ATTACHMENT 3

GUIDELINES FOR OPERATOR ACTION

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 ($\leq$ 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. Criteria for RCP Trip

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

2. Immediate Action

   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.

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.

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 A PUMP RESTART IS ALLOWED [2,3]	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 [2,3] A PUMP RESTART IS ALLOWED	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.

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.

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).

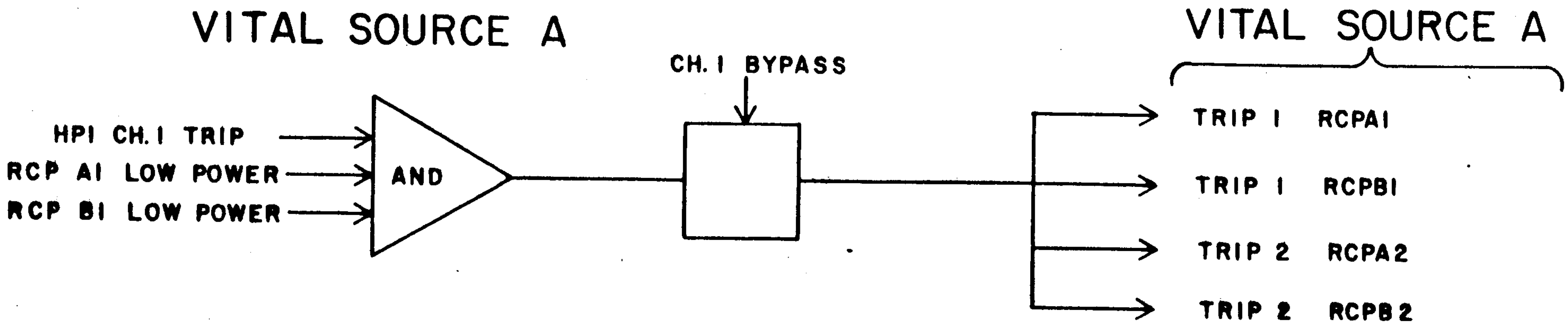
Operator action to control primary temperature (via secondary steam pressure control using the turbine bypass valves and/or atmospheric dumps) may be required to maintain pressurizer level on scale.

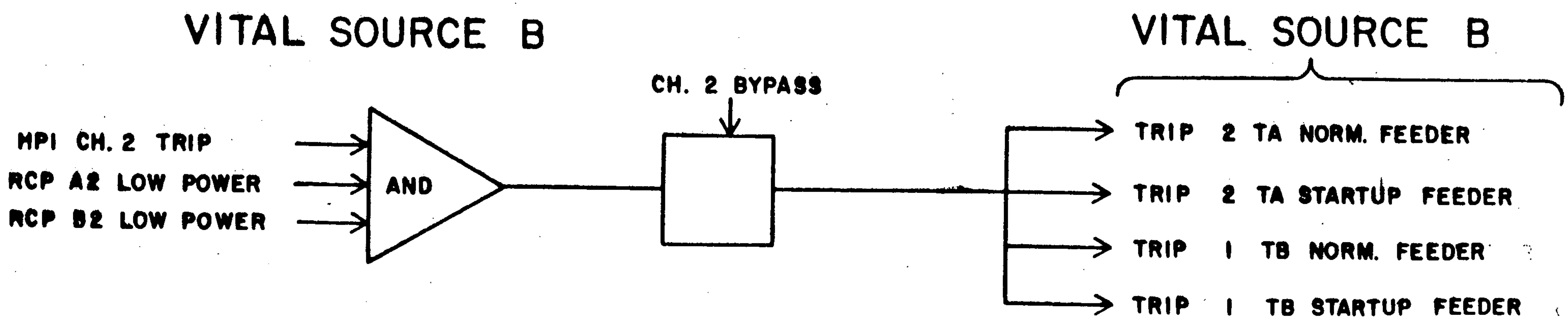
NOTE: The Operating Guidelines For Small Breaks have been modified to include Item 3 above.

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.

ATTACHMENT 4

AUTOMATIC RCP TRIP SCHEMATIC





ATTACHMENT 4

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

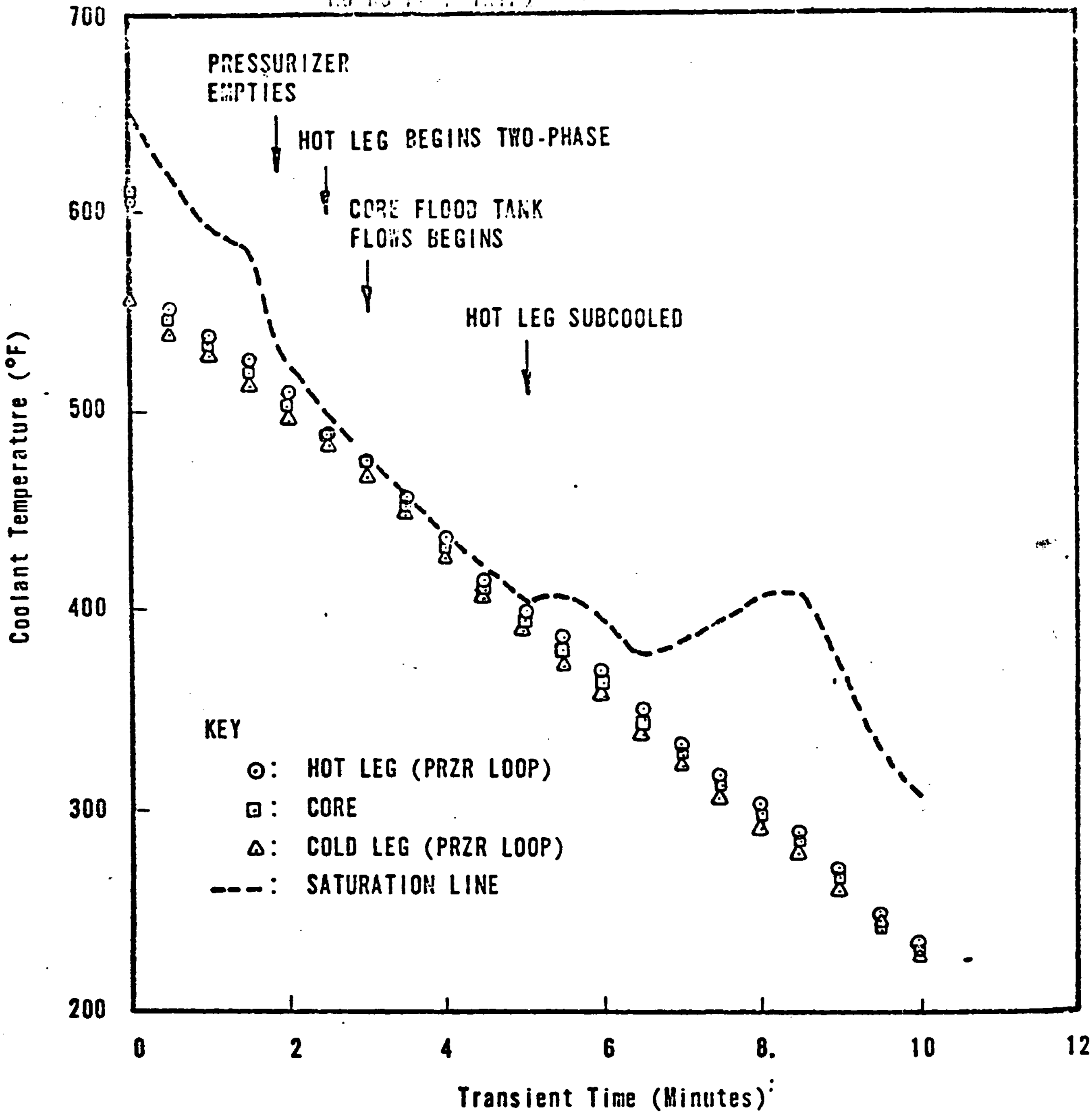


Figure 3.5

COOLANT TEMPERATURES VERSUS TRANSIENT TIME (102% FP, BEGINNING OF LIFE, 12.2 FT^2 DOUBLE END RUPTURE, UNMITIGATED STEAMLINE BREAK, RC PUMP TRIP)

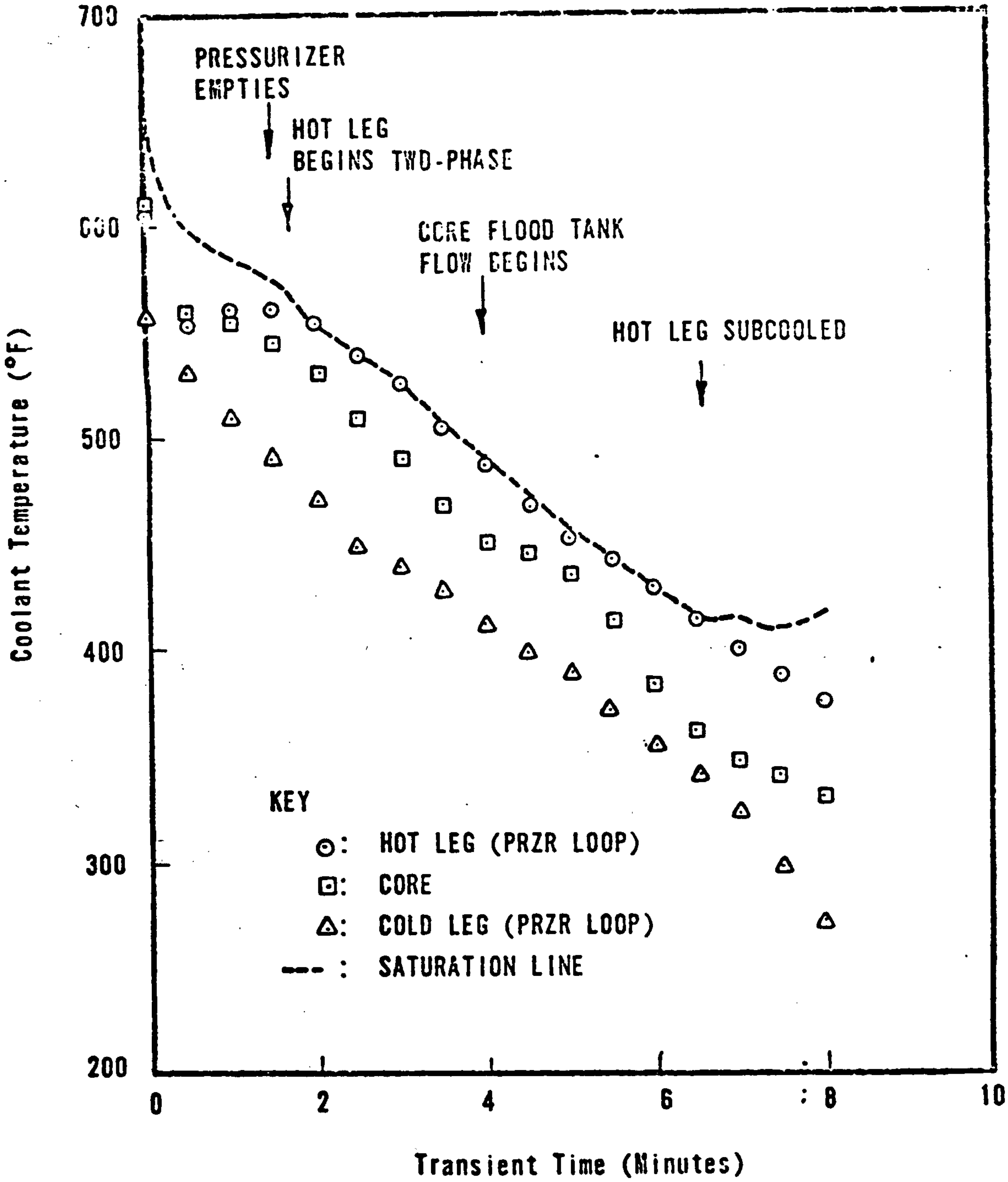


Figure 3.6

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

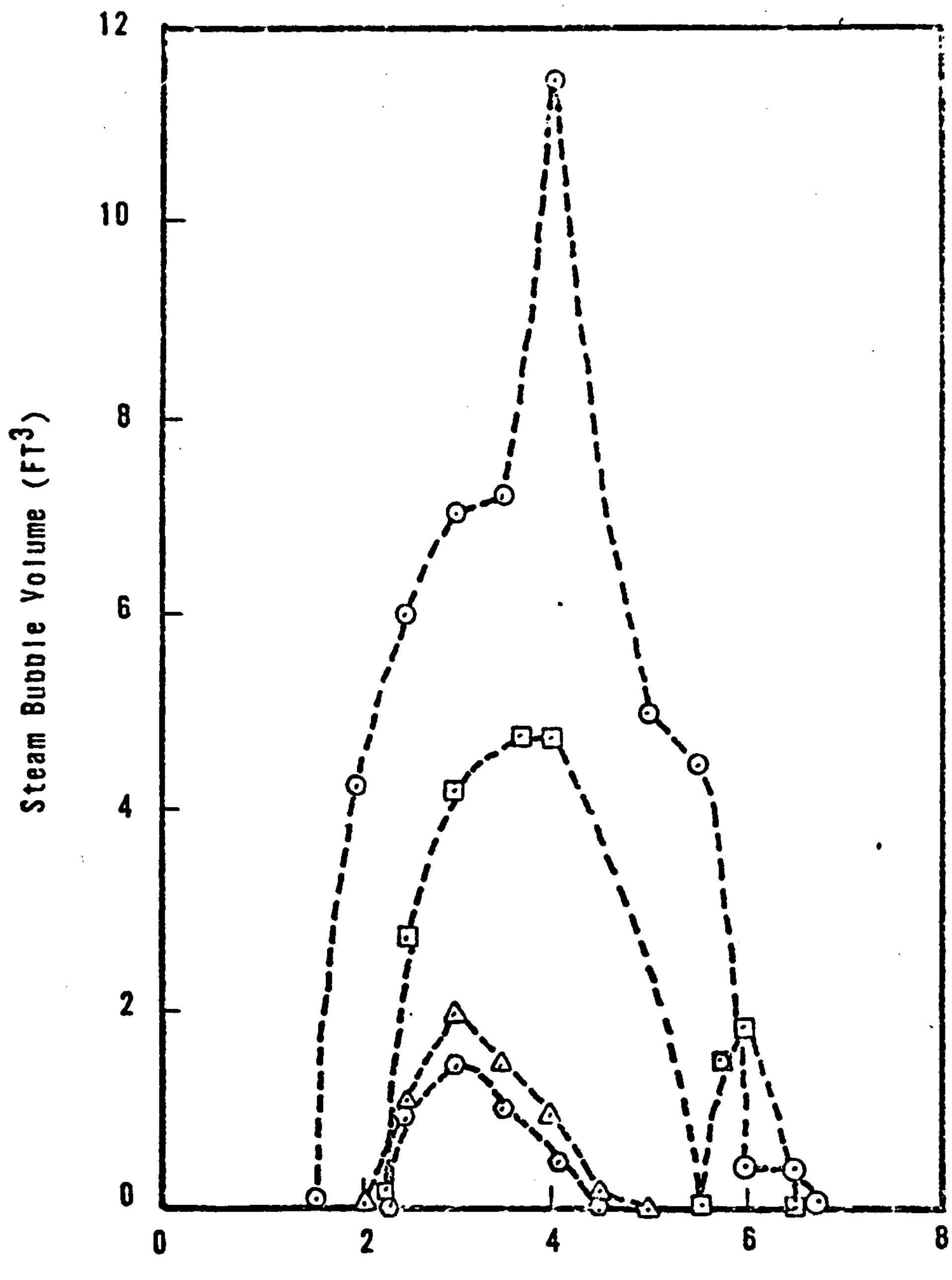


Figure 3.7

KEY

⊙: HOT LEG (PRZR) - RC PUMP TRIP

⊡: HOT LEG 'B' LOOP-RC PUMP TRIP

△: HOT LEG (PRZR LOOP) - NO TRIP

⊙: HOT LEG 'B' LOOP-NO TRIP

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

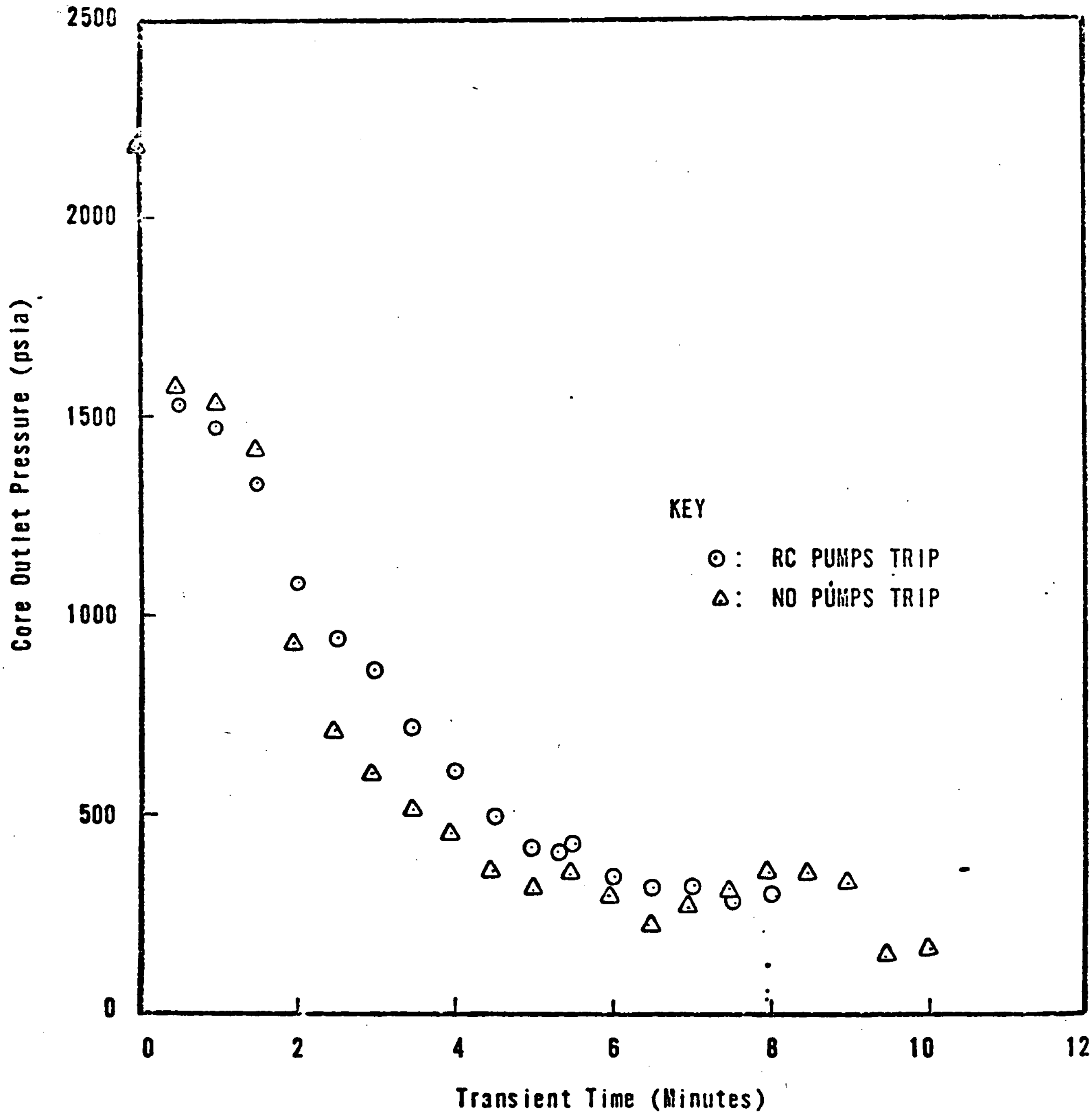


Figure 3.8

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

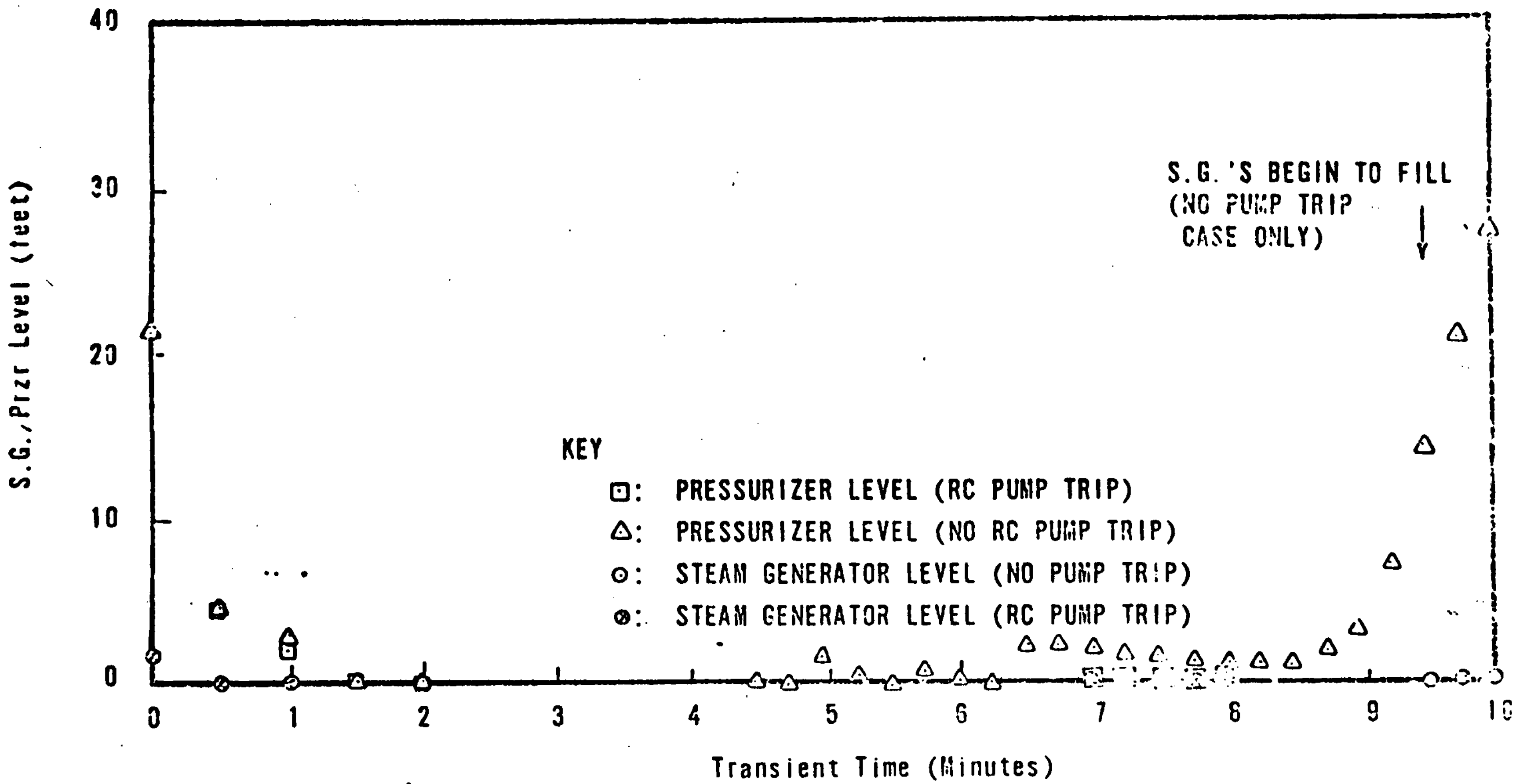


Figure 3.9

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).

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

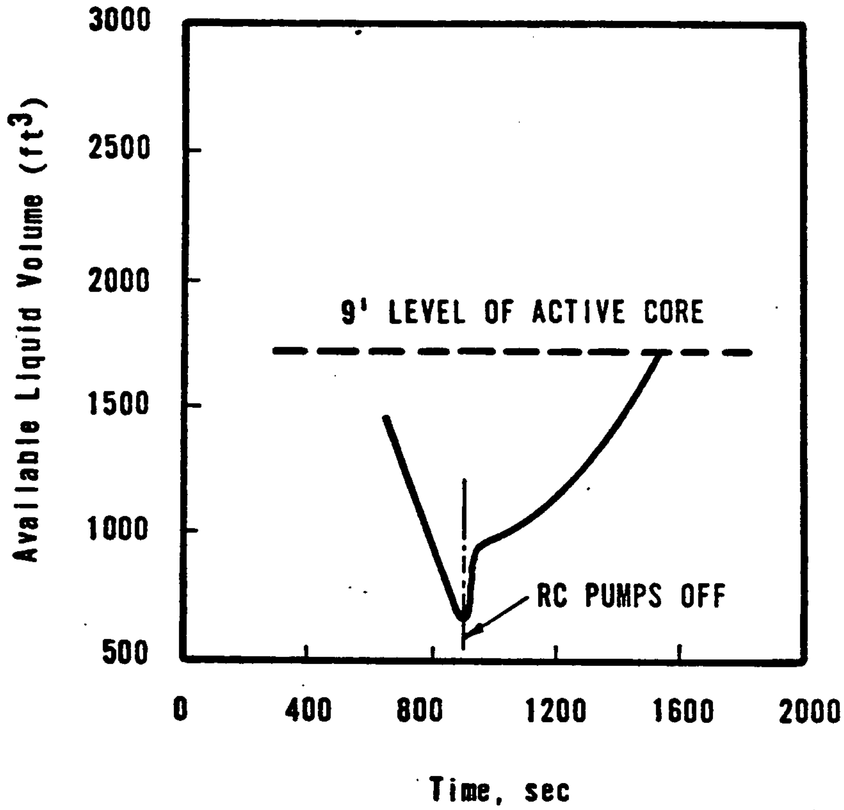


Figure 2-12

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

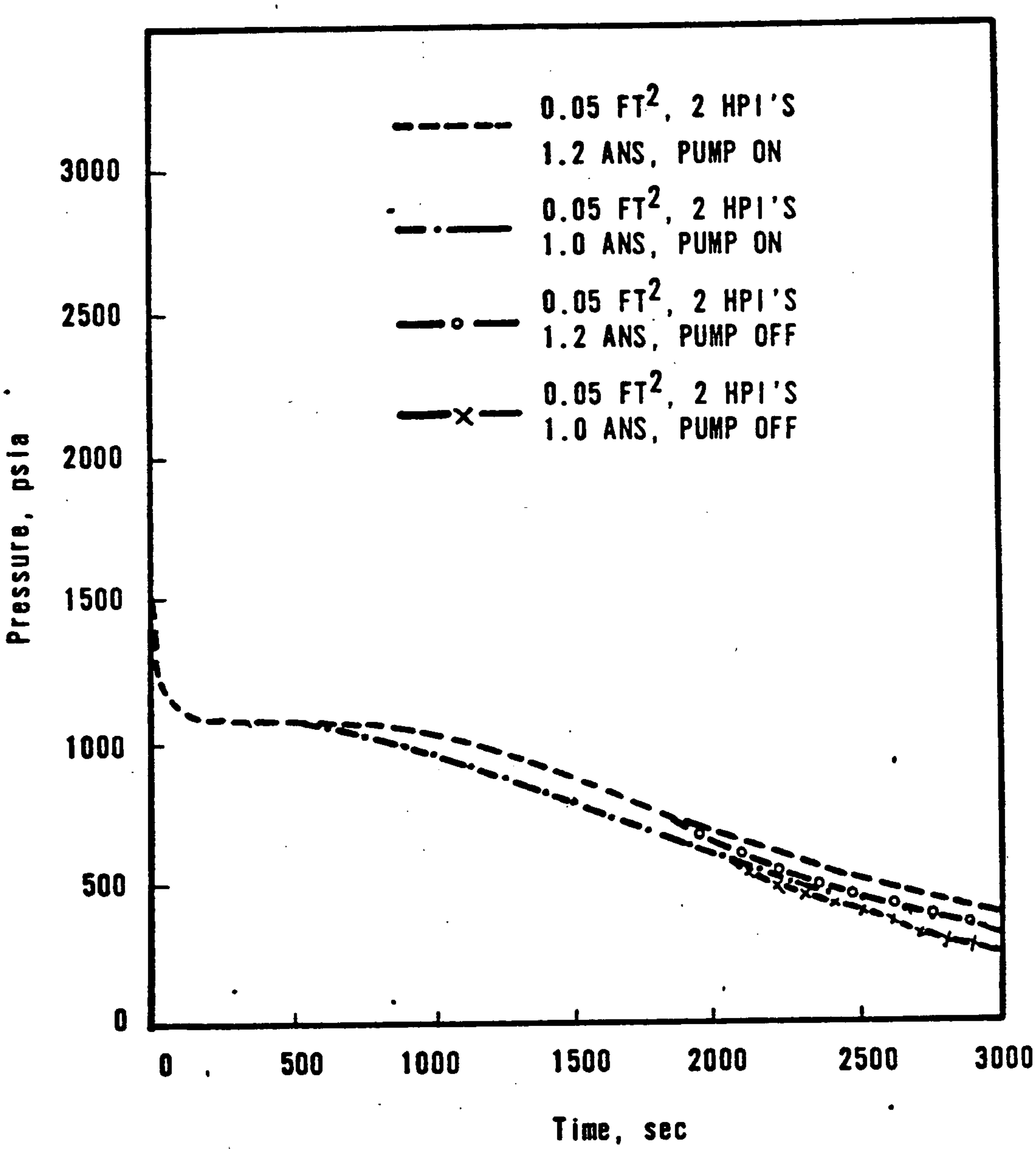


Figure 2-13

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

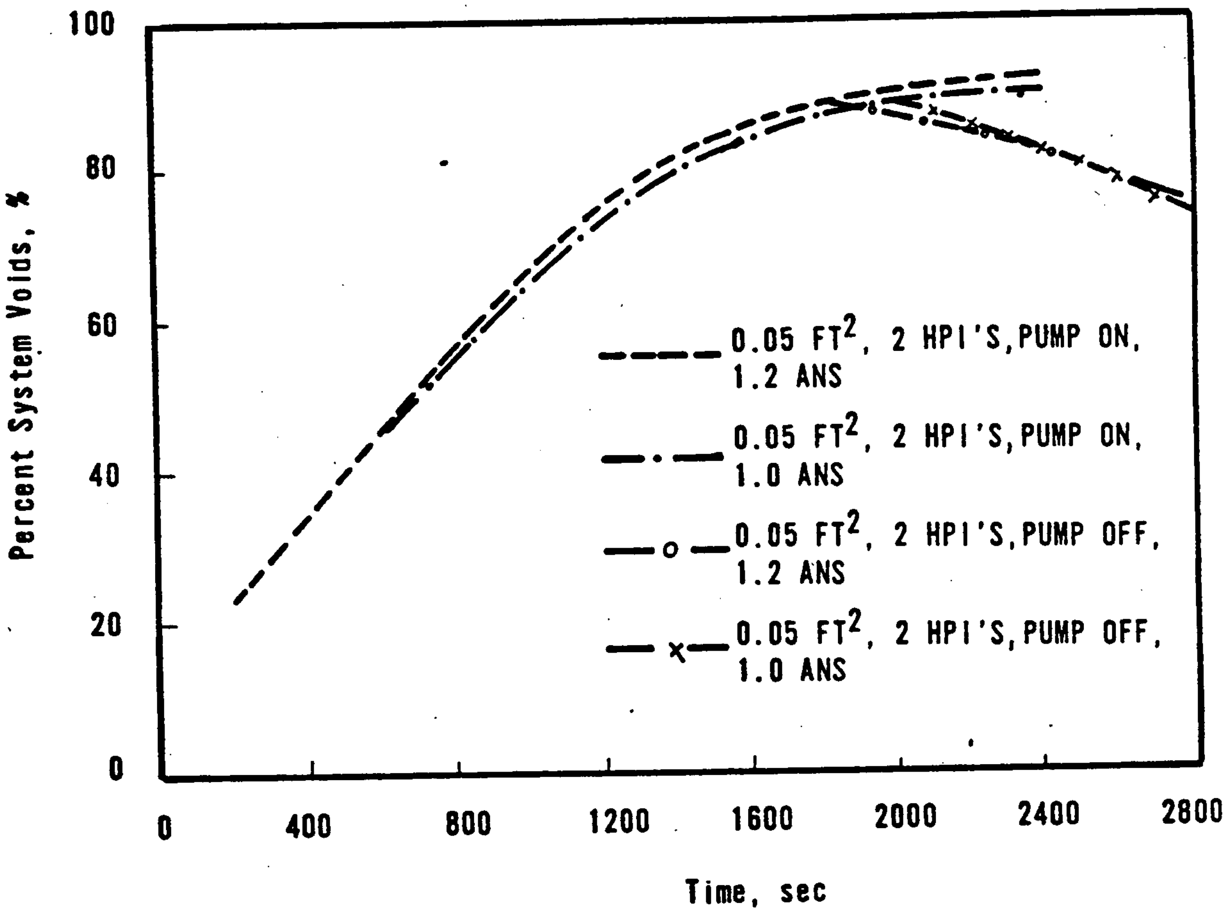


Figure 2-14

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

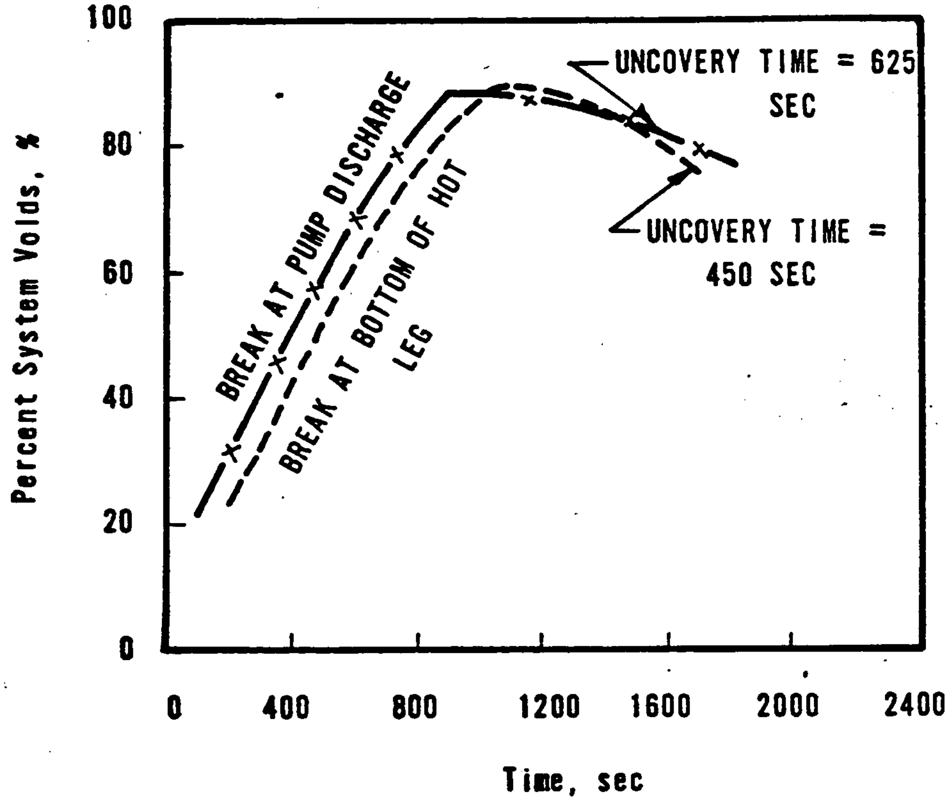


Figure 2-15

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

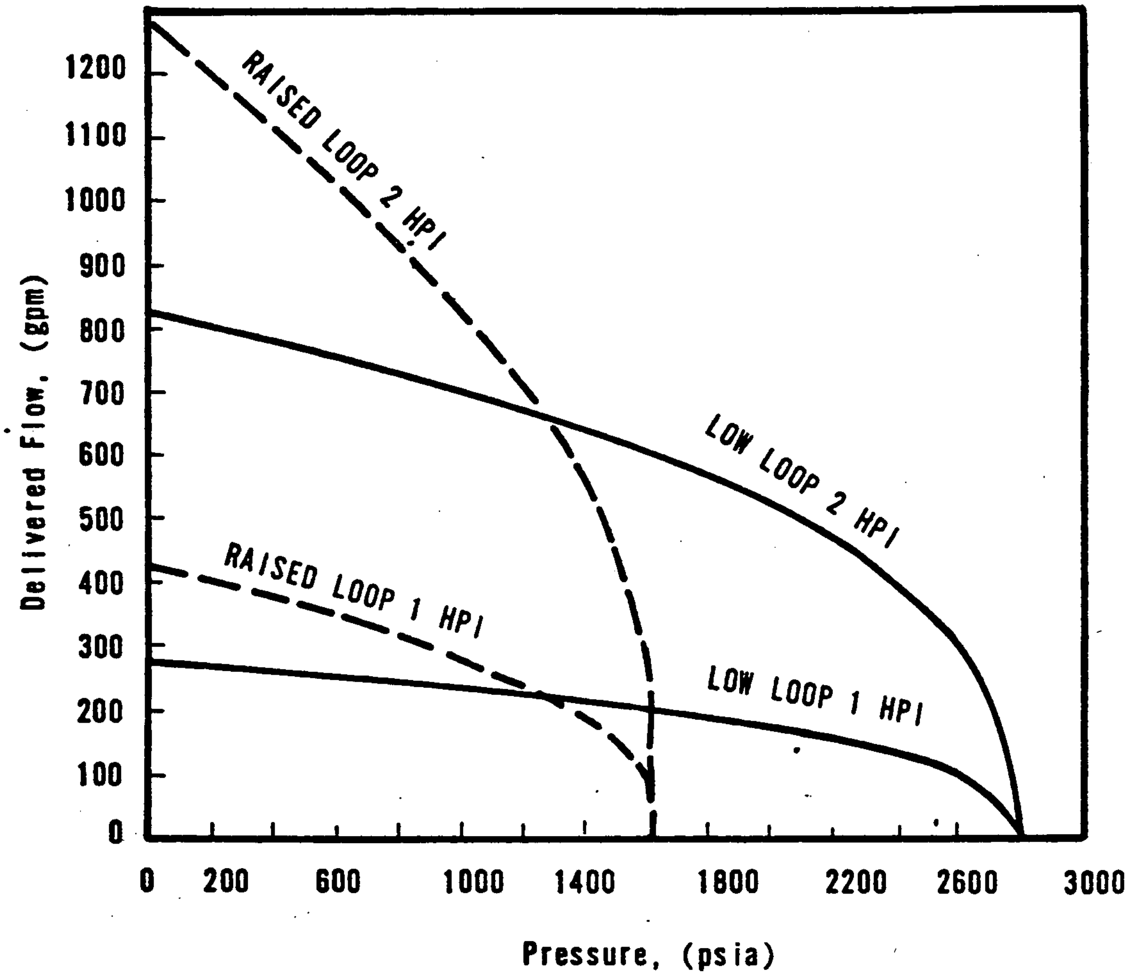


Figure 2-16

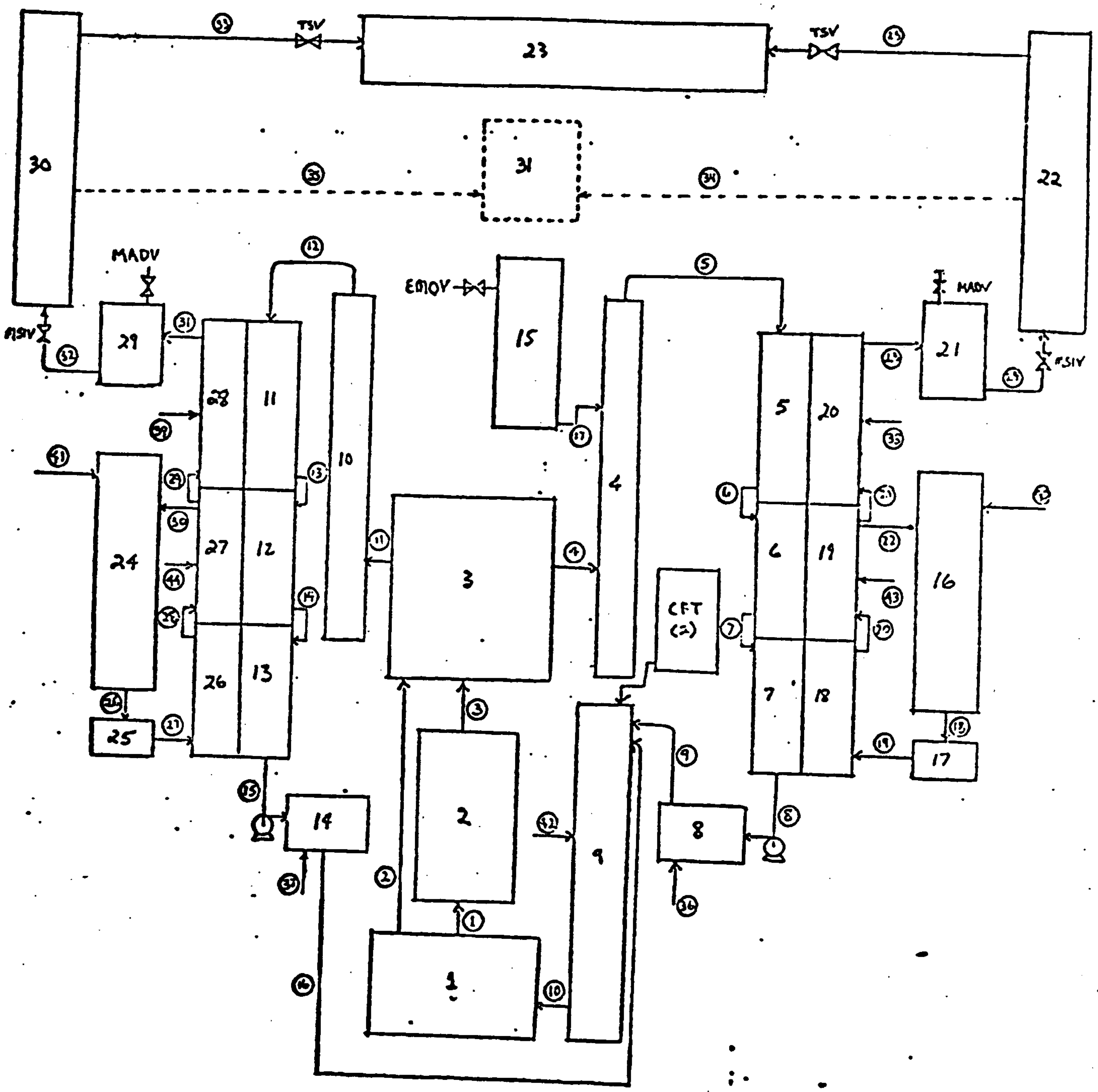


Figure
3.1

MINITRAP2 Noding and
Flow Path Scheme

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)

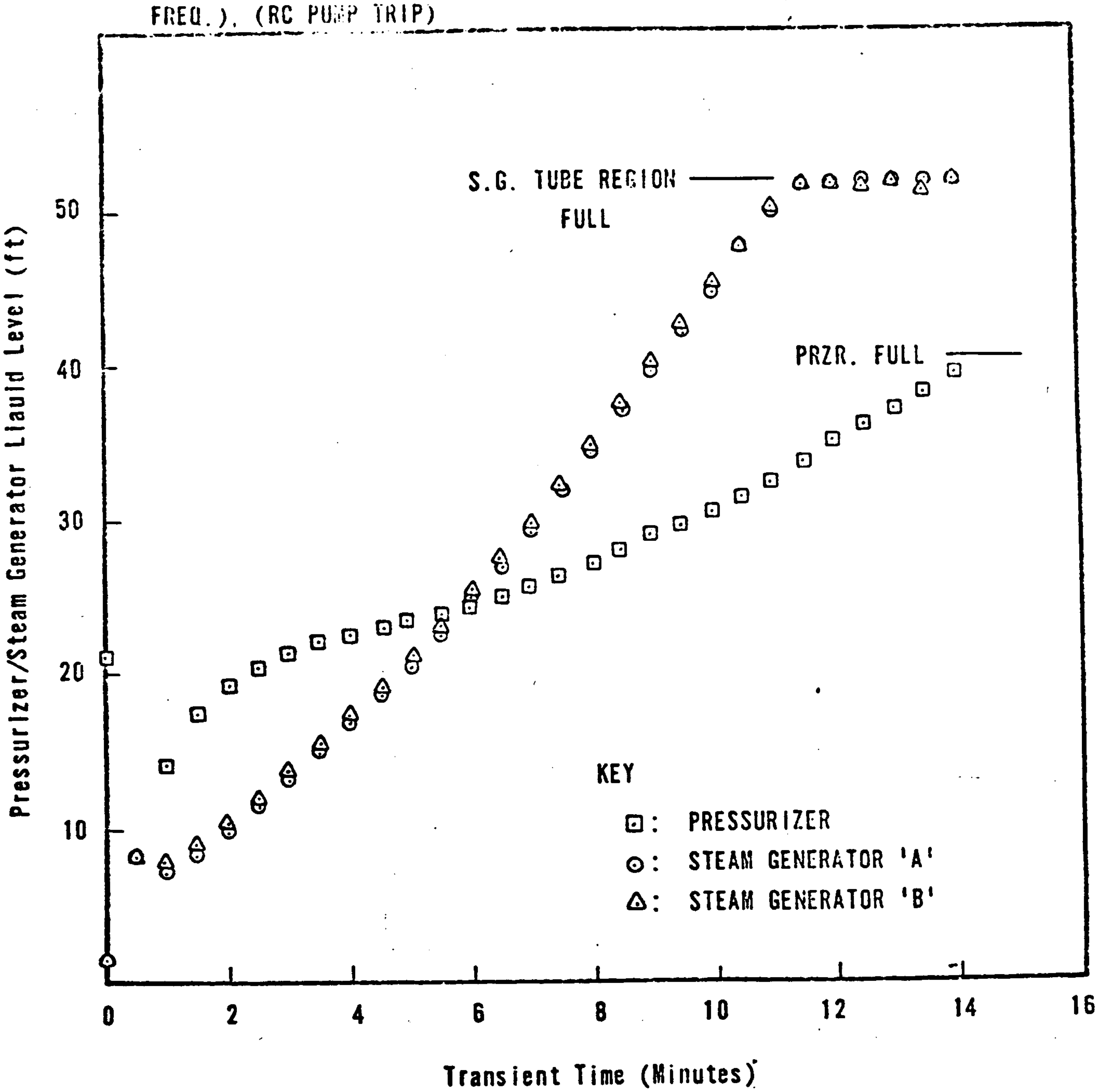


Figure 3.2

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

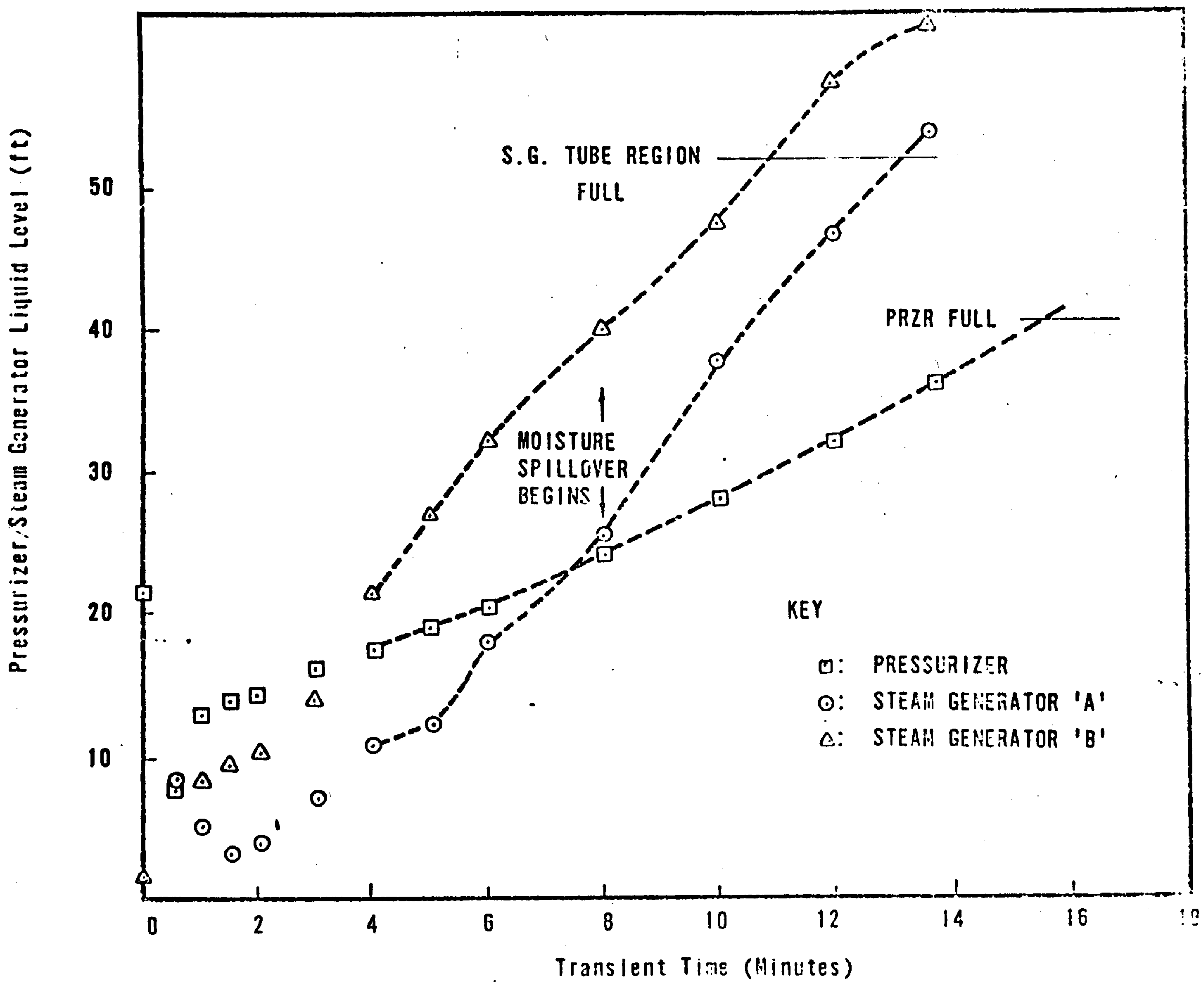


Figure 3.3

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

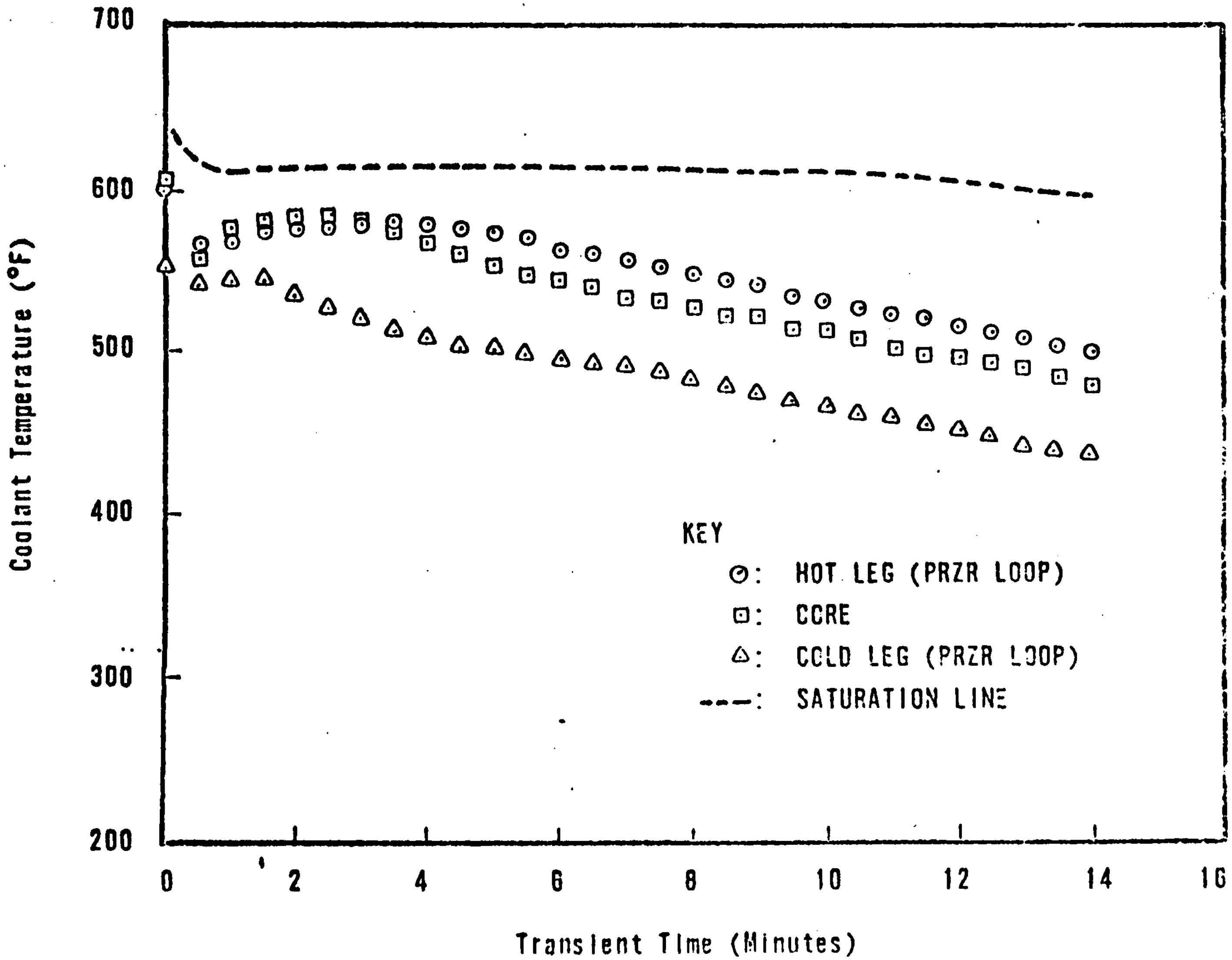
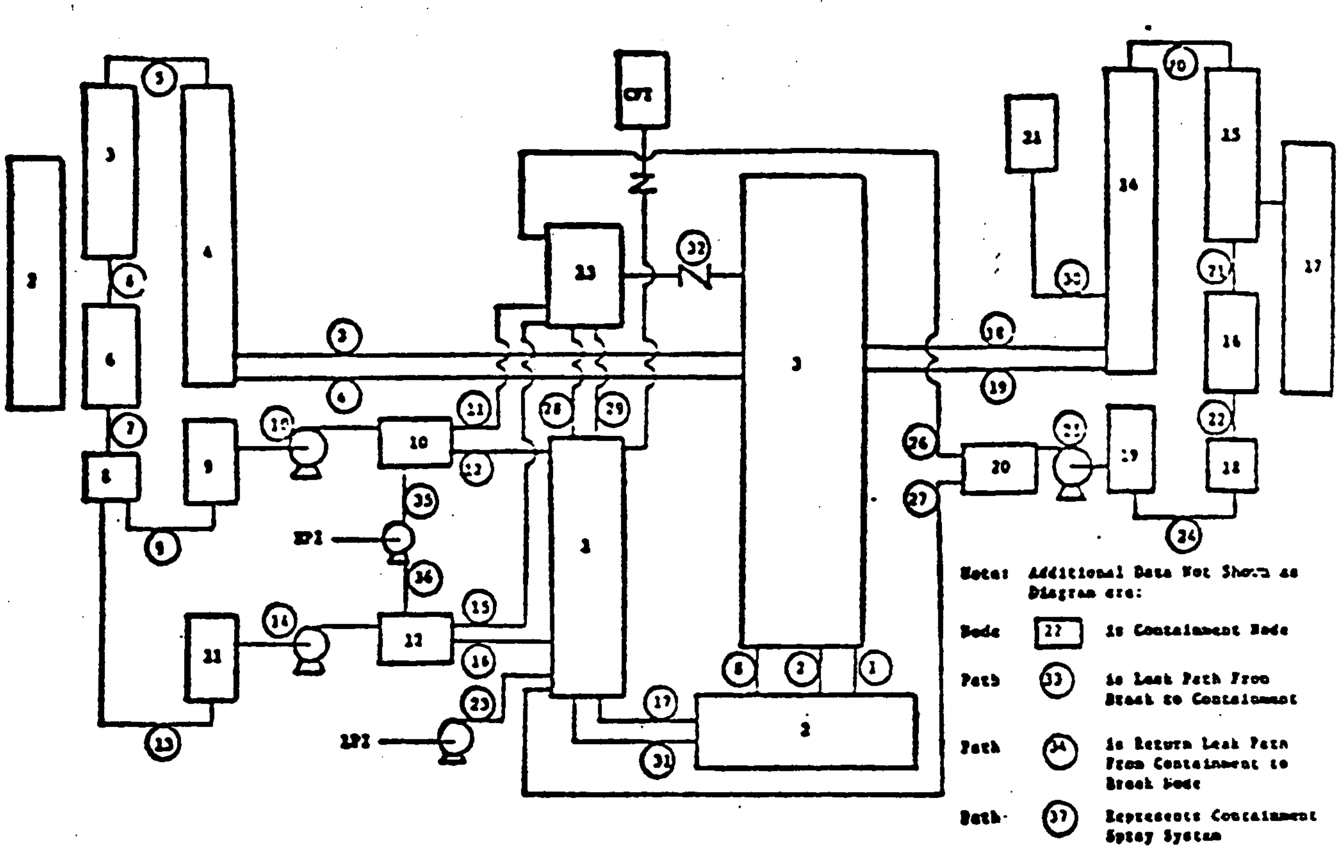


Figure 3.4

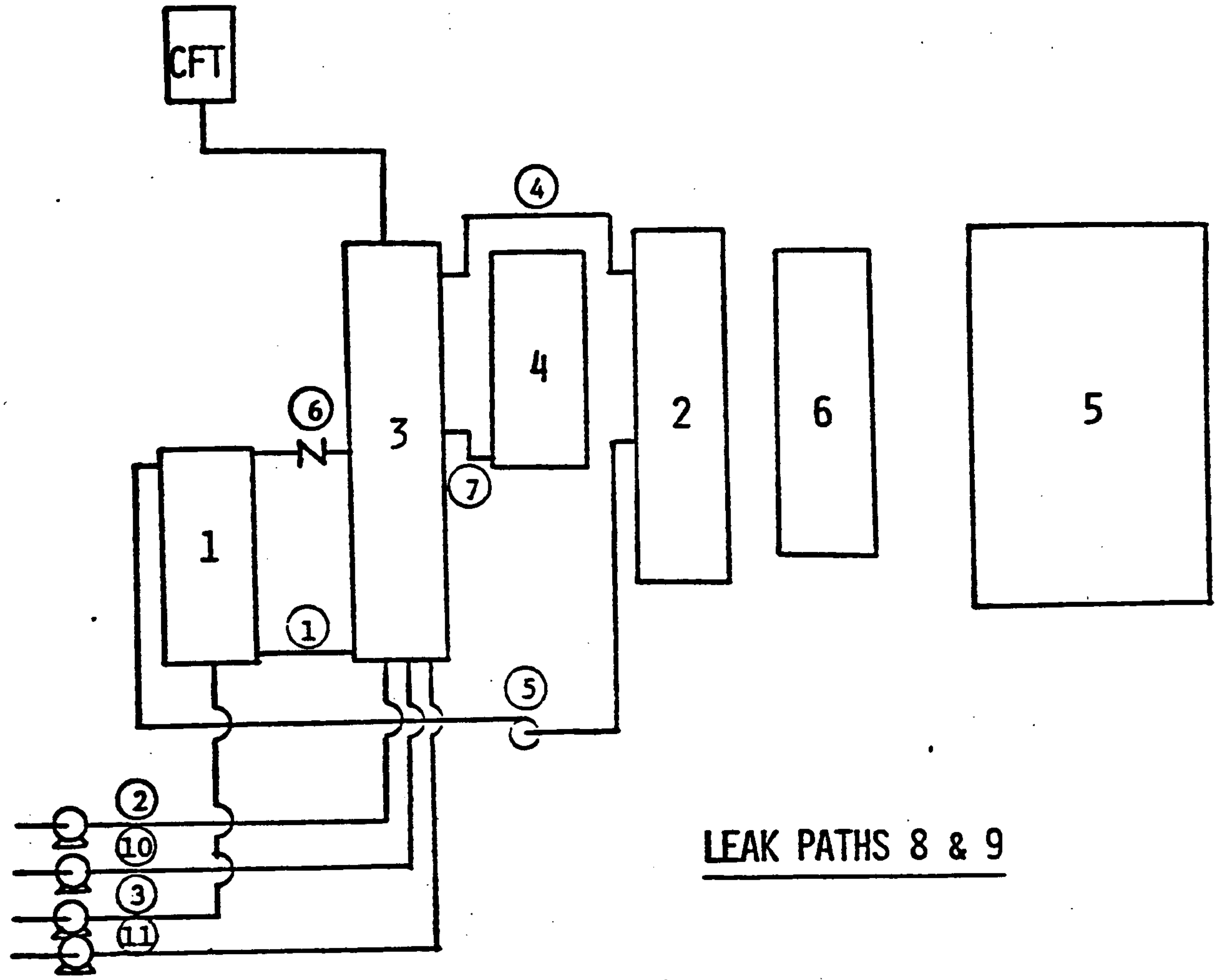
Figure 2-1. CRAFT2 Noding Diagram for Small Break



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

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

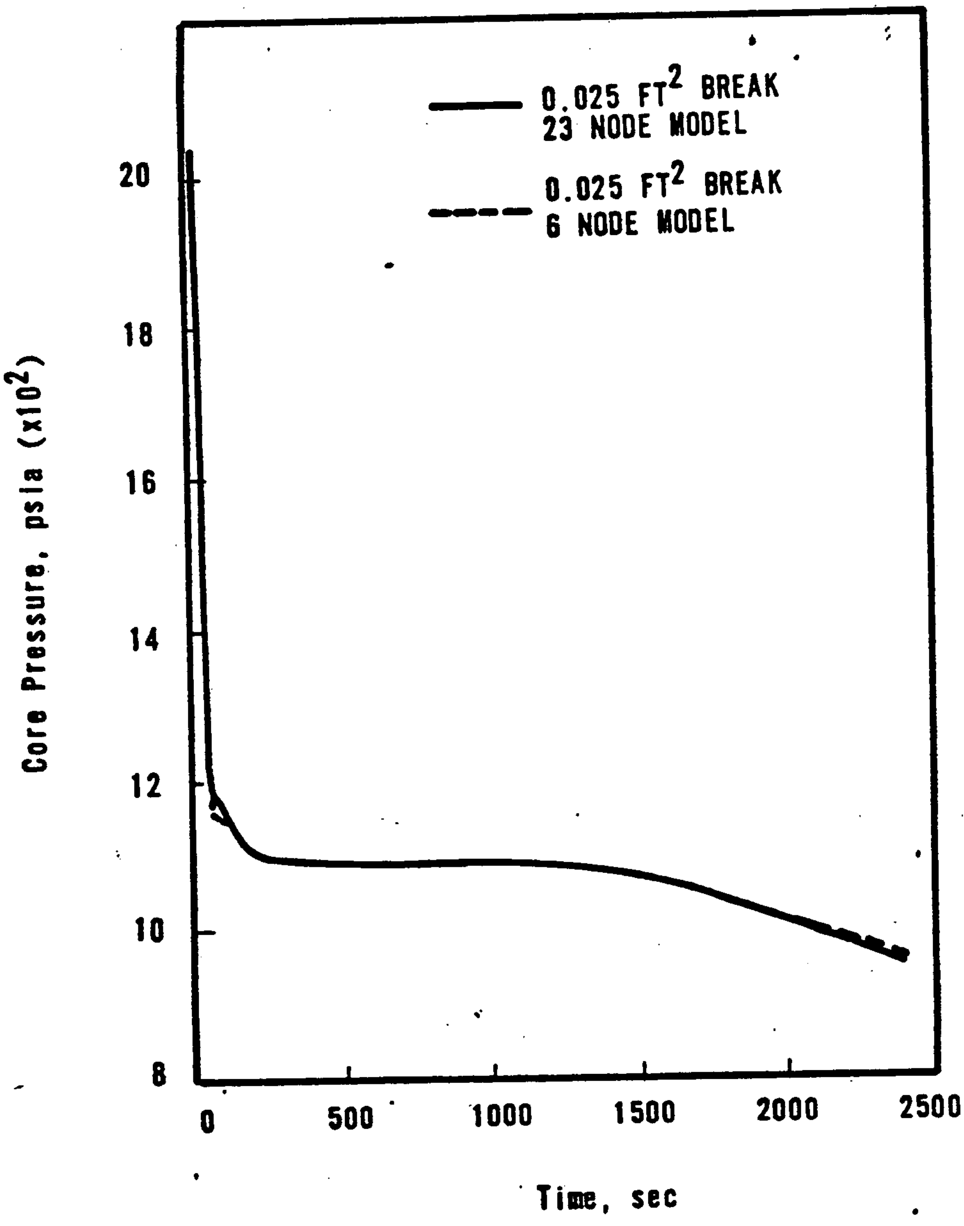


Figure 2-3

PERCENT SYSTEM VOIDS VS TIME, PUMPS ON

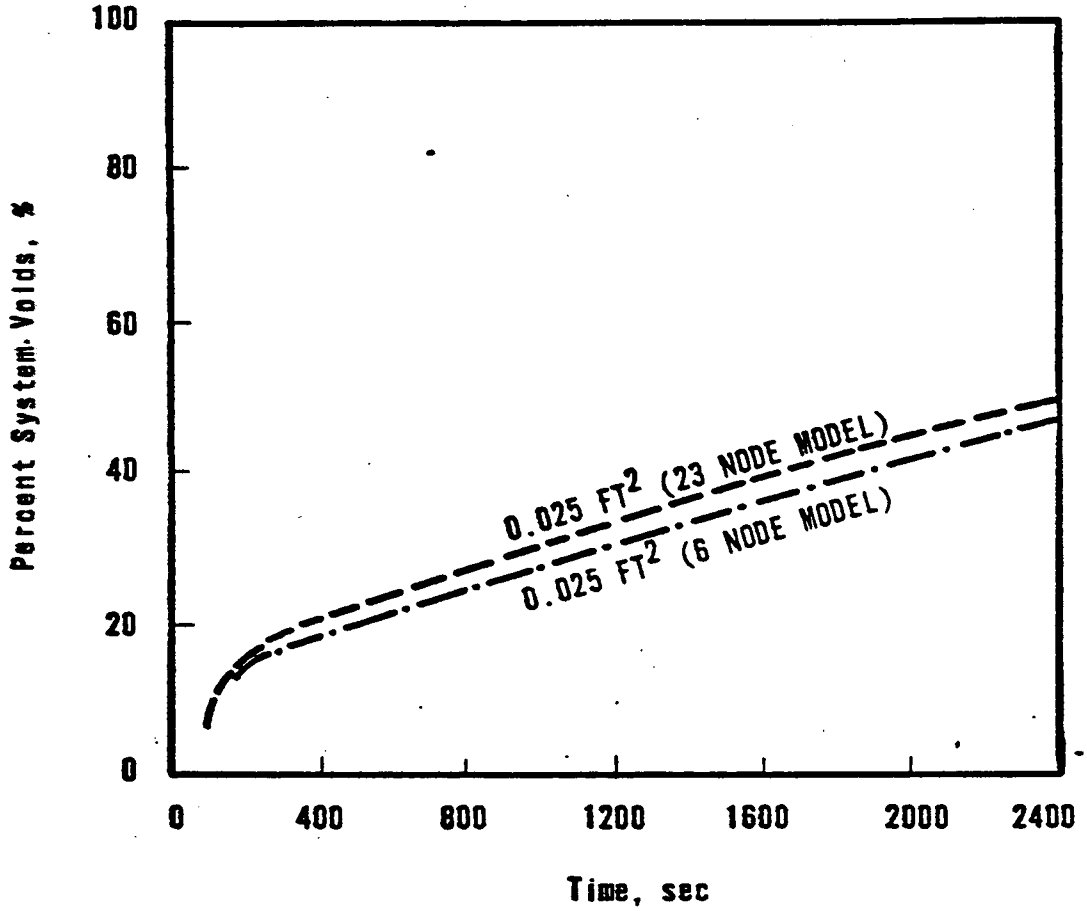


Figure 2-4

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

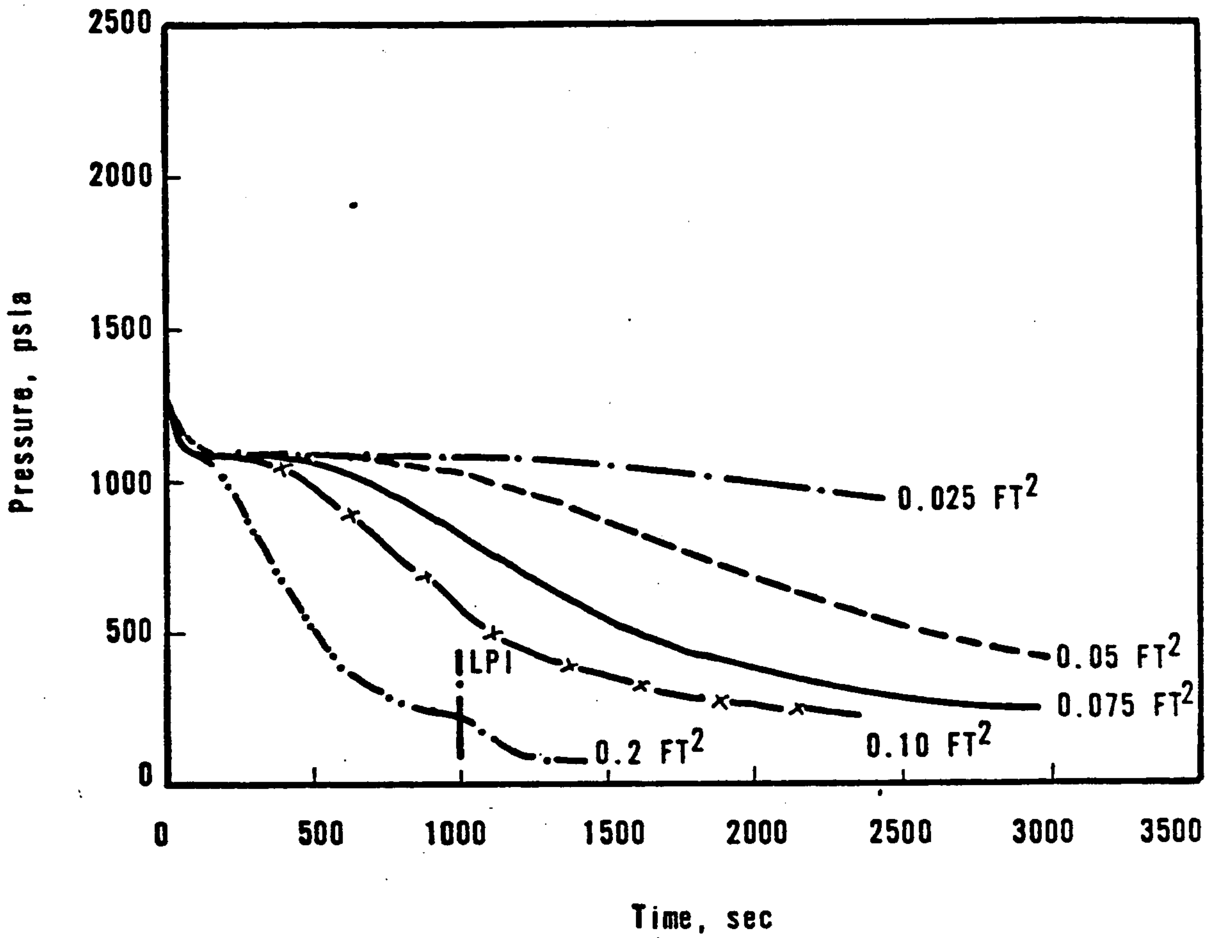


Figure 2-5

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

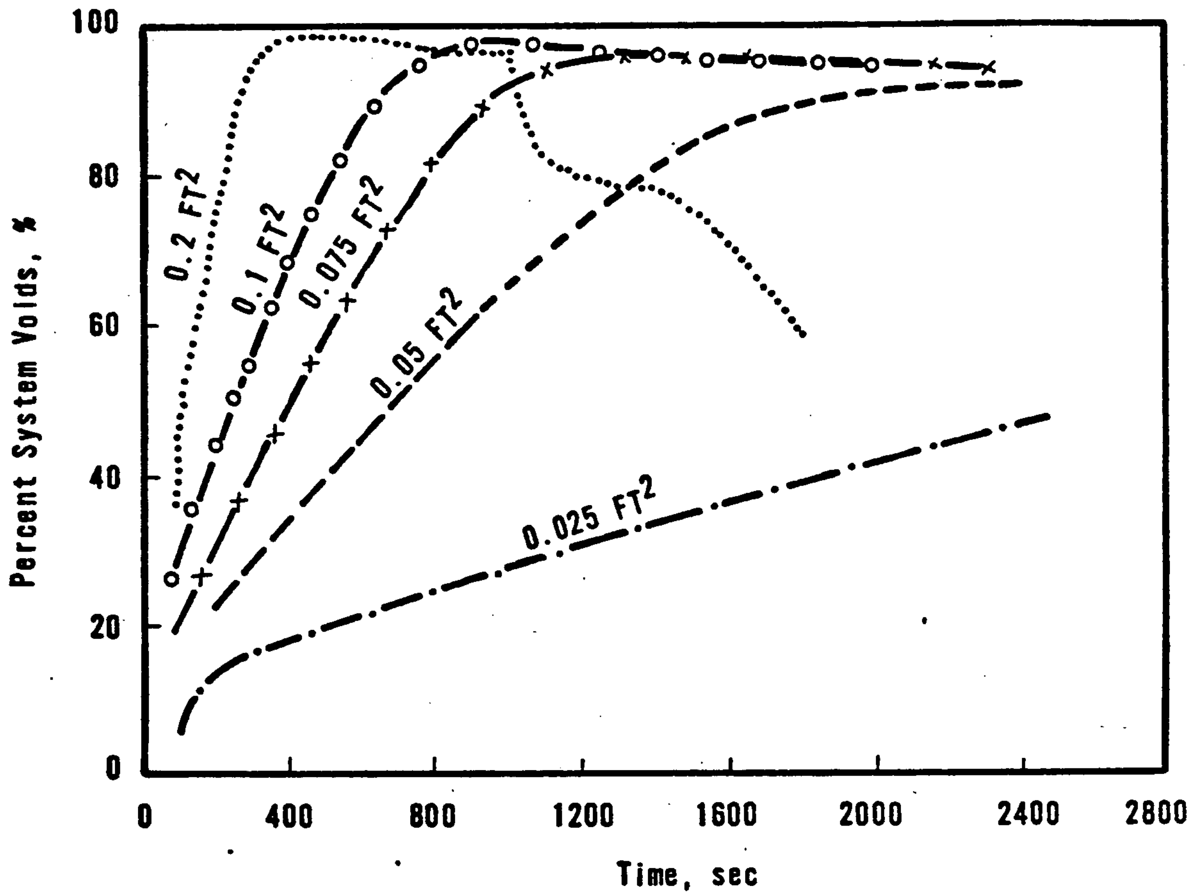


Figure 2-6

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

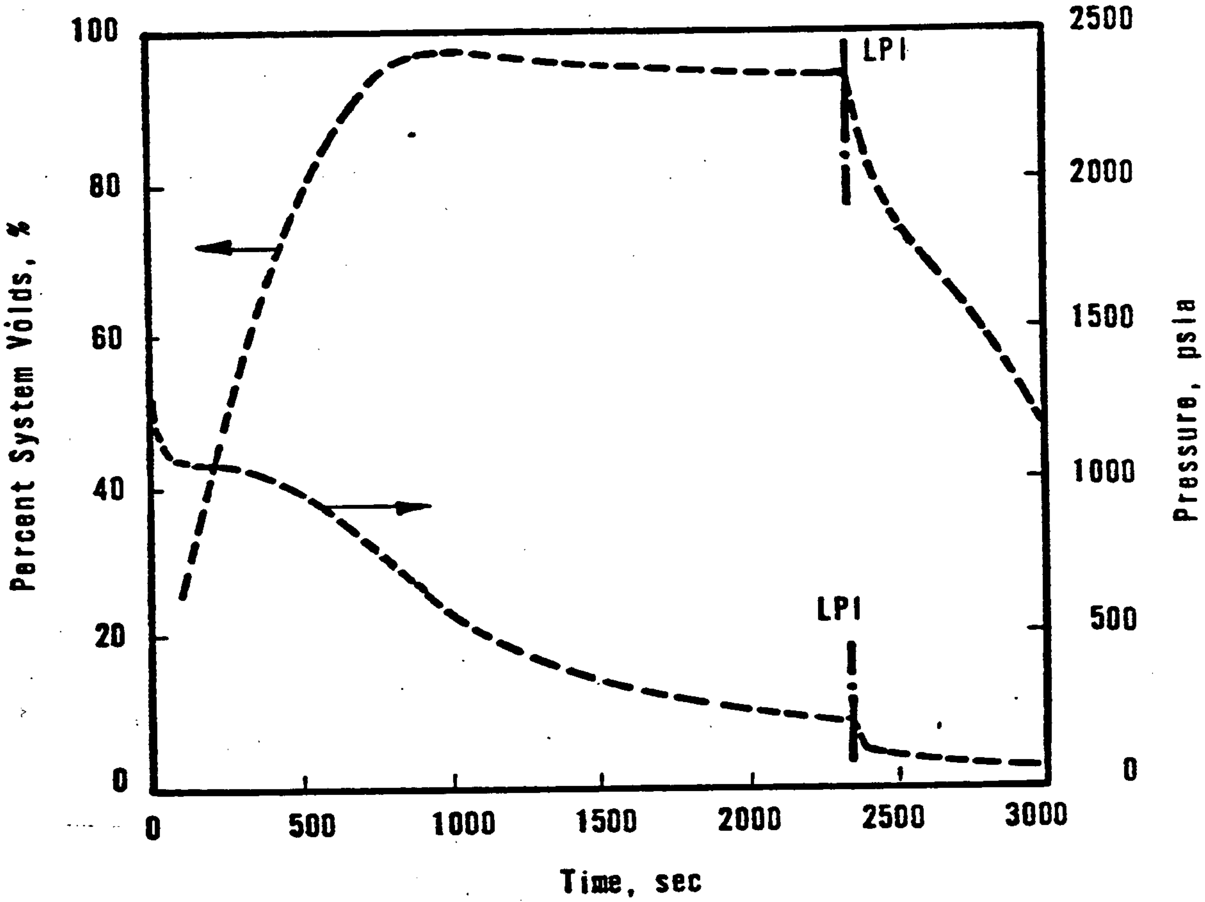


Figure 2-7

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

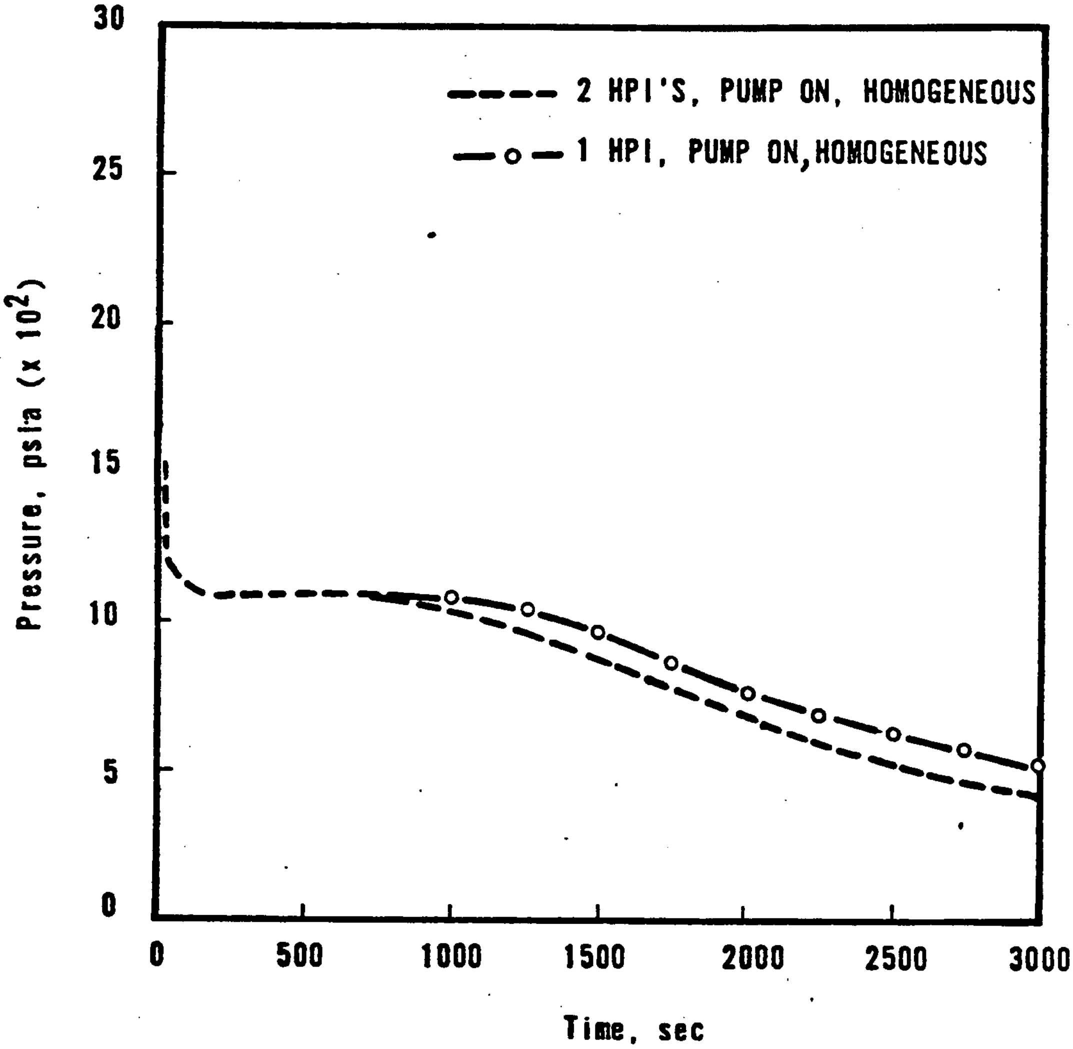


Figure 2-8

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

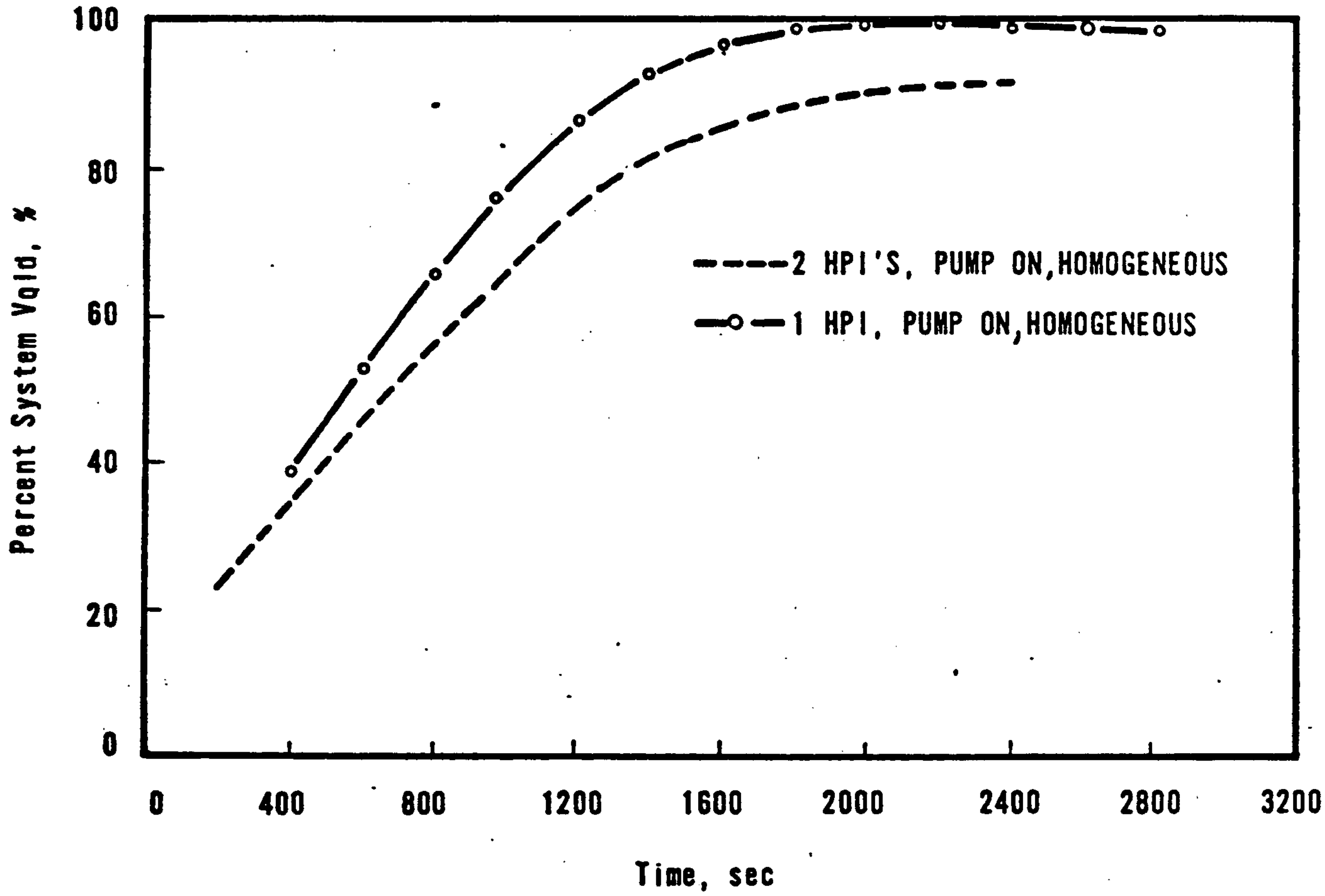


Figure 2-9

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

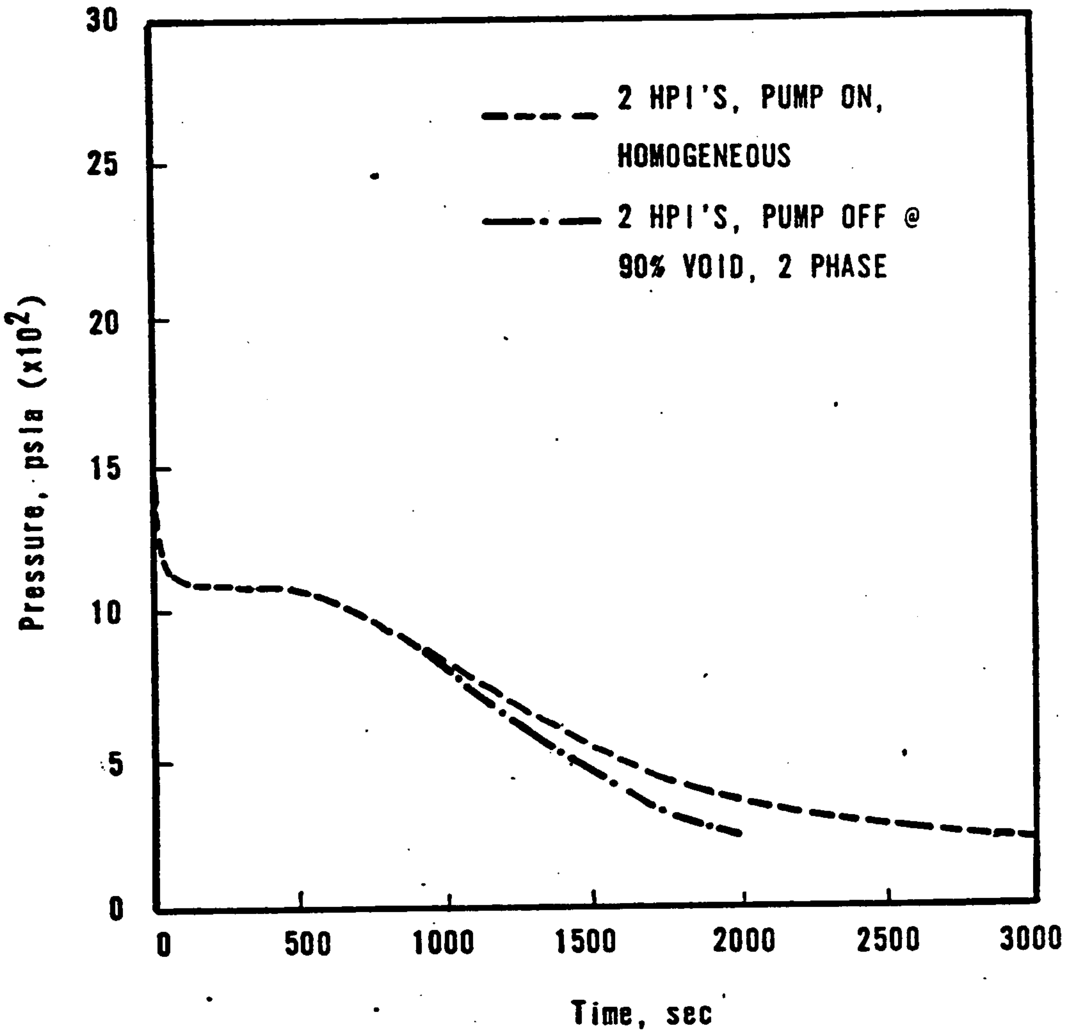


Figure 2-10

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

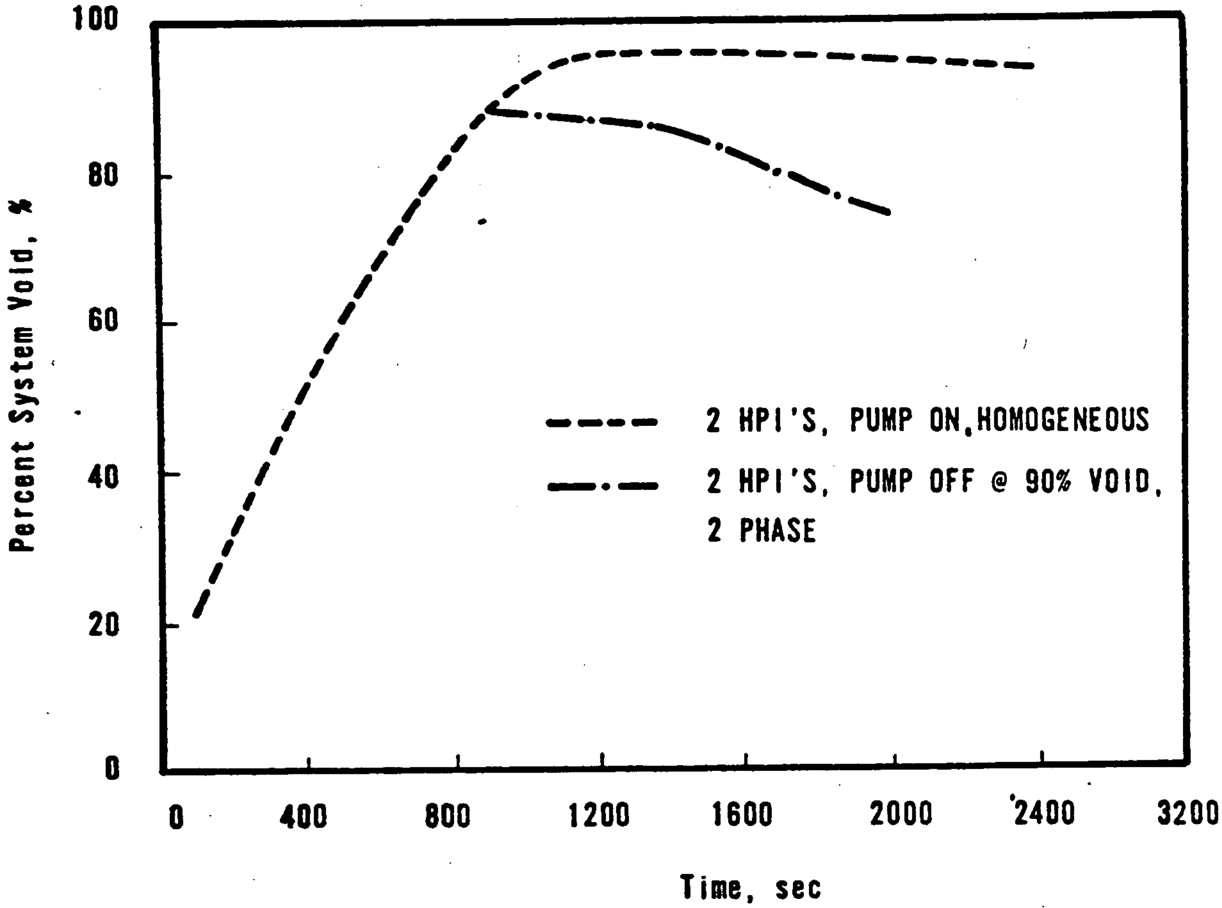


Figure 2-11

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 duration 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		

- - - - -

*Analyzed with both 1.0 and 1.2 ANS decay curves.

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

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

- - - - -

Notes: 1. Two HPIs available during the transient.

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

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

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

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

ATTACHMENT 1

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Response to IE Bulletin 79-05C

SHORT TERM ACTIONS

Item 1

On July 30, 1979, the following actions were taken:

A. The appropriate Emergency Procedures, EP/0/A/1800/4, Loss of Reactor Coolant and EP/0/A/1800/8, Steam System Leak-Rupture, were revised to require that upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, all operating reactor coolant pumps are tripped.

B. A letter was sent by the Superintendent of Operations to all Shift Supervisors requiring all Control Rooms to be manned as described in Bulletin Item 1B.

Item 2

Attachment 2, "Analysis Summary in Support of an Early RC Pump Trip" is provided in response to this item. Section 3 of this report, "Impact Assessment of a RC Pump Trip on Non-LOCA Events," is included to allow the development of preliminary non-LOCA guidelines as required by Item 3.

Item 3

Attachment 3, "Guidelines for Operator Action," provides guidelines on LOCA and non-LOCA transients.

Item 4

Following review and comment by B&W utilities, formal operating guidelines will be issued. At that time, appropriate emergency procedures will be revised and all licensed reactor operators and senior reactor operators on shift will be trained on the new guidelines. This will be completed on or before September 7, 1979.

Item 5

A preliminary response is provided in Attachment 2, Guidelines for Operator Action, Section III, "Criteria for RCP Restart." However, additional analysis is required and will be completed by October 31, 1979.

LONG TERM ACTION

Item 1

Attachment 4 provides a schematic of a conceptual design which would provide automatic tripping of operating reactor coolant pumps (RCP's) on coincident low reactor coolant system (RCS) pressure and low RCP power. The design would use existing engineered safety feature (ESF) low RCS pressure and reactor protective system (RPS) RCP power monitoring signals for input to an 'and' gate. Output would be to either the RCP breakers or the normal or startup feeder breakers. The power sources of each channel are completely independent. This RCP trip could be installed the first refueling outage for each unit following six months after NRC approval.

ATTACHMENT 2

ANALYSIS SUMMARY IN SUPPORT OF AN EARLY RC PUMP TRIP

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.

II. SMALL BREAK ANALYSES

A. Introduction

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

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

B. System Response With RC Pumps Running

1. Introduction

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

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

2. Method of Analysis

The analysis method used for this evaluation is basically that described in section 5 of BAW-10104, Rev. 3, "B&W's ECCS Evaluation Model"[1] and the letter J.H. Taylor (B&W) to S.A. Varga (NRC), dated July 18, 1978[2], which is applicable to the 177-FA lowered-loop plants for power levels up to 2772 MWt. The analysis uses the CRAFT2[3] code to develop the history of the RCS hydrodynamics. However, the CRAFT2 model used for this study is a modification of the small break evaluation model described in the above references. Figure 2-1 shows the CRAFT2 noding diagram for small breaks from the above referenced letter. The modified CRAFT2 model consists of 4 nodes to simulate the primary side, 1 node for the secondary side of the steam generator, and 1 node representing the reactor building. Figure 2-2 shows a schematic diagram of this model. Node 1 contains the cold leg pump discharge piping, downcomer, and lower plenum. Node 2 is the primary side of the SG and the pump suction piping. Node 3 contains the core, upper plenum, and the hot legs. Node 4 is the pressurizer and nodes 5 and 6 represent the reactor building and the SG secondary side, respectively. This 6 node model is highly simplified compared to those utilized in past ECCS analyses. It does, however, maintain RCS volume and elevation relationships which are important to properly evaluate the system response during a small break with the RC pumps running.

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

a. Equipment Availability

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

b. Phase Separation

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

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

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

3. Benchmarking of the 6 Node CRAFT Model

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

4. Analysis Results

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

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

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

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

where

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

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

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

Figure 2-6 shows the evolution of the system void fraction; values in excess of 90% are predicted very early (300 seconds) into the transient. For the larger breaks the system high void fractions occur early in time. For the smaller breaks it takes in the order of hours before the system evolves to high void fraction. Core cooling is maintained during a small break with continuous RC pump design of nuclear generating stations, and, at all times, one of the Chief Engineers has project management responsibility. This responsibility is transferred from the Civil-Environmental Division, to the Mechanical-Nuclear Division and then to the Electrical Division as the job progresses. The organizational structure of the Design Engineering Department is shown in Figure 13.1.1-2.

The Construction Department has the responsibility for all site construction activities. The department is organized by projects and each nuclear station has a separate project group. The Construction Department is responsible for certain field testing (e.g., preoperational hydrostatic testing).

13.1.1.3 Station Support Organization

The Steam Production Department is responsible for operation and maintenance of fossil and nuclear stations. This department provides general supervision and technical management services for the stations. The Steam Production organization is shown in Figure 13.1.1-3. The educational background and experience of key personnel is given in Table 13.1.1-1.

Personnel within the Steam Production Department have considerable nuclear experience from work associated with the Oconee Nuclear Station. As of January 1, 1978, there were approximately 300 technical/professional personnel supporting station operation. It is anticipated that this staff will expand to approximately 490 persons by 1985 in order to support startup and operation of the generating capacity now under construction.

The Nuclear Production Division has line responsibility for operating and maintaining nuclear generation facilities in a safe, economical, and reliable manner to meet Company and regulatory requirements.

The System Operation and Maintenance Group is responsible for providing support to the nuclear stations for major maintenance, materials management and other operation and maintenance functions requiring specialized services including security systems, fire protection, fuel handling equipment, and support of pre-operational and startup activities.

The System Results and Fuel Management Group is responsible for fuel management activities for new and operating generating units. Included in these activities are special nuclear materials accountability and core performance analysis.

The Technical and Environmental Services Group is responsible for providing environmental, chemical, health physics, licensing, computer and instrumentation and control support services for new and operating generating facilities. This group is also responsible for coordinating departmental reviews of nuclear station design and layout.

The Training Services Group is responsible for technical training, employee development and industrial safety programs within the department. These responsibilities include licensed operator training and requalification programs.

Other departments within the Company are available for consultation and assistance as required. The Design Engineering Department is available to furnish nuclear, mechanical, structural, electrical, thermo-hydraulic and metallurgy and materials engineering. Other departments which regularly supply assistance and services for the station are the Transmission Department and the Construction Department.

13.1.2 OPERATING ORGANIZATION

13.1.2.1 Station Organization

The organization of each nuclear station staff generally follows the pattern already proven to be successful in Duke's conventional steam stations and in the Oconee Nuclear Station. The underlying philosophy is that the station staff is to be fully capable and equipped to handle all situations involving safety of the station and public.

The nuclear station staff for one, two, and three units is shown in Figures 13.1.2-1, -2 and -3. Positions shown are functional and may not correspond to actual titles. Each nuclear station is staffed at sufficient levels prior to operation to allow for training, procedure development, and other pre-operational activities.

13.1.2.2 Personnel Functions, Responsibilities and Authorities

The functions and responsibilities of the station supervisory staff are described in the succeeding paragraphs.

(a) Station Manager

The station Manager reports to the Manager, Nuclear Production and has direct responsibility for operating the station in a safe, reliable and efficient manner. He is responsible for protection of the station staff and the general public from radiation exposure and/or any other consequences of an accident at the station. He bears the responsibility for compliance with the facility operating license. The station Manager or his designee has the authority to approve and issue Station Directives and procedures.

(b) Superintendent of Administration

The Superintendent of Administration is responsible for coordination of station administrative functions including clerical, document control, safety, fire protection, training and security. In the event of the absence of the station Manager, the Superintendent of Administration, if so designated, assumes the responsibilities and authority of the station Manager.

(c) Superintendent of Operations

The Superintendent of Operations has the responsibility for directing the actual day-to-day operation of the station. In the event of the absence of the station Manager, the Superintendent of Operations, if so designated, assumes the responsibilities and authority of the station Manager.

(d) Operating Engineer

An Operating Engineer assists the Superintendent of Operations in directing station operation and may assume complete responsibility for station operation in the absence of the Superintendent of Operations.

(e) Shift Supervisor

A Shift Supervisor is responsible for the actual operation of the station on his assigned shift. He directs the activities of the operators on his shift and is cognizant of all maintenance activity being performed while he is on duty. The Shift Supervisor on duty has both the authority and the obligation to shut down a unit if, in his opinion, conditions warrant this action.

(f) Assistant Shift Supervisor

As Assistant Shift Supervisor assists the Shift Supervisor in operation of the station on his assigned shift. The Assistant Shift Supervisor on duty has both the authority and the obligation to shut down a unit if, in his opinion, conditions warrant this action.

(g) Reactor Operator

A Reactor Operator is responsible for the actual operation of a unit on his assigned shift. The Reactor Operator has both the authority and obligation to shut down a unit if, in his opinion, conditions warrant this action.

(h) Utility Operator

A Utility Operator is responsible for the operation of equipment outside of the Control Room.

(i) Superintendent of Technical Services

The Superintendent of Technical Services is responsible for directing the activities of the Technical Services Group, which includes performance, chemistry and health physics. In the event of absence of the station Manager, the Superintendent of Technical Services, if so designated, assumes the responsibilities and authority of the station Manager.

(j) Performance Engineer

The Performance Engineer directs data gathering and evaluation in the areas of equipment and station performance. Specifically included in this are core physics and core performance, from both nuclear and thermal-hydraulic considerations. He assists in setting up fuel shuffling patterns and participates in other phases of fuel management.

(k) Station Health Physicist

The Station Health Physicist has the responsibility for conducting the health physics program. His duties include the training of personnel in use of equipment, control of radiation exposure of personnel, continuous determination of the radiological status of the station, surveillance of radioactive waste disposal operations, conducting the radiological environmental monitoring program and maintaining all required records. He has direct access to the station Manager in matters concerning any phase of radiological protection. The Station Health Physicist also has direct support as required from the System Health Physicist and his staff.

(l) Station Chemist

The Station Chemist is responsible for overall chemistry and radiochemistry requirements, with special emphasis on primary and secondary system water chemistry.

(m) Licensing and Projects Engineer

The Licensing and Projects Engineer has responsibility for coordinating station modification activities and interfaces with regulatory agencies and for providing reviews of appropriate station technical matters.

(n) Superintendent of Maintenance

The Superintendent of Maintenance is responsible for directing the activities of the Maintenance Group, which includes mechanical and electrical maintenance and instrumentation and control. In the event of absence of the station Manager, the Superintendent of Maintenance, if so designated, assumes the responsibilities and authority of the station Manager.

(o) Maintenance Engineer (Mechanical)

The Maintenance Engineer (Mechanical) has responsiblity for maintenance of mechanical equipment.

(p) Maintenance Engineer (I&E)

The Maintenance Engineer (I&E) has responsibility for maintenance of electrical equipment, instrumentation, controls, and computers. He also supervises computer maintenance.

(q) Maintenance Engineer (Planning)

The Maintenance Engineer (Planning) has responsibility for planning and scheduling of all maintenance work as well as directing all material management activities.

(r) Senior Station Quality Assurance Engineer

The functions, responsibilities and authorities of the Senior Station Quality Assurance Engineer are described in Topical Report, DUKE-1A.

13.1.2.3 Shift Crew Composition

A Shift Supervisor is responsible for operation of the station on his shift. Reporting to the Shift Supervisor are operating personnel and at least one individual qualified in radiation protection procedures. For each unit, an operating shift consists of one Assistant Shift Supervisor, who holds a Senior Reactor Operators license, two Reactor Operators both of whom hold a Reactor Operators license, and two Utility Operators who do not necessarily hold an NRC license.

For each unit containing fuel, there is at least one licensed operator in the control room at all times. During refueling operations, an additional Senior Reactor Operator who has no other concurrent responsibilities directly supervises fuel handling operations. Detailed shift crew requirements are defined in the Technical Specifications for each nuclear unit.

13.1.3 QUALIFICATIONS OF STATION PERSONNEL

The qualifications of personnel in the operating staff are in accordance with Section 4 of ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel", and are in accordance with Regulatory Guide 1.8 (Rev. 1) with the exception of those for the Operations Manager in Section 4.2.2 of ANSI N18.1-1971 and for the Radiation Protection Manager in Part C of Reg. Guide 1.8.

The Operations Manager (Superintendent of Operations) shall be an experienced professional in the operation of commercial nuclear facilities and shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of power plant experience may be fulfilled by satisfactory completion of academic training. The Superintendent of Operations shall hold or have held a Senior Reactor Operators license at this facility or other commercial nuclear facility. This minimum qualification for the Superintendent of Operators is deemed satisfactory due to the size of the operating staff and the current level of experience of operations personnel.

The RPM (Station Health Physicist) shall have a bachelor's degree in a science or engineering subject or the equivalent in experience, including some formal training in radiation protection, and shall have at least five years of professional experience in applied radiation protection of which three years shall be in applied radiation protection work in one of Duke Power Company's nuclear stations. A qualified individual who does not meet the above requirements, but who has demonstrated the required radiation protection management capabilities and has professional experience in applied radiation protection work at one of Duke Power Company's multi-unit nuclear stations, may be appointed to the position of Station Health Physicist by the station Manager, based on the recommendations of the System Health Physicist and as approved by the Manager, Nuclear Production.

Replacement personnel for positions in the nuclear stations are fully trained and qualified to fill their appointed positions.

13.1.3.1 Minimum Qualification Requirements

The minimum qualification requirements for station personnel are outlined in the suceeding paragraphs.

(a) Station Manager

The station Manager shall have a minimum of ten years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of four years of the remaining seven years of experience may be fulfilled by academic training on a one-for-one, time basis. To be acceptable, this academic training shall be in an engineering or scientific field generally associated with power production. The station Manager shall have acquired the experience and training normally required for examination by the NRC for a Senior Reactor Operator license, whether or not the examination is taken.

(b) Superintendent of Operations

The Superintendent of Operations shall have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of two years of the remaining five years of experience may be fulfilled by academic training, or related technical training, on a one-for-one, time basis. The Superintendent of Operations shall hold or have held a Senior Reactor Operator license.

(c) Superintendent of Technical Services

The Superintendent of Technical Services should have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of one year shall be nuclear station experience. A maximum of four years of the remaining seven years of experience should be fulfilled by satisfactory completion of academic training.

(d) Superintendent of Maintenance

The Superintendent of Maintenance shall have a minimum of seven years of responsible nuclear or fossil station experience, or applicable industrial experience, of which a minimum of one year shall be nuclear station experience. A maximum of two years of the remaining six years of experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. The Superintendent of Maintenance should also have non-destructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel and piping codes.

(e) Operating Engineer

An Operating Engineer shall have a minimum of a high school diploma, or equivalent, and four years of responsible nuclear or fossil station experience, of which a minimum of one year shall be nuclear station experience. A maximum of two years of the remaining three years of experience may be fulfilled by academic or related technical training on a one-for-one, time basis. An Operating Engineer shall hold a Senior Reactor Operator license.

(f) Performance Engineer

The Performance Engineer shall have a minimum of a Bachelor's degree in engineering or the physical sciences and two years of responsible nuclear power plant experience. The Performance Engineer or the Reactor Engineer shall have two years of experience in such areas as reactor physics, core measurements, core heat transfer, and core physics testing programs.

(g) Station Chemist

The Station Chemist shall have a minimum of five years of experience in chemistry, of which a minimum of one year shall be in radiochemistry. A minimum of two years of this five years of experience should be fulfilled by academic or related technical training. A maximum of four years of this five years of experience may be fulfilled by academic or related technical training.

(h) Station Health Physicist

The Station Health Physicist shall have a minimum of seven years of experience in radiation protection at a nuclear facility. A minimum of two years of this seven years of experience should be related technical training. A maximum of four years of this seven years of experience may be fulfilled by academic or related technical training.

(i) Licensing and Projects Engineer

The Licensing and Projects Engineer shall have a minimum of five years of technical experience, of which a minimum of one year shall be nuclear station experience. A minimum of four years of this five years experience may be fulfilled by related technical or academic training.

(j) Maintenance Engineer (Mechanical)

The Maintenance Engineer (Mechanical) shall have a high school diploma, or equivalent, and a minimum of four years of experience in maintenance activities.

(k) Maintenance Engineer (I&E)

The Maintenance Engineer (I&E) shall have a minimum of five years of experience in instrumentation and control of which a minimum of six months shall be in nuclear instrumentation and control. A minimum of two years of this five years of experience should be fulfilled by academic or related technical training. A maximum of four years of this five years of experience may be fulfilled by academic or related technical training.

(l) Maintenance Engineer (Planning)

The Maintenance Engineer (Planning) shall have the same qualifications as the Maintenance Engineer (Mechanical).

(m) Shift Supervisor

A Shift Supervisor shall have the same qualifications as the Operating Engineer.

(n) Assistant Shift Supervisor

An Assistant Shift Supervisor shall have the same qualifications as a Shift Supervisor.

(O) Operators

Operators to be licensed by the NRC shall have a high shcool diploma, or equivalent, and two years of nuclear or fossil station experience, of which a minimum of one year shall be nuclear station experience. In order to be acceptable for full responsibility in a job, they shall hold a Reactor Operator license.

Operators, whether or not they are to be licensed by the NRC, should have a high school diploma, or equivalent, and should possess a high degree of manual dexterity and mature judgment.

(p) Technicians

Technicians in responsible positions (i.e., individuals who direct the activities of others and who are responsible for the activities they direct) shall have a minimum of two years of experience in their specialty. These personnel should have a minimum of one year of related technical training in addition to their experience.

(q) Maintenance Personnel

Maintenance personnel in responsible positions (i.e., individuals who direct the activities of others and who are responsible for the activities they direct) shall have a minimum of three years of experience in one or more crafts. They should possess a high degree of manual dexterity and ability, and should be capable of learning and applying basic skills in maintenance operations.

13.1.3.2 <u>Qualifications of Station Personnel</u>

Qualifications of station personnel are provided in the respective FSAR's.

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.

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 ($\sim$1100 psia) about 200 seconds earlier than the case with 1.2 ANS as a result of reduced decay heat. Also, the system will evolve to a lower average void fraction as shown in Figure 2-14. After the pumps trip at 90% system void fraction, the case with 1.0 ANS decay curve has a shorter core uncovery time by approximately 200 seconds compared to 1.2 ANS case. This case demonstrates that the effect of a delayed RC pump trip may be acceptable when viewed realistically. A peak cladding temperature assessment for this case will be provided in a supplementary response planned for September 15th, to the I&E Bulletin 7905-C.

6. Effect of No Auxiliary Feedwater

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

.7 Break Location Sensitivity Study

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

8 Peak Cladding Temperature Assessment

As described previously, a RC pump trip, at the time the RCS void fraction is 90%, will result in core uncovery times of approximately 600 seconds. The peak cladding temperatures for these cases were evaluated using the small break evaluation model core power shape used to demonstrate compliance with Appendix K and 10CFR50.46. Also, an adiabatic heatup assumption during the time of core uncovery was utilized. This approach is extremely conservative in that the power shape and local power rate (kw/ft) analyzed is not expected to occur during normal plant operation. Furthermore, use of an adiabatic heatup assumption neglects any credit for the steam cooling that will occur during the core refill phase and also neglects the effect of any radiation heat transfer. Using a decay heat power level based on 1.2 ANS at 1500 seconds, the cladding will heatup at a rate will be 6.5 F/S under the adiabatic assumption. With a core uncovery period of 600 seconds and the adiabatic heatup assumption, cladding temperatures will exceed the criteria of 10CFR50.46. Use of a more realistic heat transfer approach with the extreme power shape utilized for this evaluation is also expected to result in cladding temperature in excess of the criteria. In order to ensure compliance of the 177 FA lowered loop plants to the criteria of 10CFR50.46 a prompt 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.

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-