ANALYSIS SUMMARY IN SUPPORT OF AN EARLY RC PUMP TRIP

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CONTENTS

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.

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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 ~~A.3~~ C demonstrates the applicability of the conclusions drawn in section ~~A.2~~ 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 ~~A.4~~ D. Finally, section ~~A.5~~ 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.

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 thi 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

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.

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.

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 B.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

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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 tripping 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.

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), 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.

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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 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-

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. 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.

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.

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 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 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.

Table 2-1. Analysis Scope With AFW Available

Break size, (ft^2)	Break location: Cold leg	Break location: Hot leg	Continuous RC pump operation: 2 HPI	Continuous RC pump operation: 1 HPI	RC pump trip @ 90% void: 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		

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*Analyzed with both 1.0 and 1.2 ANS decay curves.

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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

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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.

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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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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

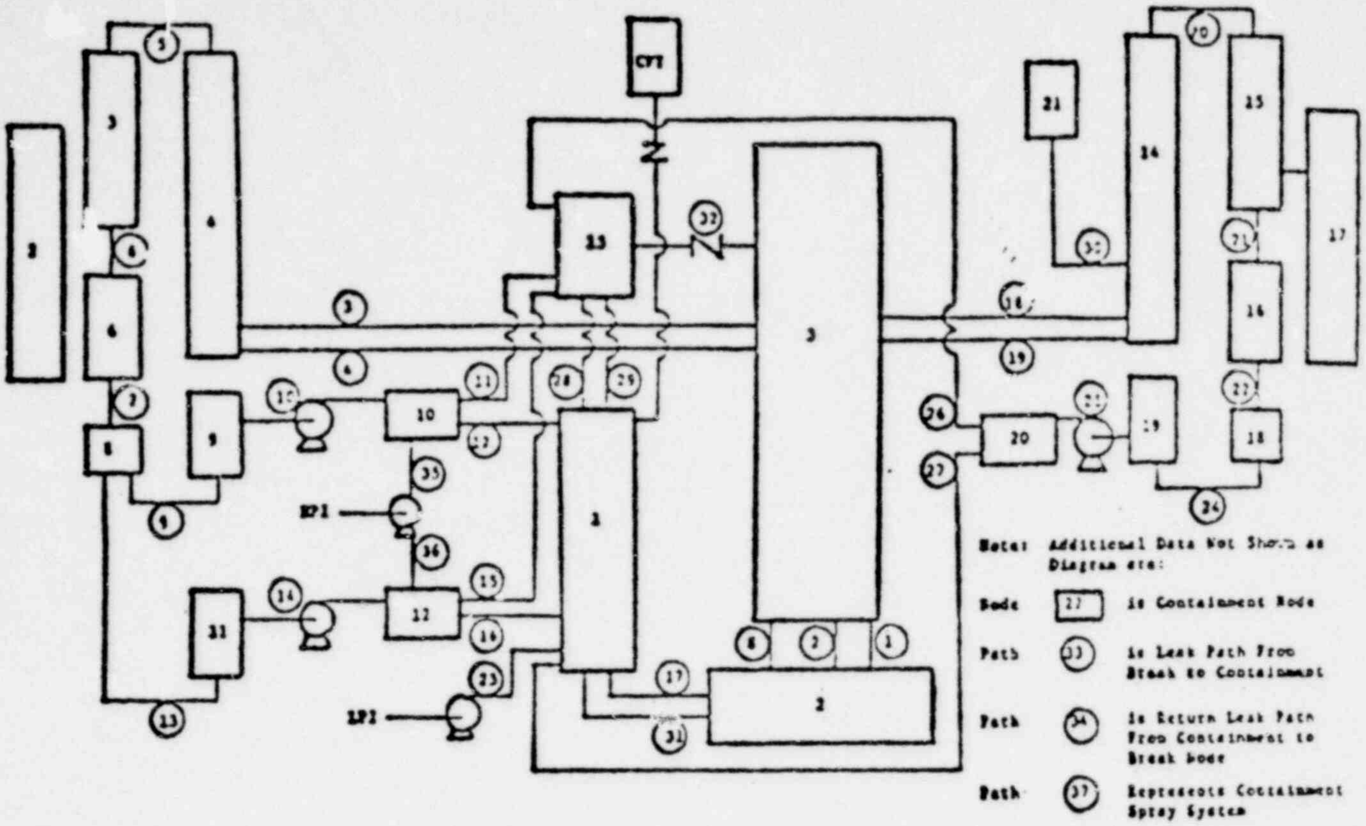
MINITRAP2 PATH DESCRIPTION

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

1054 083

Figure 2-1. CRAFT2 Noding Diagram for Small Break



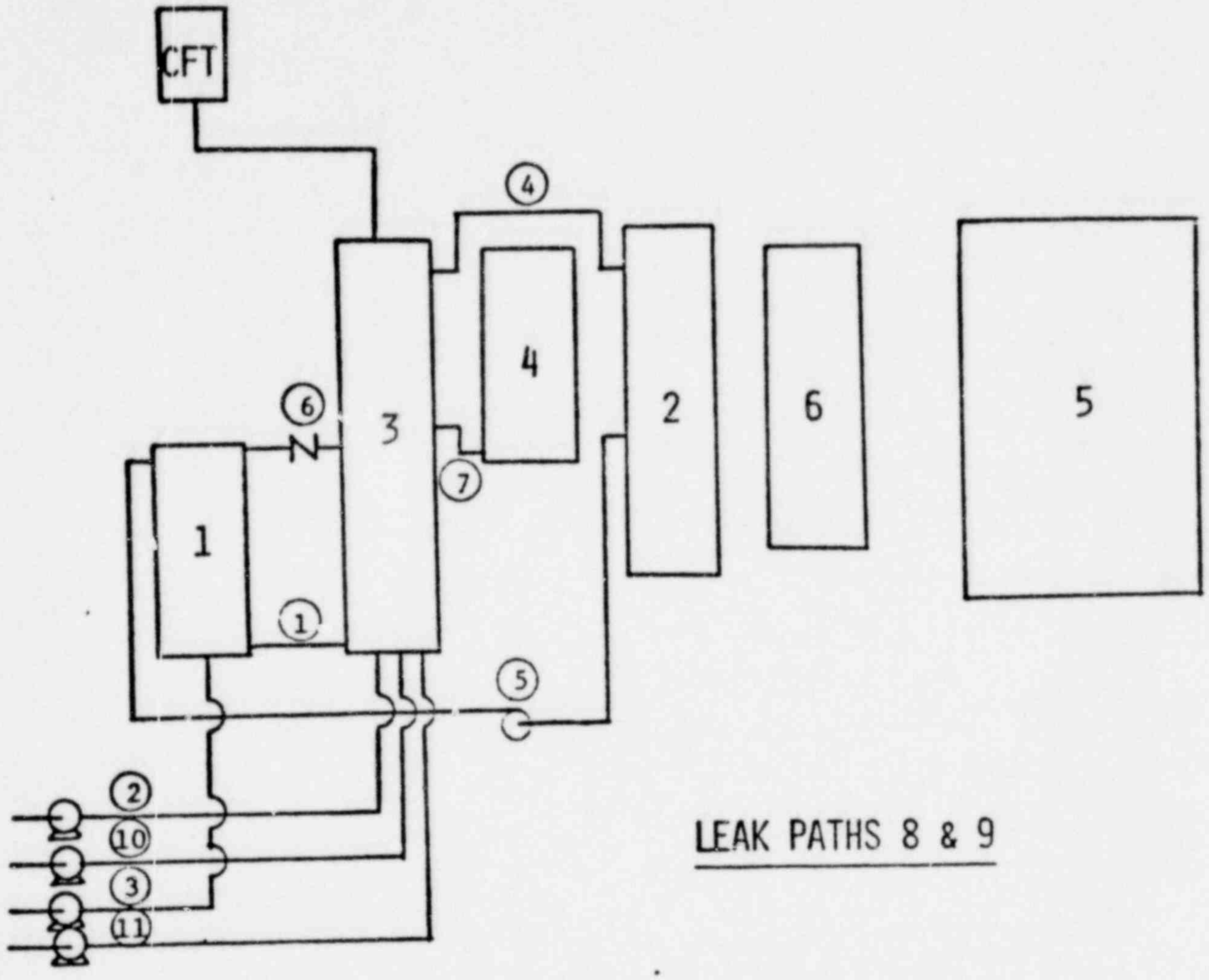
Node No.	Identification
1	Downcomer
2	Lower Plenum
3	Core, Core Bypass, Upper Plenum, Upper Head
4,14	Hot Leg Piping
5,15	Steam Generator Upper Head, SG Tubes (Upper Half)
6,16	SG Tubes (Lower Half)
8,18	SG Lower Head
9,11,19	Cold Leg Piping (Pump Suction)
10,12,20	Cold Leg Piping (Pump Discharge)
13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

POOR ORIGINAL

1054 084

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

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

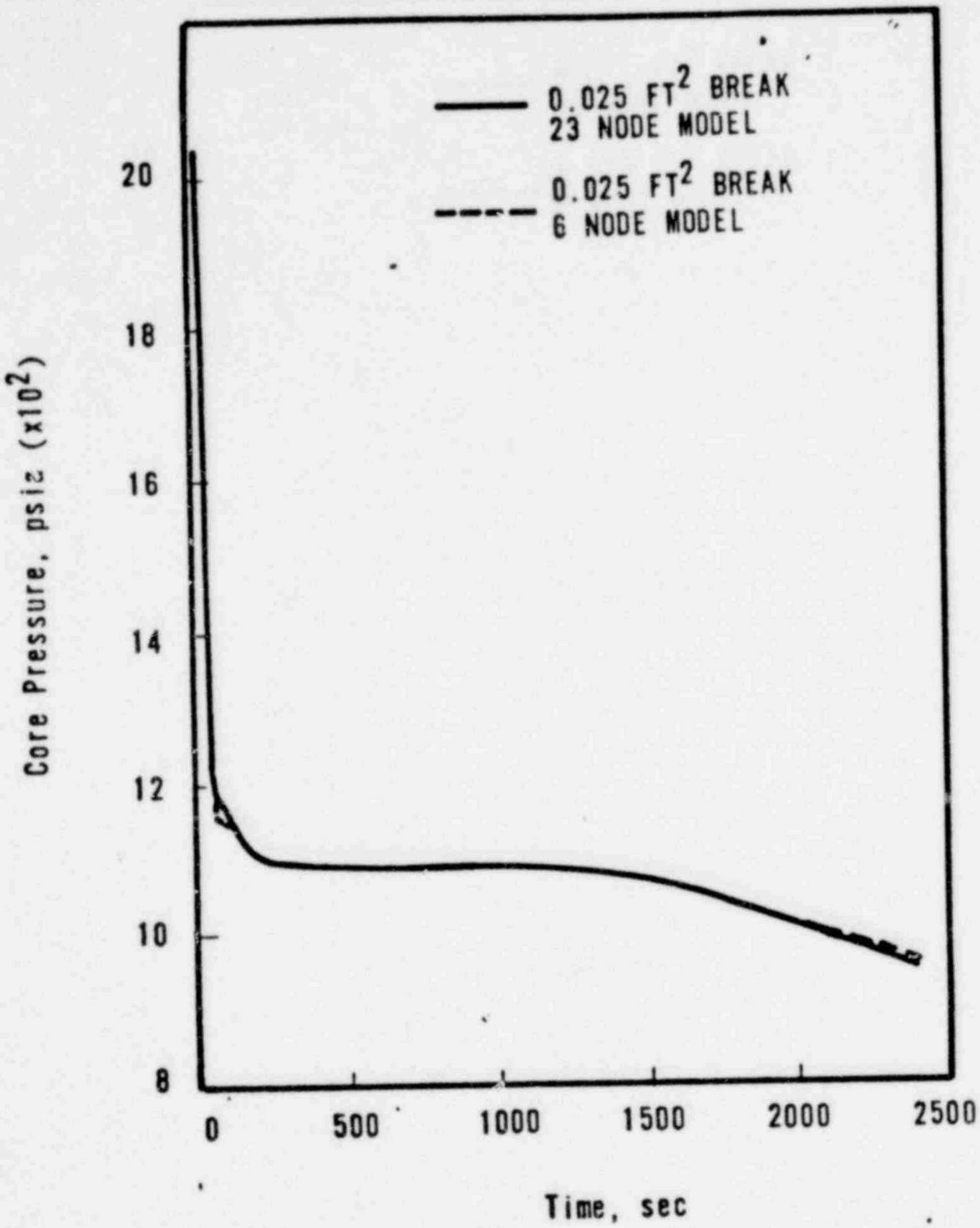


Figure 2-3

POOR ORIGINAL

1054 086

PERCENT SYSTEM VOIDS VS TIME, PUMPS ON

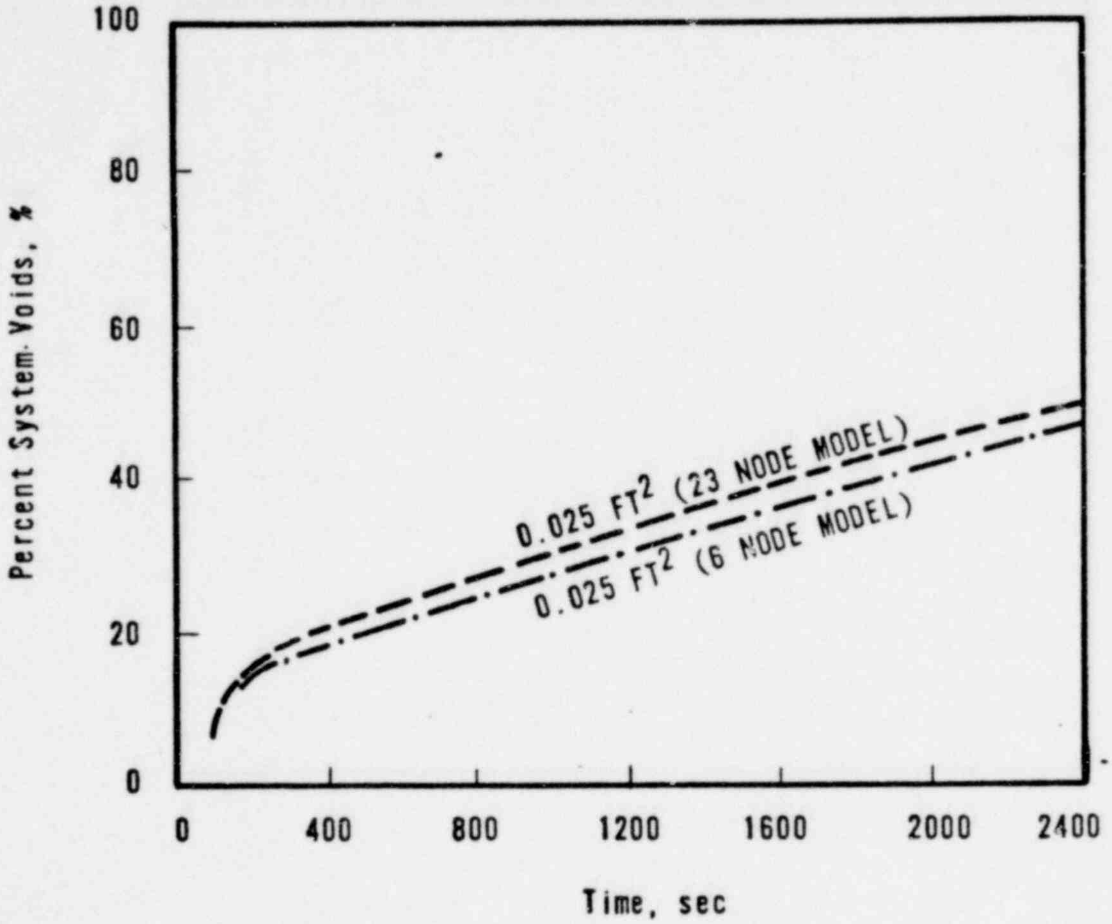


Figure 2-4

1054 089

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

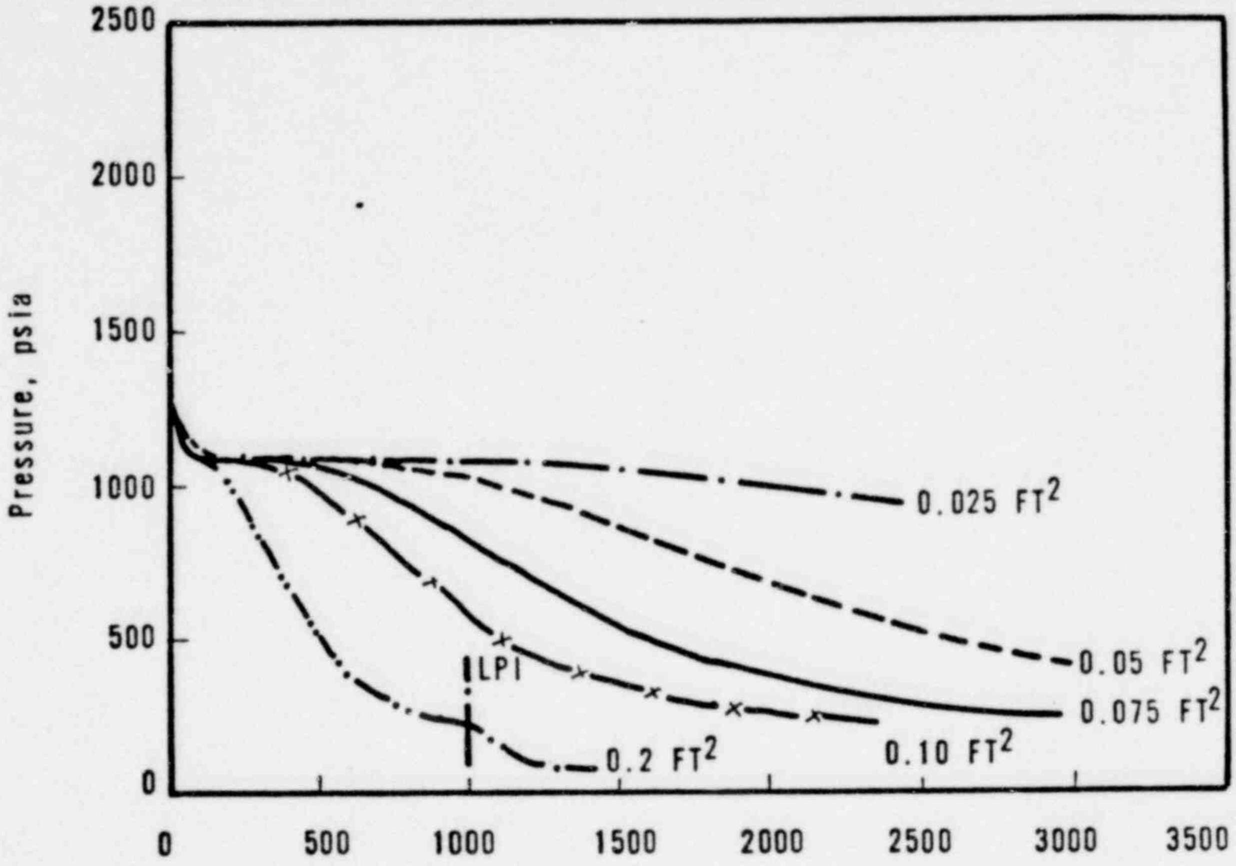


Figure 2-5

POOR ORIGINAL

1054 088

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

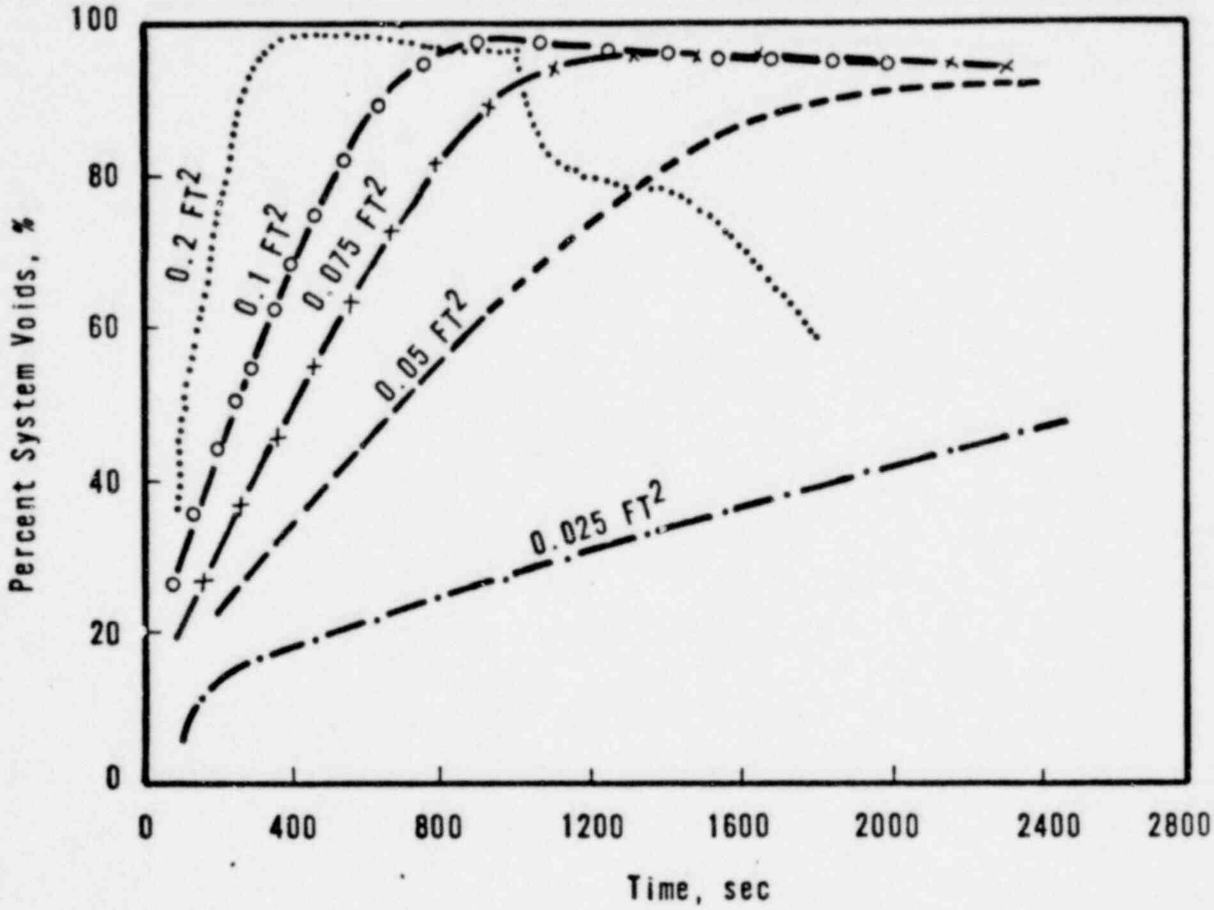


Figure 2-6

1054 089

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

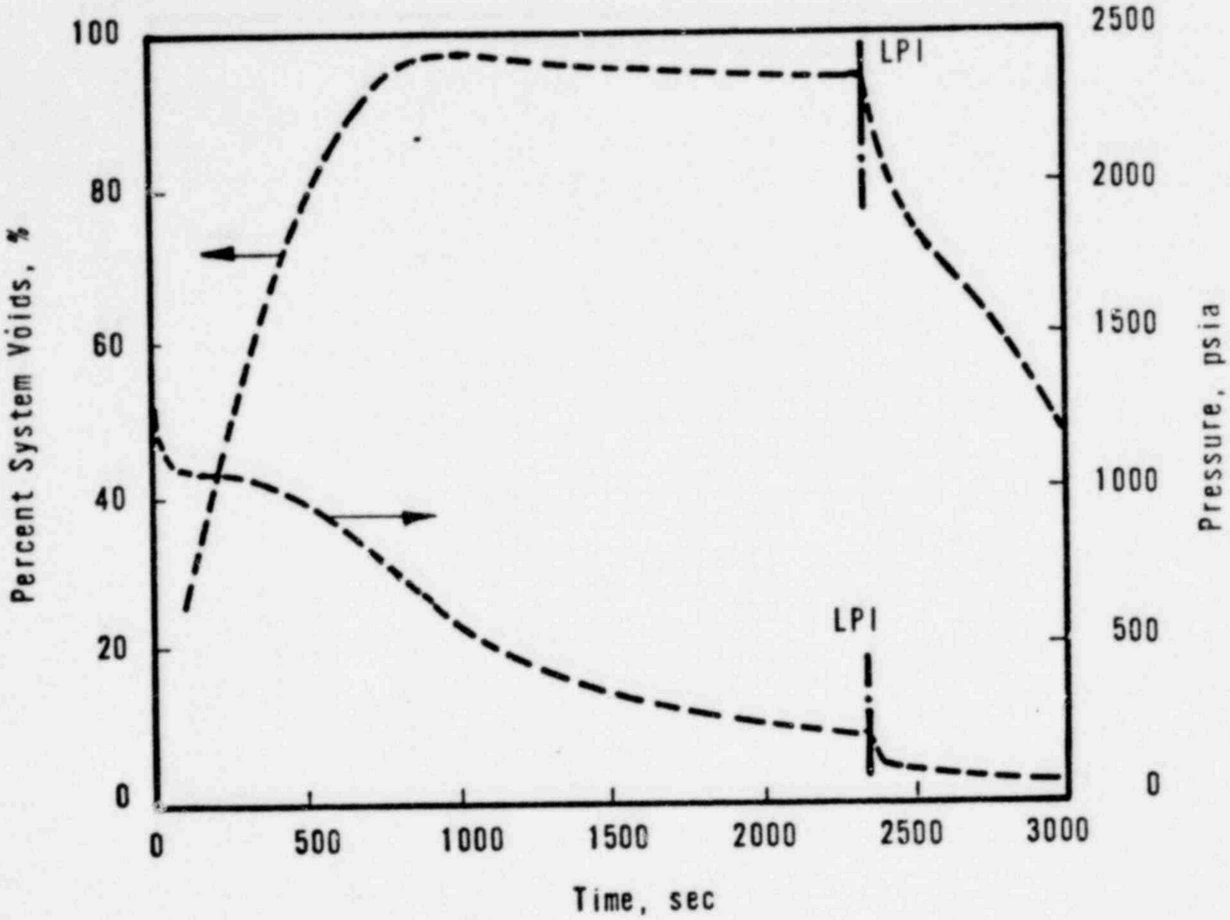


Figure 2-7

1054 090

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

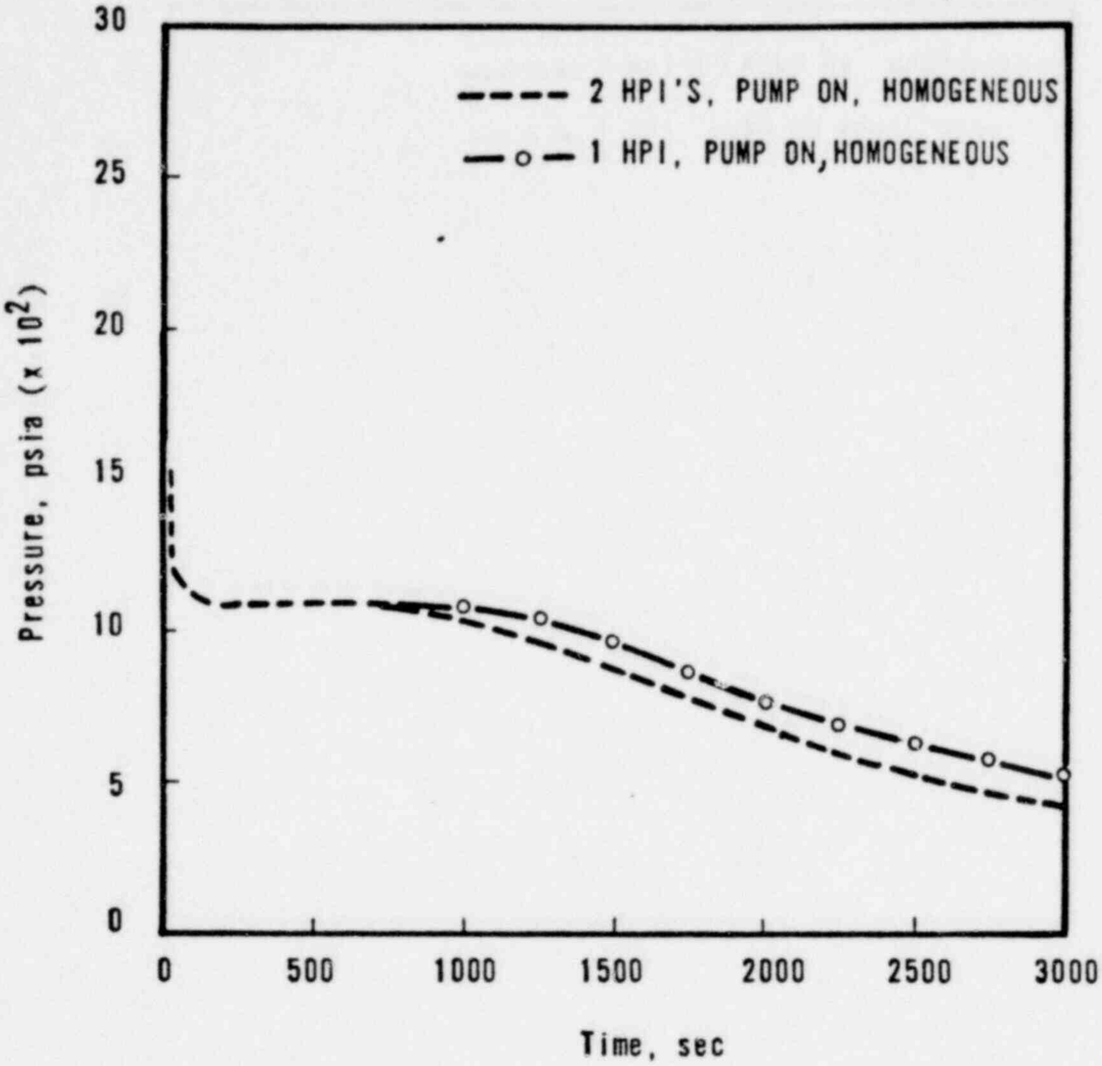


Figure 2-8

1054 09

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

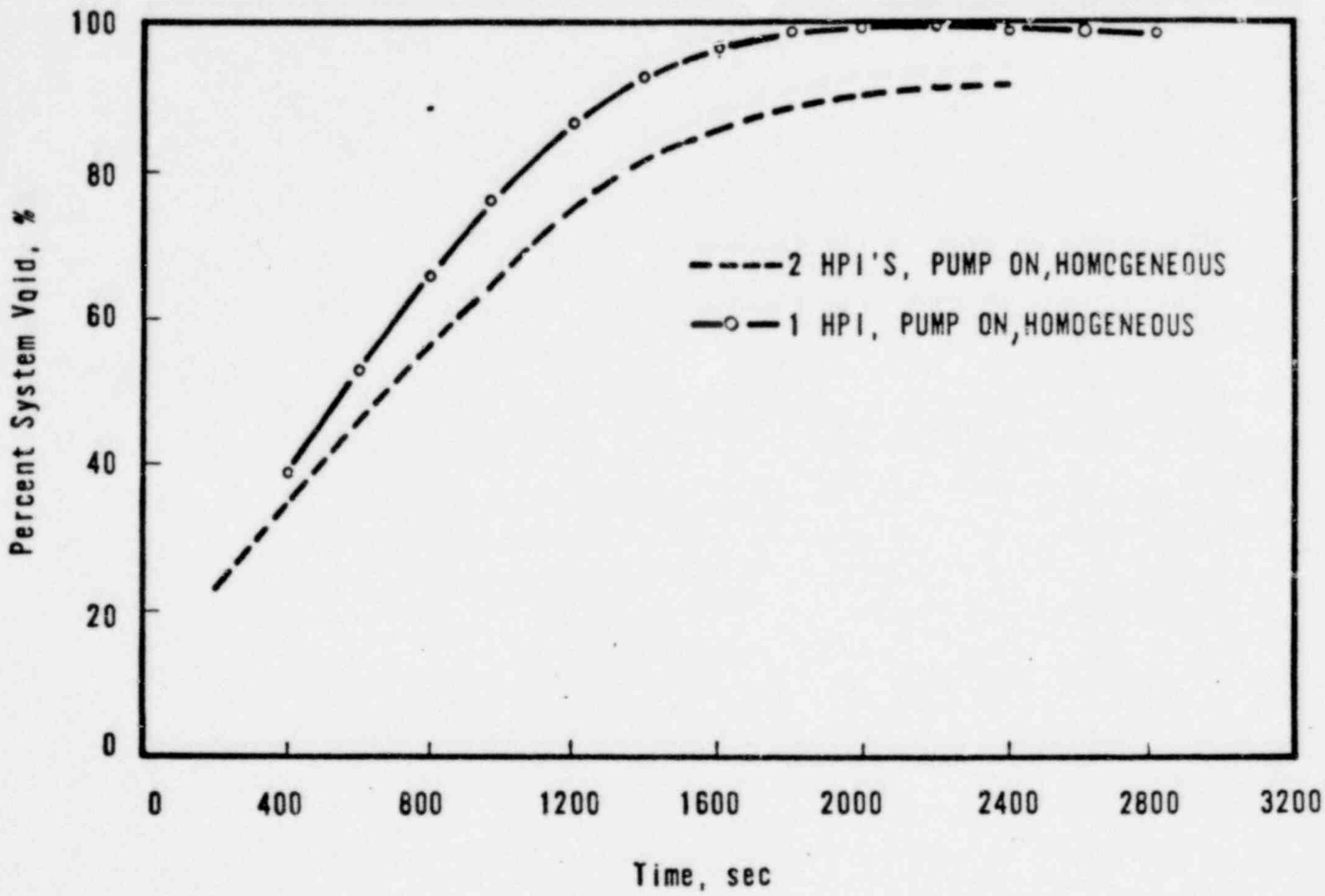


Figure 2-9

1054 092

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

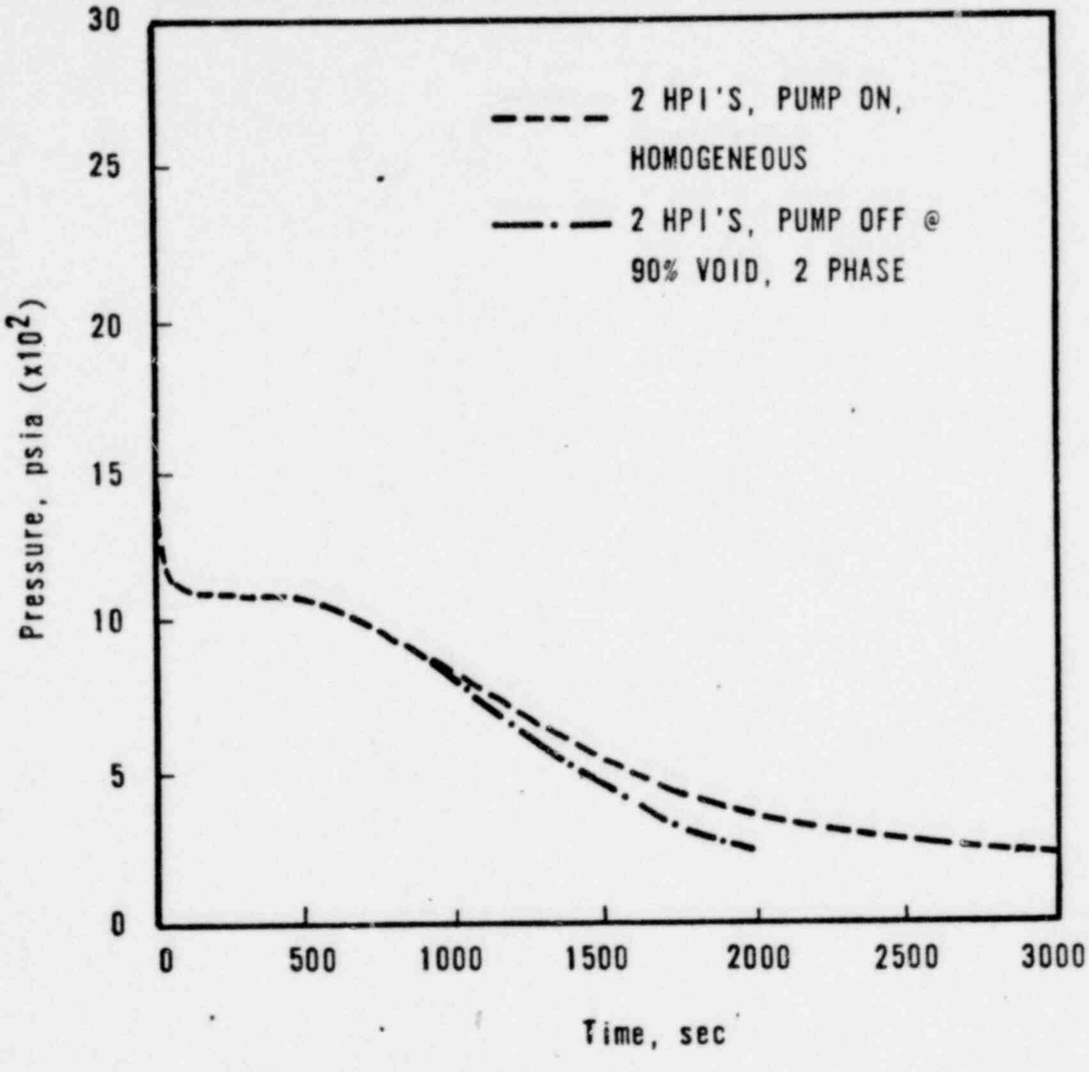


Figure 2-10

1054 093

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

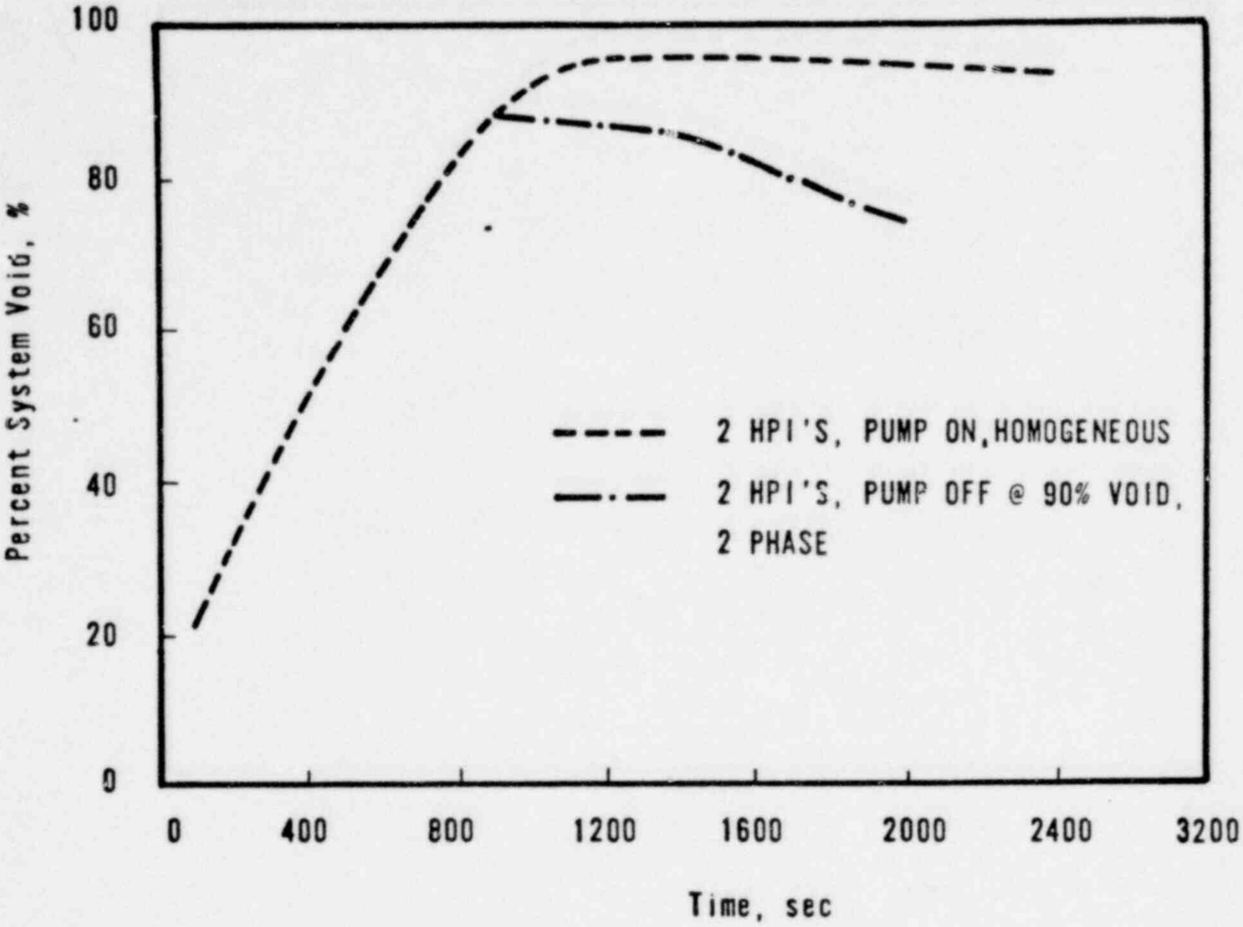


Figure 2-11

POOR ORIGINAL

1054 094

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

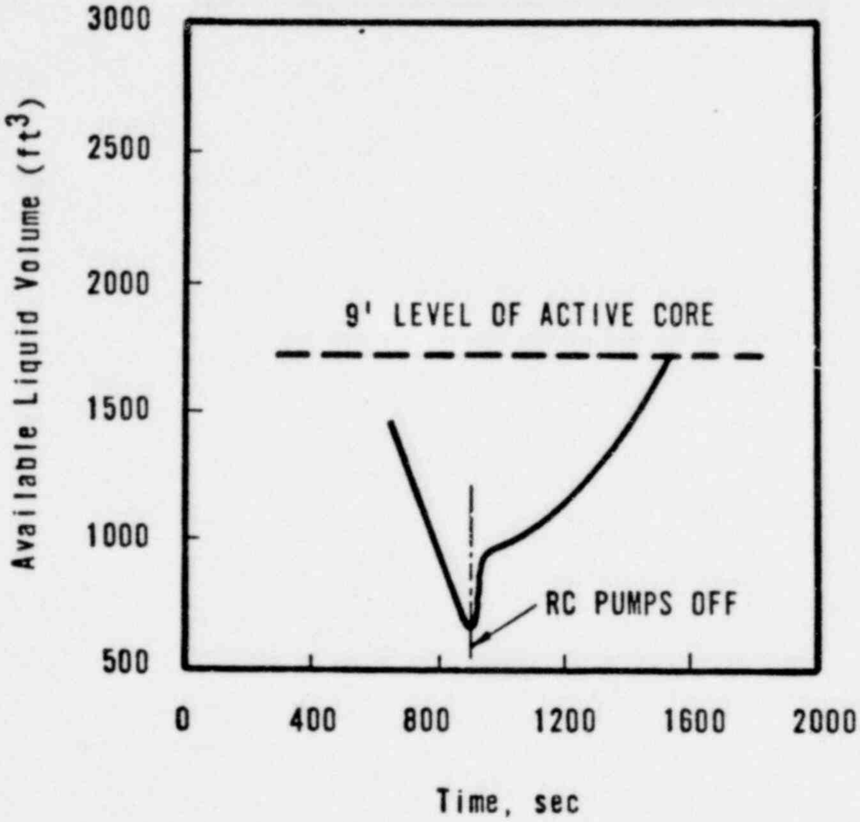


Figure 2-12

1054 095

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

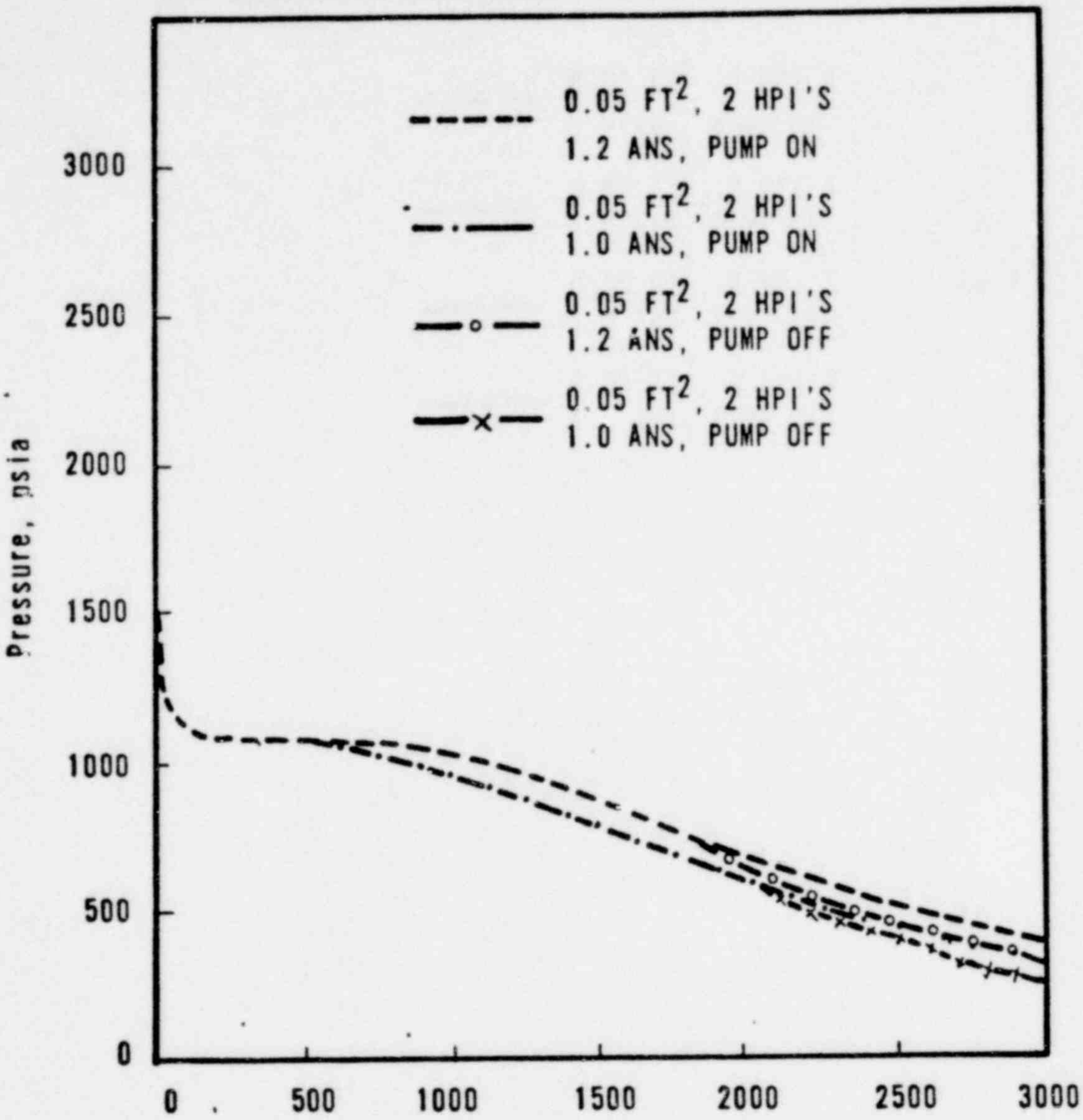


Figure 2-13

1054 096

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

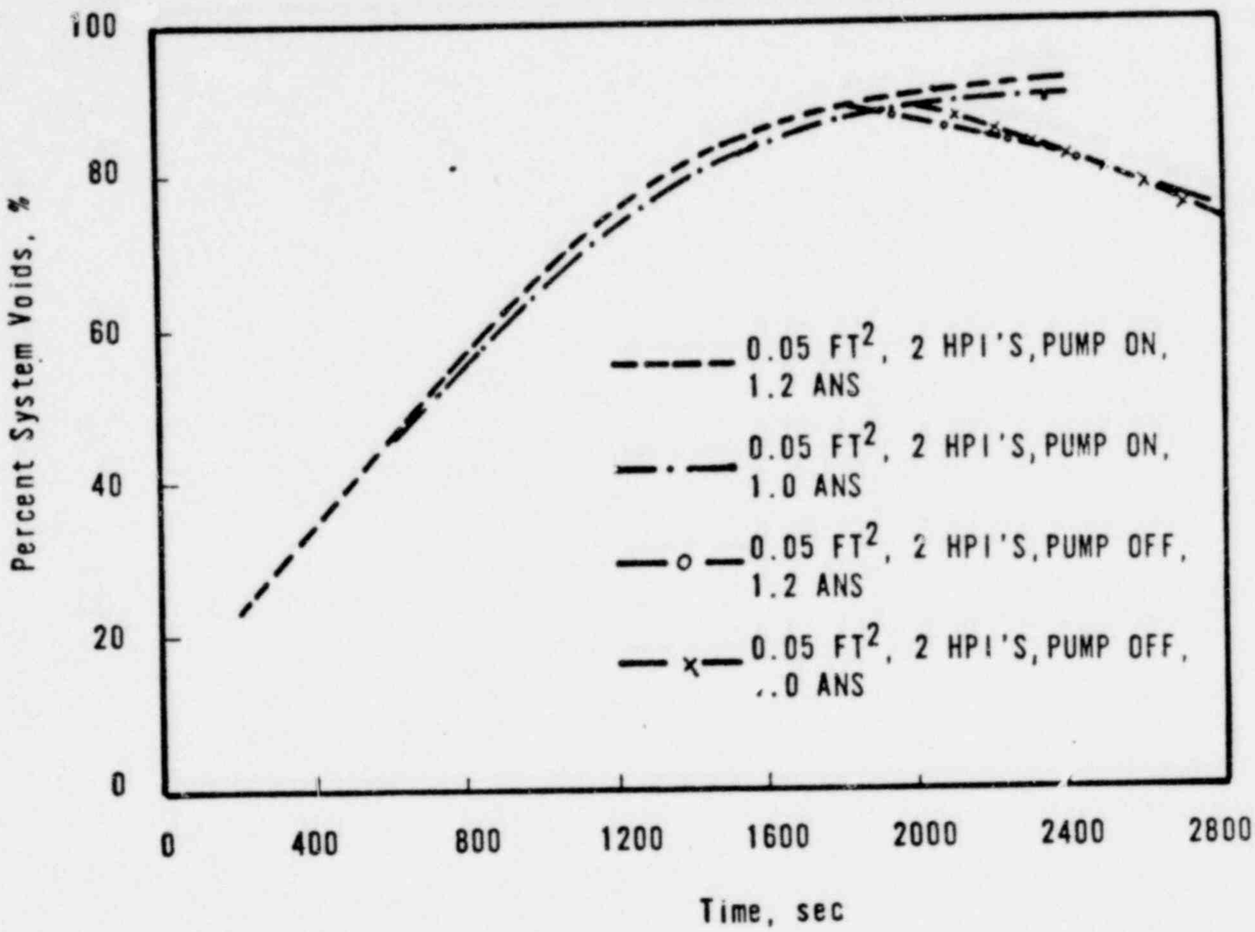


Figure 2-14

POOR ORIGINAL

1054 098

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

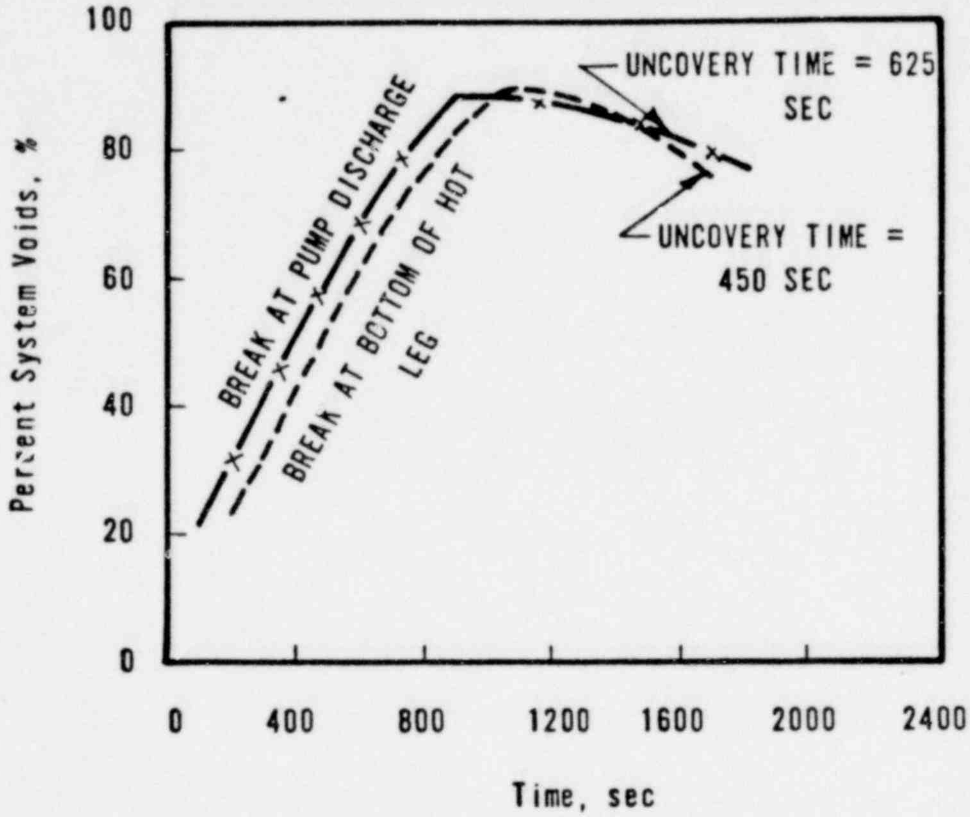


Figure 2-15

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

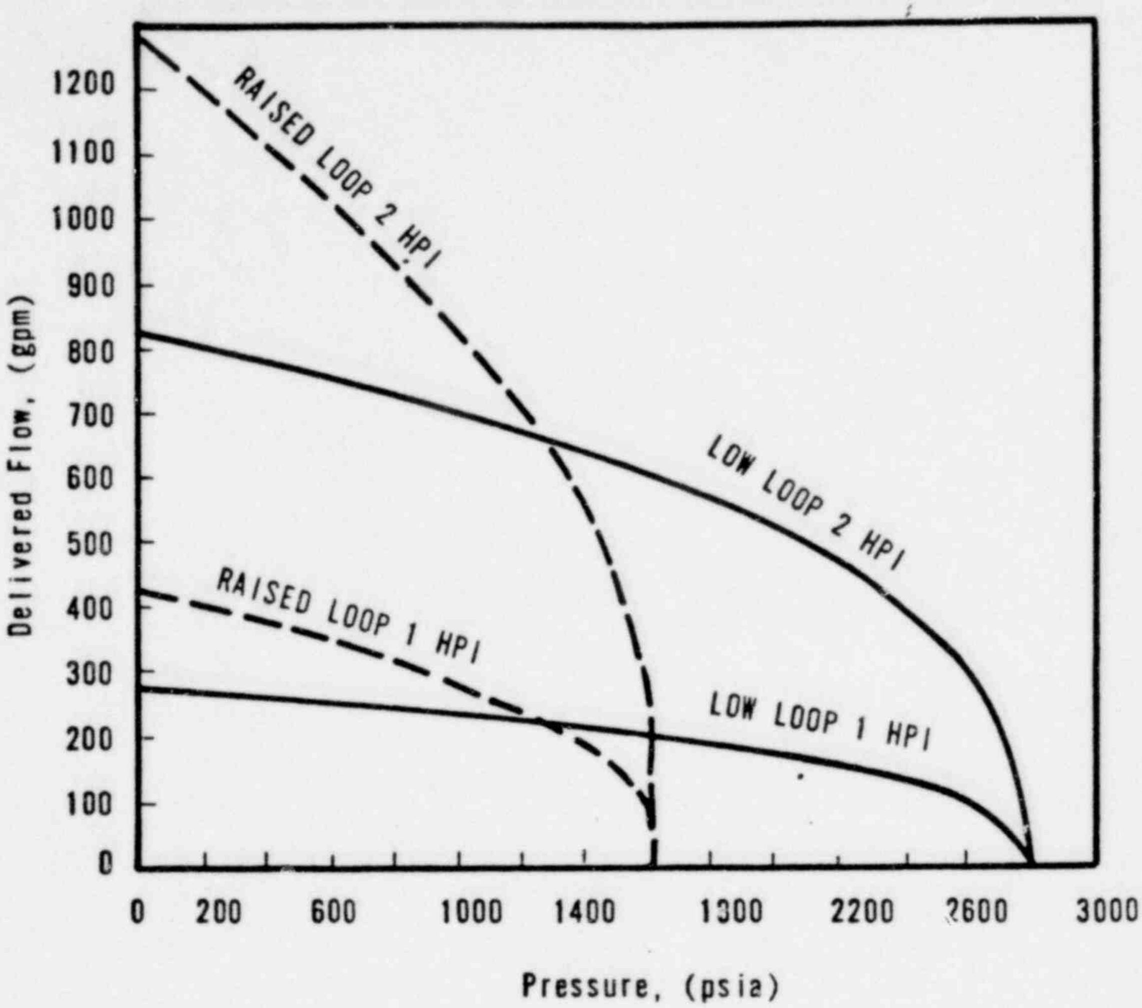


Figure 2-16

1054 099

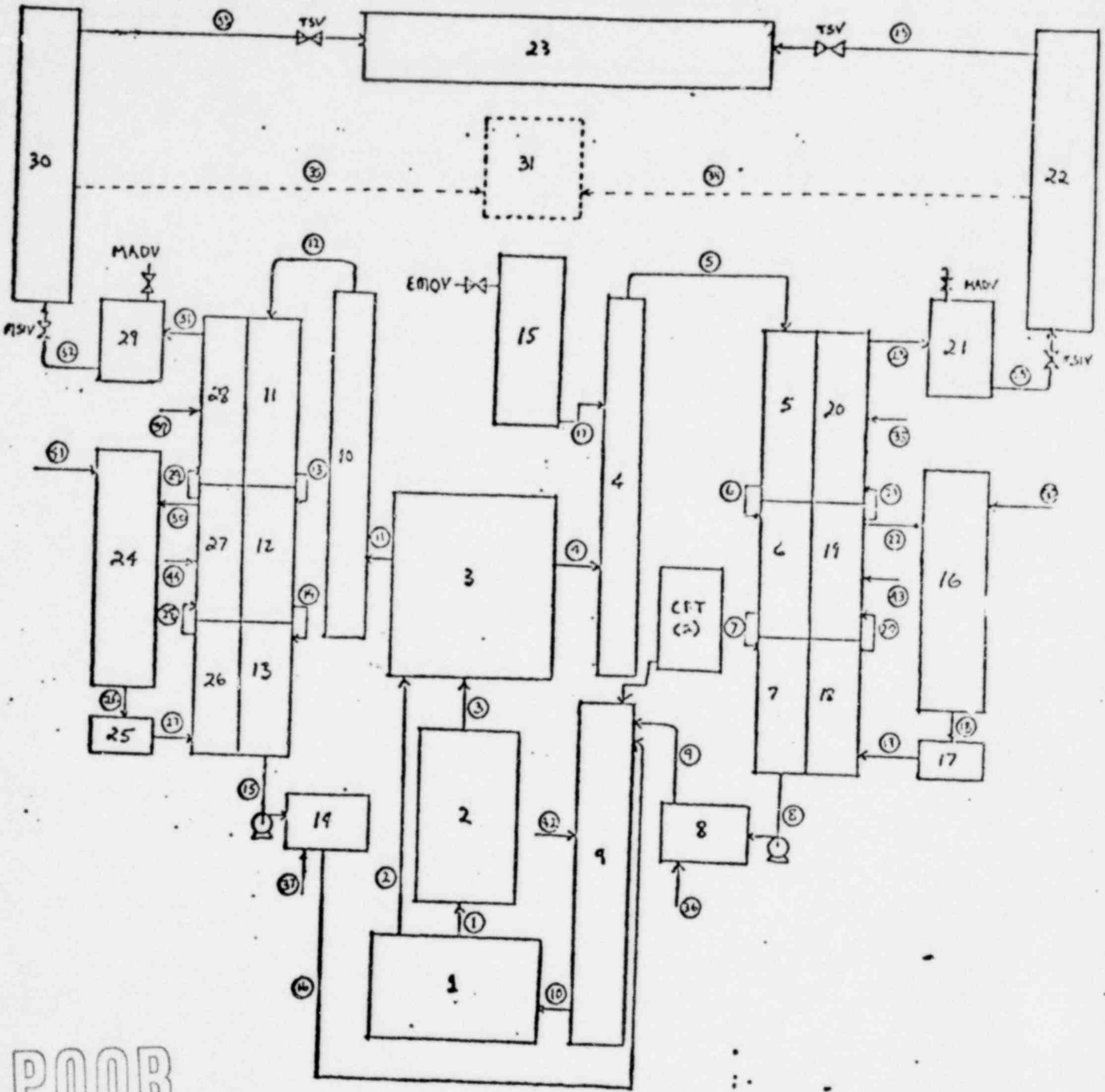


Figure 3.1

MINITRAP2 Noding and Flow Path Scheme

1054 100

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)

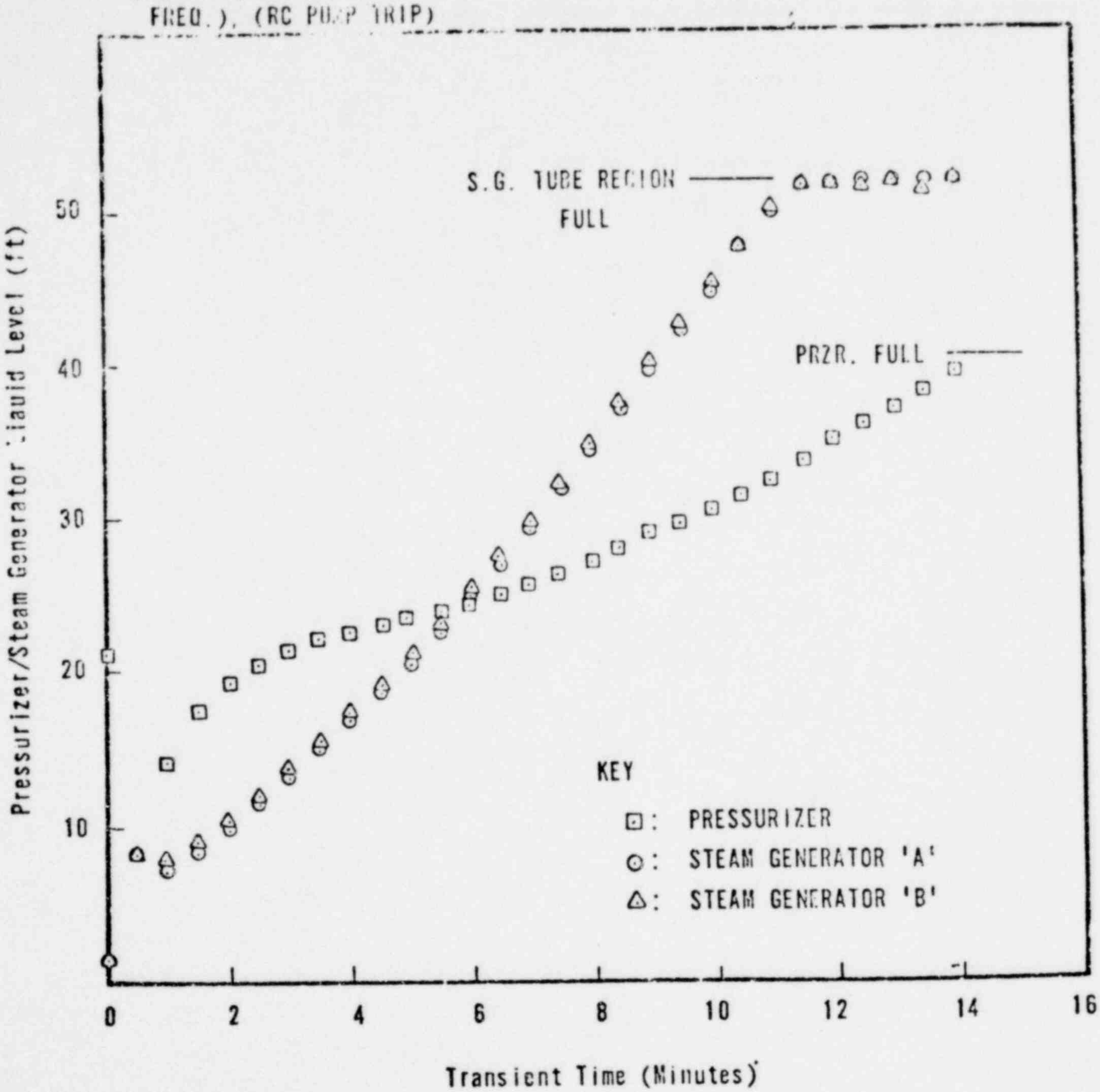


Figure 3.2

1054 10

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)

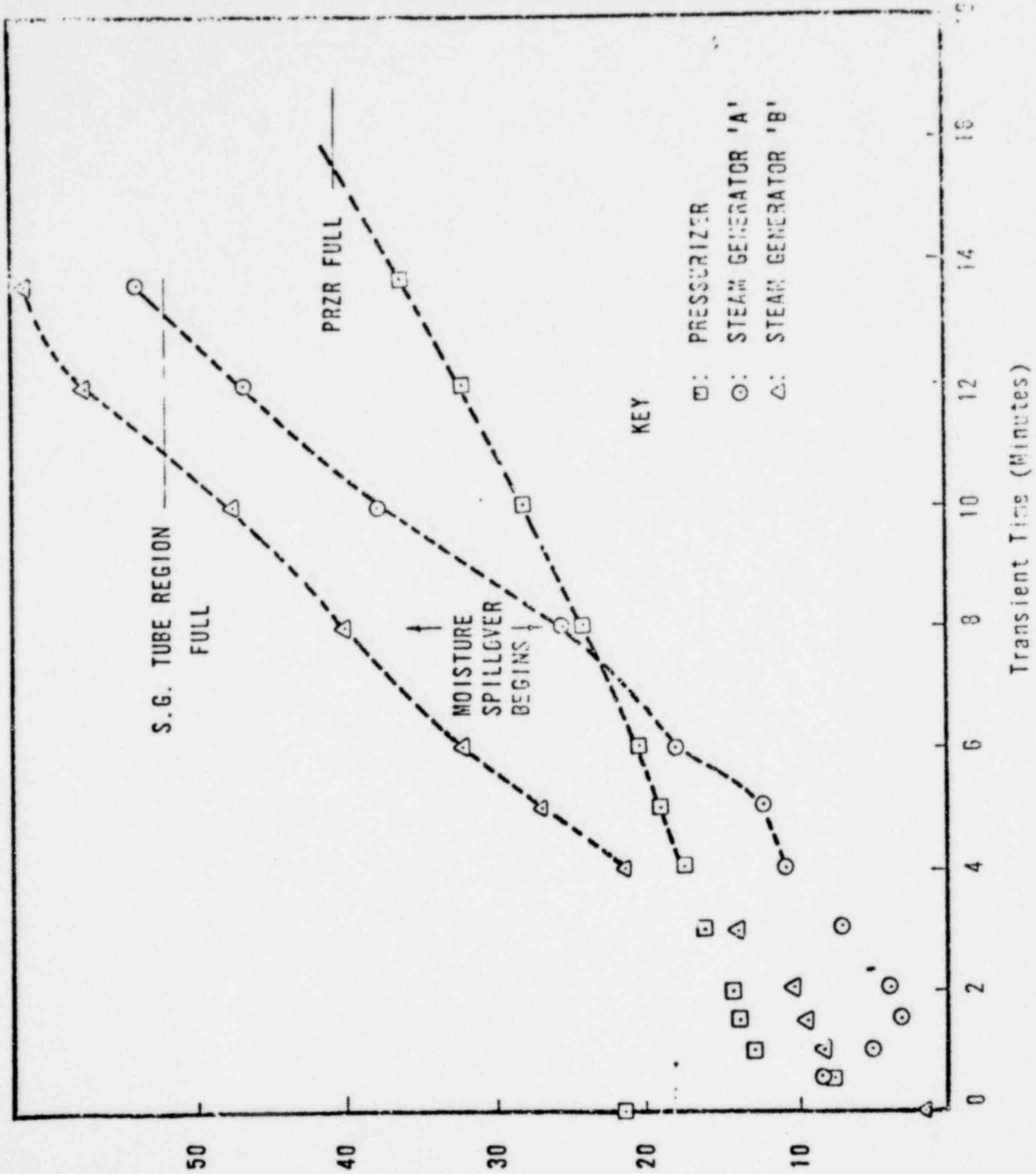
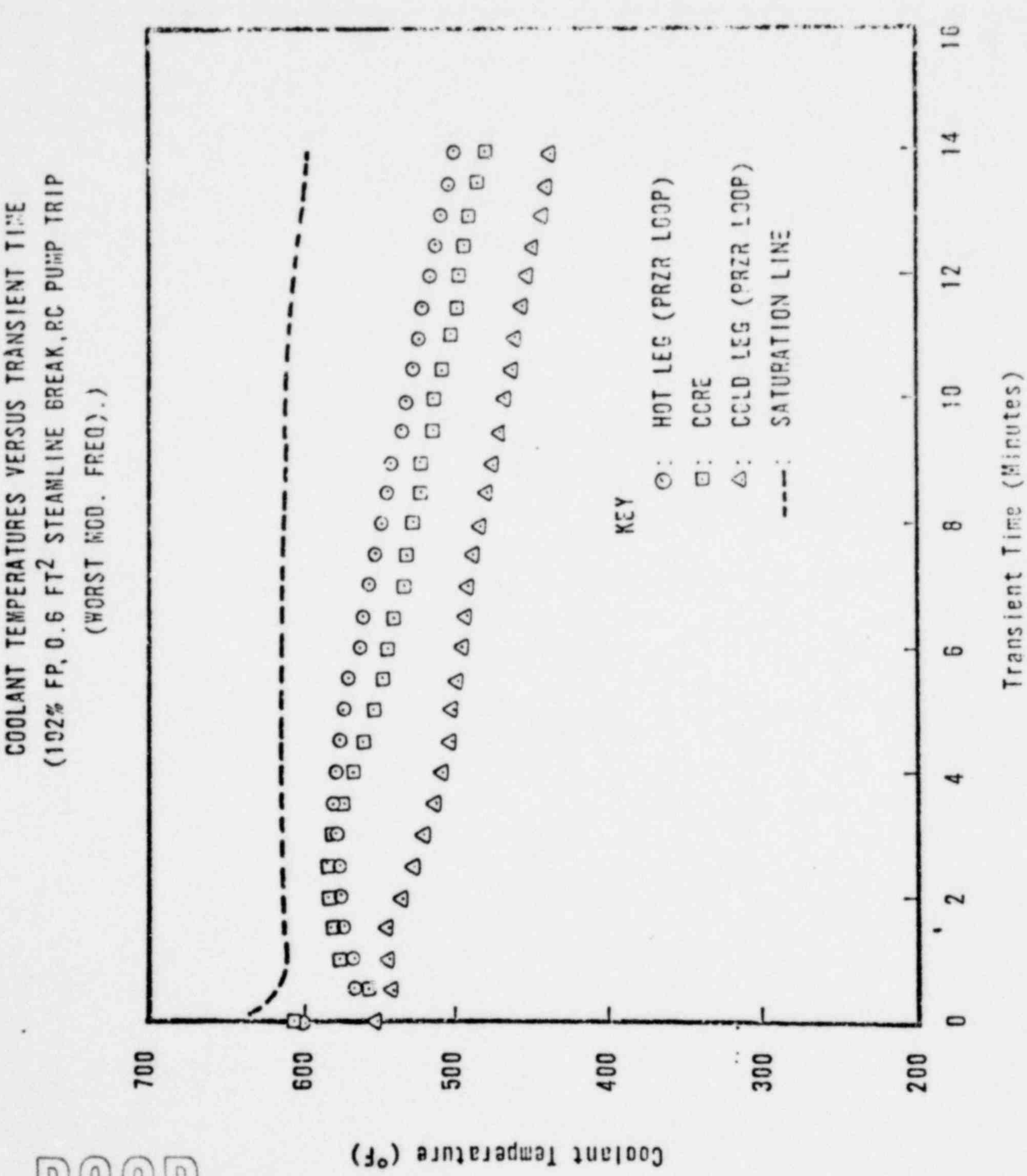


Figure 3.3



COOLANT TEMPERATURES VERSUS TRANSIENT TIME
(102% FP, 0.6 FT2 STEAMLINE BREAK, RC PUMP TRIP
(WORST MOD. FREQ).)
KEY
HOT LEG (PRZR LOOP)
CORE
COLD LEG (PRZR LOOP)
SATURATION LINE
Coolant Temperature (°F)
Transient Time (Minutes)
700
600
500
400
300
200
0
2
4
6
8
10
12
14
16

Figure 3.4

COOLANT TEMPERATURES VERSUS TRANSIENT TIME
(102% FP, BEGINNING OF LIFE, 12.2 FT² DOUBLE
END RUPTURE, STEAMLINE BREAK (UNMITIGATED)
NO RC PUMP TRIP)

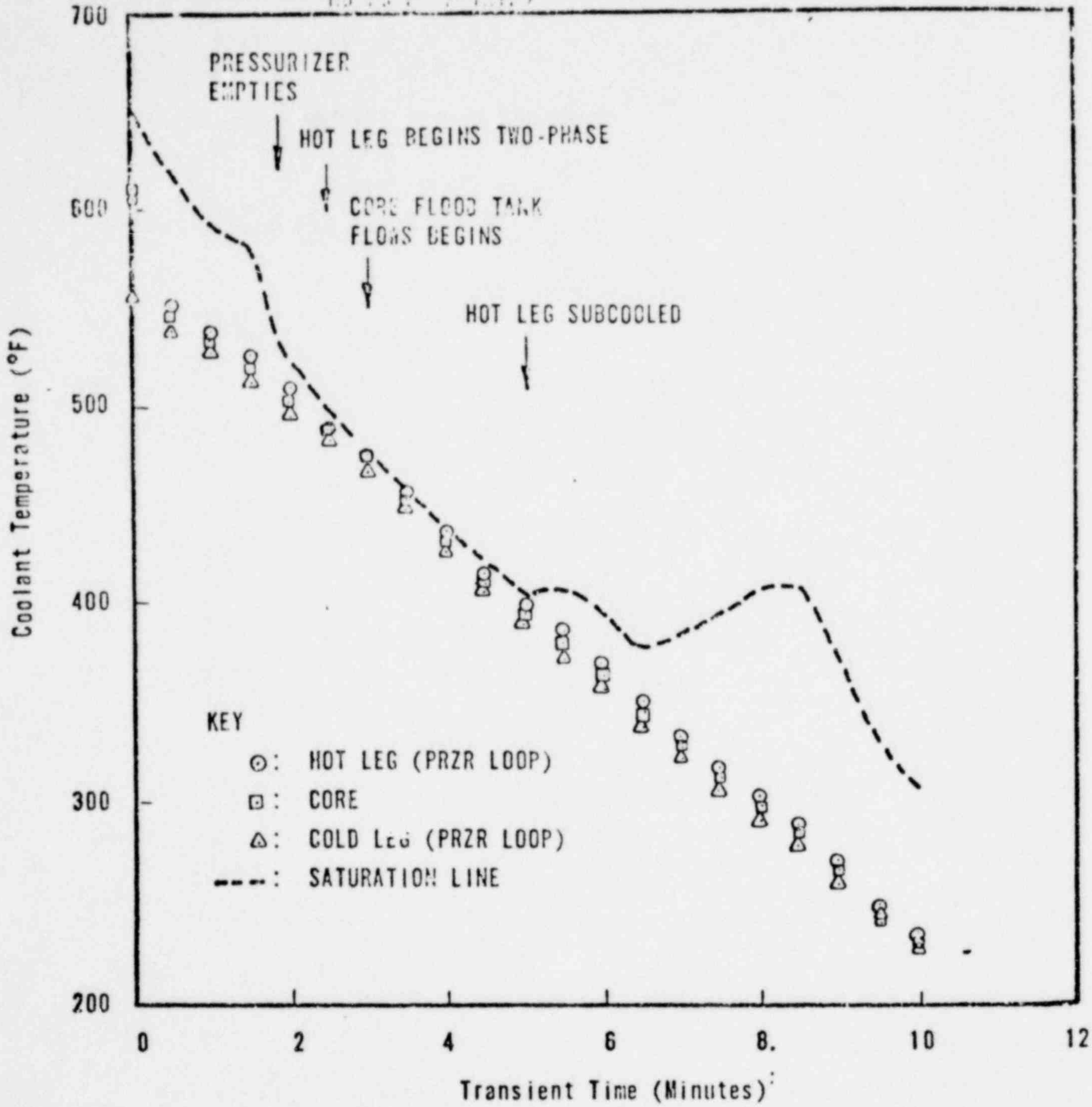


Figure 3.5

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

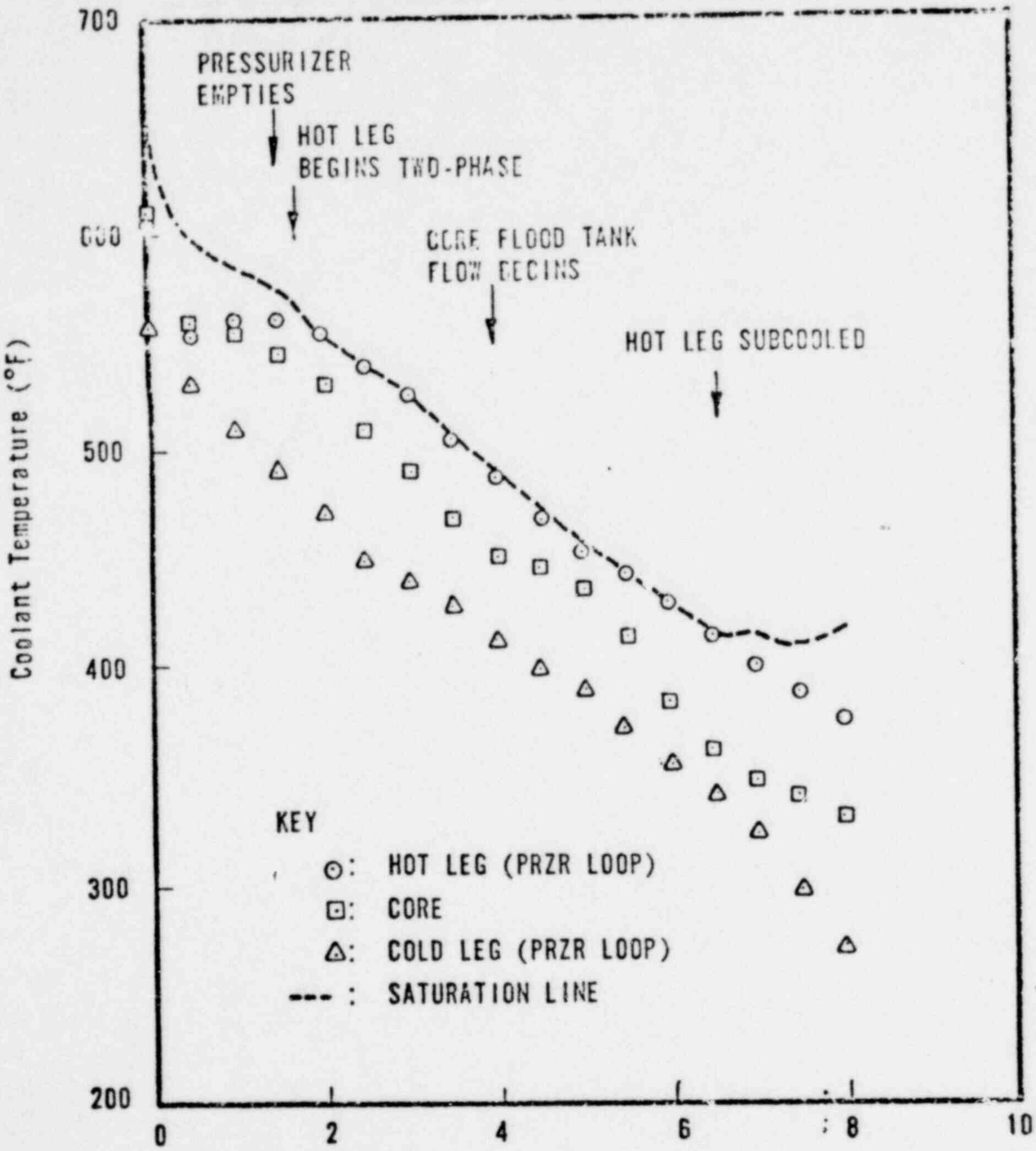


Figure 3.6

1054 105

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

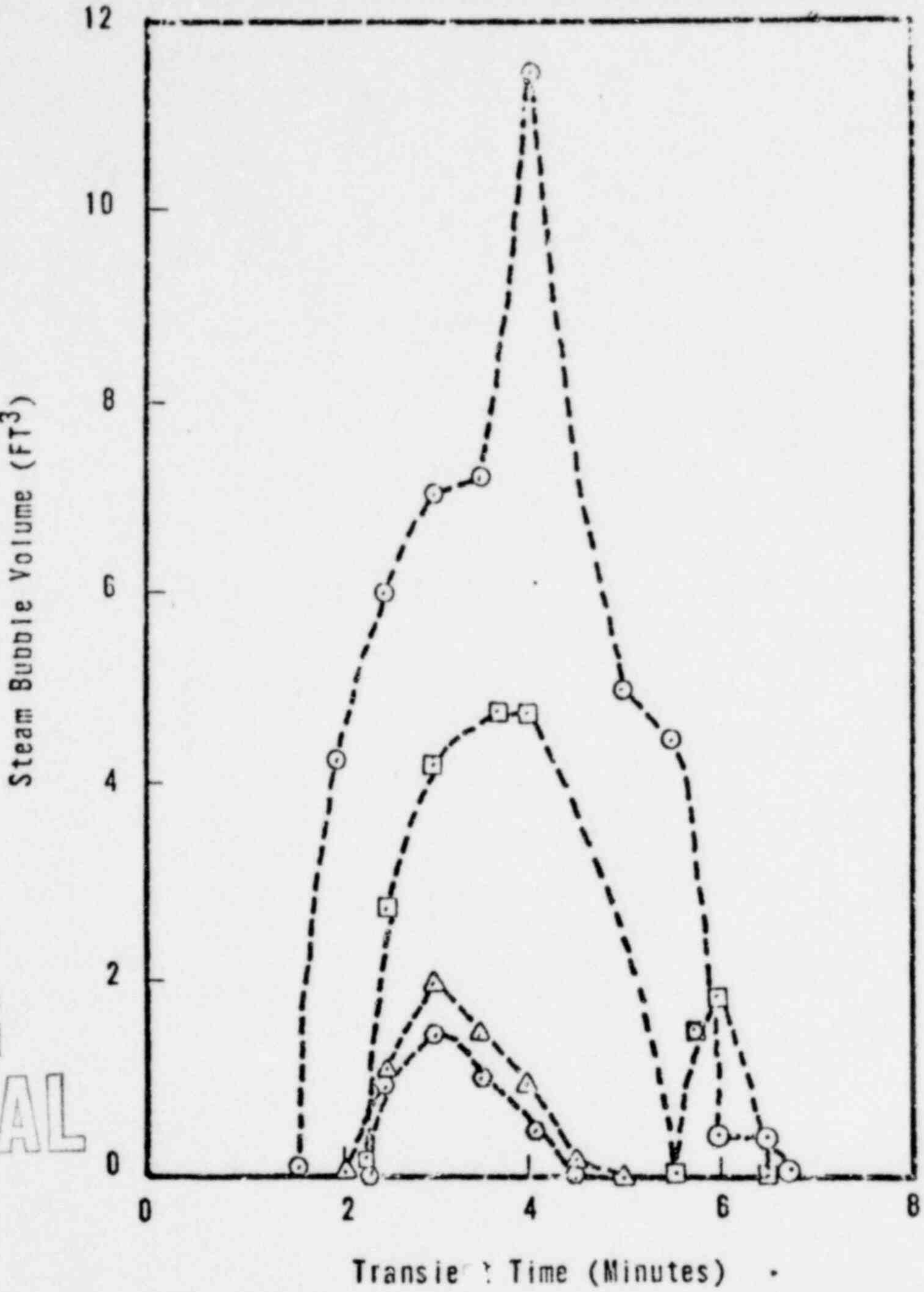


Figure 3.7

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

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

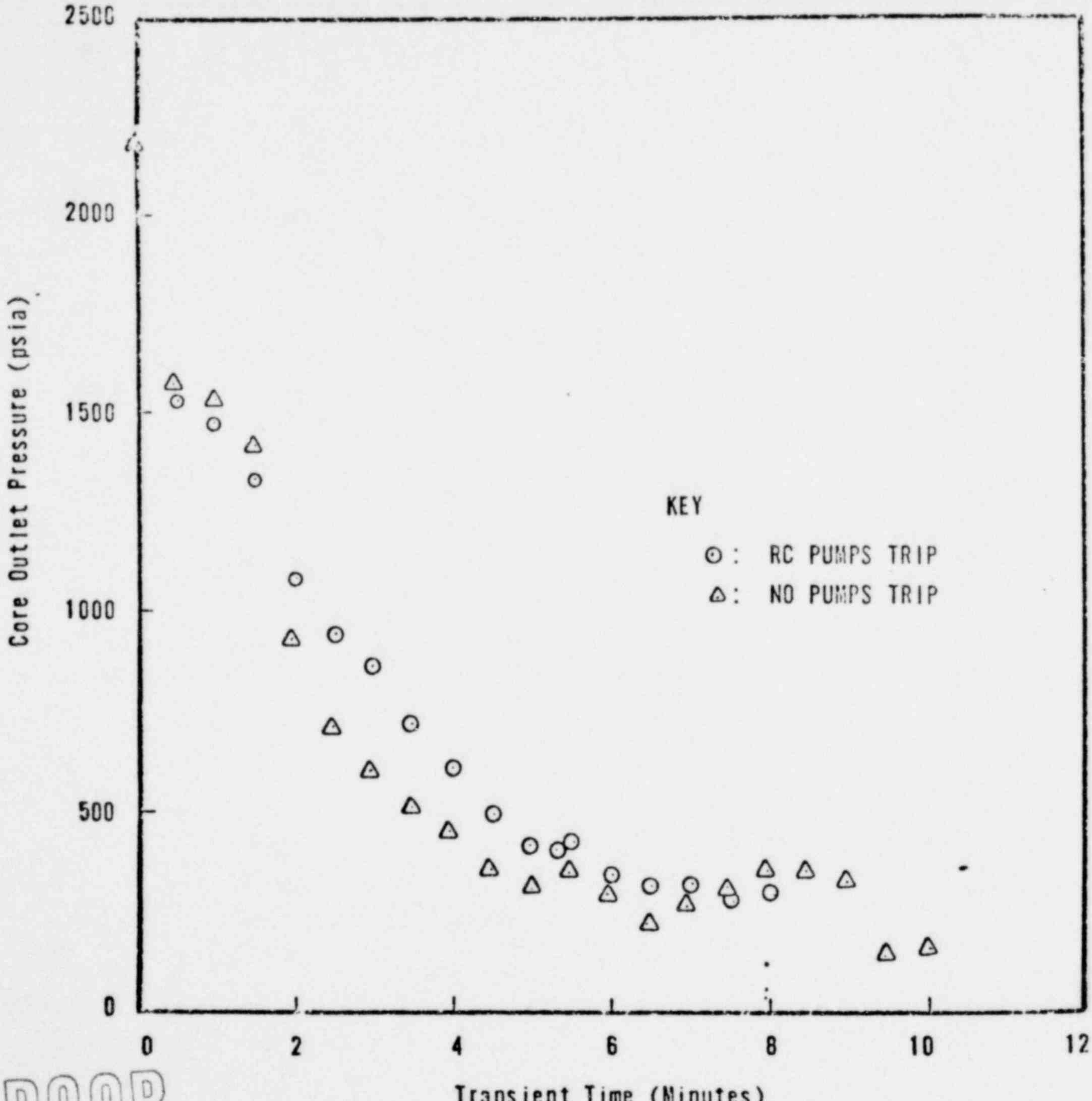

Figure 3.8

1054 10[illegible]

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

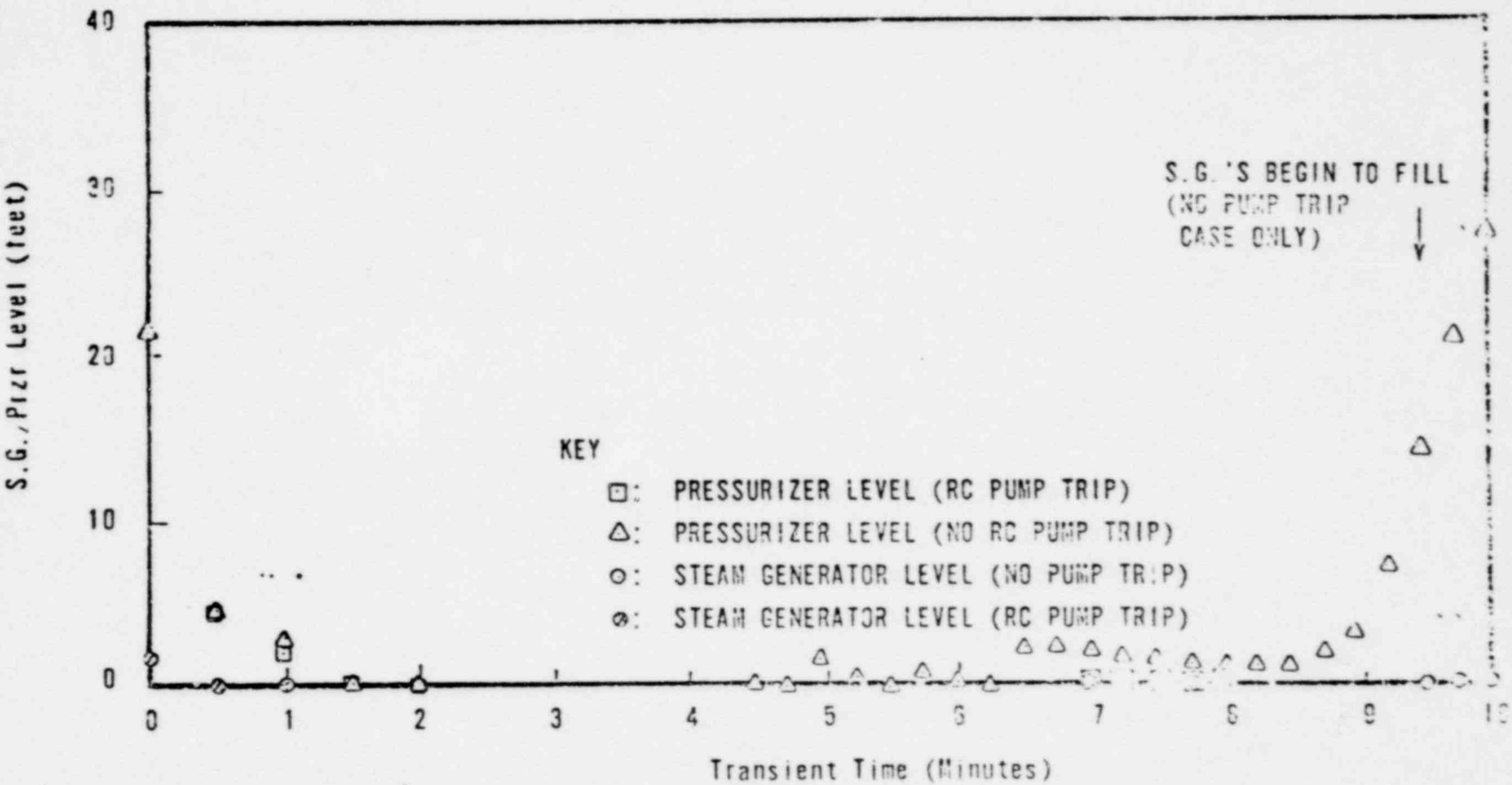

Figure 3.9

1054 108

REFERENCES

1 B.M. Dunn, et al., "B&W's ECCS Evaluation Model," BAW-10104, Rev. 3, August 1977.

2 Letter, J.H. Taylor (B&W to S.A. Varga (NRC), July 18, 1978.

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 Steam in a Bubbline Two-Phase Mixture," ANS Transactions, 5, (1962).

GUIDELINES FOR OPERATOR ACTION

I. Introduction

Guidance for operator action, during both LOCA and non-LOCA events, to account for the impact of the RC pump trip requirement of IE bulletin No. 79-05C, have been developed and are presented below. The general intent of these additional instructions is as follows:

1. To establish the basis and criteria for a RC pump trip and
2. To identify plant conditions for which a restart of the RC Pumps, if tripped, is permissable.

Section VI provides the "Operating Guidelines for Small Breaks" updated to include the impact of the RC pump trip requirements. These guidelines, in general, apply to any abnormal event where a RCP trip is required and will be used as the basis for revisions to emergency operating procedures and operator training.

II. Basis and Criteria for a RC Pump (RCP) Trip

B&W analyses of small loss-of-coolant accidents, with the RC pumps operative, indicated that the primary reactor coolant conditions evolve to high void fractions during the initial stages of the transient when the system pressure is still relatively high. The consequences of these postulated events with continouous RC pump operation are acceptable as effective core cooling is maintained due to the forced circulation of reactor coolant. For a certain range of small breaks, however, a RCP trip (by any means such as loss of power or operator action) at a time when the coolant void fraction is excessively high can lead to core uncovery and a potential for cladding temperatures in excess of 2200F.

To preclude the potential consequence of an untimely RCP trip, the RCP's will be promptly shutdown when RCS conditions indicate a small break in this size range may be in progress. This action ensures safe plant conditions as demonstrated by past small break analyses, under Appendix K assumptions, wherein the RC pumps were assumed inoperative early during the transient.

In the interim, until design changes can be made to automate the RCP trip, operating procedure will require that the operator trip the RCP's immediately following ESFAS actuation due to low RC pressure (≤ 1600 psig). Table 1 outlines the general diagnostic and confirmatory actions which will be required in addition to other immediate actions in present procedures. These immediate actions apply to any abnormal event which results in automatic ESFAS actuation on low RC pressure and will be memorized by reactor operating personnel during training programs.

As indicated above, a prompt trip of the RC pumps is required in order to maintain demonstrated conformance to 10CFR50.46. To provide good assurance that the operator will trip the RC pumps when required, the pump trip criteria (low pressure ESFAS actuation) was chosen over other possible candidates because it is a clear, simple, and early indication that a small LOCA may be in progress. The visual indication and alarms in the control room following ESFAS actuation also alert the operator to the status of the plant, and no decision process or continuous monitoring by the operator is required to decide that an RC pump trip is necessary. With procedure changes consistent with Table 1 and additional training, failure of the operate to initiate an RC pump trip when required is believed to be remote.

Table 1: IMMEDIATE ACTIONS REQUIRED FOLLOWING ESFAS ACTUATION

1. <u>Criteria for RCP Trip</u>

Upon automatic actuation of the ESFAS due to low reactor coolant system pressure, RC pump operation shall be promptly terminated.

2. <u>Immediate Action</u>

A. Upon receipt of an ESFAS actuation (indicated via audiable and visual alarms within the control room) the operator shall immediately verify that RC pressure is less than the low pressure ESFAS setpoint via examination of wide range RC pressure instrumentation or ESFAS Trip Status Indication, if available.

B. If RC pressure is less than the low pressure ESFAS setpoint, RC pump operation shall be immediately terminated by manual depressing the individual RC pump trip switches in the control room.

NOTE: If the ESFAS has been actuated due to high RB pressure, the operator shall monitor RC pressure and trip the RC pumps if pressure decreases below the ESFAS setpoint.

C. The operator shall immediately verify that the RC pumps are tripped by visual examination of RC pump status indications (status lights, motor current, etc.).

D. Following a trip of the RC pumps, the operator shall verify that the auxiliary feedwater system has been actuated and that SG level is controlled to the emergency high level control setpoint to ensure establishment of natural circulation.

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III. Criteria for RCP Restart

Plant control following abnormal events, including small breaks, is greatly improved if the RC pumps are operative. With forced circulation of reactor coolant, the steam generators and associated auxiliary systems are more effective in removing the primary system stored energy and decay heat. The plant is also placed in a more "normal" mode of operation where more familiar pressure/temperature control procedures can be employed by operating personnel. Therefore, to compliment the RC pump trip criteria provided in Section II, conditions under which an RC pump restart is allowed have also been identified. These conditions cover both LOCA and non-LOCA events and have been carefully chosen to preclude the development of excessive void fractions for small breaks where an RC pump restart is allowed.

Table 2 lists the conditions under which a RC pump restart is allowed. For each condition, typical events for which they apply and a brief discussion of the basis for the RC pump restart is provided. It should be noted that a RC pump restart is not allowed unless feedwater is available to at least one steam generator. A cross-reference to the appropriate sections of the small break guidelines where specific information can be found is also given. Furthermore, the criteria given in Table 2 are not new as each was previously issued in past small break guideline submittals. B&W has reviewed the guidelines in light of the break size and system conditions for which a RC pump trip is required and has confirmed that the RC pump restart guidance is still appropriate.

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As indicated in Table 2, system repressurization and the establishment of subcooled conditions are specified for use on non-LOCA events as criteria for which a RC pump restart is allowed. For these abnormal events, restart of the RC pumps is recommended by B&W when the Pump Restart criteria is satisfied to aid in plant recovery and control. Emergency procedures for non-

LOCA events, for which a RC pump trip may be initiated, will thus be revised to include the pump restart criteria.

TABLE 2: RC PUMP RESTART CRITERIA [1]

CONDITION FOR WHICH [2,3] A PUMP RESTART IS ALLOWED	TYPICAL EVENTS FOR WHICH A RCP RESTART IS ALLOWED	INSTRUCTION LOCATION IN SMALL BREAK GUIDELINES (SECTION)	DISCUSSION
Regain Coolant Subcooling 1. P-T conditions indicate coolant is ≥ 50F subcooled.	1. Small Leak 2. Small Break within capacity of HPI sys. 3. Isolated Small Break 4. Non-LOCA Overcooling/ depressurizing event 5. Loss-of-Offsite Power Event	4.3.4.3.2	Following any reactor trip event during which the RC pumps become inoperative (loss of power due to natural causes/ equipment failures or due to a deliberate trip initiated by the operator), the RC pumps can be restarted if RC conditions are stabilized and at least 50F of subcooling is indicated for the existing P-T state. If subcooled conditions are indicated, the primary and secondary systems are directly coupled (ie, decay heat removal via natural circulation); and if a breach of the primary pressure boundary is present also, the resulting leak will be within the capacity of the ECCS systems. The operator should restart the RC pumps (1 in. each loop) return to low SG Level control, and proceed with a plant cooldown or maintain the plant at hot shutdown if the initiating event is correctable and a return to power operation possible. NOTE: The subcooling criteria will be the principle indicator for a RCP restart for non-LOCA events.
Repressurization 1. Stable or increasing pressure with PRCS > 1600 psig.	1. Small Break within capacity of HPIS 2. Overcooling/Depressurization event 3. Isolated Small Break	4.3.4.4.1	Certain small breaks will result in a system repressurization due to momentary loss of the SG as a condensor for primary system steam (ie, the HPIS is refilling the system and a steam bubble is trapped within the hot legs above the SG tubes condensing surface). Small breaks which produce this primary system behavior are sufficiently small such that high void fractions will not evolve if the RC pumps are restarted. A RCP restart is thus allowed; this action will equalize primary and secondary pressures and temperatures and couple the primary and secondary systems such that an orderly cooldown and depressurization of the RCS can be accomplished. Section 4.3.4.4.1 of the small break guidelines would

CONDITION FOR WHICH[2,3] A PUMP RESTART IS ALLOWED	TYPICAL EVENTS FOR WHICH A RCP RESTART IS ALLOWED	INSTRUCTION LOCATION IN SMALL BREAK GUIDELINES (SECTION)	DISCUSSION
			apply to a very small break where a system repressurization would occur early (ie, prior to initiation of the secondary system depressurization). A RCP restart and resulting drop in the primary system pressure to that of the secondary side may allow the HPIS to establish a subcooled primary system. System repressurization above the low pressure ESFAS setpoint for non-LOCA events is also an acceptable condition for an RC pump restart. In most cases, increasing RC pressure will also tend to re-establish the reactor coolant subcooled margin which, as indicated above, is the principle indicator for a RCP restart for non-LOCA event. A pump restart, when system pressure is above the ESFAS setpoint when the 50F subcooled margin is not yet established, is permissable since small breaks for which a RC Pump trip is required will not produce the system behavior.
2. Increasing system pressure where PRCS > + 600 (psig) during cooldown process.	**Small Break**	4.3.4.4.2	4.3.4.4.2 of the small break guidelines applies during the cooldown process where the secondary pressure has been manually reduced below normal control (hot shutdown) setpoints. A pump bump procedure is stipulated. The intent of this action is to mix the system so that steam can be condensed to allow a system refill. If a refill and subcooled conditions are not established, the 600 psi decrease in primary system pressure will prevent high RCS void fractions with an RCP restart per the guidance provided.
Final Transition to LPI Cooling			
Stablized pressure with PSS< 100 psig and PRCS > 250 psig	**Small Break**	4.3.4.4.3	For certain small breaks, a primary system refill may not be possible until low primary system pressures are achieved. Complete depressurization may be impeded due to steam trapped within the upper hot leg piping. A bump of an RCP will depressurize the RCS such that a transition to LPI cooling per Appendix A of the small break guidelines is possible

TABLE 2 CONT'D

CONDITION FOR WHICH A PUMP RESTART IS ALLOWED[2,3]	TYPICAL EVENTS FOR WHICH A RCP RESTART IS ALLOWED	INSTRUCTION LOCATION IN SMALL BREAK GUIDELINES (SECTION)	DISCUSSION
			Continued operation of an RCP is also allowed since the LPI system will eliminate the potential for further increase in the system void fraction.
Inadequate Core Cooling	Small Break	N/A	Current considerations of the indications of and mitigating actions for inadequate core cooling may result in the potential use of the RC pumps under certain conditions. Criteria for use of the RC pumps, if required, will be developed consistent with the schedule requirement of Item 5 (short term) of 79-05C.

NOTE: 1. An RC Pump restart is allowed only if feedwater is available to at least one steam generator.

2. Standard precautions to be observed prior to pump restart.

   A. CCW has been maintained or will be reinstated prior to starting the RC pumps.
   B. Seal injection flow has been maintained to all RC pumps.
   C. Seal return is maintained or is reinstated prior to starting pumps.
   D. Prcs $\geq$ 250 psig.

3. Emergency operating limits for continued pump operation.

   A. Shaft runout (vibration) shall not exceed 30 mils.
   B. Frame vibration as measured on the lower motor mounting flange shall not exceed 5 mils.

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IV. Operating Guidelines for Small Break

Part I and Part II of the "Operating Guidelines for Small Breaks" have been revised to include the RC pump trip requirement of IE Bulletin 79-05C and are attached. This information will serve as the basis for revisions to emergency procedures and additional operator training.

PART I – OPERATING GUIDELINES FOR SMALL BREAKS

1.0 SYMPTOMS AND INDICATIONS (IMMEDIATE INDICATIONS)

1.1 EXCESSIVE REACTOR COOLANT SYSTEM (RCS) MAKEUP*

1.2 DECREASING RCS PRESSURE

1.3 REACTOR TRIP

1.4 DECREASING PRESSURIZER LEVEL*

1.5 ESFAS ACTUATION*

1.6 LOW MAKEUP TANK LEVEL*

*MAY NOT OCCUR ON ALL SMALL BREAKS

2.0 IMMEDIATE ACTIONS

o 2.1 IF THE ESFAS HAS BEEN INITIATED AUTOMATICALLY BECAUSE OF LOW RC PRESSURE, IMMEDIATELY SECURE ALL RC PUMPS.

2.2 VERIFY CONTROL ROOM INDICATIONS SUPPORT THE ALARMS RECEIVED, VERIFY AUTOMATIC ACTIONS, AND CARRY OUT STANDARD POST-TRIP ACTIONS.

2.3 BALANCE HIGH-PRESSURE INJECTION (HPI) FLOW BETWEEN ALL INJECTION LINES WHEN HPI IS INITIATED.

o 2.4 VERIFY THAT APPROPRIATE ONCE-THROUGH STEAM GENERATOR (OTSG) LEVEL IS MAINTAINED BY FEEDWATER CONTROL (LOW LEVEL LIMIT WITH RC PUMPS OPERATING, EMERGENCY LEVEL WITHOUT RC PUMPS OPERATING).

o 2.5 MONITOR SYSTEM PRESSURE AND TEMPERATURE. IF SATURATED CONDITIONS OCCUR, INITIATE HPI.

3.0 PRECAUTIONS

o 3.1 IF THE ESFAS HAS BEEN INITIATED ON LOW RC PRESSURE, TERMINATION OF RC PUMP OPERATION TAKES PRECEDENCE OVER ALL OTHER IMMEDIATE ACTIONS.

NOTE: IF ESFAS HAS BEEN ACTUATED ON HIGH RB PRESSURE, THEN MONITOR RC PRESSURE AND TRIP RC PUMPS ONCE PRESSURE DECREASES BELOW THE ESFAS LOW PRESSURE SETP(T.

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PART I — OPERATING GUIDELINES FOR SMALL BREAKS

1.0 SYMPTOMS AND INDICATIONS (IMMEDIATE INDICATIONS)

1.1 EXCESSIVE REACTOR COOLANT SYSTEM (RCS) MAKEUP*

1.2 DECREASING RCS PRESSURE

1.3 REACTOR TRIP

1.4 DECREASING PRESSURIZER LEVEL*

1.5 ESFAS ACTUATION*

1.6 LOW MAKEUP TANK LEVEL*

*MAY NOT OCCUR ON ALL SMALL BREAKS

2.0 IMMEDIATE ACTIONS

o 2.1 IF THE ESFAS HAS BEEN INITIATED AUTOMATICALLY BECAUSE OF LOW RC PRESSURE, IMMEDIATELY SECURE ALL RC PUMPS.

2.2 VERIFY CONTROL ROOM INDICATIONS SUPPORT THE ALARMS RECEIVED, VERIFY AUTOMATIC ACTIONS, AND CARRY OUT STANDARD POST-TRIP ACTIONS.

2.3 BALANCE HIGH-PRESSURE INJECTION (HPI) FLOW BETWEEN ALL INJECTION LINES WHEN HPI IS INITIATED.

o 2.4 VERIFY THAT APPROPRIATE ONCE-THROUGH STEAM GENERATOR (OTSG) LEVEL IS MAINTAINED BY FEEDWATER CONTROL (LOW LEVEL LIMIT WITH RC PUMPS OPERATING, EMERGENCY LEVEL WITHOUT RC PUMPS OPERATING).

o 2.5 MONITOR SYSTEM PRESSURE AND TEMPERATURE. IF SATURATED CONDITIONS OCCUR, INITIATE HPI.

3.0 PRECAUTIONS

o 3.1 IF THE ESFAS HAS BEEN INITIATED ON LOW RC PRESSURE, TERMINATION OF RC PUMP OPERATION TAKES PRECEDENCE OVER ALL OTHER IMMEDIATE ACTIONS.

NOTE: IF ESFAS HAS BEEN ACTUATED ON HIGH RB PRESSURE, THEN MONITOR RC PRESSURE AND TRIP RC PUMPS ONCE PRESSURE DECREASES BELOW THE ESFAS LOW PRESSURE SETPOINT.

°Revised

3.2 IF ESFAS HAS BEEN INITIATED, THE RC PUMP TRIPPED, AND THE RCS DETERMINED TO BE AT LEAST 50 F SUBCOOLED. THE OPERATOR SHOULD ESTABLISH AS QUICKLY AS POSSIBLE IF THE CAUSE FOR THE DEPRESSURIZATION IS DUE TO EITHER A LOCA OR NON-LOCA (OVERCOOLING) EVENT. PROCEED TO STEP 4.4 FOR NON-LOCA EVENTS.

3.3 IF THE HPI SYSTEM HAS ACTUATED BECAUSE OF LOW PRESSURE CONDITIONS, IT MUST REMAIN IN OPERATION UNTIL ONE OF THE FOLLOWING CRITERIA IS SATISFIED:

1. THE LPI SYSTEM IS IN OPERATION AND FLOWING AT A RATE IN EXCESS OF 1000 GPM IN EACH LINE AND THE SITUATION HAS BEEN STABLE FOR 20 MINUTES.

OR

2. ALL HOT AND COLD LEG TEMPERATURES ARE AT LEAST 50 F BELOW THE SATURATION TEMPERATURE FOR THE EXISTING RCS PRESSURE, THE HOT LEG TEMPERATURES ARE NOT MORE THAN 50 F HOTTER THAN THE SECONDARY SIDE SATURATION TEMPERATURE, AND THE ACTION IS NECESSARY TO PREVENT THE INDICATED PRESSURIZER LEVEL FROM GOING OFF-SCALE HIGH. IF 50 F SUBCOOLING CANNOT BE MAINTAINED, THE HPI SHALL BE REACTIVATED. THE DEGREE OF SUBCOOLING BEYOND 50 F AND THE LENGTH OF TIME HPI IS IN OPERATION SHALL BE LIMITED BY THE PRESSURE/TEMPERATURE CONSIDERATIONS FOR THE VESSEL INTEGRITY (SEE SECTION 3.4). °

3.4 WHEN THE REACTOR COOLANT IS $\geq$ 50 F SUBCOOLED, THE REACTOR VESSEL DOWNCOMER PRESSURE/TEMPERATURE (P-T) COMBINATION SHALL BE KEPT BELOW AND TO THE RIGHT OF THE LIMIT CURVE SHOWN IN FIGURE 1. THE DOWNCOMER TEMPERATURE SHALL BE DETERMINED AS FOLLOWS: °

3.4.1 WITH ONE OR MORE RC PUMPS OPERATING USE ANY COLD LEG RTD AS AN INDICATION OF REACTOR VESSEL DOWNCOMER TEMPERATURE.

3.4.2 WITH NO RC PUMPS OPERATING THE RV DOWNCOMER TEMPERATURE SHALL BE DETERMINED BY AVERAGING THE FIVE LOWEST INCORE THERMOCOUPLE TEMPERATURE READINGS AND SUBTRACTING 150 F FROM THE AVERAGE INCORE THERMOCOUPLE TEMPERATURE VALUE.

°Revised

$$T_{DWN} = \frac{\sum^{5} T_{tc}}{5} - 150 \text{ F}$$

WHERE T_{DWN} = AVERAGE RV DOWNCOMER TEMPERATURE, F

$\sum^{5} T_{tc}$ = SUM OF THE 5 LOWEST INCORE THERMOCOUPLE TEMPERATURE READINGS.

NOTE: FIGURE 1 IS APPLICABLE ONLY UNDER LOCA CONDITIONS. THE P/T CURVE IN THE TECHNICAL SPECIFICATION IS VALID FOR ALL OTHER OPERATING CONDITIONS.

NOTE: WHEN THE REACTOR COOLANT IS LESS THAN 50 F SUBCOOLED, THE REACTOR VESSEL DOWNCOMER PRESSURE TEMPERATURE COMBINATION WILL INHERENTLY BE BELOW AND TO THE RIGHT OF THE LIMIT CURVE. THEREFORE, NO OPERATOR ACTION WILL BE REQUIRED TO PREVENT EXCEEDING THE REACTOR VESSEL INTEGRITY LIMITS UNTIL AFTER A $\geq$ 50 F SUBCOOLED MARGIN EXISTS.

NOTE: WHEN THE REACTOR COOLANT IS ≥ 50°F SUBCOOLED, RC PRESSURE CAN BE REDUCED BY REDUCING THE HPI FLOW RATE TO AVOID EXCEEDING THE RV INTEGRITY LIMITS.

3.5 PRESSURIZER LEVEL MAY BE INCREASING DUE TO RCS REACHING SATURATED CONDITIONS OR A BREAK ON TOP OF THE PRESSURIZER.

3.6 IF HIGH ACTIVITY IS DETECTED IN A STEAM GENERATOR, ISOLATE THE LEAKING GENERATOR. IT IS RECOMMENDED THAT BOTH STEAM GENERATORS NOT BE ISOLATED.

3.7 OTHER INDICATIONS WHICH CAN CONFIRM THE EXISTENCE OF A LOCA:

3.7.1 RC DRAIN TANK (QUENCH TANK) PRESSURE (RUPTURE DISK MAY BE BLOWN).

3.7.2 INCREASING REACTOR BUILDING SUMP LEVEL.

3.7.3 INCREASING REACTOR BUILDING TEMPERATURE.

3.7.4 INCREASING REACTOR BUILDING PRESSURE.

3.7.5 INCREASING RADIATION MONITOR READINGS INSIDE CONTAINMENT

3.7.6 REACTOR COOLANT SYSTEM TEMPERATURE BECOMING SATURATED RELATIVE TO THE RCS PRESSURE.

3.7.7 HOT LEG TEMPERATURE EQUALS OR EXCEEDS PRESSURIZER TEMPERATURE.

3.8 HPI COOLING REQUIREMENTS COULD DEPLETE THE BORATED WATER STORAGE TANK, AND INITIATION OF LPI FLOW FROM THE REACTOR BUILDING SUMP TO THE HPI PUMPS WOULD BE REQUIRED.

3.9 ALTERNATE INSTRUMENT CHANNELS SHOULD BE CHECKED AS AVAILABLE TO CONFIRM KEY PARAMETER READINGS (IE, SYSTEM TEMPERATURES, PRESSURES AND PRESSURIZER LEVEL).

3.10 MAINTAIN A TEMPERATURE VERSUS TIME PLOT AND A CORRESPONDING TEMPERATURE PRESSURE PLOT ON A SATURATION DIAGRAM. THESE PLOTS WILL MAKE IT POSSIBLE TO TRACK THE PLANT'S CONDITION THROUGH PLANT COOLDOWN. PRIMARY TEMPERATURE AND PRESSURE WILL DECREASE ALONG THE SATURATION CURVE UNTIL SUBCOOLED CONDITIONS ARE ESTABLISHED. THIS WILL BE INDICATED BY PRIMARY SYSTEM PRESSURE NO LONGER FOLLOWING THE SATURATION CURVE, AS PRIMARY SYSTEM TEMPERATURE DECREASES. WHEN THIS OCCURS, PRIMARY SYSTEM PRESSURE SHOULD BE CONTROLLED BY ADJUSTING HPI FLOW, TO MAINTAIN 50°F SUBCOOLING. o THE DEGREE OF SUBCOOLING BEYOND 50°F SHALL BE CONTROLLED WITHIN THE LIMITS DEFINED IN SECTION 3.4.

o 3.11 COMPONENT COOLING WATER (CCW) AND SEAL INJECTION SHOULD BE MAINTAINED TO THE RC PUMPS TO INSURE CONTINUED SERVICE OR THE ABILITY TO RESTART THE PUMPS AT A LATER TIME.

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o 3.11.1 IF CCW IS LOST AND THE RC PUMPS ARE OPERATIVE, CCW MUST BE RESTORED WITHIN 30 MINUTES OR THE RC PUMPS MUST BE MANUALLY TRIPPED.

o 3.11.2 IF THE RC PUMPS ARE TRIPPED FOR ANY REASON, SEAL INJECTION SHOULD BE MAINTAINED TO ENSURE LONG TERM SEAL INTEGRITY.

4.0 FOLLOWUP ACTIONS

4.1 IDENTIFICATION AND EARLY CONTROL

4.1.1 IF HPI HAS INITIATED BECAUSE OF LOW PRESSURE, CONTROL HPI IN ACCORDANCE WITH STEP 3.2.

4.1.2 IF BOTH HPI TRAINS HAVE NOT ACTUATED ON ESFAS SIGNAL, START SECOND HPI TRAIN IF POSSIBLE. BALANCE HPI FLOWS.

4.1.3 IF RC PRESSURE DECREASES CONTINUOUSLY, VERIFY THAT CORE FLOOD TANKS (CFTs) AND LOW PRESSURE INJECTION (LPI) HAVE ACTUATED AS NEEDED, AND BALANCE LPI.

o 4.1.4 IF CAUSE FOR COOLDOWN/DEPRESSURIZATION IS DETERMINED TO BE DUE TO A NON-LOCA OVERCOOLING EVENT AND THE RCS IS AT LEAST 50 F SUBCOOLED THEN PROCEED TO SECTION 4.4.

4.1.5 ATTEMPT TO LOCATE AND ISOLATE LEAK IF POSSIBLE. LETDOWN WAS ISOLATED IN STEP 2.1. OTHER ISOLATABLE LEAKS ARE PORV (CLOSE BLOCK VALVE) AND BETWEEN VALVES IN SPRAY LINE (CLOSE SPRAY AND BLOCK VALVE).

4.1.6 DETERMINE AVAILABILITY OF REACTOR COOLANT PUMPS (RCPs) AND MAIN AND AUXILIARY FEEDWATER SYSTEMS. IF FEEDWATER IS NOT AVAILABLE GO TO 4.2. IF FEEDWATER IS AVAILABLE GO TO 4.3.

4.2 ACTIONS IF FEEDWATER IS NOT AVAILABLE

4.2.1 THROUGHOUT THE FOLLOWING STEPS MAINTAIN MAXIMUM HPI FLOW AND RESTORE FEEDWATER AS SOON AS POSSIBLE.

o 4.2.2 IF RCPs ARE OPERATING, GO TO ONE PUMP PER LOOP. IF RCPs ARE NOT OPERATING, GO TO STEP 4.2.5 BELOW.

4.2.3 IF RCS PRESSURE INCREASES, OPEN PORV AND LEAVE OPEN.

NOTE: IF THE PORV CANNOT BE ACTUATED, THE SAFETIES WILL RELIEVE PRESSURE.

o Revised

o 4.2.4 WHEN FEEDWATER IS RECOVERED, RESTORE OTSG LEVELS IN A CONTROLLED MANNER. CLOSE PORV OR BLOCK VALVE, IF POSSIBLE, AND PROCEED TO STEP 4.3.2.

4.2.5 IF NO RCPs ARE OPERATING, OPEN PORV AND MAINTAIN HPI FLOW.

NOTE: IF THE PORV CANNOT BE ACTUATED, THE SAFETIES WILL RELIEVE PRESSURE.

4.2.6 WHEN FEEDWATER FLOW IS RESTORED, RAISE OTSG LEVELS TO 95% ON THE OPERATE RANGE, CLOSE PORV OR BLOCK VALVE, IF POSSIBLE.

NOTE: <u>OTSG LEVEL SHOULD BE MONITORED PERIODICALLY DURING THE FILL PROCESS. LEVELS $\geq$ 95% ON THE OPERATING RANGE MUST BE AVOIDED TO PRECLUDE FEEDWATER CARRYOVER TO THE STEAMLINES.</u>

4.2.7 VERIFY NATURAL CIRCULATION IN THE RCS BY OBSERVING:

4.2.7.1 COLD LEG TEMPERATURE IS SATURATION TEMPERATURE OF SECONDARY SIDE PRESSURE WITHIN APPROXIMATELY 5 MINUTES.

4.2.7.2 PRIMARY ΔT (THOT - TCOLD) BECOMES CONSTANT

4.2.8 GO TO STEP 4.3.4.1.

4.3 <u>ACTIONS WITH FEEDWATER AVAILABLE TO ONE OR BOTH GENERATORS</u>

4.3.1 MAINTAIN ONE RCP RUNNING PER LOOP (STOP OTHER RCPs). IF NO RCPs OPERATING (DUE TO A LOSS OF OFFSITE POWER OR DUE TO MANUAL SECUREMENT PER SECTION 2.0), GO TO STEP 4.3.4 BELOW.

4.3.2 ALLOW RCS PRESSURE TO STABILIZE.

4.3.3 ESTABLISH AND MAINTAIN OTSG COOLING BY ADJUSTING STEAM PRESSURE VIA TURBINE BYPASS AND/OR ATMOSPHERIC DUMPS. COOLDOWN AT 100 F PER HOUR TO ACHIEVE AN RC PRESSURE OF 250 PSIG. REFER TO PRECAUTION 3.10 FOR DEVELOPMENT OF TEMPERATURE AND PRESSURE PLOTS. ISOLATE CORE FLOOD TANKS WHEN 50 F SUBCOOLING IS ATTAINED AND RC PRESSURE IS LESS THAN 700 PSIG. GO INTO LPI COOLING PER APPENDIX A.

o 4.3.4 IF RCPs ARE NOT OPERATING:

4.3.4.1 ESTABLISH AND CONTROL OTSG LEVEL TO 95% ON THE OPERATE RANGE. VERIFY THE CONDITIONS IN STEP 4.2.7.

NOTE: <u>OTSG LEVELS GREATER THAN 95% ON THE OPERATING RANGE MUST BE AVOIDED TO PRECLUDE FEEDWATER CARRYOVER INTO THE STEAMLINES.</u>

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4.3.4.2 IF RC PRESSURE IS DECREASING, WAIT UNTIL IT STABILIZES OR BEGINS INCREASING. IF IT BEGINS INCREASING, GO TO STEP 4.3.4.4.

4.3.4.3 PROCEED WITH A CONTROLLED COOLDOWN AT 100 F/HR BY CONTROLLING STEAM GENERATOR SECONDARY SIDE PRESSURE. MONITOR RC PRESSURES AND TEMPERATURES DURING COOLDOWN AND PROCEED AS INDICATED BELOW:

4.3.4.3.1 IF RC PRESSURE CONTINUES TO DECREASE, FOLLOWING SECONDARY SIDE PRESSURE DECREASES AND WITH PRIMARY SYSTEM TEMPERATURES INDICATING SATURATED CONDITIONS, CONTINUE COOLDOWN UNTIL AN RC PRESSURE OF 150 PSI IS REACHED, AND PROCEED TO STEP A.4 OF APPENDIX A.

4.3.4.3.2 IF RC PRESSURE STOPS DECREASING IN RESPONSE TO SECONDARY SIDE PRESSURE DECREASE AND REACTOR SYSTEM BECOMES <u>SUBCOOLED</u>, CHECK TO SEE THAT THE FOLLOWING CONDITIONS ARE BOTH SATISFIED:

A) ALL HOT AND COLD LEG TEMPERATURES ARE BELOW THE SATURATION TEMPERATURE FOR THE EXISTING RCS PRESSURE.

AND B) RCS HOT LEG TEMPERATURES ARE NOT MORE THAN 50 F HOTTER THAN THE STEAM GENERATOR SECONDARY SIDE SATURATION TEMPERATURE.

IF THESE CONDITIONS ARE SATISFIED AND REMAIN SATISFIED, CONTINUE COOLDOWN TO ACHIEVE AN RCS TEMPERATURE (COLD LEG) OF 280 F, AND PROCEED TO STEP A.1 OF APPENDIX A.

NOTE: IF THE CONDITIONS ABOVE ARE MET BELOW 700 PSIG, THE CORE FLOOD TANKS SHOULD BE ISOLATED.

NOTE: IF THE PRIMARY SYSTEM IS 50 F SUBCOOLED IN BOTH HOT AND COLD LEGS AND PRIMARY

SYSTEM PRESSURE IS ABOVE 250 PSIG, STARTING A REACTOR COOLANT PUMP IS PERMISSIBLE. IF SYSTEM DOES NOT RETURN TO AT LEAST 50 F SUBCOOLING IN TWO MINUTES, TRIP PUMPS. IF FORCED CIRCULATION IS ACHIEVED, PROCEED TO STEP 4.3.

4.3.4.3.3 IF RC PRESSURE STOPS DECREASING AND THE CONDITIONS OF 4.3.4.3.2 ARE NOT MET OR CEASE TO BE MET OR IF RC PRESSURE BEGINS TO INCREASE, THEN PROCEED TO STEP 4.3.4.4 BELOW.

4.3.4.4 RESTORE RCP FLOW (ONE PER LOOP) WHEN POSSIBLE PER THE INSTRUCTIONS BELOW. IF RC PUMPS CANNOT BE OPERATED AND PRESSURE IS INCREASING, GO TO STEP 4.3.4.6.

4.3.4.4.1 IF PRESSURE IS INCREASING, STARTING A PUMP IS PERMISSIBLE AT RC PRESSURE GREATER THAN 1600 PSIG.

4.3.4.4.2 IF REACTOR COOLANT SYSTEM PRESSURE EXCEEDS STEAM GENERATOR SECONDARY PRESSURE BY 600 PSIG OR MORE "BUMP" ONE REACTOR COOLANT PUMP FOR A PERIOD OF APPROXIMATELY 10 SECONDS (PREFERABLY IN OPERABLE STEAM GENERATOR LOOP). ALLOW REACTOR COOLANT SYSTEM PRESSURE TOSTABILIZE. CONTINUE COOLDOWN. IF REACTOR COOLANT SYSTEM PRESSURE AGAIN EXCEEDS SECONDARY PRESSURE BY 600 PSI, WAIT AT LEAST 15 MINUTES AND REPEAT THE PUMP "BUMP". BUMP ALTERNATE PUMPS SO THAT NO PUMP IS BUMPED MORE THAN ONCE IN AN HOUR. THIS MAY BE REPEATED, WITH AN INTERVAL OF 15 MINUTES, UP TO 5 TIMES. AFTER THE FIFTH "BUMP," ALLOW THE REACTOR COOLANT PUMP TO CONTINUE IN OPERATION.

4.3.4.4.3 IF PRESSURE HAS STABILIZED FOR GREATER THAN ONE HOUR, SECONDARY PRESSURE IS LESS THAN 100 PSIG AND PRIMARY PRESSURE IS GREATER THAN 250 PSIG, BUMP A PUMP, WAIT 30 MINUTES, AND START AN ALTERNATE PUMP.

4.3.4.5 IF FORCED FLOW IS ESTABLISHED, GO TO STEP 4.3.3.

4.3.4.6 IF A REACTOR COOLANT PUMP CANNOT BE OPERATED AND REACTOR COOLANT SYSTEM PRESSURE REACHES 2300 PSIG, OPEN PRESSURIZER PORV TO REDUCE REACTOR COOLANT SYSTEM PRESSURE. RECLOSE PORV WHEN RCS PRESSURE FALLS TO 100 PSI ABOVE THE SECONDARY PRESSURE. REPEAT IF NECESSARY. IF PORV IS NOT OPERABLE, PRESSURIZER SAFETY VALVES WILL RELIEVE OVERPRESSURE.

4.3.4.7 MAINTAIN RC PRESSURE AS INDICATED IN 4.3.4.6 IF PRESSURE INCREASES. MAINTAIN THIS COOLING MODE UNTIL AN RC PUMP IS STARTED OR STEAM GENERATOR COOLING IS ESTABLISHED AS INDICATED BY ESTABLISHING CONDITIONS DESCRIBED IN 4.3.4.3.1 OR 4.3.4.3.2. WHEN THIS OCCURS, PROCEED AS DIRECTED IN THOSE STEPS. GO TO STEP 4.3.2 IF FORCED FLOW IS ESTABLISHED.

4.4 NON-LOCA OVERCOOLING TRANSIENT WITH FEEDWATER AVAILABLE

4.4.1 IMMEDIATELY RESTART A RC PUMP IN EACH LOOP IF THE RCS IS 50 F SUBCOOLED.

4.4.2 RESTORE CONTROL OF FEEDWATER FLOW AND GENERATOR LEVEL AND CONTROL STEAM PRESSURE VIA TURBINE BYPASS OR ATMOSPHERIC DUMP VALVES TO STABILIZE OR CONTROL PLANT HEATUP.

NOTE: CONSIDERABLE HPI MAY HAVE BEEN ADDED TO THE RCS. THEREFORE, TO PREVENT RCS FROM GOING SOLID, THE ABOVE ACTION MAY BE NECESSARY.

4.4.3 AS LONG AS THE RCS IS MAINTAINED 50 F SUBCOOLED, THROTTLE HPI AND LETDOWN FLOW TO MAINTAIN PZR LEVEL AT 100 INCHES.

4.4.4 USING TURBINE BYPASS VALVES AND FEEDWATER SYSTEM, CONTROL STEAM GENERATORS AS NEEDED TO LIMIT PLANT HEATUP UNTIL RC PRESSURE CONTROL CAN BE RE-ESTABLISHED WITH THE PRESSURIZER.

NOTE: COLD RCS WATER MAY HAVE BEEN ADDED TO THE PRESSURIZER; THEREFORE, A PERIOD OF TIME MAY ELAPSE BEFORE NORMAL RC PRESSURE CONTROL CAN BE ESTABLISHED WITH THE PRESSURIZER HEATERS.

4.4.5 ONCE PRESSURE CONTROL IS RE-ESTABLISHED, USE NORMAL HEATUP/COOLDOWN PROCEDURE TO ESTABLISH DESIRED PLANT CONDITIONS.

APPENDIX A

LPI COOLING

A.1 DETERMINE IF PRIMARY COOLANT IS AT LEAST 50 F SUBCOOLED. IF NOT GO TO STEP A.3.

A.1.1 START LPI PUMPS. IF BOTH PUMPS ARE OPERABLE GO TO STEP A.2. FOR ONE LPI PUMP OPERABLE MAINTAIN OTSG COOLING AS FOLLOWS. THE OPERABLE LPI PUMP WILL BE USED TO MAINTAIN SYSTEM INVENTORY.

A.1.2 OBTAIN PRIMARY SYSTEM CONDITIONS OF 280 F AND 250 PSIG.

A.1.3 ALIGN THE DISCHARGE OF THE OPERABLE LPI PUMP TO THE SUCTIONS OF THE HPI PUMPS AND TAKE SUCTION FROM THE BWST. IF THE BWST IS AT THE LOW LEVEL ALARM, ALIGN LPI SUCTION FROM THE RB SUMP AND SHUT SUCTION FROM BWST.

A.1.4 START THE OPERABLE LPI PUMP SPECIFIED ABOVE. THE HPI-LPI SYSTEMS WILL NOW BE IN "PIGGY BACK" AND HPI FLOW IS MAINTAINING SYSTEM PRESSURE.

A.1.5 GO TO SINGLE RC PUMP OPERATION.

A.1.6 WHEN THE SECOND LPI PUMP IS AVAILABLE ALIGN IT IN THE DECAY HEAT MODE AND COMMENCE DECAY HEAT REMOVAL. (DECAY HEAT SYSTEM FLOW GREATER THAN 1000 GPM). SECURE REMAINING RC PUMP WHEN DECAY HEAT REMOVAL IS ESTABLISHED.

<u>CAUTION</u>: VERIFY THAT ADEQUATE NPSH EXISTS FOR THE DECAY HEAT PUMP IN THE DH REMOVAL MODE. IF INADEQUATE, TRANSFER TO LPI MODE.

A.1.7 REDUCE REACTOR COOLANT PRESSURE TO 150 PSIG BY THROTTLING HPI FLOW. CONTROL RC TEMPERATURE USING THE DECAY HEAT SYSTEM COOLER BYPASS TO MAINTAIN SYSTEM PRESSURE AT LEAST 50 PSI ABOVE SATURATION PRESSURE, TO ASSURE THAT NPSH REQUIREMENTS FOR THE DECAY HEAT PUMP ARE MAINTAINED.

A.1.8 SECURE THE HPI PUMP AND SHIFT THE LPI PUMP SUPPLYING IT TO THE LPI INJECTION MODE.

A.1.9 REDUCE REACTOR COOLANT TEMPERATURE TO 100 F BY CONTROLLING THE DECAY HEAT SYSTEM COOLER BYPASS.

<u>NOTE</u>: IF ONE OF THE LPI/DECAY HEAT PUMPS IS LOST, RETURN TO OTSG COOLING USING NATURAL CIRCULATION OR ONE REACTOR COOLANT PUMP (A1).

A.2. <u>COOLDOWN ON TWO LPI PUMPS</u>

A.2.1 MAINTAIN RCS PRESSURE AT 250 PSIG AND REDUCE RCS TEMPERATURE TO 280 F.

A.2.2 ALIGN ONE LPI PUMP IN THE DECAY HEAT REMOVAL MODE.

A.2.3 SECURE ONE RC PUMP IF TWO ARE OPERATING.

A.2.4 START THE DECAY HEAT PUMP IN THE DECAY HEAT REMOVAL MODE, AND WHEN DECAY HEAT SYSTEM FLOW IS GREATER THAN 1000 GPM, SECURE THE RUNNING RC PUMP.

A.2.5 REDUCE RC PRESSURE TO 150 PSIG BY THROTTLING HPI FLOW. CONTROL RC TEMPERATURE TO MAINTAIN AT LEAST 50 PSI MARGIN TO SATURATION PRESSURE.

A.2.6 START THE SECOND LPI PUMP IN THE LPI INJECTION MODE. SECURE HPI PUMP.

A.2.7 SHIFT LPI SUCTION FROM THE BWST TO THE REACTOR BUILDING SUMP WHEN SUFFICIENT NPSH IS AVAILABLE.

<u>NOTE</u>: THIS IS DESIRABLE TO AVOID UNNECESSARY QUANTITIES OF WATER IN CONTAINMENT.

A.2.8 REDUCE REACTOR COOLANT TEMPERATURE TO 100 F BY CONTROLLING THE DECAY HEAT SYSTEM COOLER BYPASS.

<u>NOTE</u>: IF ONE OF THE LPI/DECAY HEAT PUMPS IS LOST, RETURN TO OTSG COOLING USING NATURAL CIRCULATION OR ONE RC PUMP PER A.1.

A.3 COOL DOWN RC SYSTEM AT SATURATION

A.3.1 MAINTAIN RC PRESSURE AT 250 PSIG.

A.3.2 ALIGN ONE LPI PUMP TO SUCTION OF THE HPI PUMPS AND THE SUCTION TO THE REACTOR BUILDING SUMP. (SHUT BWST SUCTION VALVE FOR THIS PUMP.)

A.3.3 WHEN THE BWST LEVEL REACHES THE LO-LO LEVEL LIMITS, START THE LPI PUMP AND SHUT THE HPI PUMP SUCTION FROM THE BWST.

A.3.4 WHEN PRIMARY SYSTEM TEMPERATURE BECOMES SUBCOOLED BY AT LEAST 50 F, GO TO A.1.1.

A.4 COOLDOWN WITHOUT REACTOR COOLANT PUMPS

A.4.1 RCS INITIAL CONDITIONS ARE: PRESSURE 150 PSI, TEMPERATURE AT SATURATION.

A.4.2 ALIGN LOW PRESSURE INJECTION SYSTEM FOR SUCTION FROM REACTOR BUILDING SUMP AND PLACE INTO SERVICE.

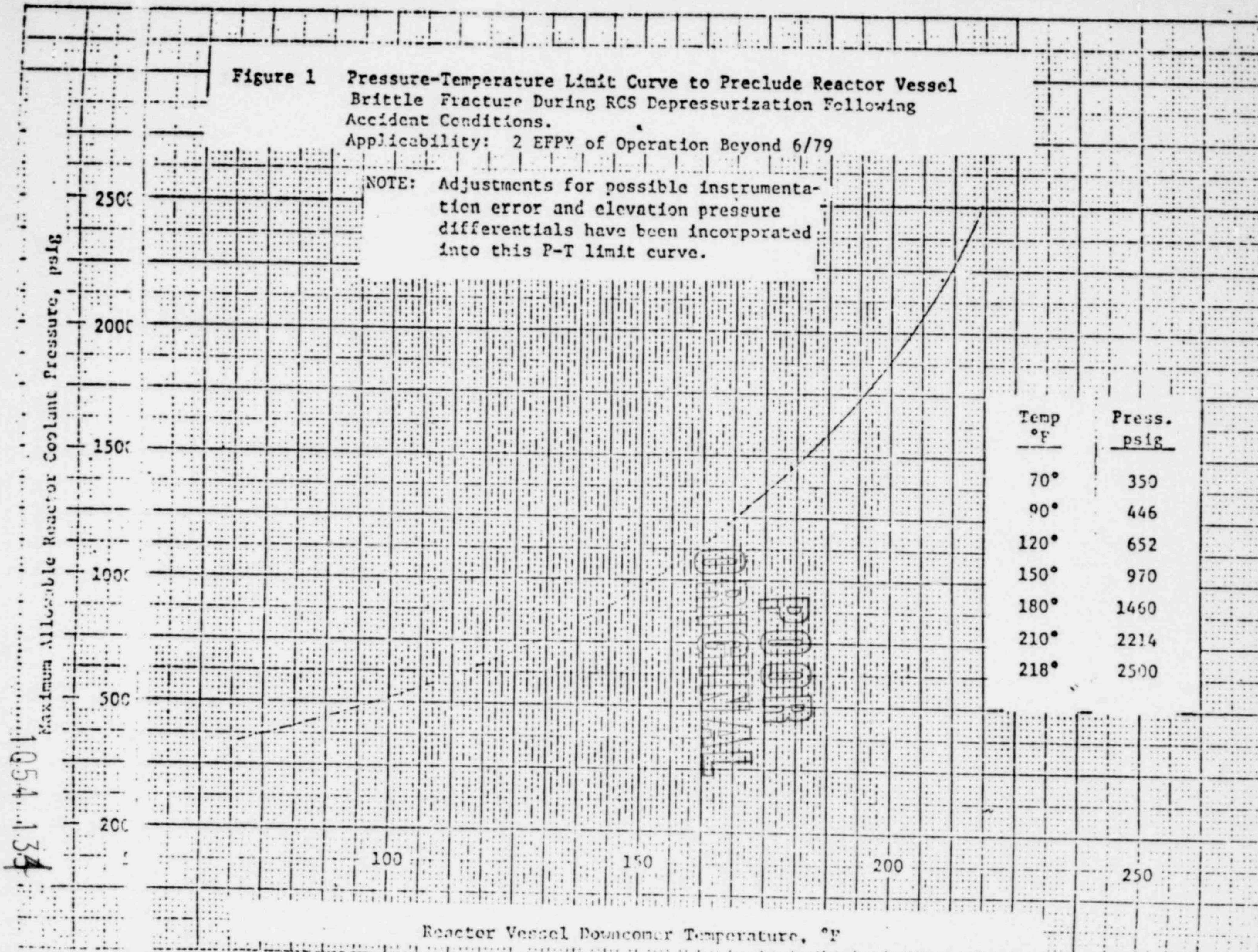
A.4.3 BALANCE LPI INJECTION AND CONTROL RC TEMPERATURE WITH DECAY HEAT COOLERS.

A.4.4 ISOLATE CORE FLOOD TANKS.

A.4.5 GO TO STEP A.1.1 AND FOLLOW THE PROCEDURE GIVEN THERE, IGNORING THE INSTRUCTIONS RELATING TO RC PUMP OPERATION.

Figure 1 Pressure-Temperature Limit Curve to Preclude Reactor Vessel Brittle Fracture During RCS Depressurization Following Accident Conditions.
Applicability: 2 EFPY of Operation Beyond 6/79

NOTE: Adjustments for possible instrumentation error and elevation pressure differentials have been incorporated into this P-T limit curve.



Temp °F	Press. psig
70°	350
90°	446
120°	652
150°	970
180°	1460
210°	2214
218°	2500

Part II: Small Break Phenomena - Description of Plant Behavior

1. Introduction

A loss-of-coolant accident is a condition in which liquid inventory is lost from the reactor coolant system. Due to the loss of mass from the reactor coolant system, the most significant short-term symptom of a loss-of-coolant accident is an uncontrolled reduction in the reactor coolant system pressure. The reactor protection system is designed to trip the reactor on low pressure. This should occur before the reactor coolant system reaches saturation conditions. The existence of saturated conditions within the reactor system is the principal longer-term indication of a LOCA and requires special consideration in the development of operating procedures.

Following a reactor trip, it is necessary to remove decay heat from the reactor core to prevent damage. However, so long as the reactor core is kept covered with cooling water, core damage will be avoided. The ECCS systems are designed to respond automatically to low reactor coolant pressure conditions and take the inital actions to protect the reactor core. They are sized to provide sufficient water to keep the reactor core covered even with a single failure in the ECCS systems. Subsequent operator actions are required ultimately to place the plant in a long-term cooling mode. The overall objective of the automatic emergency core cooling system and the followup operator actions is to keep the reactor core cool.

A detailed discussion of the small break LOCA phenomenalogy is presented in this section. This discussion represents Part II of the operating procedure guidelines for the development of detailed operating procedures. Part I presents the more detailed step-by-step guidelines.

The response of the primary system to a small break will greatly depend on break size, its location in the system, operation of the reactor coolant pumps, the number of ECCS trains functioning, and the availability of secondary side cooling. RCS pressure and pressurizer level histories for various combinations of parameters are presented in order to indicate the wide range of system behavior which can occur for small LOCA's.

2. Impact of RC Pump Operation on a Small LOCA

With the RC pumps operating during a small break, the steam and water will remain mixed during the transient. This will result in liquid being discharged out the break continuously. Thus, the fluid in the RCS can evolve to a high void fraction as shown in Figure 1. The maximum void fraction that the system evolves to, and the time it occurs, is dependent on the break size and location. Continued RC pump operation, even at high system void fractions, will provide sufficient core flow to keep cladding temperatures within a few degrees of the saturated fluid temperature.

Since the RCS can evolve to a high void fraction for certain small breaks with the RC pumps on, a RC pump trip by any means (i.e., loss of offsite power, equipment failure, etc.) at a high void fraction during the small break transient may lead to inadequate core cooling. That is, if the RC pumps trip at a time period when the system void fraction is greater than approximately 70%, a core heatup will occur because the amount of water left in the RCS would not be sufficient to keep the core covered. The cladding temperature would increase until core cooling is re-established by the ECC systems. For certain break sizes and times of RC pump trip, acceptable peak cladding temperatures during the event could not be assured and the core could be damaged. Thus, prompt operator action to trip the RC pumps upon receipt of a low pressure ESFAS signal is required in order

to ensure that adequate core cooling is provided. Following the RC pump trip, the small break transient will evolve as described in the subsequent sections.

3. Small Breaks with Auxiliary Feedwater

There are four basic classes of break response for small breaks with auxiliary feedwater. These are:

1. LOCA large enough to depressurize the reactor coolant system
2. LOCA which stabilizes at approximately secondary side pressure
3. LOCA which may repressurize in a saturated condition
4. Small LOCA which stabilizes at a primary system greater than secondary system pressure

The system transients for these breaks are depicted in Figure 2.

3.1 LOCA Large Enough to Depressurize Reactor Coolant System: Curves 1 and 2 of Figure 2 show the response of RCS pressure to breaks that are large enough in combination with the ECCS to depressurize the system to a stable low pressure. ECCS injection easily exceeds core boil-off and ensures core cooling. Curves 1 and 2 of Figure 3 show the pressurizer level transient. Rapidly falling pressure causes the hot legs to saturate quickly. Cold leg temperature reaches saturation somewhat later as RC pumps coast down or the RCS depressurizes below the secondary side saturation pressure. Since these breaks are capable of depressurizing the RCS without aid of the steam generators, they are essentially unaffected by the availability of auxiliary feedwater. Upon receipt of a low pressure ESFAS signal, the operator must trip all RC pumps and verify that all ESFAS actions have been completed. The operator must also balance HPI flows such that flow is available through all HPI injection nozzles even if only one HPI is available. The operator should also balance LPI flows, should the system be actuated, to ensure flow through both lines. The operator needs to take no further actions to bring the system to a safe shutdown

condition. Rapid depressurization of the steam generators would only act to accelerate RCS depressurization. It is, however, not necessary. Restarting of the RC pumps is not desirable for this class of break.

Long-term cooling will require the operator to shift the LPI pump suction to the reactor building sump.

3.2 LOCA Which Stabilizes at Approximately Secondary Side Pressure. Curve 3 of Figure 2 shows the pressure transient for a break which is too small in combination with the operating HPI to depressurize the RCS. The steam generators are, therefore, necessary to remove a portion of core decay heat. Although the system pressure will initially stabilize near the secondary side pressure, RCS pressure may eventually begin falling as the decay heat level decreases. Curve 3 of Figure 3 shows pressurizer level behavior. The hot leg temperature quickly equalizes to the saturated temperature of the secondary side and controls primary system pressure at saturation. The cold leg temperature may remain slightly subcooled. If the HPI refills and repressurizes the RCS, the hot legs can become subcooled. The immediate operator action is to trip the RC pumps upon receipt of the low pressure ESFAS signal and then verify ESFAS functions. The operator must then balance HPI in order to ensure flow through each high pressure injection line.

Followup action by the operator is to raise the emergency feedwater level to 95% on the operating range and check for established natural circulation. This is done by gradually depressurizing the steam generators. If this test fails, intermittent bumping of a RC pump should be performed as soon as one is available. Continued depressurization of the steam generators with natural circulation leads to cooling and depressurization of the RCS. The operator's goal is to depressurize the RCS to a pressure that enables the ECCS to exceed core boil-off, possibly refill the RCS, and to ultimately establish long-term cooling.

3.3 LOCA Which May Repressurize in a Saturated Condition. Curve 4 of Figure 2 shows the behavior of a small break that is too small, in combination with the HPI, to depressurize the primary system. Although steam generator feedwater is available, the loss of primary system coolant and the resultant RCS voiding will eventually lead to interruption of natural circulation. This is followed by gradual repressurization of the primary system. It is possible that the primary system could repressurize as high as the pressurizer safety valve setpoint before the pressure stabilizes. This is shown by the dashed line in Curve 4. Once enough inventory has been lost from the primary system to allow direct steam condensation in the regions of the steam generators contacting secondary side coolant, the primary system is forced to depressurize to the saturation pressure of the secondary side.

Since the cooling capabilities of the secondary side are needed to continue to remove decay heat, RCS pressure will not fall below that on the secondary side. HPI flow is sufficient to replace the inventory lost to boiling in the core, and condensation in the steam generators removes decay heat energy. The RCS is in a stable thermal condition and it will remain there until the operator takes further action. The pressurizer level response is characterized by Curve 3 of Figure 3 during the depressurization, and Curve 4 of Figure 3 during the temporary repressurization phase. The dashed line indicates the level behavior if pressure is forced up to the pressurizer safety valve setpoint. During this transient, hot leg temperature will rapidly approach saturation with the initial system depressurization, and it will remain saturated during the whole transient. Cold leg temperature will approach saturation as circulation is lost, but may remain slightly subcooled during the repressurization phase of the transient. Later RCS depressurization could cause the cold leg temperatures to reach saturation. Subsequent refilling of the primary

system by the HPI might cause temporary interruption of steam condensation in the steam generator as the primary side level rises above the secondary side level. If the depressurization capability of the break and the HPI is insufficient to offset decay heat, the primary system will once more repressurize. This decreases HPI flow and increases loss through the break until enough RCS coolant is lost to once more allow direct steam condensation in the steam generator. This cyclic behavior will stop once the HPI and break can balance decay heat or the operator takes some action.

The operator's immediate action is to trip the RC pumps upon receipt of the low pressure ESFAS signal and verify the completion of all ESFAS functions. The operator should then balance HPI flow. Following that, he should raise the steam generator level to 95% of the operating range and check for natural circulation. If it is positive, he should depressurize the steam generators, cool and depressurize the primary system, and attempt to refill it and establish long-term cooling. If the system fails to go into natural circulation, he should open the PORV long enough to bring and hold the RCS near the secondary side pressure. Once natural circulation is established or a RC pump can be bumped, he will be able to continue depressurizing the RCS with the steam generators and establish long-term cooling.

3.4 Small LOCA Which Stabilizes at $P > P_{sec}$. Curve 5 of Figure 2 shows the behavior of the RCS pressure to a break for which high pressure injection is being supplied and exceeds the leak flow before the pressurizer has emptied. The primary system remains subcooled and natural circulation to the steam generator removes core decay heat. The pressurizer never empties and continues to control primary system pressure. The operator needs to trip the RC pumps and ensure that ESFAS actions have occured. Throttling of HPI is permitted only after RCS subcooling of 50°F has been established, the pressurizer has refilled, and natural or forced circulation has been

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verified. A restart of the RC pumps under these conditions is desirable for plant control.

3.5 Small Breaks in Pressurizer. The system pressure transient for a small break in the pressurizer will behave in a manner similar to that previously discussed. The initial depressurization, however, will be more rapid as the initial inventory loss is entirely steam.

The pressurizer level response for these accidents will initially behave like a very small break without auxiliary feedwater. The initial rise in pressurizer level shown in Figure 4 will occur due to the pressure reduction in the pressurizer and an insurge of coolant into the pressurizer from the RCS. Once the reactor trips, system contraction causes a decreasing level in the pressurizer. Flashing will ultimately occur in the hot leg piping and cause an insurge into the pressurizer. This ultimately fills the pressurizer. For the remainder of the transient, the pressurizer will remain full. Toward the later stages of the transient, the pressurizer may contain a two-phase mixture and the indicated level will show that the pressurizer is only partially full. Except for closing the PORV block valve, operator actions and system response are the same for these breaks as for similar breaks in the loops.

4. Small Breaks Without Auxiliary Feedwater

There are three basic classes of break response for small breaks without auxiliary feedwater. These are:

1. Those breaks capable of relieving all decay heat via the break.
2. Breaks that relieve decay heat with both the HPI injection and via the break.
3. Breaks which do not automatically actuate the HPI and result in system repressurization.

The system pressure transients for these breaks are depicted in Figure 5.

4.1 LOCA's Large Enough to Depressurize Reactor Coolant System. For Class 1 (curve 1 of Figure 5), RC system pressure decreases smoothly throughout the transient. For the larger breaks in this class, CFT actuation and LPI injection will probably occur. For the smaller breaks of this class only, CFT actuation will occur. Auxiliary feedwater injection is not necessary for the short-term stabilization of these breaks. The pressurizer level for this transient rapidly falls off scale. Operator action and plant response are similar to those described for this class of breaks with a feedwater supply.

4.2 LOCA's Which Reach a Semi-Stabilized State. For Class 2 (Curve 2 of Figure 5) breaks, the RC pressure will rapidly reach the low pressure ESFAS trip signal (about two to three minutes). With the HPI's on, a slow system depressurization will be established coincident with the decrease in core decay heat. No CFT actuation is expected. Auxiliary feedwater is not necessary for the short-term stabilization of these breaks. The pressurizer level for this transient rapidly falls off scale.

The operator needs to trip the RC pumps upon the low pressure ESFAS signal, verify completion of all ESFAS functions, and try to establish secondary side cooling. Balancing of the HPI must also be performed. If steam generator feedwater cannot be obtained and RCS pressure is increasing, the operator should open the PORV and provide all the HPI and makeup capability possible. The goal is to depressurize and cool the core with the ECCS, the PORV, and the break. If secondary side cooling is again established, the operator should verify natural circulation, and if unavailable, bump a RC pump to complete RCS cooldown with the steam generators. At this point, the PORV can be closed, the system refilled, and long-term cooling established.

4.3 Small LOCA's Which do not Actuate the ESFAS. Automatic ESFAS actuation will not occur for Class 3 (Curve 3 of Figure 5) breaks. Once the SG secondary side inventory is boiled off, system repressurization will occur as the break is not capable of removing all the decay heat being generated in the core. System repressurization to the PORV or the pressurizer safety valves will occur for smaller breaks in this class. For the "zero" break case, repressurization to the PORV will occur in the first five minutes. Operator action is required within the first 20 minutes to ensure core coverage throughout the transient. For the 177-FA lowered-1990 plants, this action can be either manual actuation of the auxiliary feedwater system or the HPI system.

The establishment of auxiliary feedwater will rapidly depressurize the RCS to the ESFAS actuation pressure, and system pressure will stabilize at either the secondary side SG pressure or at a pressure where the HPI equals the leak rate. Upon receipt of the low pressure ESFAS signal, the operator must trip the RC pumps.

For the raised loop Davis-Besse plant (which has a safety-grade auxiliary feedwater system) operator action is necessary at some time greater than 20 minutes (approximately 40 minutes) as there is increased inventory in the loops that is available to drain into the reactor vessel. However, because the plant is equipped with low shut-off head HPI pumps, the operator must establish auxiliary feedwater in order to depressurize the RCS.

For the Class 3 breaks, pressurizer level response will be as shown in Figure 6. The minimum refill time for the pressurizer is that for the "zero" break and is shown in Figure 6. After initially drawing inventory from the pressurizer, the system repressurization will cause the pressurizer level to increase, possibly to full pressurizer level. Once the operator action to restore auxiliary feedwater has been taken, the system

depressurization will result and cause an outsurge from the pressurizer. Complete loss of pressurizer level may result. For the smaller breaks in Class 3 which result in a system repressurization following the actuation of the HPI system, pressurizer level will increase and then stabilize.

Without auxiliary feedwater, both the hot and cold leg temperatures will saturate early in the transient and, for the Class 1 and 2 breaks, will remain saturated. For the Class 3 breaks, once auxiliary feedwater is established, the cold leg temperatures will rapidly decrease to approximately the sautration temperature corresponding to the SG secondary side pressure and will remain there throughout the remainder of the transient. Hot leg temperatures will remain saturated throughout the event.

The operator needs to manually initiate all ESFAS actions, balance HPI flow, and attempt to restore secondary side cooling. In the meantime, he should actuate the makeup pump and open the PORV in order to cool the core and limit the RCS repressurization. Once feedwater is available, he can close the PORV and continue the RCS cooldown and depressurization with the steam generators. If natural circulation has not been established, he can bump a RC pump to cause forced circulation. The goal is to depressurize to where the ECCS can refill the RCS and guarantee long-term cooling.

4.4 Small Breaks in Pressurizer. See the writeup for small breaks in pressurizer with feedwater.

Small breaks in the pressurizer will differ from those in the loops in the same manner as those previously described in the section addressing small breaks in the pressurizer with auxiliary feed.

5. Transients with Initial Response Similar to a Small Break

Several transients give initial alarms similar to small breaks. These transients will be distinguished by additional alarms and indications or subsequent system response.

Overcooling transients such as steam line breaks, increased feedwater

flow, and steam generator overfill can cause RCS pressure decreases with low-pressure reactor trip and ESFAS actuation. But steam line breaks actuate low steam pressure alarms for the affected steam generator, and steam generator overfills result in high steam generator level indications. The overcooling transients will repressurize the primary system because of HPI actuation, and will return to a subcooled condition during repressurization. The immediate actions for both overcooling and small break transients are the same, including tripping of the RC pumps.

The operator will recognize overcooling events during repressurization, if not sooner, and is instructed to throttle HPI and restart the RC pumps, if subcooled conditions are established, by the small break operating instructions.

A loss-of-feedwater transient will result in a high reactor system pressure alarm but does not give an ESFAS actuation alarm.

A loss of integrated control system power transient starts with a high RC pressure trip. After the reactor trip, this becomes an overcooling transient and will give low reactor system pressure and possible ESFAS actuation. Steam generator levels remain high and the system becomes subcooled during repressurization.

Design features of the B&W NSS provide automatic protection during the early part of small break transients, thereby providing adequate time for small breaks to be identified and appropriate action taken to protect the system. The only prompt manual operator action required is to trip the RC pumps once the low pressure ESFAS signal is reached.

6. Transients that might Initiate a LOCA

There are no anticipated transients that might initiate a LOCA since the PORV has been reset to a higher pressure and will not actuate during anticipated transients such as loss of main feedwater, turbine trip, or loss of offsite power.

However, if the PORV should lift and fail to reset, there are a number of indications which differentiate this transient from the anticipated transients identified above. These include:

- o ESFAS actuation
- o Quench tank pressure/temperature alarms
- o Saturated primary system
- o Rising pressurizer level

These additional signals will identify to the operator that in addition to the anticipated transient, a LOCA has occurred. In the unlikely event that small breaks other than a malfunctioning PORV occur after a transient, they can be identified by initially decreasing RCS pressure and convergence to saturation conditions in the reactor coolant. Small break repressurization, if it occurs, will follow saturation conditions. By remaining aware of whether the reactor coolant remains subcooled or becomes saturated after transients, the operator is able to recognize when a small break has occurred.

7. HPI Throttling

For small LOCA's, the HPI system is needed to provide makeup to the RCS and must remain operable unless specific criteria are satisfied. The basis for these criteria are described below.

For certain small breaks, system depressurization will result in LPI actuation. Since the LPI is designed to provide injection at a greater capacity than the HPI, termination of the HPI is allowed. However, this action should only be taken if the flow rate through each line is at least 1000 gpm and the situation has been stable for 20 minutes. The 20-minute time delay is included to ensure that the system will not repressurize and result in a loss of the LPI fluid. In the event of a core flooding line break, the LPI fluid entering the broken core flooding line will not reach the vessel. Thus, in order to ensure that fluid is continually being injected to the RV for all breaks, the LPI must be providing fluid through both lines. The 1000 gpm is equivalent to the flow from

two HPI pumps and ensures that upon termination of the HPI pumps, adequate flow is being delivered to the RV.

Throttling or termination of the HPI flow is also allowed if all the following criteria are met:

A. Hot and cold leg temperatures are at least 50°F below the saturation temperatures for the existing RCS pressure.

B. Hot leg temperatures are no more than 50°F hotter than the secondary side saturation temperature (This ensures that heat is being removed via the SG.)

C. The action is necessary to prevent the indicated pressurizer level from going off-scale high.

Under these conditions, the primary system is solid. Continued HPI flow at full capacity may result in a solid pressurizer and would result in a lifting of the PORV and/or the pressurizer code safety valves. This may in turn lead to a LOCA. Thus, HPI flow should be throttled to maintain a stable inventory in the RCS. However, if the 50°F subcooling cannot be maintained, the HPI shall be immediately reactivated.

HPI flows should also be throttled to prevent violation of the nil ductility temperature (NDT) for the reactor vessel.

This concern can only arise if the fluid temperature within the reactor vessel is at least 50°F subcooled. A curve of the allowable downcomer temperature for a given RCS pressure is provided within the operating guidelines. The downcomer temperature is determined by one of two methods:

1. If one or more RC pumps are operative, the cold leg RTD reading will be essentially the same as the reactor vessel downcomer temperature.

2. Without the RC pumps operating, the cold leg RTD's may not provide

temperature readings indicative of the actual RV downcomer temperature, as a stagnant pool of water may exist at these locations. The incore thermocouples will provide the best indicator of the downcomer temperature and should be utilized if no RC pumps are available. In order to account for heat added to the fluid from the core, 150°F must be subtracted from the incore thermocouple readings to reflect the downcomer temperature. This method will result in temperatures which will be lower than the expected downcomer temperature. Thus, use of this methodology assumes that NDT will not be a problem.

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

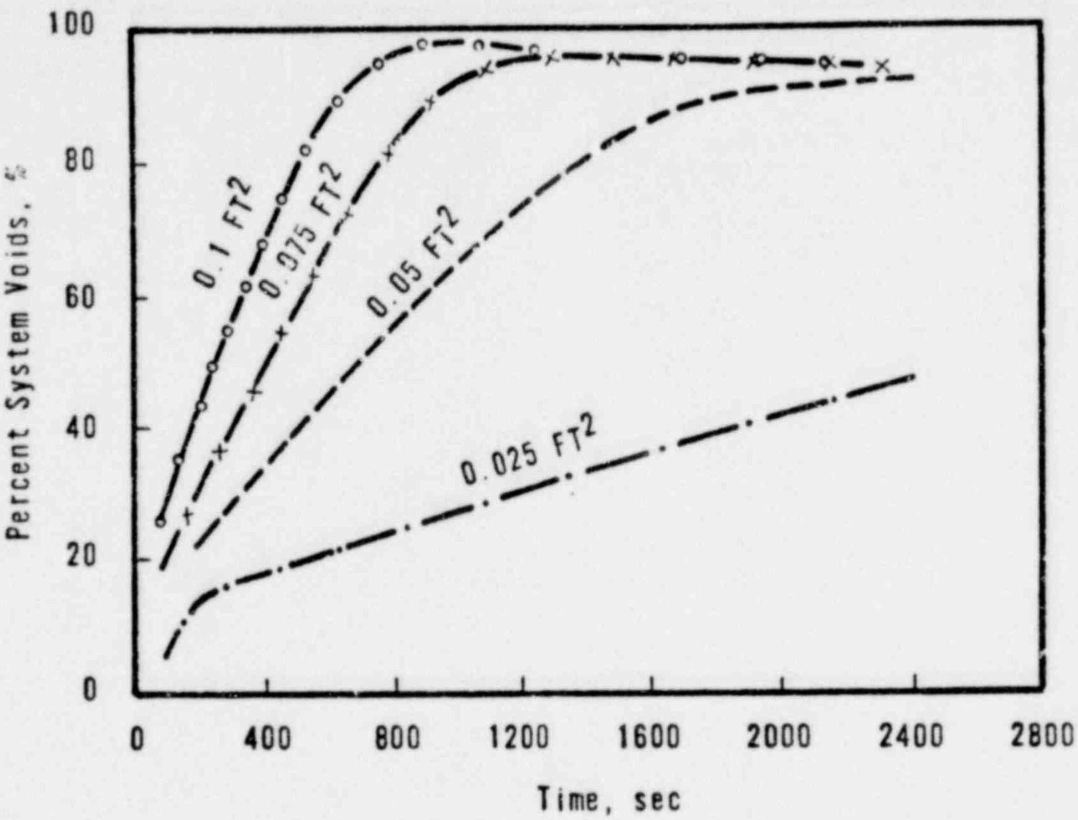

Figure 1

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PRESSURE VS TIME-SMALL BREAKS WITH AUXILIARY FEEDWATER

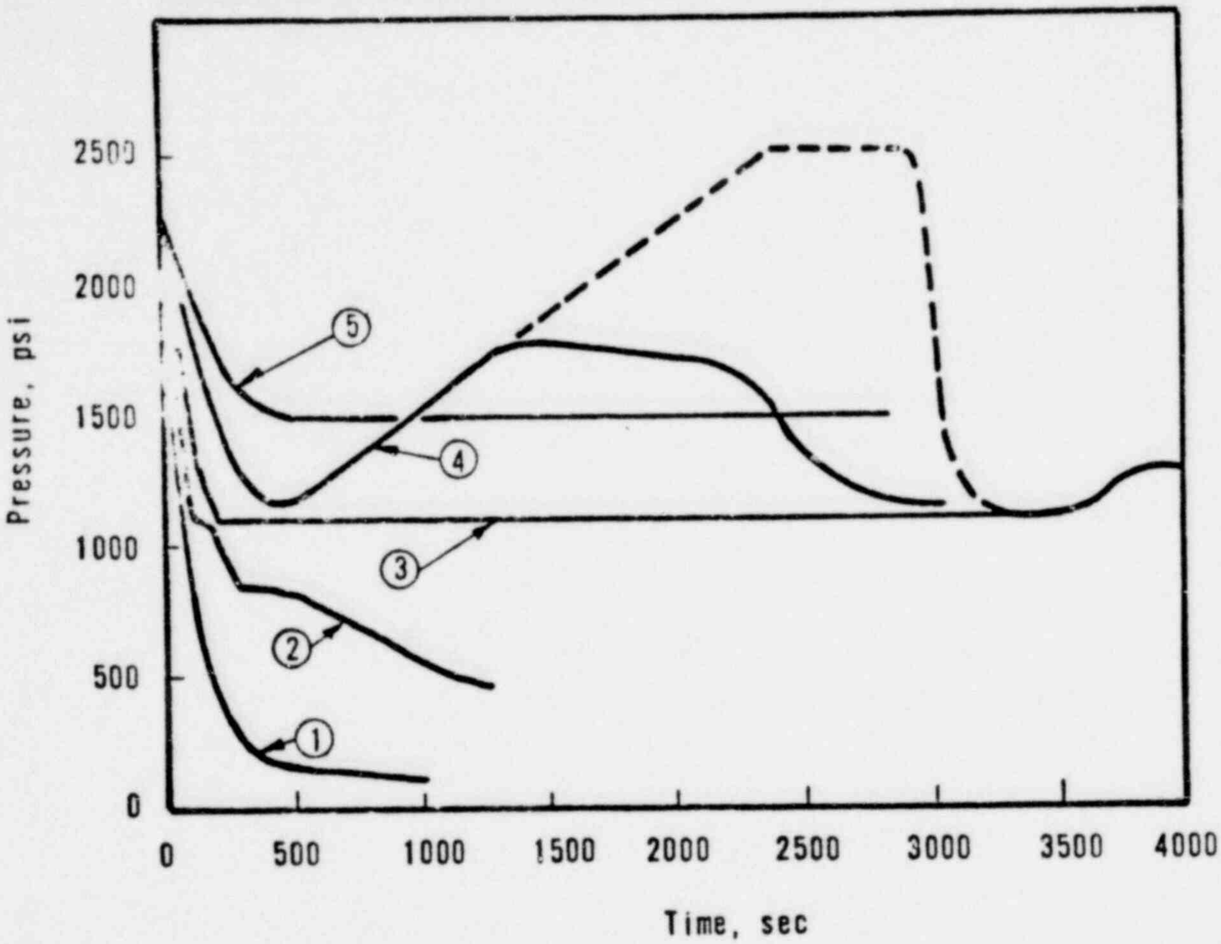

Figure 2

PRESSURIZER LEVEL VS TIME-SMALL BREAKS WITH AUXILIARY FEEDWATER

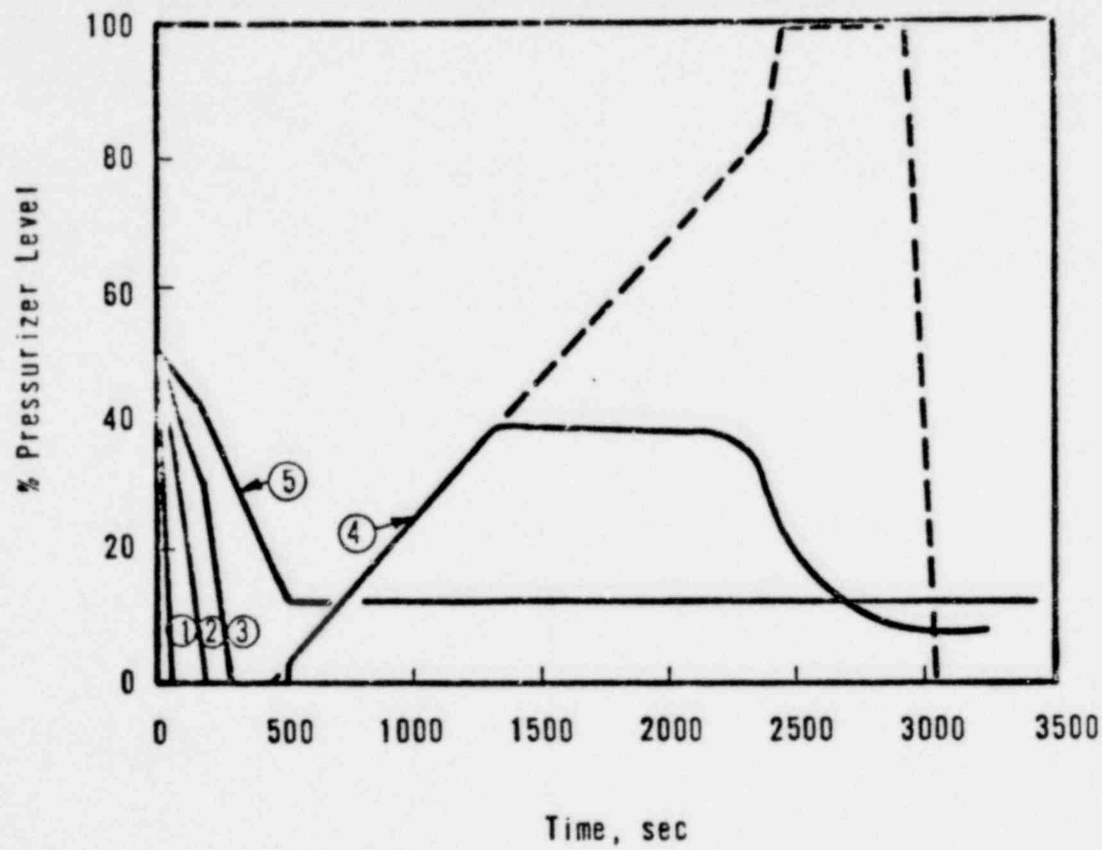

Figure 3

PRESSURIZER LEVEL VS TIME FOR SMALL BREAK IN PRESSURIZER

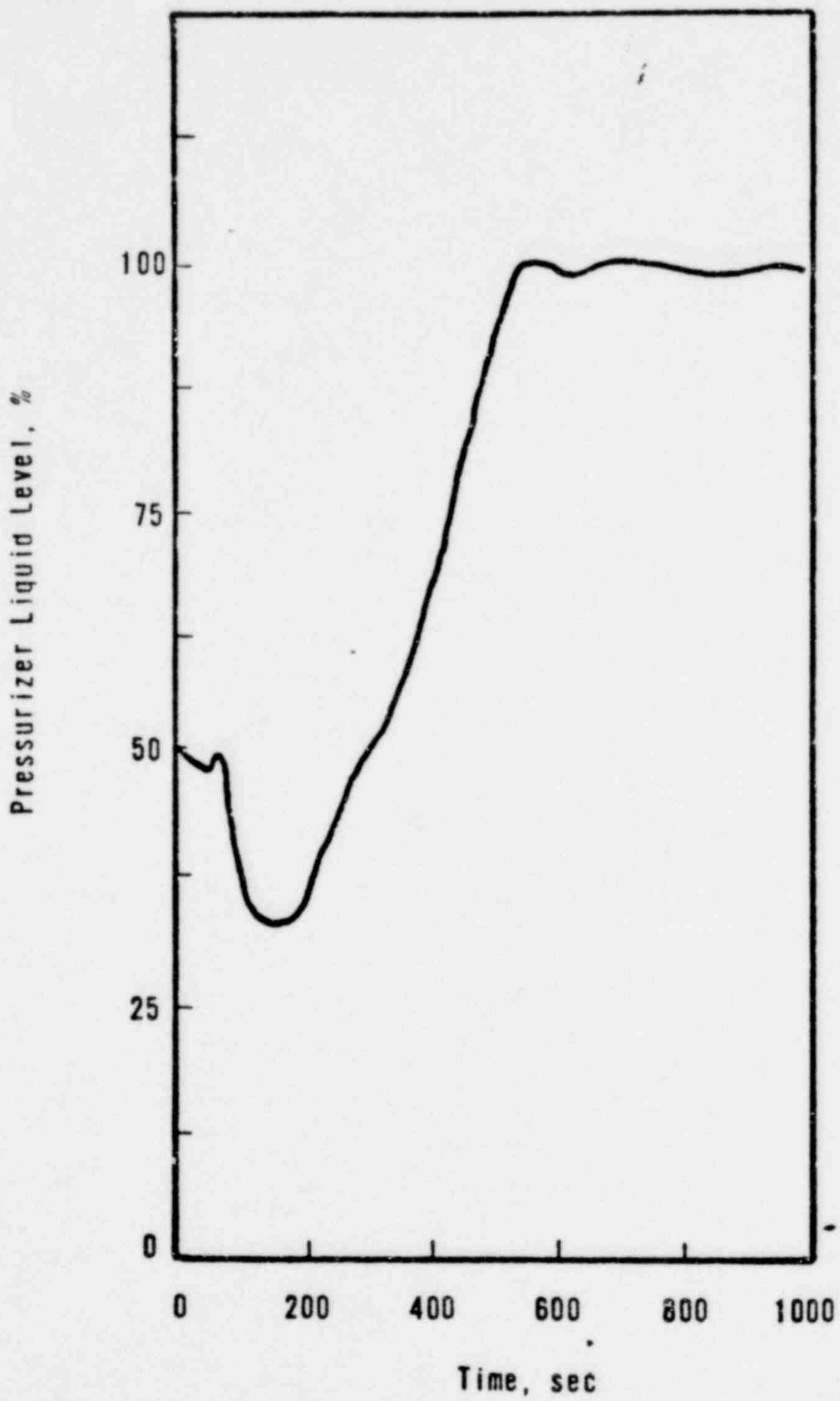

Figure 4

SYSTEM PRESSURE VS. TIME-SMALL BREAKS
W/O AUXILIARY FEEDWATER

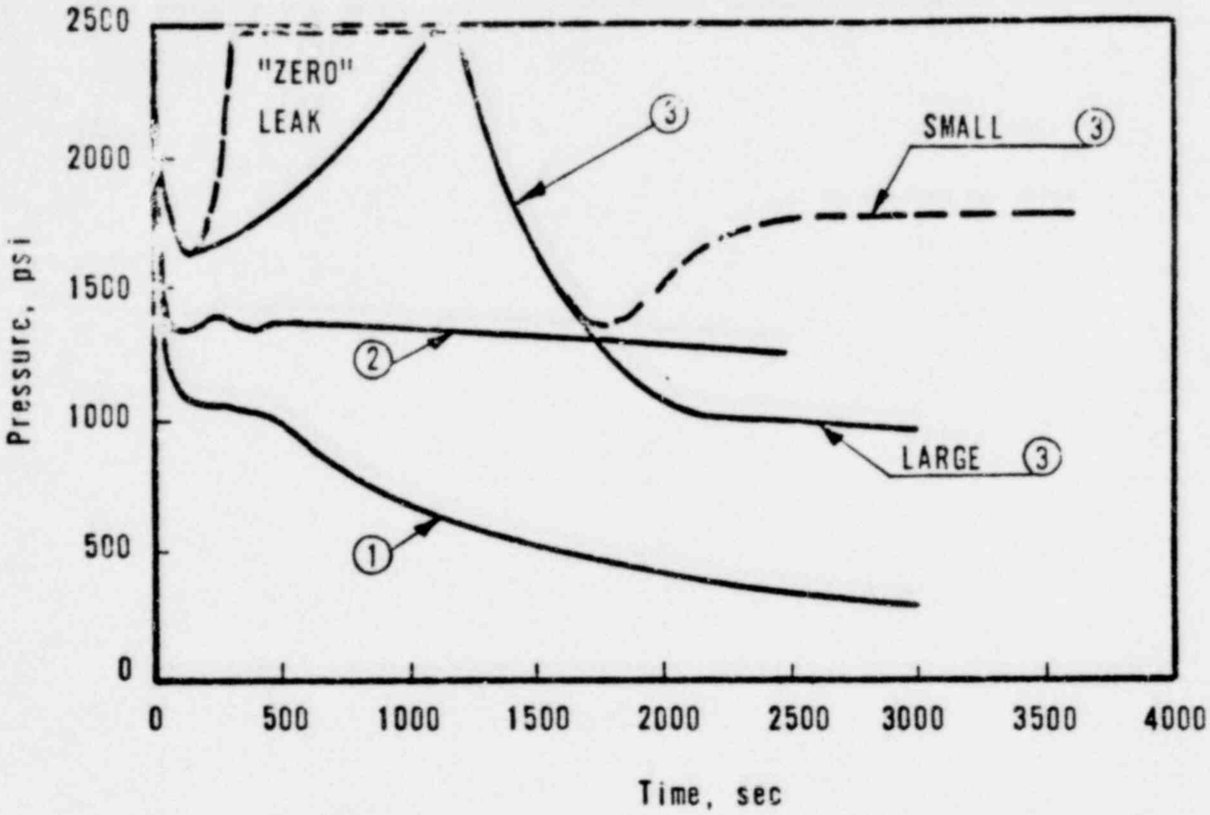

Figure 5

PRESSURIZER LEVEL VS. TIME-CLASS 3
BREAKS W/O AUXILIARY FEEDWATER

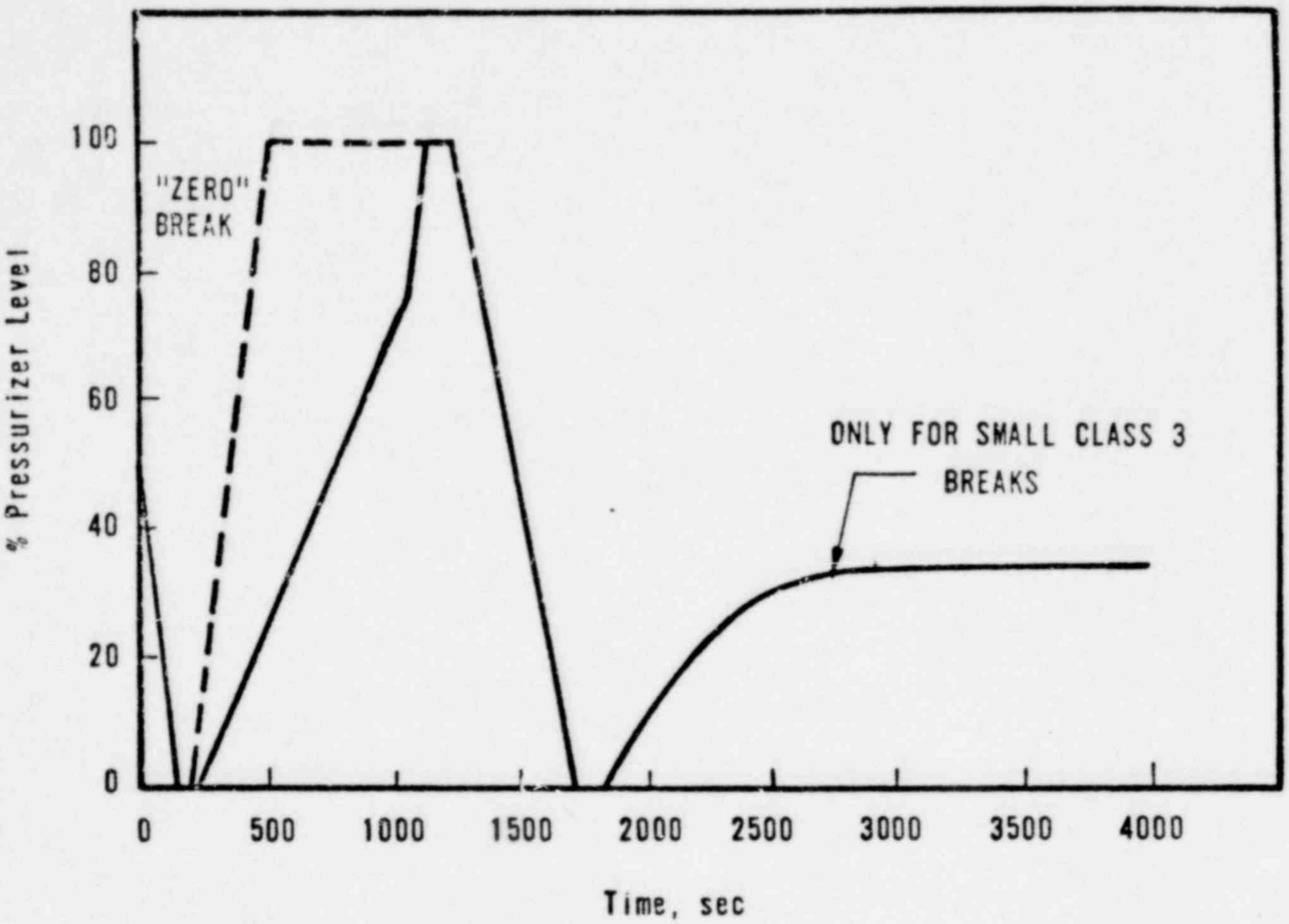

Figure 6

V. Guidelines for Non-LOCA Events

Because of the broad spectrum of system conditions covered by the small break guidelines, the operator actions and precautions identified to bring the plant to a long term cooling mode apply, in general, to any abnormal event which results in a decrease in RCS pressure. The small break guidelines will thus be utilized to update the emergency procedures for non-LOCA events; at a minimum, the following pertinent sections of the small break guidelines will be incorporated:

1. RC Pump Trip Criteria and SG Level Control actions to promote natural circulation.
2. RC pump Restart Criteria
3. HPI Control Criteria
4. The need to monitor system subcooling limits.

The items will be supplemented by the additional instructions/precautions to the effect that:

1. For non-LOCA events, a restart of the RC pumps (1 per loop) and termination of SG fill is prudent to minimize system overcooling due to addition of cold AFW to the OTSG's.

   Note: The establishment of a subcooled condition (>50F) is a clean indication that a non-LOCA event or a LOCA for which a RCP trip is not required is not in progress.

2. HPI should be throttled, when 50F subcooling is established, to avoid a pressurizer overfill.
3. During severe overcooling events, sufficient HPI water may be added, prior to achieving a subcooled condition ($\geq$ 50F) and a pressurizer level (on-scale), such that the system may evolve to water solid state when the RC temperature recovers to a hot shutdown condition ($\sim$ 530F).

Operator action to control primary temperature (via secondary steam pressure control using the turbine bypass valves and/or atmospheric dumps) may be required to maintain pressurizer level on scale.

NOTE: The Operating Guidelines For Small Breaks have been modified to include Item 3 above.

With operator training in the post-LOCA recovery methods in conjunction with modification of existing emergency procedures based on the small break guidelines, plant recovery and control can be achieved for any abnormal event for which an RCP trip is required.

REACTOR COOLANT PUMP TRIP ON HIGH PRESSURE INJECTION SIGNAL

In response to NRC Bulletins 79-05C and 79-06C, dated July 26, 1979, Florida Power Corporation wishes to respond to Short-term Action Item 1A on page 2 of 3. This item requires that "upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating reactor coolant pumps (RCP's)".

Florida Power Corporation proposes that a high pressure injection signal (HPI) be provided to immediately trip all operating RCP's. The proposed design is outlined as follows:

1. A new Clark type PM relay and a solid state switch, Hamlin type P/N7, in parallel with existing relays 63Y1/RC1 and 63Y2/RC1. These relays and solid state switches are the same types that are presently used in the Engineered Safety Features Actuation System.

2. The new components described above would be installed in both HPI ES Actuation "A" and "B" and operate by becoming de-energized when the 1500 psi bistable becomes de-energized below 1500 psi and is not bypassed.

3. One normally closed contact from each of the relays in the HPI ES Actuation "A" and "B" would be seriesed and placed in the trip circuits of each of the RCP's. The purpose of using contacts in series is to permit testing of a channel without tripping the RCP's.

4. In addition, two (2) alarms would be added to the main control board to alert the operator that (1) an HPI signal has tripped the RCP's or (2) an alarm would alert the operator that one of the ES Actuation "A" or B" signals has been actuated.

5. The schedule for implementation of this trip function is September 30, 1979.

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