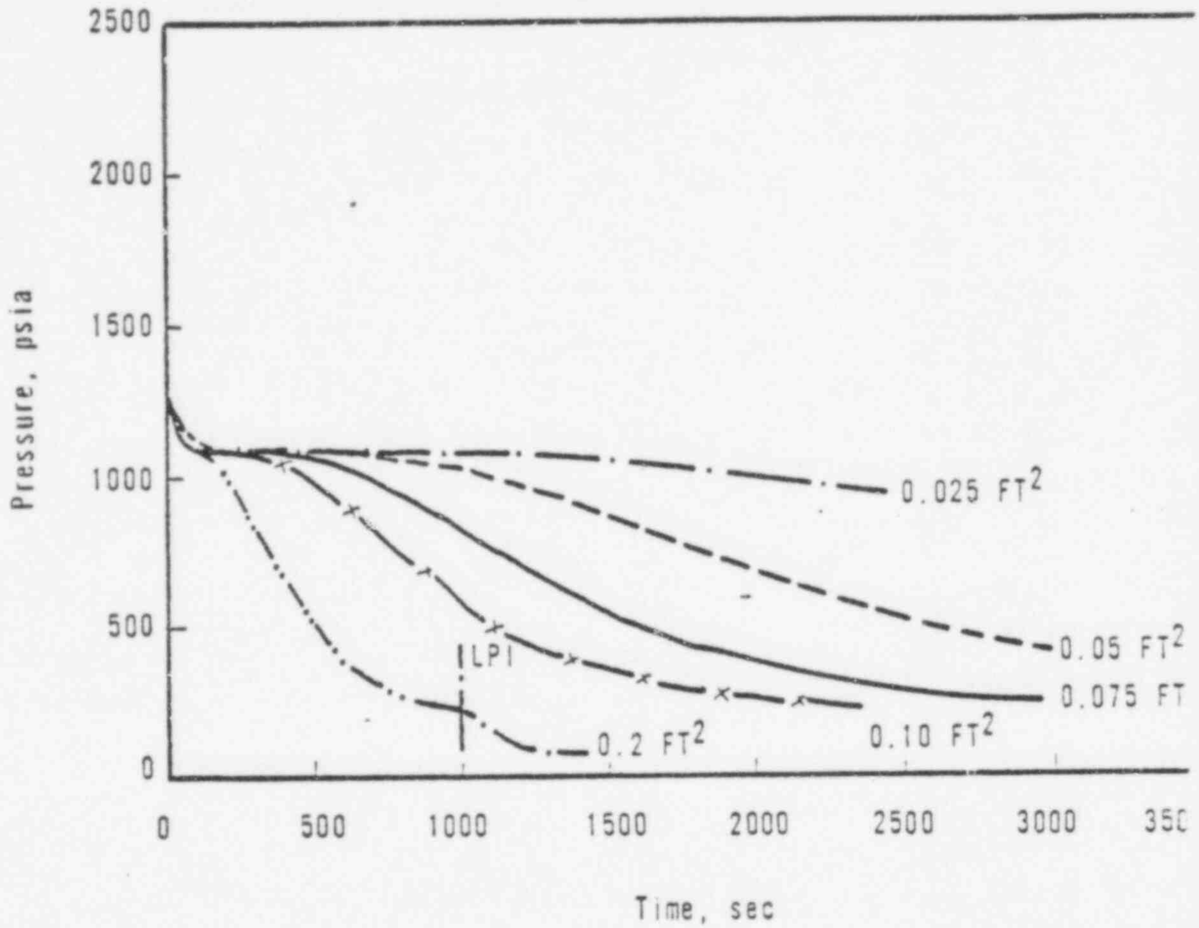

Figure 2-5

POOR ORIGINAL

946087

BREAK SPECTRUM-AVERAGE SYSTEM VOID FRACTION
WITH THE RC PUMPS OPERATIVE AND 2 HPI PUMPS

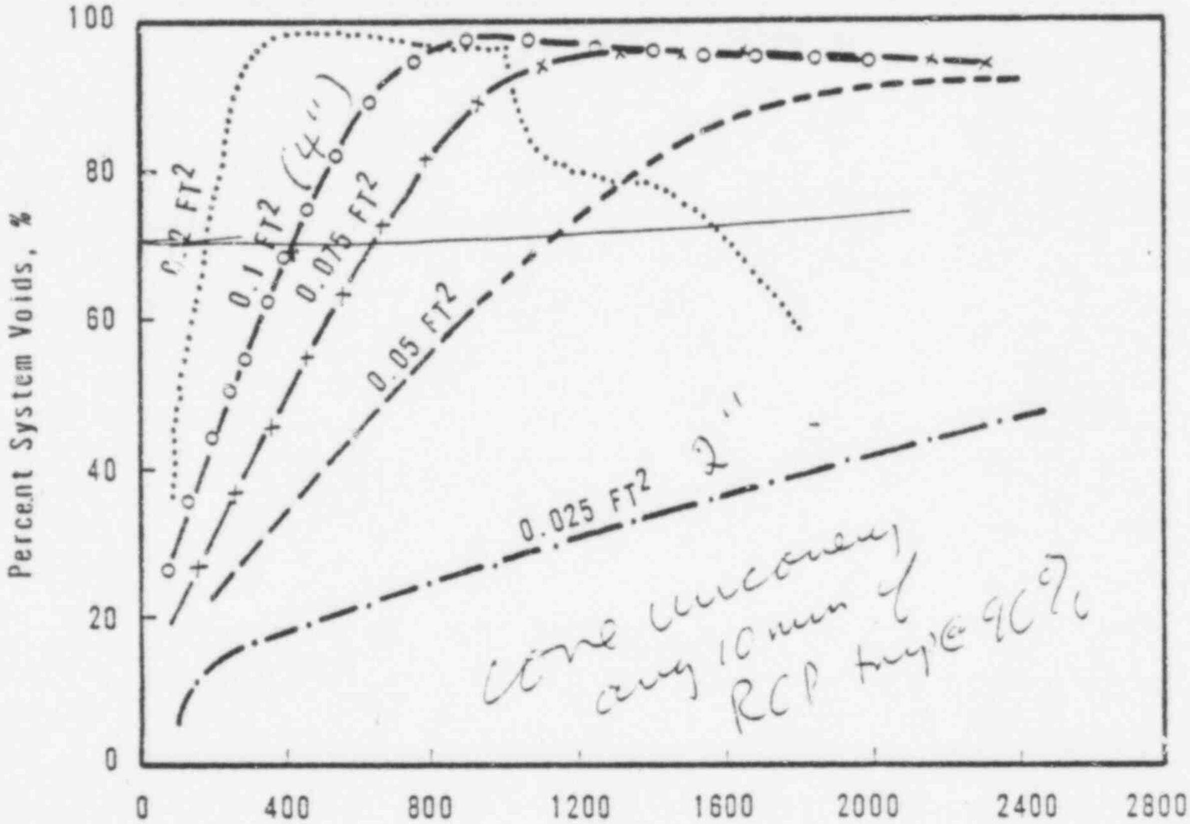

Figure 2-6

0.1 FT^2 BREAK WITH CONTINUOUS RC PUMP
OPERATION AND 2 HPI PUMPS

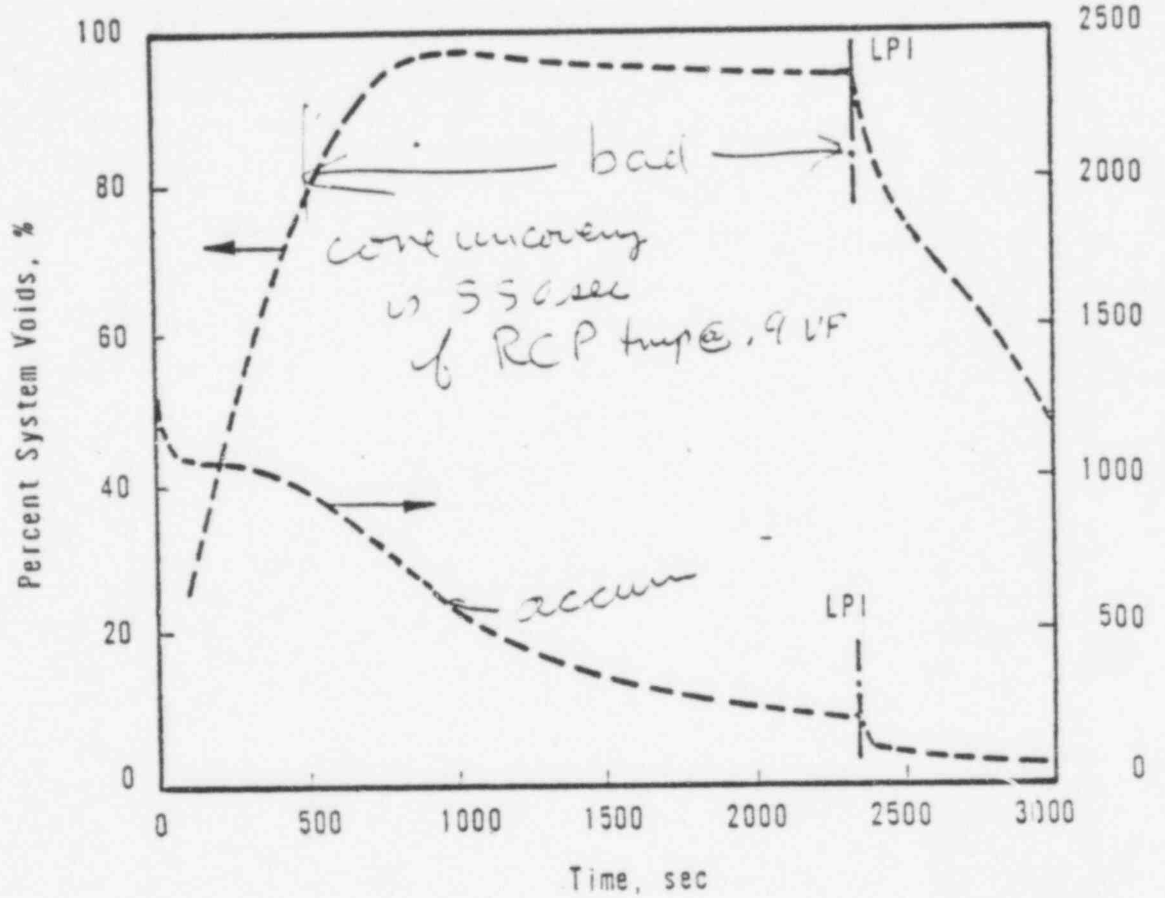

Figure 2-7

946089

RC PRESSURE FOR 0.05 FT^2 BREAK
AVAILABLE 1 HPI VS 2 HPI'S

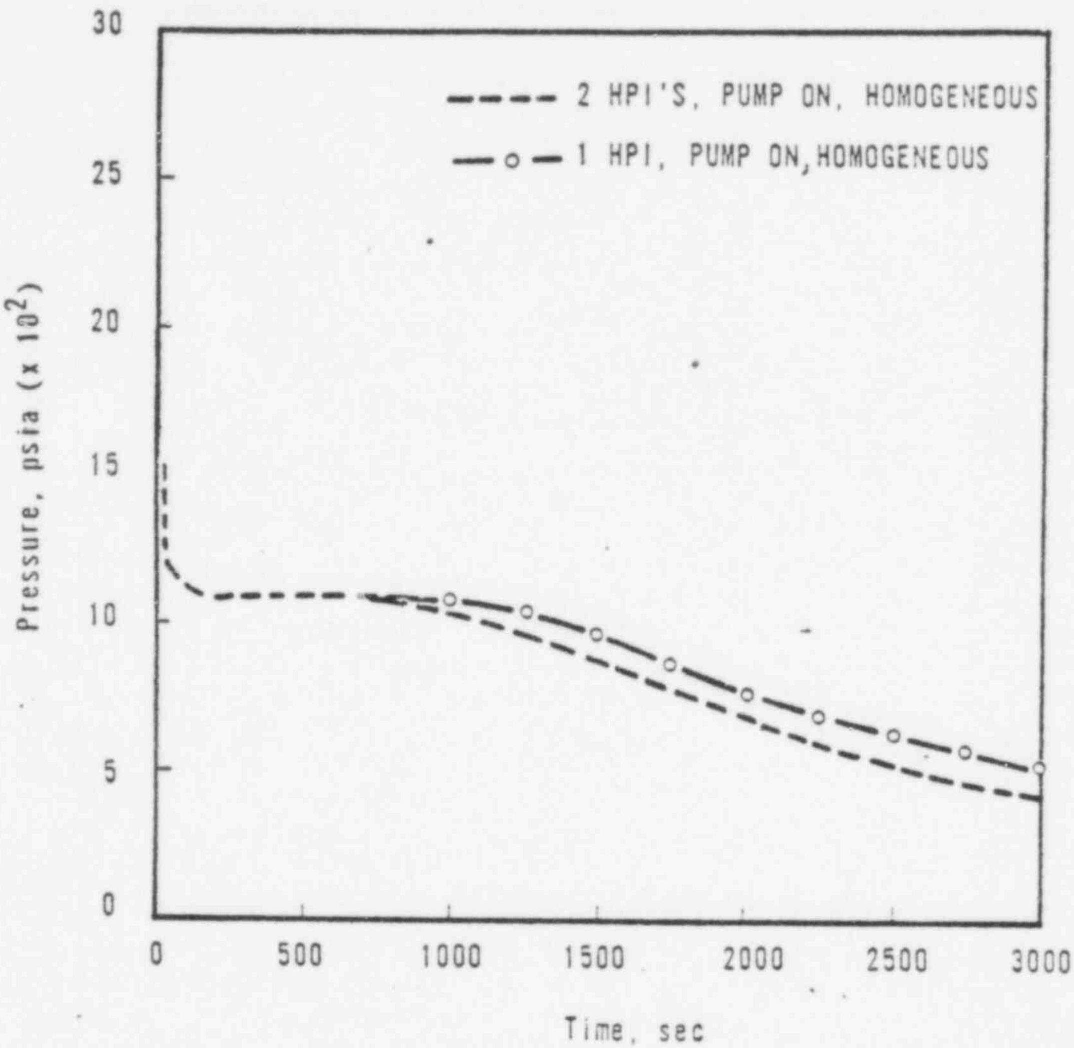

Figure 2-8

POOR ORIGINAL

946090

AVERAGE SYSTEM VOID FRACTION FOR 0.05 FT^2
AVAILABLE 1 HPI VS 2 HPI'S

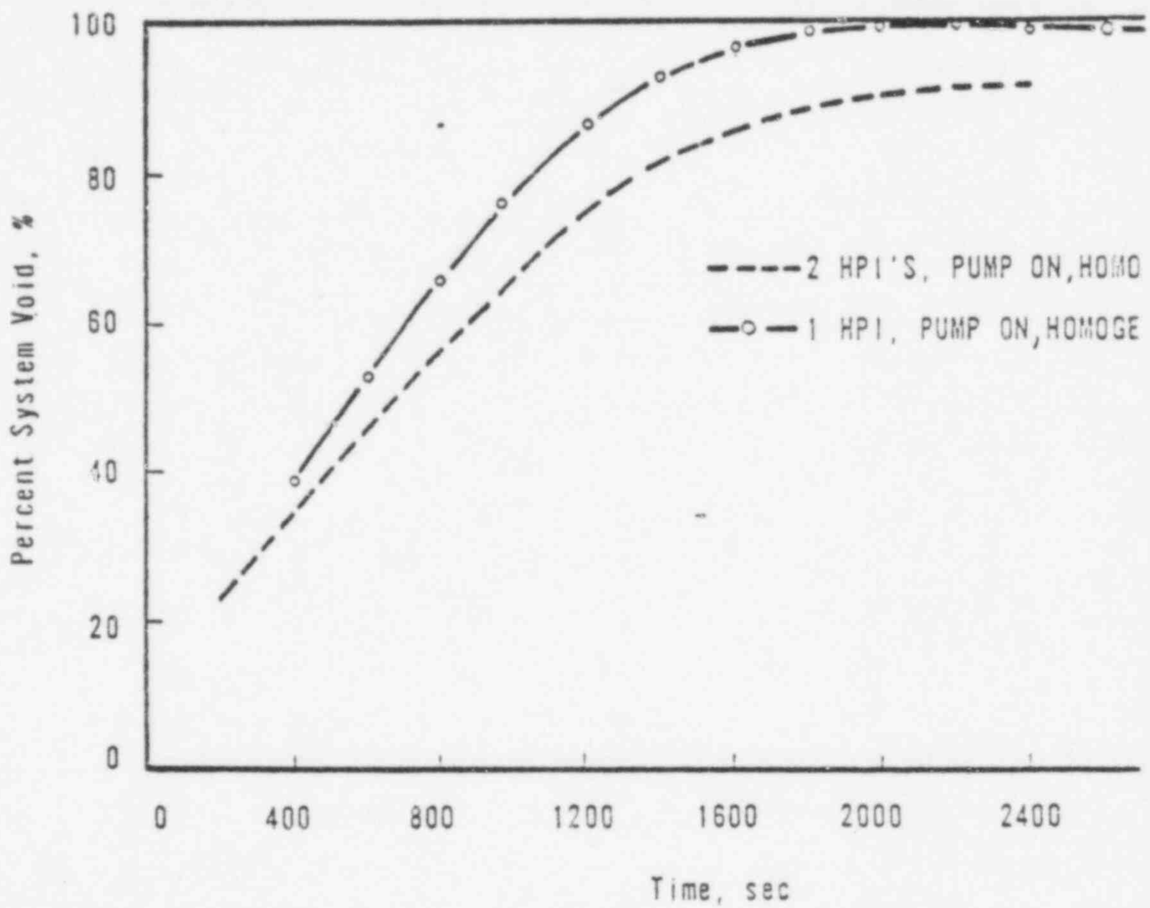

Figure 2-9

946091

RC PRESSURE FOR 0.075 FT^2, PUMPS OFF @ 90% SYSTEM VOID

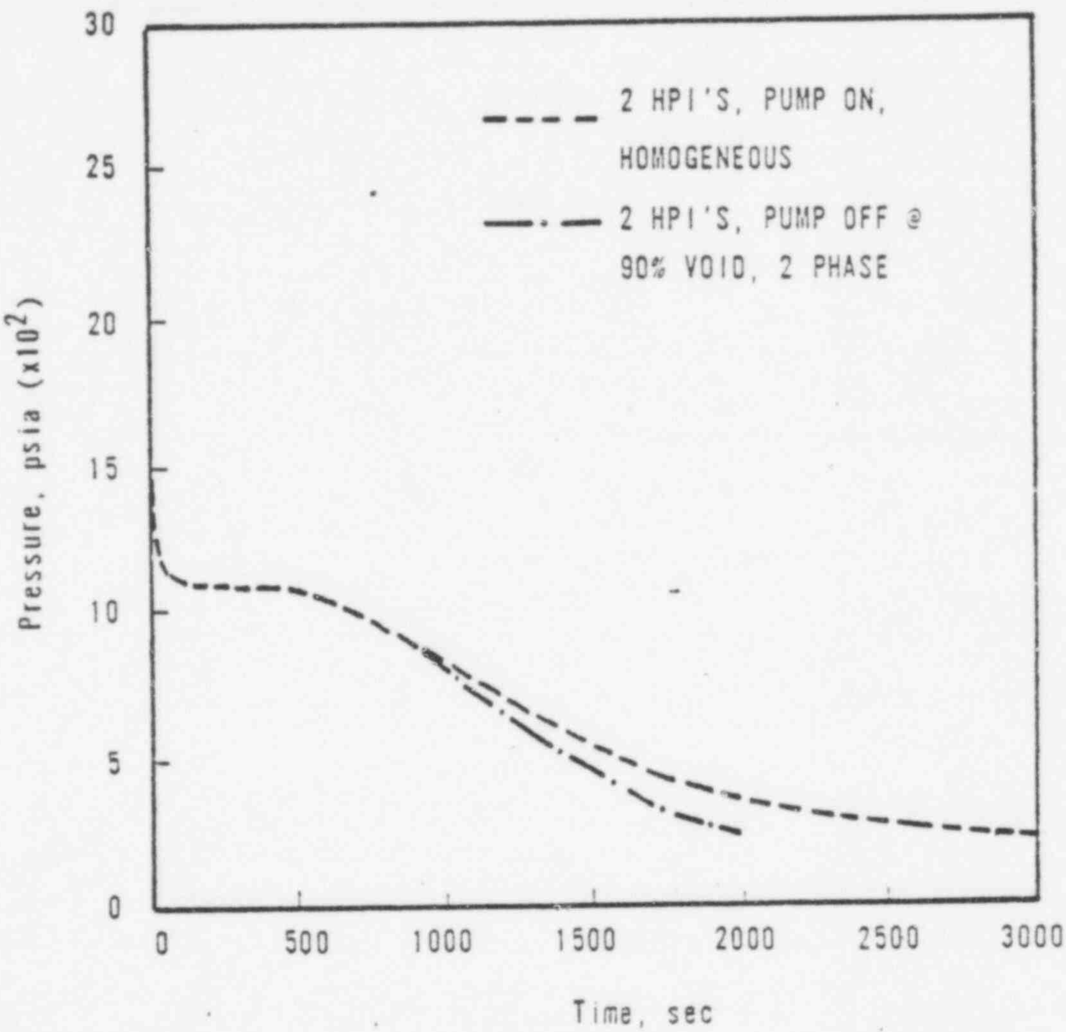

Figure 2-10

POOR ORIGINAL

346092

AVERAGE SYSTEM VOID FRACTION FOR
0.075 FT^2, PUMPS OFF @ 90% SYSTEM VOID

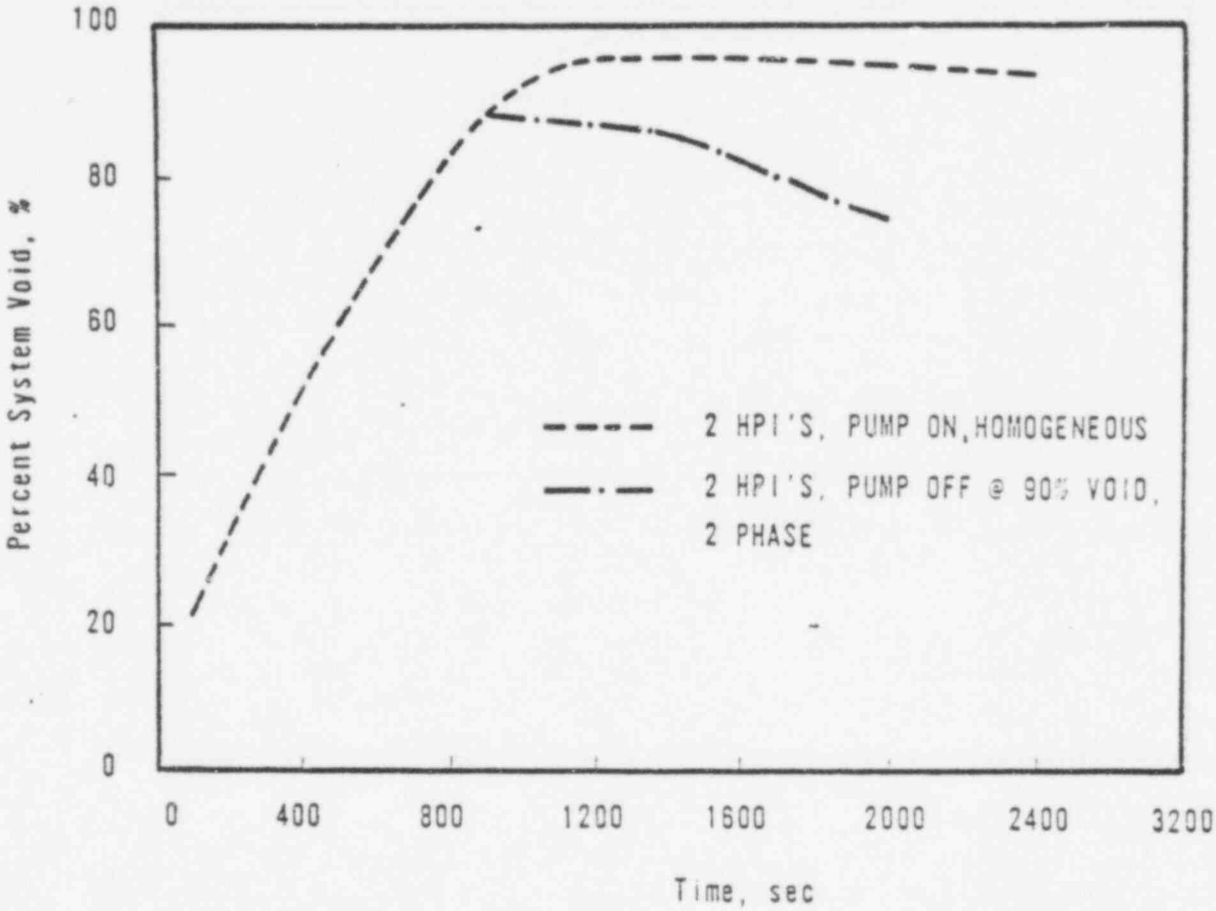

Figure 2-11

POOR
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946093

AVAILABLE LIQUID VOLUME VS TIME
FOR 0.075 FT^2 BREAK WITH 1.2 ANS
DECAY HEAT CURVE

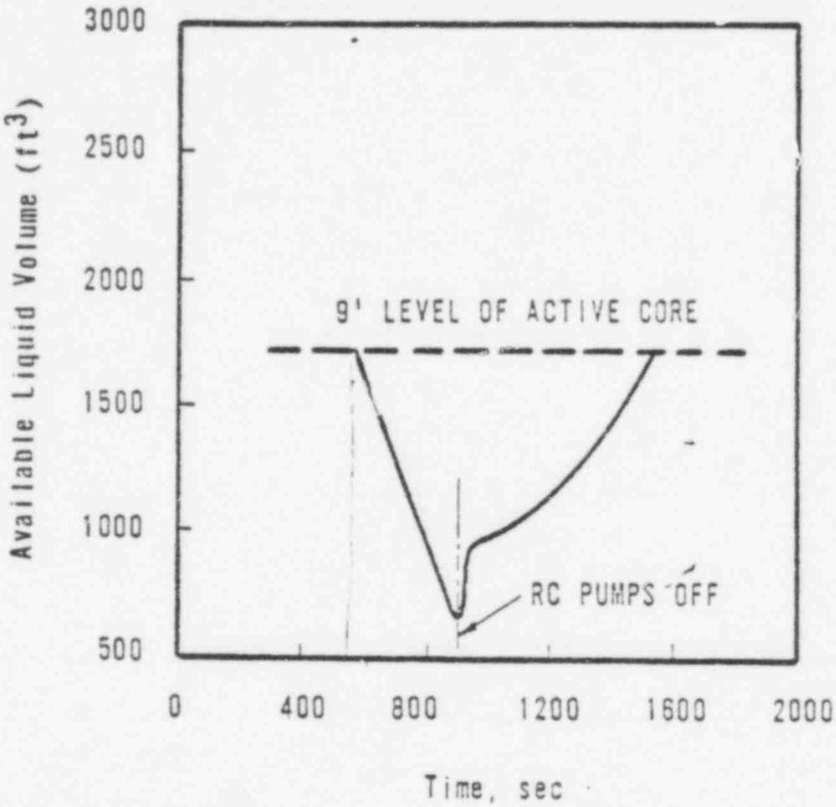

Figure 2-12

RC PRESSURE VS TIME FOR 0.05 FT^2
BREAK WITH 1.0 AND 1.2 ANS BEFORE
AND AFTER PUMP TRIP

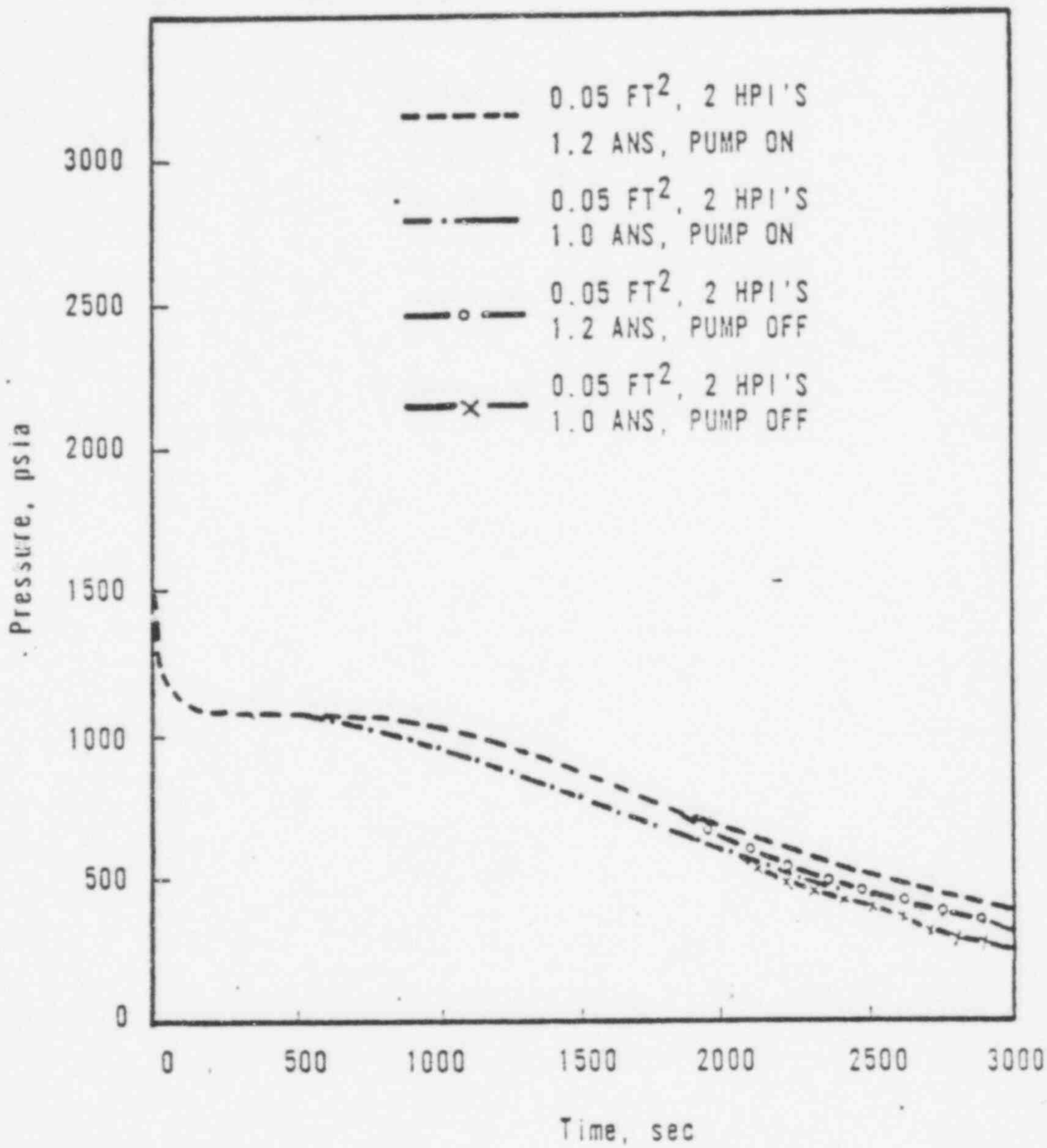

Figure 2-13

PERCENT SYSTEM VOID FRACTION FOR 0.05 FT^2
BREAK WITH 1.0 AND 1.2 ANS BEFORE AND AFTER PUMP TRIP

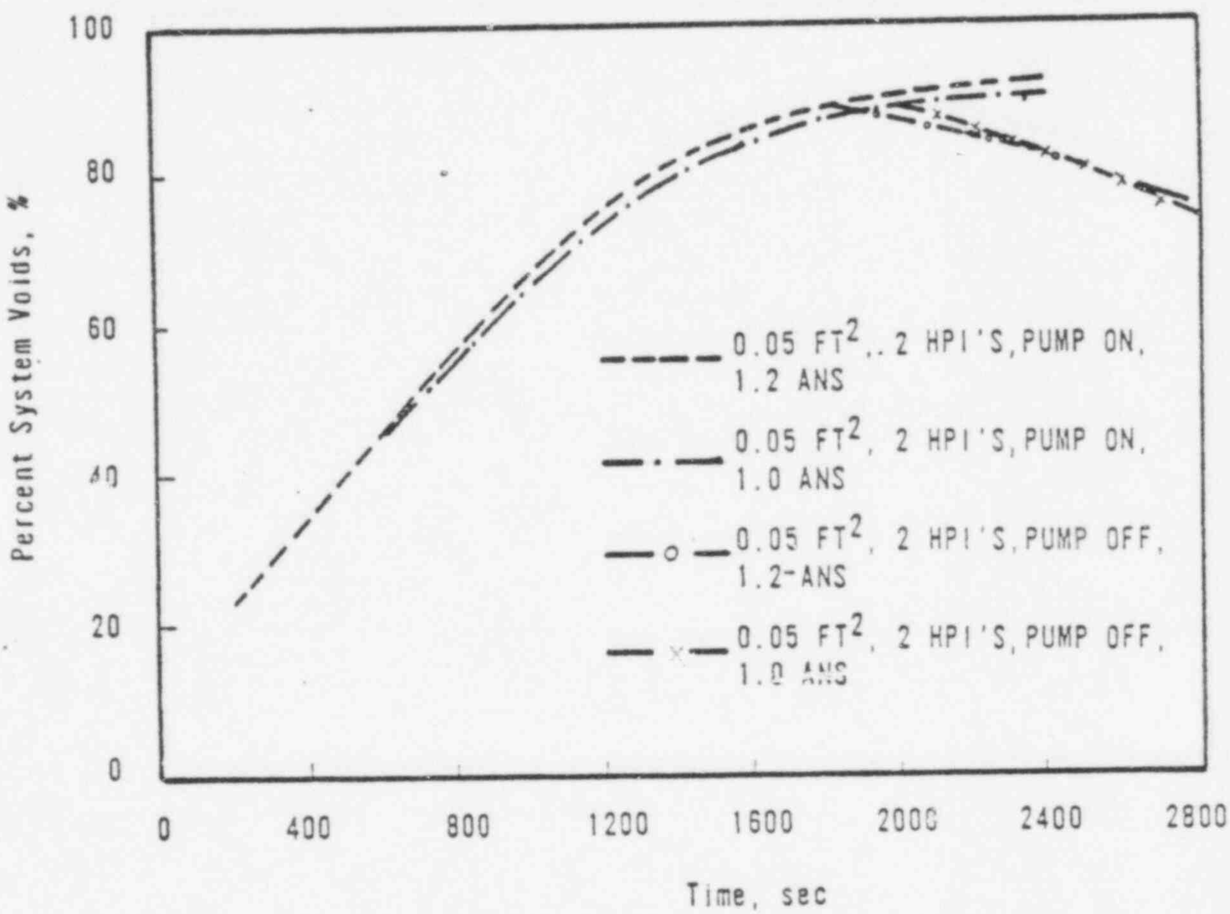

Figure 2-14

POOR ORIGINAL

346096

AVERAGE SYSTEM VOID FRACTION VS TIME FOR A 0.075 FT^2 BREAK, BREAK LOCATION COMPARISON PUMPS OFF @ 90% VOID

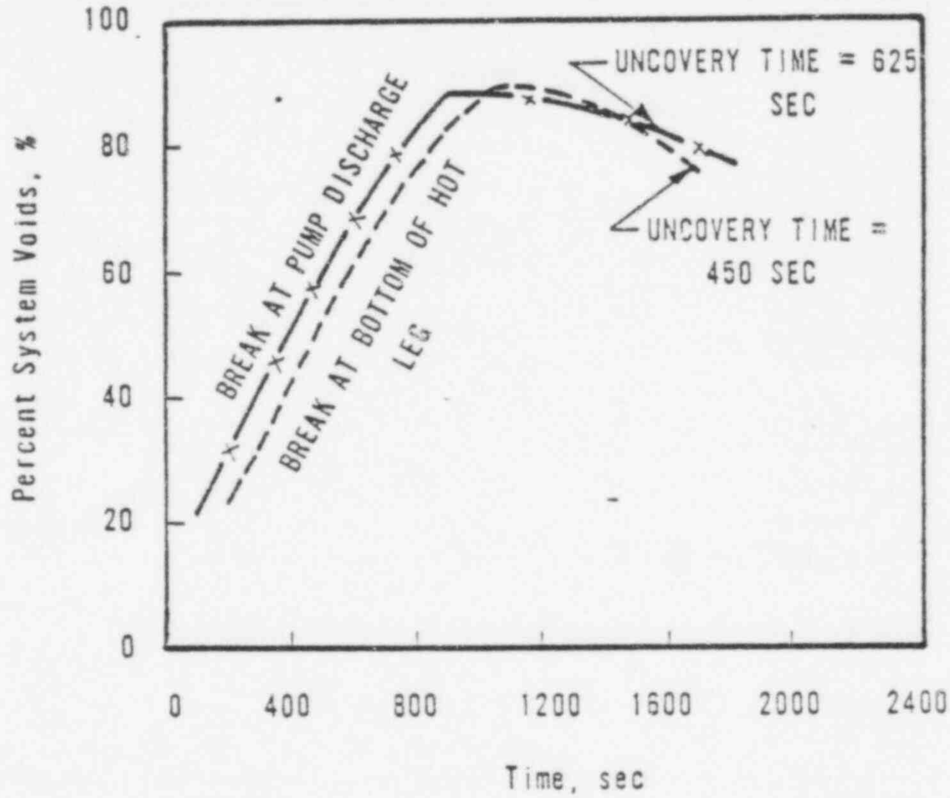

Figure 2-15

COMPARISON OF DELIVERED HIGH PRESSURE
INJECTION FLUID TO RV FOR PUMP DISCHARGE BREAK

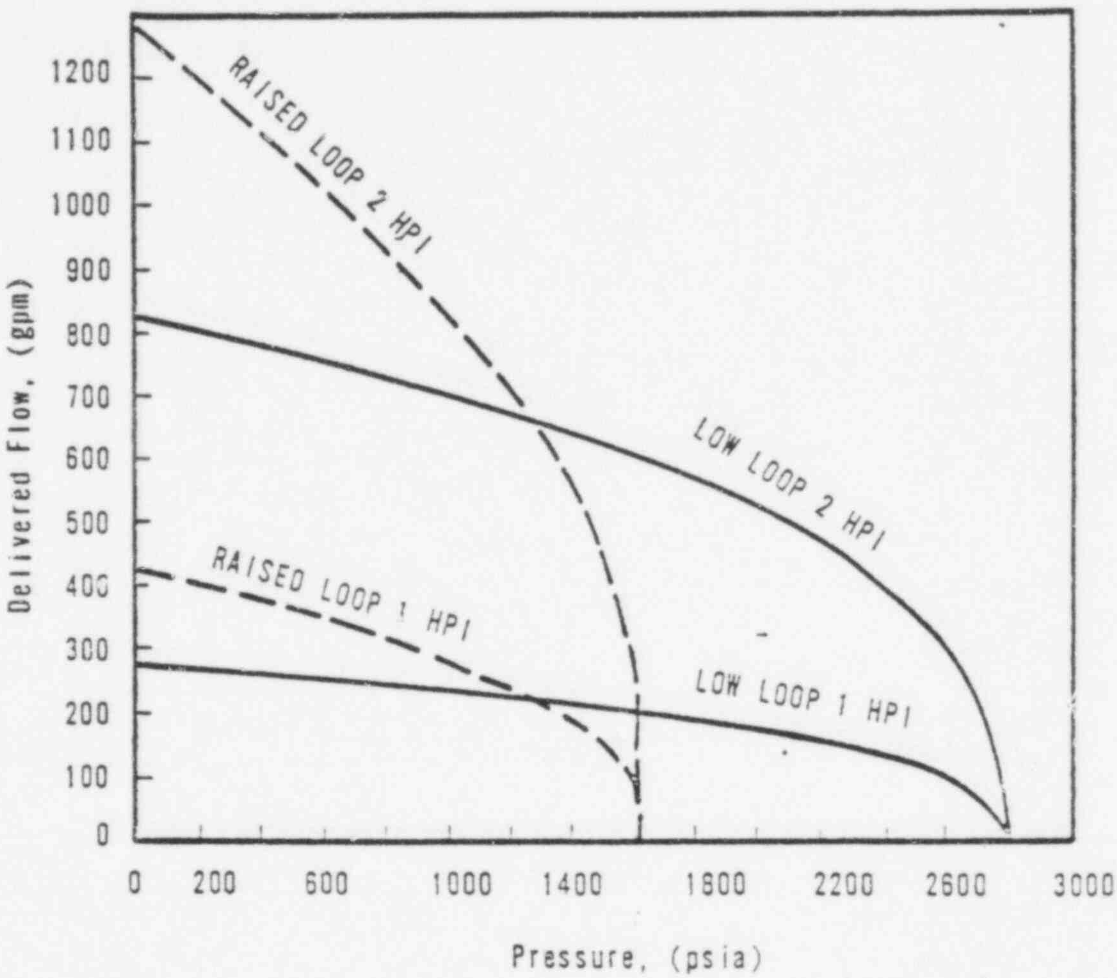

Figure 2-16

POOR
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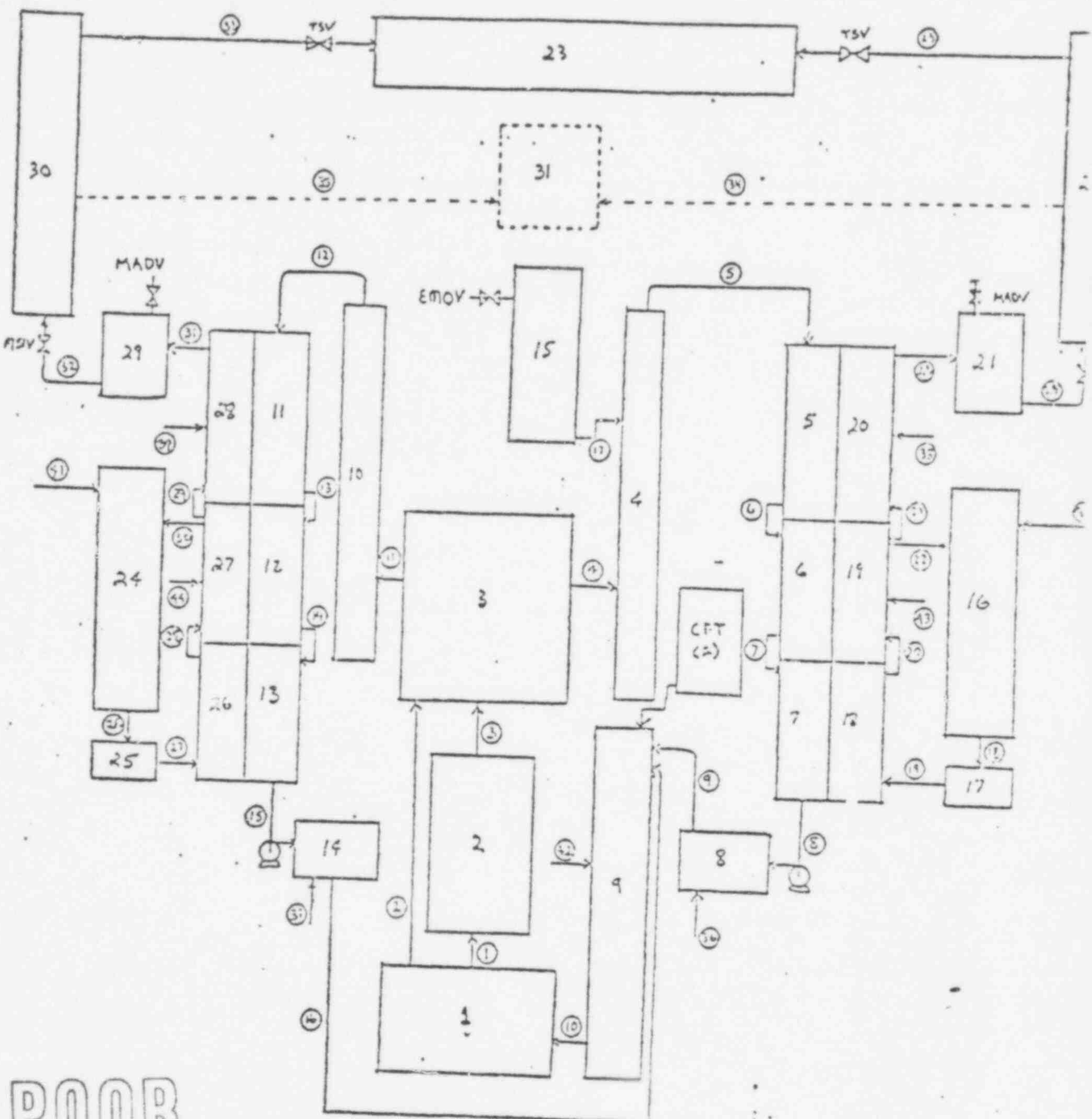
946098



POOR ORIGINAL

Figure
3.1
MINITRAP2 Noding and
Flow Path Scheme

946099

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)

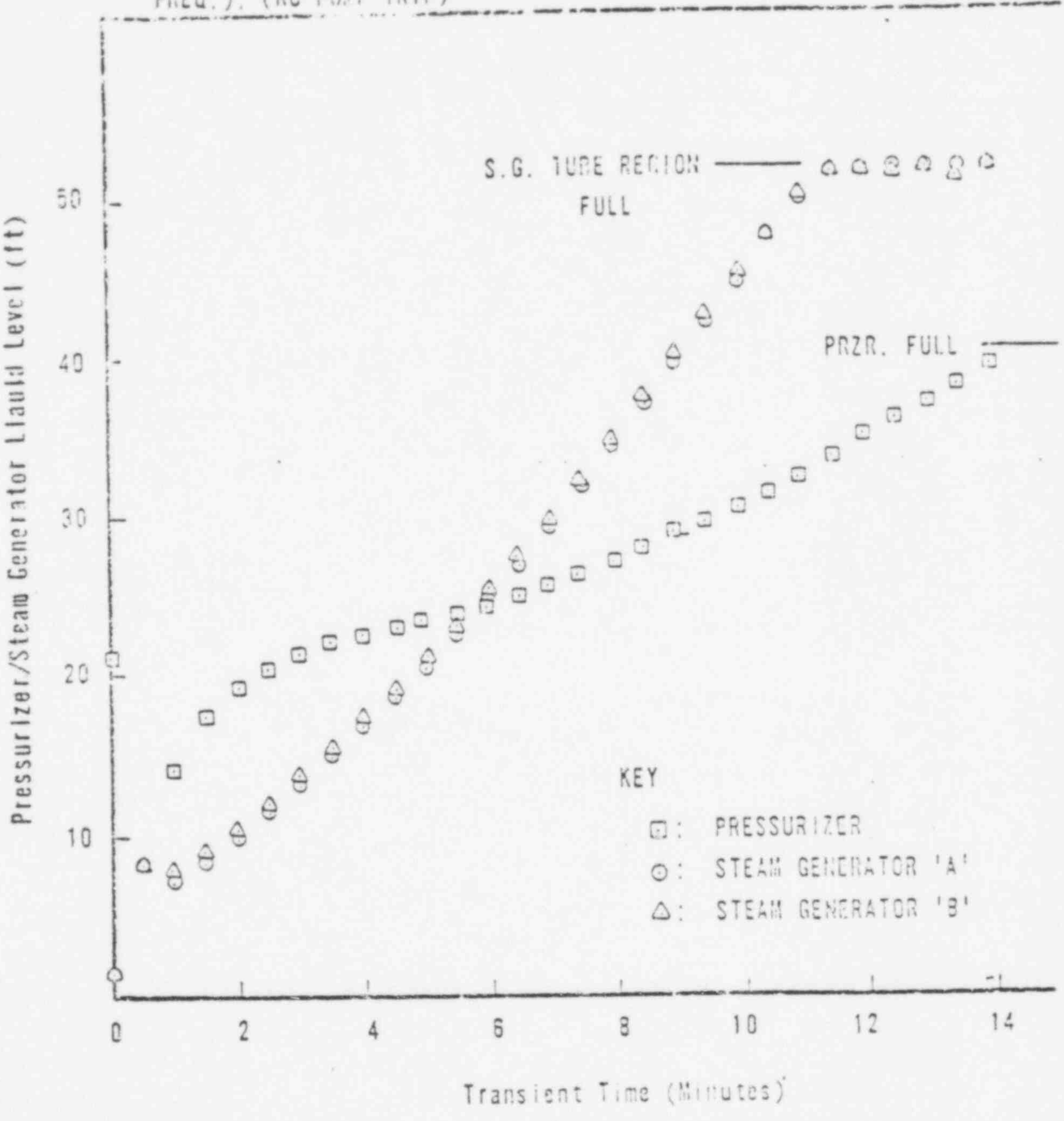


Figure 3.2

946100

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)

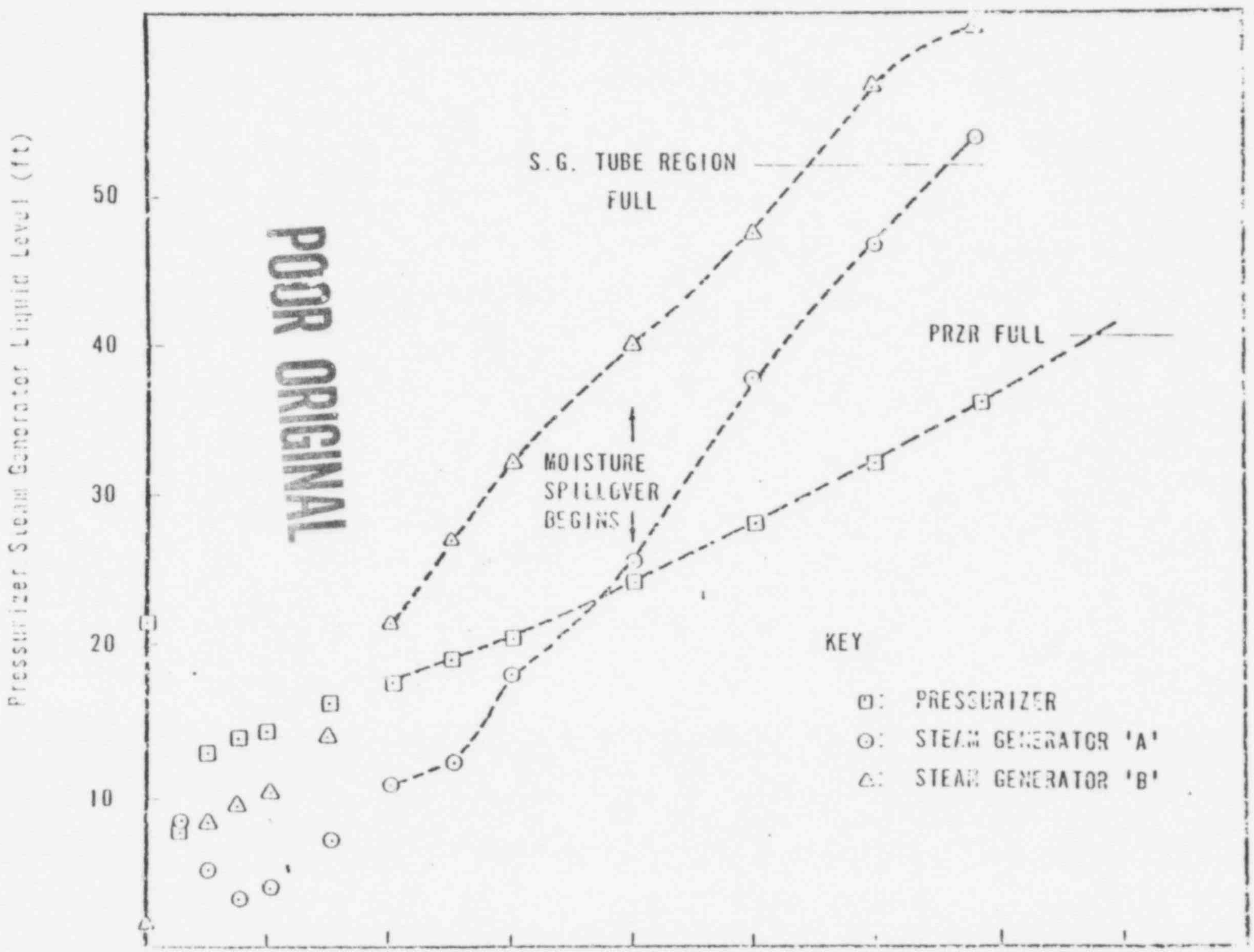


Figure 3.3

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

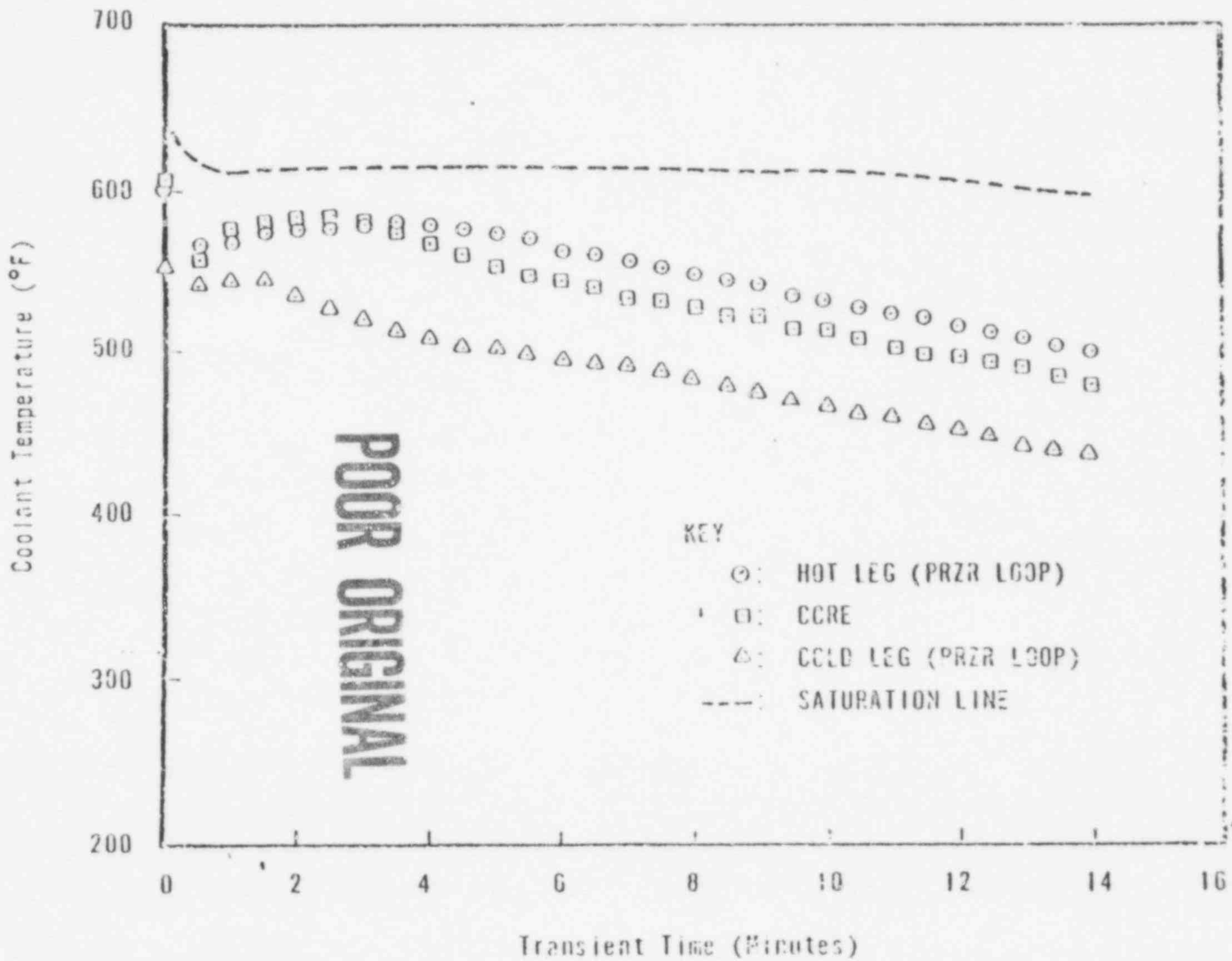


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Figure 3.4

946102

COOLANT TEMPERATURES VERSUS TRANSIENT TIME
(102% FP, BEGINNING OF LIFE, 12.4 FT^2 DOUBLE END RUPTURE, STEAMLINE BREAK (UNMITIGATED) NO RC PUMP TRIP)

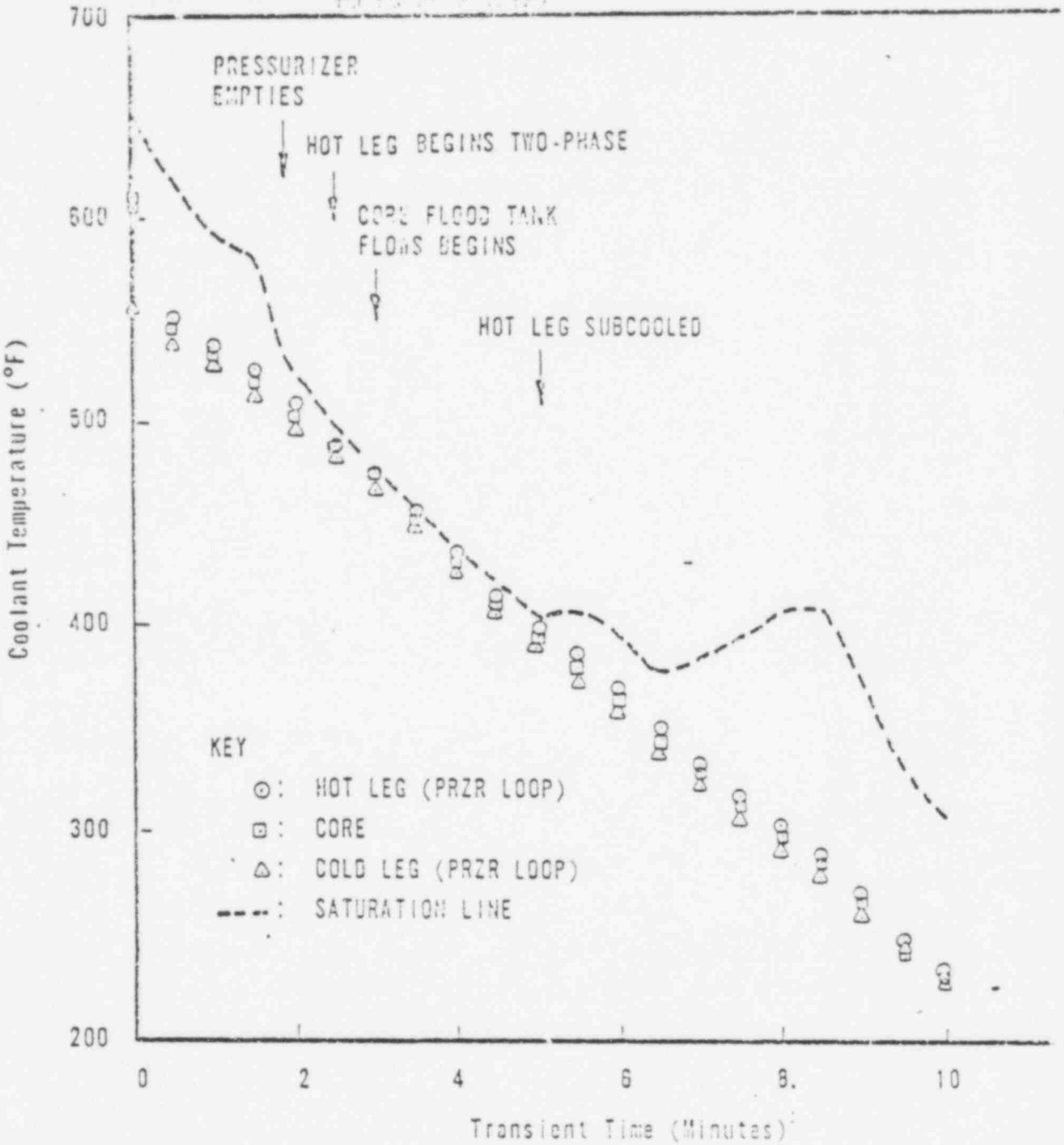

Figure 3.5

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946103

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

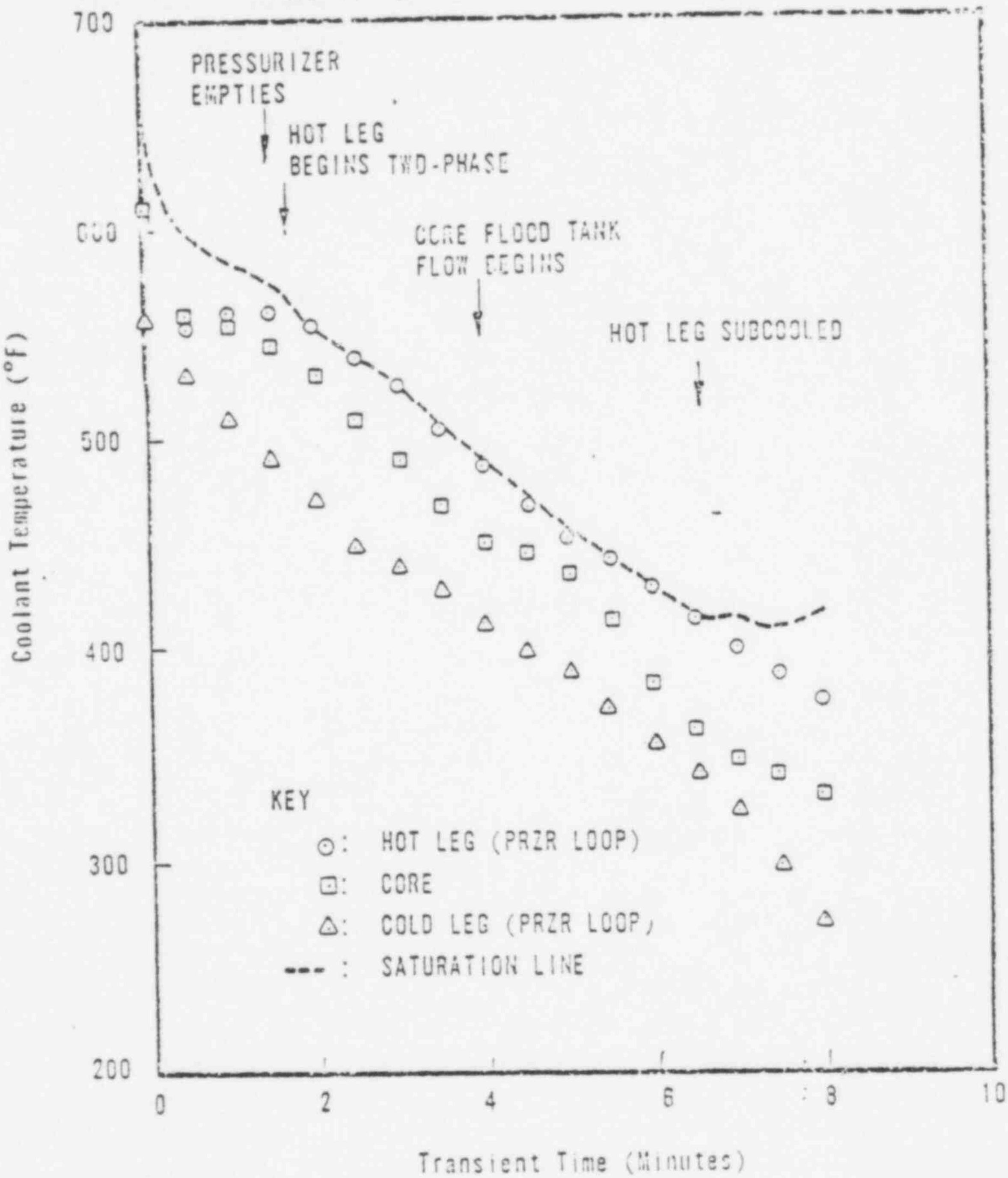

Figure 3.6

STEAM BUBBLE VOLUME VERSUS TRANSIENT TIME
(102% FP, BEGINNING OF LIFE, 12.2 FT^2 DOUBLE
END RUPTURE, UNMITIGATED STEAMLINE BREAK)

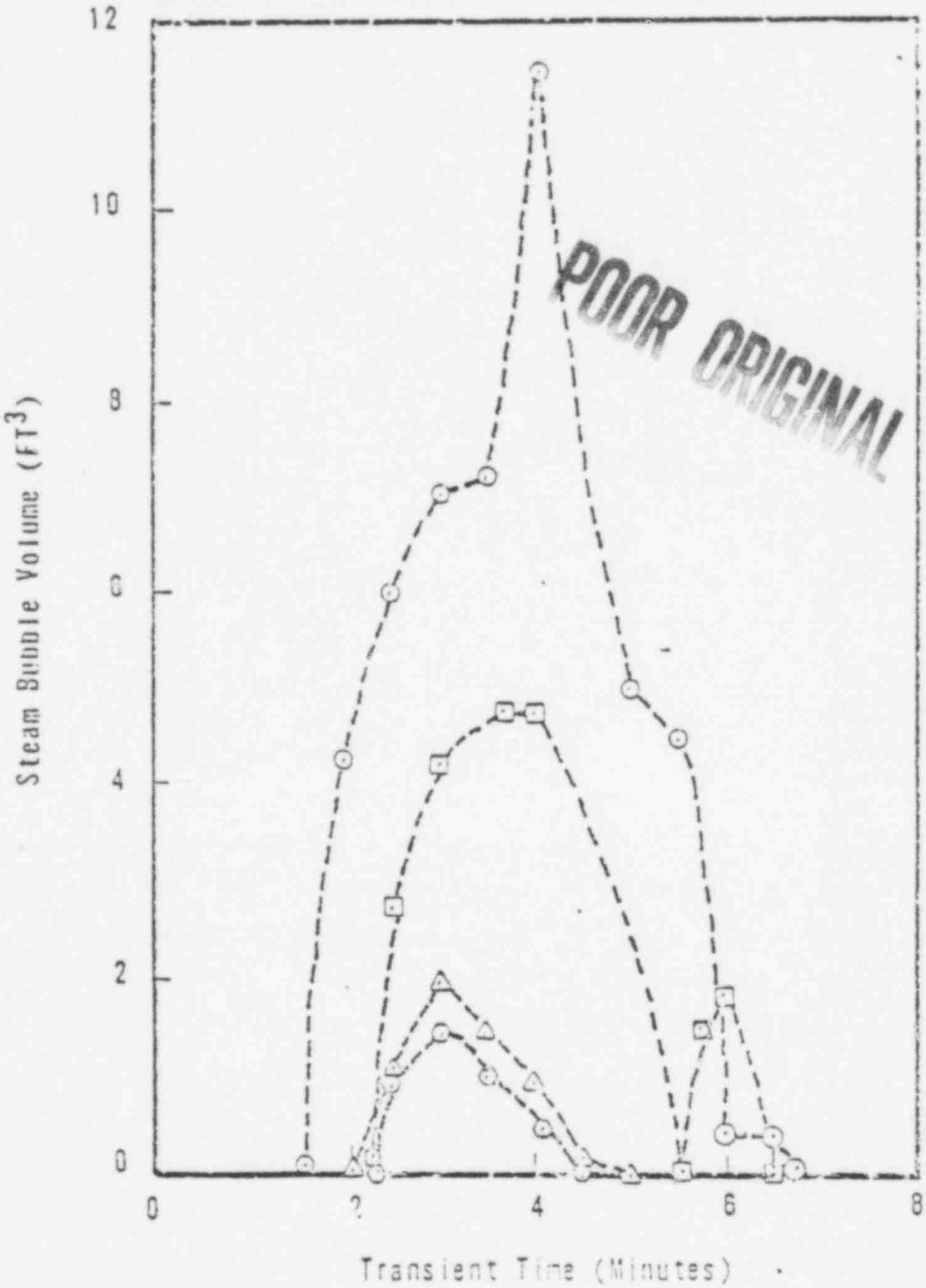

Figure 3.7

KEY

⊙: HOT LEG (PRZR) - RC PUMP TRIP

⊡: HOT LEG 'B' LOOP-RC PUMP TRIP

△: HOT LEG (PRZR LOOP) - NO TRIP

⊙: HOT LEG 'B' LOOP-NO TRIP

946105

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

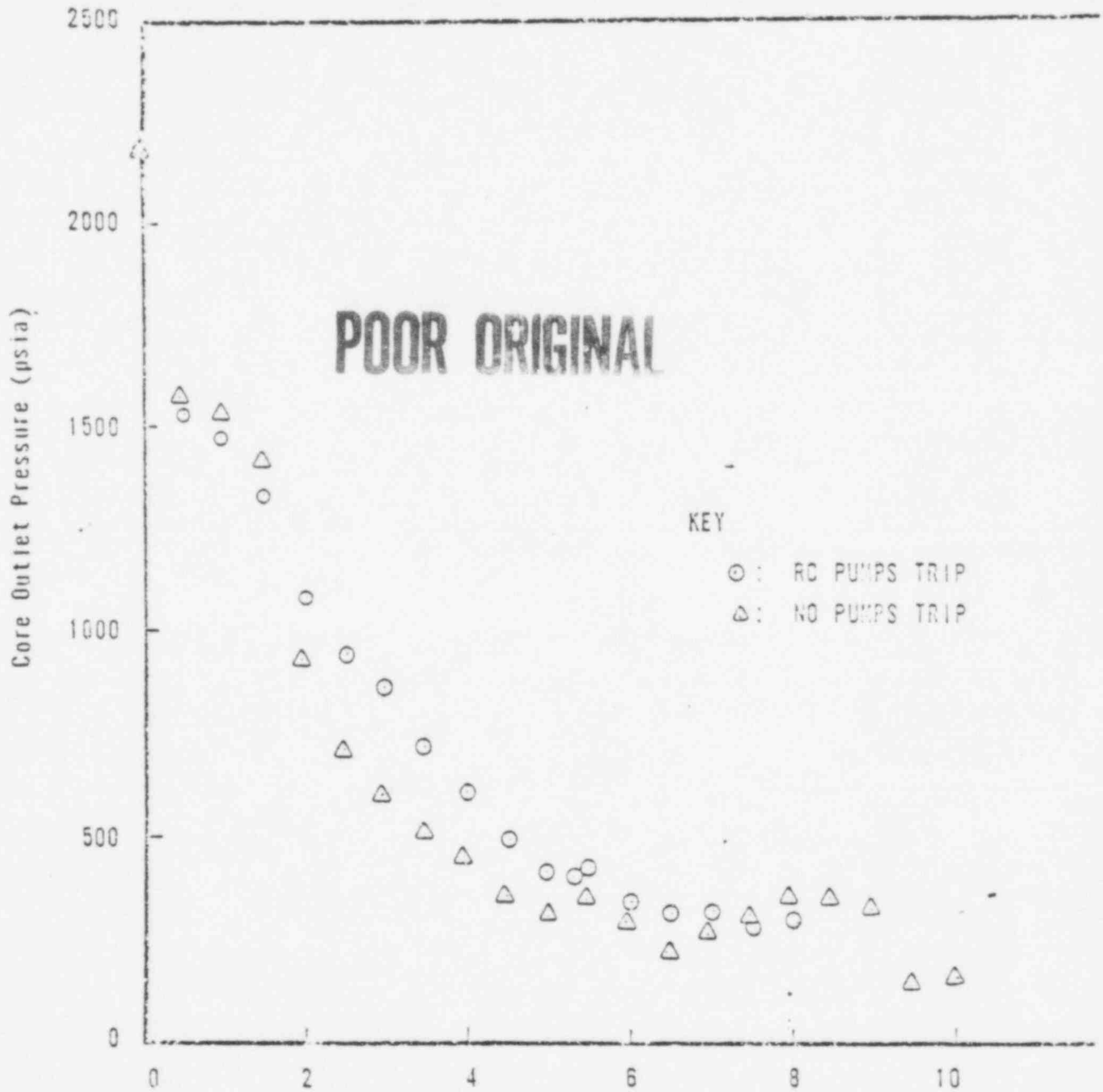

Figure 3.8

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

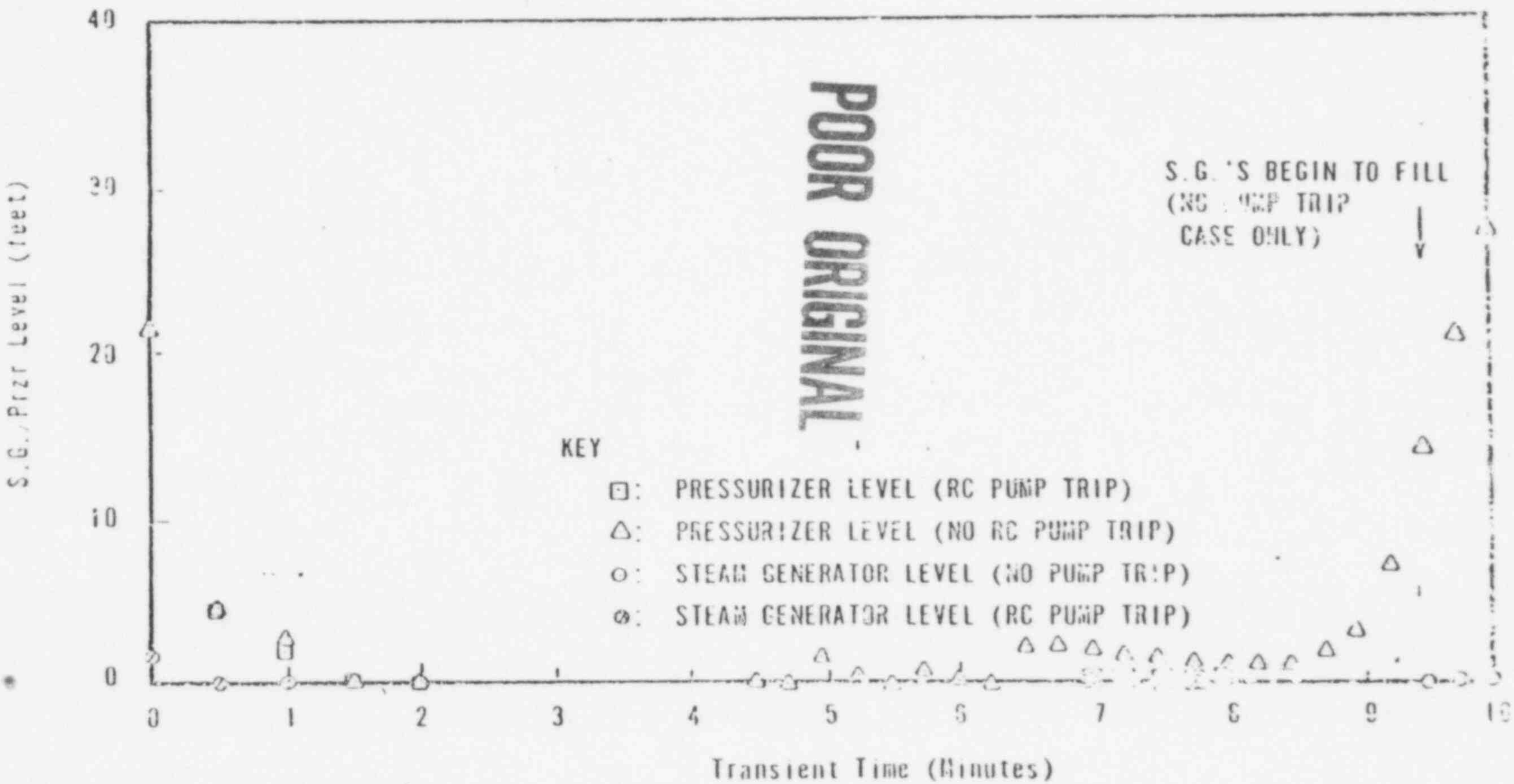

Figure 3.9

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3 R.A. Hedrick, J.J. Cudlin, and R.C. Foltz, "CRAFT2 - Fortran Program for Digital Simulation of a Multinode Reactor Plant During Loss-of-Coolant," BAW-10092, Rev. 2, April 1975.

4 J.F. Wilson, R.J. Grenda, and J.F. Patterson, "The Velocity of Rising Ste in a Bubbline Two-Phase Mixture," ANS Transactions, 5, (1962).

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