Safety Analysis Input to Startup Team Safety Assessment Report

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In a review of emergency diesel generator (EDG) capacities and emergency feedwater (EFW) dependencies, Crystal River Unit-3 (CR-3) determined that insufficient margins could exist for some Emergency Safeguards equipment in mitigation of small break loss-of-coolant accidents (SBLOCAs) subject to coincident loss of offsite power and specific assumed single failures.

The CR-3 final safety analysis report (FSAR), the improved technical specifications, and various supporting design bases documents were reviewed to identify the limiting transient(s) and single failures that could challenge the EDGs and the ability to preserve/maintain EFW flow to the OTSGs. The limiting transients were narrowed to a single event:, a SBLOCA with a coincident loss of offsite power (LOOP) occurring on reactor trip (Figure 1). The single failures of concern were identified as (1) the loss of battery "A" (LOBA), (2) loss of battery "B" (LOBB), and (3) failure of the turbine-driven EFW pump.

A successful mitigation path was identified for each of the three single failures. These "solution sets" are depicted in Figures 2 through 4. In each of the solution sets, certain challenges were identified. This document presents the Framatome Technologies, Inc. (FTI) safety analyses and evaluations addressing those challenges.

Record Of Revisions

Rev.	Description	Affected Pages
00	Original Release	All
01	Addition Comments	All

Table of Contents

List of Tables

List of Figures

List of Figures (Continued)

List of Acronyms

ADVs	Atmospheric Dump Valves
ANS	American Nuclear Society
BWST	Borated Water Storage Tank
CC	Control Complex
CLPD	Cold Leg Pump Discharge
CR-3	Crystal River Unit 3
ECCS	Emergency Core Coolant System
EDG	Emergency Diesel Generator
EFIC	Emergency Feedwater Integrated Control
EFP-1	Emergency Feedwater Pump-1 (Motor-driven)
EFP-2	Emergency Feedwater Pump-2 (Turbine-driven)
EFW	Emergency Feedwater
EOPs	Emergency Operating Procedures
FPC	Florida Power Corporation
FPS	Full Power Seconds
ESAS	Engineered Safeguards Actuation System
FSAR	Final Safety Analysis Report
FTI	Framatome Technologies, Inc.
HPI	High Pressure Injection
LOBA	Loss of Battery 'A'
LOBB	Loss of Battery 'B'
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
LPI	Low Pressure Injection
LSCM	Loss of Subcooling Margin
MSSV	Main Steam Safety Valve
MUP	Makeup Pump
MUV	Makeup Valve
MWt	Mega-Watts Thermal
OTSG	Once-Through Steam Generator
PSV	Pressurizer Safety Valve
Q_{core}	Core Decay Heat
Q_{hpi}	Energy Absorption of HPI Fluid
RB	Reactor Building

List of Acronyms (Continued)

RBES	Reactor Building Emergency Sump
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulation Guide
SBLOCA	Small Break LOCA
SCM	Subcooling Margin
W_{hpi}	HPI Mass Flow Rate

1.0 BACKGROUND

In a review of emergency diesel generator (EDG) capacities and emergency feedwater (EFW) dependencies, Crystal River Unit-3 (CR-3) determined that insufficient margins could exist for some Engineered Safeguards equipment in mitigation of small break loss-of-coolant accidents (SBLOCAs) subject to coincident loss of offsite power and specific assumed single failures.

Certain SBLOCAs require cooling via EFW to the once-through steam generators (OTSGs) for some period of time for successful mitigation. At CR-3, EFW is provided by a motor-driven pump (EFP-1), powered by the 'A'-Train emergency diesel generator, and by a turbine-driven pump (EFP-2), powered by steam from the OTSGs. The 'A'-Train emergency diesel generator (EDG-1A) cannot support concurrent operation of the EFP-1 with either the 'A'-Train control complex (CC) cooling system operating, or with the 'A'-Train low pressure injection (LPI) pump operating coincident with reactor building (RB) spray pump operation. This represents an operational limitation associated with EDG-1A that, in combination with specific single failures, could pose challenges to the mitigation of accident events requiring EFW.

The CR-3 final safety analysis report (FSAR), the improved technical specifications, and various supporting design bases documents were reviewed to identify the limiting transient(s) and single failures that could challenge the EDGs and the ability to preserve or maintain EFW flow to the OTSGs. The limiting transients were narrowed to a SBLOCA with a coincident loss of offsite power (LOOP) occurring on reactor trip (Figure 1). The single failures of concern were identified as (1) the loss of battery 'A' (LOBA), (2) loss of battery "B" (LOBB), and (3) failure of the turbine-driven EFW pump. (Note that a battery failure will disable the associated EDG and the equipment powered by that EDG.) Specific descriptions of these postulated single failures and accident mitigation challenges are contained in Reference 1.

All event sequences begin with the plant operating at power in Mode 1 when a SBLOCA occurs in the reactor coolant cold leg pump discharge (CLPD) piping region, or a high pressure injection (HPI) line breaks between the last check valve and the CLPD piping, or a core flood line breaks between the last check valve and the reactor vessel. The CLPD break region is the most limiting relative to available HPI flow for core cooling because a portion of the emergency core cooling system (ECCS) injection fluid is lost directly through the break. Due to the loss of subcooled liquid, the reactor coolant system (RCS) begins to depressurize. The reactor trips on low RCS pressure with a coincident LOOP. Subcooling margin will be lost and, if it is not automatically actuated on low RCS pressure, the operators will manually initiate high pressure injection (HPI) flow. The emergency feedwater initiation and control (EFIC) system actuates, and EFW will begin to fill the OTSGs to the required setpoint. These events are consistent with assumptions modeled in the SBLOCA design basis analyses.

A success path was identified for each of the three single failures. These "solution sets" are depicted in Figures 2 through 4 and are described in Reference 1. In each of the solution sets, certain challenges were identified. This document presents the Framatome Technologies, Inc. (FTI) safety analyses and evaluations addressing some of those challenges.

2.0 Solution Set Challenges and Resolution Summary

The solution sets for successful mitigation of SBLOCA rely upon the operability and capacities of the emergency feedwater and safety injection systems subject to specific single failures. In developing the solution sets, six challenges to successful SBLOCA mitigation were identified. Each of these challenges are discussed below, along with a summary of the resolution to each challenge. Additional discussion of how each challenge was resolved is provided in the subsequent sections (Sections 3 through 8).

1. With two HPI pumps, is EFW required for SBLOCA mitigation? (Section 3.0)

   This challenge is related to the single failure of EFP-2, when both HPI pumps would be available. If the flow from two HPI pumps is not sufficient to provide core cooling for all break sizes, then EFW will be necessary for some period of time.

   As discussed in Section 3.0, for certain small breaks, steam generator cooling would be required for a period of time--depending upon the break type and size--to assure adequate HPI flow and core cooling. These small breaks were identified as those smaller than approximately 0.04 ft^2 with one HPI pump available and smaller than roughly 0.01 ft^2 with two HPI pumps operating. If EFW is lost for these break sizes, the RCS could repressurize to the pressurizer safety valve setpoint before HPI alone is capable of removing core decay heat, i.e., break/HPI cooling.

2. Under realistic and Appendix K decay heat assumptions, with one and two operating HPI pumps, how long is emergency feedwater required? What is the EFW mission time? (Section 4.0)

   In Section 4.0, EFW mission times are calculated to determine the minimum times--for each of the three significant single failures--that feedwater would be required before break/HPI cooling could match core decay heat. These mission times define how long EFW must be available to ensure adequate core cooling for each of the three single failures. Beyond these EFW mission times, break/HPI cooling alone will provide adequate core cooling.

   With one HPI pump (for either a LOBA or LOBB single failure), the BWST will be drained before the HPI flow into the RCS is sufficient to remove the core decay heat. Therefore, HPI flow will be switched to recirculation from the RB sump, via the LPI pumps (piggy-back operation) with EFW still required. Accounting for the

higher enthalpy of the injection fluid during the recirculation mode, the time for the HPI flow to match the core decay heat is 35 hours, using conservative decay heat. With realistic core decay heat, this time is significantly reduced to 14.2 hours. However, the EFW mission time with realistic decay heat is established by the BWST drain time of 18.84 hours.

With two HPI pumps operating (for a single failure of EFP-2), the core decay heat can be matched at 1.6 hours post-LOCA when in the recirculation mode. For realistic decay heat levels, this time is reduced to less than one hour. For the EFP-2 single failure, the HPI flow--based on taking suction from the BWST--will not match the core decay heat before the BWST is drained. With the CR-3 HPI flows, the BWST will not reach the sump switchover level until after 2.7 hours. Therefore, EFP-1 can be secured at that time because HPI flow alone will provide adequate core cooling at the time of 'A'-Train LPI pump startup for recirculation mode, piggy-back operation. That is, the EFW mission time is 2.7 hours for those break sizes that require EFW.

3. Will EFP-2 be operable through the required mission time? (Section 5.0)

For the LOBA single failure, EFP-1 will be unavailable, and EFP-2 must provide the required EFW to ensure adequate core cooling. For the LOBB single failure, EFP-2 must be crosstied to the 'A'-powered EFW control valves to provide the necessary EFW before EFP-1 is secured. This is because EFP-1 must be secured to allow loading of the CC cooling system, or the 'A' Train LPI pump on the 'A' Train EDG for recirculation, or if the EFW/LPI interlock trips EFP-1 at 500 psig in the RCS during an operator-initiated cooldown. In any case, EFP-2 must be able to provide EFW at the reduced OTSG pressures that may occur over the EFW mission time.

FPC and FTI were advised by the Terry Turbine division of Dresser-Rand that there have been no changes to the operation of the Terry Turbine since the original purchase specification. The purchase specification for the EFP-2 pump and driver called for a capability to provide EFW to the steam generators for heat removal and cooldown over the NSS steam pressure operating range from the steam generator design pressure of 1050 psig down to about 20 psig. This steam pressure is low enough to cool the RCS to the decay heat removal system operating temperature of 280°F.

Recent testing performed by Ingersoll-Dresser (Reference 18) on a pump similar to the turbine-driven pump at CR-3 confirms that operation of EFP-2 down to a turbine inlet pressure (OTSG pressure assumed to be the same as the turbine inlet pressure) of 20 psig will support a turbine speed of 1080 rpm. Under these conditions, EFP-2 can provide flow rates between 50 and 320 gpm, including the pump recirculation flow. Based on this information, EFP-2 can perform the intended safety function at OTSG pressures as low as 20 psig.

To demonstrate RCS and OTSG response to the break areas requiring the longest EFW mission times, a set of representative RELAP5 calculations were performed. In the cases for the smallest break areas, single-phase natural circulation was maintained and the HPI flow was throttled to maintain subcooling margin. Both continuous and cyclic (closing steam admission valves at an OTSG pressure of 185 psig and opening the valves when either the pressure increased to 1000 psig or the SG level decreased below the RCP spillover elevation) operation of EFP-2 were modeled, demonstrating continued RCS cooling throughout the EFW mission time. For slightly larger breaks, decoupling of the OTSGs and RCS may occur for relatively brief intervals, but break/HPI cooling was sufficient to continue depressurization of the RCS to the matchup time.

4. What are the cooldown restrictions for the EFP-2 single failure? (Section 6.0)

   With a single failure of EFP-2, only EFP-1 will be available to provide the EFW required to ensure adequate core cooling. If the RCS pressure decreases to 500 psig during post-LOCA RCS cooldown and depressurization, the EFW/LPI interlock could trip EFP-1 at a time when EFW flow is still required (i.e., flow from the two available HPI pumps is not sufficient to provide core cooling). Also, with EFP-2 unavailable, if the EFW/LPI interlock at 500 psig can be reached when EFW is still required to provide adequate core cooling, restrictions must be placed on the RCS cooldown to ensure that the RCS pressure is not reduced below 500 psig until HPI flow alone is sufficient to provide core cooling.

   For an operator initiated cooldown, guidance for the operators should be provided, when both HPI pumps are available, to start the RCS cooldown at a reasonable time, maintain the cooldown rate within technical specification limits, and limit the OTSG pressure to 500 psig. This guarantees that the RCS pressure will remain above the 500 psig RCS pressure EFW/LPI interlock setpoint until an assured long-term source of EFW is available for the break sizes that require EFW for mitigation. For break sizes that are large enough to depressurize the RCS to the EFW/LPI interlock setpoint without operator action, EFW will not be required for mitigation because the break area is large enough to pass all of the core steam production, and break/HPI cooling will be sufficient to cool the core.

5. What are the broken HPI line isolation criteria when both HPI pumps are operating and EFW is lost? (Section 7.0)

   As indicated in Item 1, even with both HPI pumps available, EFW will be required for a period of time for breaks of approximately 0.01 ft^2 and less. The current HPI line isolation criterion are evaluated at only one time during the event. If all EFW were subsequently lost (e.g., EFP-1 is secured to allow operation of the 'A'-Train LPI pump for recirculation from the RBES, along with a single failure of EFP-2), the available HPI flow may not be adequate to ensure core cooling, depending on the decay heat level. If all EFW is lost and the break is in an HPI injection line, flow to

the broken HPI line must be isolated to ensure adequate core cooling. Therefore, revised HPI line isolation criterion were developed to ensure that the broken HPI line is isolated when necessary to ensure adequate core cooling.

A spectrum of breaks, single failures, and plant configurations were evaluated to develop a revised isolation criterion. Based on the evaluation, the revised isolation criterion provides sufficient HPI flow to the core for these breaks and single failures to demonstrate that core cooling will be assured. The new criterion also provides additional benefits over the existing criterion. The new criterion is as follows:

At any time in an unthrottled condition, IF the highest-reading HPI line indicates flow > 50 gpm higher than the next highest-reading HPI line, THEN isolate the high flow HPI line.

This new criterion should be applied when all four injection lines are fully-open, with normal makeup isolated. If normal makeup cannot be isolated, or if an HPI injection valve in the broken line is failed open, operator action to isolate the high flow line still needs to be taken (if possible), and additional operator actions to preserve EFW or to split the MUP discharge header may be required. In any case, the isolation criterion will identify the line that needs to be isolated. If subcooling margin has been maintained or restored before the criterion is met, and the HPI flow has been throttled, no isolation action should be taken for SBLOCA mitigation. If subcooling margin is subsequently lost, the operators should maximize the HPI flow before taking any action to isolate a broken line.

6. What inventory of emergency feedwater is required until break/HPI cooling is adequate to cool the core? (Section 8.0)

As discussed in Item 1, EFW is required for a period of time until break/HPI cooling is sufficient to remove the core energy. The amount of emergency feedwater inventory available must be sufficient to support EFW operation over the required time period to ensure adequate core cooling. An assessment was made of the inventory of emergency feedwater required until break/HPI cooling is adequate to cool the core. The required EFW inventory was calculated by integrating the core decay heat over time with an end time based on the calculated mission times for EFW.

Since not all break sizes require emergency feedwater for the same period of time, a series of calculations for different break areas were performed and are presented. These calculations provide an approximate calculation of the required EFW flow for various break sizes. Break sizes above approximately 0.04 ft^2 effectively do not need EFW to mitigate the transient and the inventory required is that necessary to fill both OTSGs (32,093 gallons). For the smallest break area, about 35 hours are required to reach HPI/break/PSV cooling. Approximately 339,000 gallons of EFW

inventory are used over the 35-hour period. This is based on the mission time calculated when one HPI pump is available, in the recirculation mode, with the limiting pinch break HPI flow rates.

The 339,000 gallon inventory required over the 35-hour EFW mission time is available on site. The total capacity of the EFW tank is 184,000 gallons, with a 150,000 gallon minimum capacity, as required by the Technical Specifications. In addition, water in the condensate storage tank (200,000 gallon capacity) and in the condenser hotwells (200,000 gallon combined capacity) is also available.

Additional discussions of how each of the above challenges were resolved is provided in the following sections (Sections 3 through 8).

3.0 EFW Requirements for SBLOCA

Small break LOCA safety analyses assume that uninterrupted EFW flow is provided to the steam generators until OTSG level reaches the required setpoint (error-adjusted 95 percent on the operating range). Thereafter, EFW flow was modeled in the analyses (as needed) to maintain OTSG level. This configuration was relied on until long term cooling (decay heat removal system operation) conditions were established. Mitigation of the transient with acceptable consequences was demonstrated with only one train of ECCS equipment (one HPI pump and one LPI pump) and one EFW pump available.

The SBLOCA spectrum includes breaks where the break flow rate exceeds the capacity of the makeup system at normal RCS pressures, approximately 0.001 ft^2 based on a net makeup flow of 160 gpm into the RCS at 2200 psig, up to and including an area of 0.5 ft^2. The larger break sizes, above approximately 0.04 ft^2, based on conservative core decay heat values, do not require EFW flow to mitigate the transient. In these cases, all of the steam generated in the core passes out the break removing decay heat (Reference 2). The break is large enough that the RCS pressure decreases below the OTSG pressure due to the break flow, and additional OTSG cooling is not required. The RCS pressure will not increase for these breaks, even if EFW is lost at some point during the transient, and ECCS injection flow will still be adequate to remove the core energy.

For smaller break areas, however, EFW flow is required for some period of time. The HPI/core decay heat matchup time, or the EFW mission time, is when the energy absorption by the HPI fluid is sufficient to match core decay heat when boiled at the maximum expected RCS pressure. This time is based on the number of operating HPI pumps, break location, and the RCS pressure response. Using conservative decay heat with one operating HPI pump, break areas of roughly 0.04 ft^2 and smaller require EFW for mitigation. With two HPI pumps, break sizes less than approximately 0.01 ft^2 require EFW. If EFW is lost before the HPI injection flow can absorb core decay heat and the break size is small enough, the RCS can repressurize, possibly as high as the

pressurizer safety valve (PSV) lift setpoint. As the RCS pressure increases, the HPI flow rate will decrease. At these high pressures, depending on the break location, there may be insufficient ECCS injection flow to remove core decay heat.

For both the LOBA and LOBB single failures, the feedwater flow required for successful mitigation must be provided by EFP-2, the turbine-driven emergency feedwater pump. For the LOBA scenario, EFP-1 is unavailable and EFP-2 alone must provide the required EFW. For the LOBB scenario, both EFP-1 and EFP-2 will initially be operating. However, EFP-2 must be crosstied to the 'A'-powered EFW control valves to provide the EFW before EFP-1 is secured. EFP-1 can not operate concurrent with either the CC cooling system operating, or the LPI pump when the RB sprays are operating, since this would overload EDG-1A. EFP-1 could also be tripped due to actuation of the 500 psig EFW/LPI interlock during an operator-initiated cooldown.

With a single failure of EFP-2 (the turbine-driven EFW pump) and loss of offsite power coincident with reactor trip, EFP-1 would be available to supply feedwater flow to the steam generators, and would be automatically loaded on EDG-1A. EFP-1 may need to be tripped manually in order to start the 'A'-Train LPI pump and align the 'A'-HPI pump for recirculation mode (piggy-back operation). In addition, EFP-1 could also be tripped due to actuation of the 500 psig EFW/LPI interlock during an operator initiated cooldown. Therefore, EFW flow to the steam generators could be interrupted. The need to initiate CC cooling will not effect EFP-1 operation with a single failure of EFP-2 because CC chillers can be loaded on the 'B'-Train EDG, as part of EDG-1A load management.

Depending on the break size and location, the RCS may repressurize if all EFW flow is terminated. If the RCS repressurizes, timely identification and isolation of a broken HPI line is necessary to assure adequate core cooling for certain size HPI line pinch breaks that repressurize (i.e., greater than approximately 1600 psig which is the corresponding saturation pressure for the initial hot leg temperature). This requires a revision to the existing HPI line isolation criterion currently implemented at CR-3, Reference 3. A discussion of the bases for a new criterion is provided in Section 7.0 of this report.

Based on the number of operating HPI pumps and their respective RCS flow delivery profiles, a time can be calculated for each SBLOCA sequence at which HPI alone can absorb all of the core decay heat. After this time, cooling from the OTSGs—EFW flow—is not necessary, but availability of EFW is desirable to continue to cool down the RCS. This HPI core decay heat matchup time is a function of the core decay heat (10CFR50 Appendix K versus realistic) and the expected RCS pressure response to the postulated SBLOCA transient.

4.0 Mission Time for Emergency Feedwater

The EFW mission time is a function of the available HPI flow that reaches the core, which in turn is based on the source for the HPI flow, i.e., the BWST or the RBES in recirculation mode, the number of operating HPI pumps, and location of the break. Once the HPI flow rates are known, the energy absorption of the injection fluid can be calculated. Based on the initial core power level, the core decay heat as a function of time after reactor trip is known and can be equated to the HPI absorption energy. This defines the EFW mission time, the time at which HPI flow matches decay heat, without the need for OTSG cooling.

Before the HPI/core decay heat matchup time can be determined, several parameters need to be calculated: Section 4.1 contains a list of assumptions for boundary conditions. Section 4.2 provides the CR-3 specific HPI flow rates for different breaks. In Section 4.3 the BWST drain time is calculated. Finally, the EFW mission time is calculated in Section 4.4, and a summary is provided in Section 4.5.

4.1 Assumptions

In order to determine a conservative time at which HPI alone can match core decay heat, a number of assumptions are made:

1. Temperature of fluid in the BWST is 120°F, Reference 4 (conservatively greater than the 100°F maximum design basis temperature).

2. Temperature of the fluid at the exit of the decay heat cooler(s) is 140°F, Reference 5.

3. EFW flow will be lost at some time and the break is small enough to cause the RCS to instantaneously repressurize to the PSV lift setpoint.

4. A +3 percent lift tolerance on the nominal PSV lift setpoint of 2500 psig will be included, i.e. 2575 psig.

5. HPI pump stop-check valves in their current position

6. A 10 percent equivalent head degradation on the head-flow curves for the HPI pumps.

Calculations are performed using both realistic (90 percent of the 1971 ANS 5.1 decay heat standard for infinite irradiation) and Appendix K (120 percent of the ANS standard) decay heats. Since it is expected that the HPI core decay heat matchup time will be after the BWST is drained, calculations for both one- and two-operating HPI pumps

taking suction from the RBES will be performed. The HPI injection temperature in the recirculation mode through the RBES corresponds to the exit fluid temperature of the decay heat coolers.

Decay Heat Calculation

Appendix K to 10CFR50 specifies the sources of heat that must be considered following a LOCA. The regulation states that, "...it must be assumed that the reactor has been operating continuously at a power level of 1.02 times the licensed power level (to allow for such uncertainties as instrumentation error)." In addition, the radioactive decay of actinides and fission product must be considered. An extra 20 percent--for conservatism--is added to the fission product decay contribution. A curve fit to this data is contained in Reference 6.

For non-Appendix K analyses, or best-estimate calculations, that support operator actions, a conservative, but more realistic core decay heat model is used (1979 ANS 5.1 decay heat standard, Reference 26). A comparison of the 1979 to 100 percent of the 1971 data is presented in Table 1. A ratio of the decay heat fractions is also included. From this data, it is reasonable to conclude that using 90 percent of the 1971 data is appropriate to use for non-Appendix K applications.

Table 1. Comparison of 1979 and 100 percent of the 1971 ANS 5.1 Standard

Time	Decay Heat Fraction		Ratio
(sec)	1979	1.0 * 1971	1979/1971
1	0.062505	0.065822	0.9496
10	0.048125	0.053011	0.9078
50	0.036168	0.041356	0.8745
100	0.031623	0.036734	0.8609
500	0.022319	0.024864	0.8976
1000	0.019001	0.021664	0.8771
1500	0.016951	0.019726	0.8593
2000	0.015484	0.018197	0.8509
2500	0.014361	0.016965	0.8465
3000	0.013469	0.015965	0.8437
4000	0.012155	0.014480	0.8395
5000	0.011259	0.013464	0.8362
7500	0.009988	0.011990	0.8331
10000	0.009338	0.011177	0.8355
20000	0.008052	0.009370	0.8593
30000	0.007258	0.008264	0.8783
40000	0.006709	0.007526	0.8914
50000	0.006328	0.007016	0.9019

4.2 HPI Flow Rates

The HPI flow versus RCS pressure is provided by Florida Power Corporation (FPC) and is taken from a CR-3-specific hydraulics model of the makeup (MU) system (Reference 5) using the PIPF-PC code. A BWST drain time calculation using this data is performed for the LOBA and LOBB single failures in Section 4.3. The cases are similar in that only one HPI pump will be available. Another calculation is performed for the EFP-2 single failure case with two HPI pumps available. In each instance, the calculations are for CLPD breaks and for HPI line pinch breaks. The flow rates for a double-ended break of an HPI line are included, but the operator can recognize these breaks and will take action to isolate the broken line. The double-ended break of an HPI line is bounded by the CLPD breaks because, when the broken line is isolated, all of the HPI pump flow will be available to cool the core (all ECCS flow is to the intact legs).

The HPI flow rates at different RCS pressures for use in the LOBA and LOBB evaluations are taken from Reference 5 and are listed in Table 2. Table 3 contains a comparison of the HPI flow rates for recognizable pinch breaks at 1100, 1800, and 2575 psig. HPI flow rates for the EFP-2 single failure case for two HPI pumps flowing are included in Table 4.

Table 2. HPI Flow Rates Into the RCS and Total Flow for One-Pump Operation
(after 20 minutes with 10% equivalent head degradation)

RCS	CLPD Break		Double Ended HPI Line Break		HPI Line Pinch Break[1]	
Pressure	From BWST	From RBES	From BWST	From RBES	From BWST	From RBES
(psig)	(RCS/total gpm)[2]	(RCS/total gpm)	(RCS/total gpm)	(RCS/total gpm)	(RCS/total gpm)	(RCS/total gpm)
	(Ref. 5, Tables 1 & 8)	(Ref. 5, Table 7)	(Ref. 5, Table 13)	(Ref. 5, Table 13)	(Ref. 5, Table 9)	(Ref. 5, Table 6)
0	399.3 / 549.5					
200	388.0 / 534.0					
400	375.9 / 517.4					
600	363.3 / 500.0					
800	349.5 / 481.0					
1000	334.9 / 460.9					
1100	327.0 / 450.0[3]	338.4 / 465.6	0.0 / 459.4	12.3 / 473.8	370.4 / 447.7	380.3 / 462.4
1200	319.1 / 439.1	330.7 / 455.1			356.7 / 436.9	367.6 / 452.8
1400	301.2 / 414.5	313.7 / 431.7			328.1 / 413.9	340.1 / 431.2
1600	283.0 / 389.4	296.3 / 407.7			296.8 / 387.8	310.7 / 407.3
1800	262.0 / 360.6	277.0 / 381.[illegible]			263.7 / 359.7	278.3 / 380.2
2000	238.2 / 327.9	255.2 / 351.2			228.0 / 328.7	244.7 / 351.6
2200	211.1 / 290.5	230.4 / 317.1			186.9 / 292.1	206.9 / 318.6
2400	178.9 / 246.2	201.7 / 277.5			139.0 / 248.6	163.5 / 279.8
2575	144.3 / 198.6	171.8 / 236.4			88.4 / 201.7	119.4 / 239.6

1. The pinch break is assumed to be in the B1 cold leg. The isolation criterion is based on a 50 gpm difference between the highest and next highest flowing HPI lines at an RCS pressure of 2575 psig. Flow measurement uncertainties are taken from References 7 and 8.
2. The flow into the RCS is equal to the sum of the three lowest-flowing lines. The total flow is the summation of all four injection lines.
3. Data point was interpolated.

Table 3. HPI Flow Rates Into The RCS and Total Flow For Different Pinch Break Areas[1]
(after 20 minutes with 10% equivalent head degradation)

RCS Pressure (psig)	Pinch Recognizable at 1100 psig		Pinch Recognizable at 1800 psig		Pinch Recognizable at 2575 psig	
	From BWST (RCS/total gpm) (Ref. 5, Table 9)	From RBES (RCS/total gpm) (Ref. 5, Table 6)	From BWST (RCS/total gpm) (Ref. 5, Table 9)	From RBES (RCS/total gpm) (Ref. 5, Table 6)	From BWST (RCS/total gpm) (Ref. 5, Table 9)	From RBES (RCS/total gpm) (Ref. 5, Table 6)
0						
200						
400						
600						
800						
1000						
1100	278.8 / 452.3	289.7 / 467.2			370.4 / 447.7	380.3 / 462.4
1200	261.7 / 442.1	273.0 / 457.5			356.7 / 436.9	367.6 / 452.8
1400	225.9 / 419.3	238.6 / 436.4			328.1 / 413.9	340.1 / 431.2
1600	188.4 / 394.1	202.4 / 412.7			296.8 / 387.8	310.7 / 407.3
1800	148.6 / 366.0	164.3 / 386.5	212.4 / 362.4	227.6 / 383.0	263.7 / 359.7	278.3 / 380.2
2000	105.4 / 334.0	123.0 / 356.6	174.0 / 331.5	191.0 / 354.2	228.0 / 328.7	244.7 / 351.6
2200	57.3 / 296.8	78.8 / 323.4	130.1 / 294.9	150.6 / 321.3	186.9 / 292.1	206.9 / 318.6
2400	2.7 / 252.6	28.9 / 284.1	79.4 / 251.2	104.5 / 282.4	139.0 / 248.6	163.5 / 279.8
2575	0.0 / 250.3	0.0 / 264.2	25.6 / 203.4	57.5 / 241.5	88.4 / 201.7	119.4 / 239.6

1. The pinch break is assumed to be in the B1 cold leg. The isolation criterion is based on a 50 gpm difference between the highest and next highest flowing HPI lines at an RCS pressure of 2575 psig. Flow measurement uncertainties are taken from References 7 and 8.

Table 4. HPI Flow Rates Into the RCS and Total Flow for Two-Pump Operation
(after 20 minutes with 10% equivalent head degradation)

RCS Pressure (psig)	CLPD Break From BWST (RCS/total gpm)[2] (Ref. 5, Table 12)	CLPD Break From RBES (RCS/total gpm) (Ref. 5, Table 10)	Double Ended HPI Line Break From BWST (RCS/total gpm) (Ref. 5, Table 13)	Double Ended HPI Line Break From RBES (RCS/total gpm) (Ref. 5, Table 13)	HPI Line Pinch Break[1] From BWST (RCS/total gpm) (Ref. 5, Table 11)	HPI Line Pinch Break[1] From RBES (RCS/total gpm) (Ref. 5, Table 10)
0	759.7 / 1048.0					
200	737.4 / 1017.2					
400	713.4 / 984.1					
600	687.4 / 948.2					
800	661.1 / 912.0					
1000	632.9 / 873.1					
1100	617.8 / 852.3	639.7 / 882.4	404.8 / 892.2	429.0 / 920.1	710.8 / 829.1	727.5 / 860.3
1200	602.2 / 830.7	624.4 / 861.2	370.2 / 873.4	396.4 / 903.2	687.9 / 809.2	705.1 / 841.3
1400	568.4 / 784.1	593.1 / 818.1	298.6 / 832.8	327.7 / 865.1	639.7 / 766.9	659.8 / 802.5
1600	531.6 / 733.4	559.6 / 771.8	221.7 / 786.1	255.0 / 822.3	587.0 / 719.7	610.1 / 759.0
1800	493.0 / 680.1	522.2 / 720.3	138.7 / 733.2	177.2 / 774.0	531.8 / 669.9	556.5 / 711.4
2000	448.5 / 618.7	480.8 / 663.2	0.0 / 625.0	91.7 / 718.1	469.7 / 613.0	498.0 / 658.7
2200	397.7 / 548.6	434.2 / 598.9		0.0 / 655.5	399.7 / 548.0	432.9 / 599.2
2400	337.6 / 465.8	380.5 / 524.9			317.9 / 471.2	358.8 / 530.6
2575	273.1 / 376.8	324.9 / 448.2			230.3 / 387.8	282.5 / 459.0

1. The pinch break is assumed to be in the B1 cold leg. The isolation criterion is based on a 50 gpm difference between the highest and next highest flowing HPI lines at an RCS pressure of 2575 psig. Flow measurement uncertainties are taken from References 7 and 8.
2. The flow into the RCS is equal to the sum of the three lowest-flowing lines. The total flow is the summation of all four injection lines.

The flow rates for the pinch breaks that are presented in Table 3 are based on the break being in the B1 HPI line with the makeup pump 1A (MUP-1A) operating. The B1 break is also taken in Tables 2 And 4. MUP1A and MUP1C are assumed operating in Table 4. Calculations performed in Reference 5 have demonstrated that breaks in this line and this configuration are slightly more limiting than breaks in the other HPI lines.

Relative to EFW mission time, the limiting pinch break is one that just meets the isolation criterion (see Section 7.0, which discusses the isolation criterion) at the highest expected RCS pressure, i.e.. 2575 psig, or the longest time before the operators will take action to isolate the line. This results in the least HPI flow to the core and the minimum HPI energy absorption capability. This maximizes the EFW mission time. For pinch breaks that can be recognized at lower RCS pressures, operator action will be taken to isolate the pinched line, which will result in a larger amount of HPI flow available for core cooling, once the pinched line is isolated, due to the lower RCS pressure. The higher HPI flow rates result in a shorter EFW mission time.

Larger effective pinch areas or full-size HPI line breaks--on the HPI pump side--are more limiting in terms of core cooling if EFW is lost and the broken HPI line is not isolated. These breaks are larger than breaks which are just recognizable at higher RCS pressures and result in less HPI flow to the RCS due to increased flow out the break. If EFW is lost for those break sizes that EFW is required to provide core cooling, the RCS may still repressurize, and the HPI flow to the RCS will be further reduced. If the broken line is not isolated, there will be inadequate HPI flow to cool the core. The repressurization increases the difference in flow rates between the broken and intact HPI lines, and as the RCS repressurizes, the isolation criterion will be met. If the operators isolate the high flow HPI line, core cooling will be assured.

4.3 BWST Drain Time

The time required for the BWST to drain is a function of the initial BWST inventory, the HPI flow, and if actuated, the RB spray flow rate. By conservatively modeling these flow rates and initiation times for the HPI and RB spray, the limiting time required to drain the BWST can be calculated.

Tables 2 through 4 represent HPI flow rates for various break types and include 10 percent equivalent head degradation on the pump head versus flow curve. In the cases where an early BWST drain time is conservative, the nominal HPI pump H-Q curve was used, with no pump degradation, to determine the total pump flow (Reference 5). The flow rates are higher and result in an earlier BWST drain time. For the cases where it is conservative to maximize the BWST drain time, the flow rates in Table 2 are used with no adjustment.

LOBA Single Failure

In the LOBA single failure, EFP-2 is relied on to supply EFW throughout the transient. The mission time for EFP-2 is defined by the time at which HPI flow can match core decay heat without the need for continued EFW. This mission time will be the longest when the HPI flow is at a minimum. Since the HPI flow rates will be smaller when taking suction from the BWST, it is conservative to maximize the time over which HPI is taking suction from the BWST by assuming that RB sprays are not actuated, and using the maximum BWST inventory available.

In order to calculate conservatively the time when the BWST is drained, the maximum technical specification inventory, less the inventory at the level at which RBES swapover must be accomplished (5.5 ft from Reference 9), must be determined. From Reference 9, the maximum BWST inventory defined in the CR-3 Improved Technical Specification is 449,000 gallons which corresponds to a level of 47.72 ft. There is an uncertainty in the level indication of 0.26 ft. The maximum level could be as high as 47.98 ft. After conservatively rounding to 48 ft., the maximum inventory is 451,533 gallons, Reference 10. The volume at the minimum swapover level of 5.5 ft is 51,702 gallons, Reference 10. The net inventory that can be drained from the BWST is :

BWST Volume = (451,533 - 51,702) gallons
= 399,831 gallons

The RCS pressure may not always decrease to the low RCS pressure engineered safeguards actuation system (ESAS) setpoint and therefore an automatic actuation is not expected for all of the break sizes of interest. However, on loss of subcooling margin (SCM), the operators manually initiate HPI. This will occur early in the transient, but it will be conservatively assumed that HPI is not started by the operator until 20 minutes after the break opens.

The HPI flow rate will be taken from Table 2. The RCS pressure response for 0.0025, 0.003, and 0.0035 ft^2 break cases are presented in Figures 5, 6, and 7, Reference 11. The RCS pressure is less than approximately 1800 psig for most of the transient because EFW was available. The minimum total HPI pump flow when injection fluid is supplied from the BWST at 1800 psig is approximately 360 gpm. If the RB sprays are not actuated, the equivalent time into the transient that the BWST will be emptied for the LOBA is:

Drain Time after LOCA = 399,831 gallons / 360 gpm + 20 min
= 1130.6 minutes (or 18.84 hours)

LOBB Single Failure

In the LOBB single failure, EFP-1 may be relied on to supply EFW flow for a portion of the transient. EFP-2, however, must be crosstied to the 'A'-powered EFW control valves to provide the required EFW before EFP-1 is secured. EFP-1 may be secured due to one of three operational limitations: First, the CC cooling system must be loaded on the A-EDG at approximately one hour (Reference 12), and EFP-1, the motor-driven EFW pump, must be secured to prevent exceeding the allowable EDG loads. Second, if the BWST is drained, EFP-1 must be secured to allow the decay heat pump to be started so that recirculation from the RBES can be initiated. (This issue is addressed below.) Third, an operator-initiated cooldown below the 500 psig RCS pressure EFW/LPI interlock would cause EFP-1 to be tripped, and the decay heat pump to be started. In this single failure only one OTSG atmospheric dump valve (ADV) is available. With a restricted relief capacity, the RCS can not be cooled to the 500 psig RCS pressure EFW/LPI interlock setpoint before CC cooling needs to be established.

In any case, EFP-2 must also be able to provide EFW at the reduced OTSG pressures that may occur over the EFW mission time. If EFW is lost due to any of these operational limitations before break/HPI cooling is adequate to remove core decay heat, the RCS can repressurize, possibly resulting in inadequate core cooling.

In the LOBB single failure case, it is conservative to calculate an early drain time for the BWST since this will result in the earliest loss of EFP-1 due to the need to initiate recirculation from the RBES. To calculate how long it takes for the BWST to drain, the minimum technical specification inventory, less the inventory at the RBES swapover alarm, will be used. From Reference 13, the minimum inventory in the CR-3 Improved Technical Specification minus the measurement uncertainty is 413,976 gallons. The RBES swapover alarm occurs at a BWST level of 15 feet, Reference 9. Accounting for the uncertainty in level measurement of one foot, Reference 13, the minimum swapover inventory is 150,440 gallons. The net minimum inventory that can be drained from the BWST is:

$$\text{BWST Minimum Volume} = (413{,}976 - 150{,}440) \text{ gallons} = 263{,}536 \text{ gallons}$$

The largest break size that will require EFW, with one operating HPI pump, is approximately 0.04 ft^2. This is based on the RCS pressure response for the 0.04 ft^2 cold leg break contained in Reference 2, shown as Figure 8. Larger break areas are capable of depressurizing the RCS to below the OTSGs and are large enough to pass all of the steam generated in the core. The RCS will not

repressurize for these larger breaks and the ECCS injection flow will be sufficient to remove the core decay heat. From Figure 8, the minimum expected RCS pressure will be greater than 600 psig. For this case, the flow rate, based on a nominal H-Q curve for one operating pump, should be used. Reference 5 provides a nominal flow rate for one pump and is 518.2 gpm at 600 psig, rounded to 520 gpm.

In addition to the HPI, RB sprays will also increase the rate that the BWST is drained. For the limiting break size with this single failure, the RB spray is expected to actuate. Actuation of the RB sprays increase the rate that the BWST is drained which is conservative for this single failure evaluation. From Reference 14, the maximum RB spray flow rate during the initial drawdown of the BWST is throttled to 1500 gpm per pump. Once the transition to recirculation from the RBES is made, the flow is further throttled to 1200 gpm per pump. Corresponding conservative values, which include allowance for uncertainty, are 1600 and 1300 gpm (Reference 16). Since the time of interest is before the BWST is drained, the RB spray flow rate for suction from the BWST--1600 gpm--will be used.

Revised analyses for the 0.01 and 0.04 ft^2 (the approximate break sizes requiring EFW for 2 HPI pumps available and 1 HPI pump available, respectively) breaks were performed in Reference 15. Included in these analyses are cases with one and two RB fan coolers in operation. From Reference 16, when considering instrument uncertainty and repeatability errors, the RB spray actuation setpoint may be as low as 21.6 psig. Figures 9 through 12 provide the RB pressure responses for the 0.01 and 0.04 ft^2 breaks with one and two RB fan coolers. A summary of the results is presented in Table 5.

Table 5. Error-Adjusted RB Spray Actuation Time
for 0.01 and 0.04 ft^2 CLPD Breaks

CLPD Break	Number of Fan coolers	RB Spray Actuation Time (sec)
0.01 ft^2	1	3600
	2	8000
0.04 ft^2	1	675
	2	800

The break sizes requiring EFW flow for accident mitigation are a function of the number of operating HPI pumps, which depends on the single failure that is being considered. For the LOBB solution set, only one HPI pump and one RB fan cooler will be available, and the break sizes requiring EFW are for areas of approximately 0.04 ft^2 and less. The RB spray actuation time for the LOBB case will be 675 seconds (conservatively 10 minutes. This value was generated based on conservative--Appendix K--core decay heat values. Generally, if realistic decay heat were used, the energy addition rate to the RB would be less and the resulting pressure increase would be slower, resulting in a longer time to actuate the RB sprays. Therefore, the values in Table 5 are conservative for spray actuation times.

For the LOBB single failure, the minimum BWST drain time for the limiting break size of 0.04 ft^2 is:

(520 gpm)*(10 min) + (520 + 1600 gpm)*(X) = 263,536 gal

X = 121.9 minutes

Drain Time = 10 + 121.9 = 131.9 minutes (2.2 hours)

This time is longer than that required to initiate CC cooling, i.e., one hour. Therefore, the draining of the BWST will not set the operability limit for EFP-1.

EFP-2 Single Failure

For the EFP-2 single failure case, an early drain time is conservative because of the operability limitations on EFP-1. These limitations are initiation of recirculation mode of injection or operator initiated cooldown to the 500 psig EFW/LPI interlock. In this case, it must be demonstrated that break/HPI cooling, with two operating HPI pumps, will be adequate at one of these operability limits to remove core decay heat.

Using realistic decay heat values and assuming that an operator may initiate an RCS cooldown, the RCS pressure may be reduced to near 600 psig, Figure 21 (Reference 11). A conservative total pump flow rate at 600 psig will be used. Reference 5 provides a nominal 2-pump HPI flow at 600 psig of 981.5 gpm, rounded to 985 gpm.

The RCS pressure may not always decrease to the low RCS pressure setpoint for the ESAS. Therefore, an automatic actuation is not expected for all of the break sizes. However, on loss of SCM, the operators will manually initiate HPI. Loss of SCM will occur near the time that the reactor trips on low RCS pressure. This is early in the transient, within 2 to 3 minutes after the break is opened.

Therefore, it will conservatively be assumed that HPI starts when the break opens.

The net minimum inventory that can be drained from the BWST is the same as for the LOBB single failure, 263,536 gallons. The break sizes requiring EFW flow for accident mitigation are a function of the number of operating HPI pumps, which depends on the single failure that is being considered. With EFP-2 being the single failure, two HPI pumps and two RB fan coolers are operating, and the break areas of interest will be less than roughly 0.01 ft^2. For the EFP-2 single failure case, the RB spray actuation time will be 8000 seconds (conservatively 130 minutes). This value was generated based on conservative--Appendix K--core decay heat values. Generally, if realistic decay heat were used, the energy addition rate to the RB would be less and the resulting pressure increase would be slower, resulting in a longer time to actuate the RB sprays. Therefore, the values in Table 5 are conservative for spray actuation times.

Using the RB spray actuation setpoint for the 0.01 ft^2 case described above, the minimum BWST drain time for a single failure of EFP-2 is:

$$(985 \text{ gpm})*(130 \text{ min}) + (985 + 3200 \text{ gpm})*(X) = 263{,}536 \text{ gal}$$

$$X = 32.4 \text{ minutes}$$

Drain Time = 130 + 32.4 = 162.4 minutes (2.7 hours)

If one of the operating RB spray pumps is stopped, the drain time can be extended to 182.4 minutes, or roughly three hours. A summary of the drain times for the single failures are presented in Table 6.

Table 6. BWST Drain Times

Single Failure	Break Size (ft^2)	No. of RB Spray Pumps	BWST Empty Time (hours)	Comment
LOBA	<0.005	0	18.84	Long drain time is conservative
LOBB	0.04	1	2.2	Short drain time is conservative
EFP-2	0.01	1	3.0	Short drain time is conservative
EFP-2	0.01	2	2.7	Short drain time is conservative

4.4 Matchup Time Calculation

The matchup time will correspond to the time at which the energy absorption capacity of the HPI fluid that reaches the core through the intact HPI lines matches core decay heat. For these small breaks, if the RCS repressurizes, the hot leg piping will drain before the core will uncover and saturated steam will pass through the PSVs. Core cooling will be through HPI/break/PSV cooling with the core acting as if it were a single pass heat exchanger.

The fluid temperature in the BWST is 120°F (conservatively greater than the 100°F maximum design basis temperature). At atmospheric pressure, the enthalpy is 88 Btu/lbm. The fluid temperature at the exit of the decay heat coolers is conservatively taken as 140°F (Reference 5) and at a pressure of 135 psig, the enthalpy will be 108.3 Btu/lbm. The saturated steam enthalpy at 2575 psig (2590 psia) is 1083.2 Btu/lbm.

From Table 2, the break configuration resulting in the least amount of HPI flow at the PSV lift pressure, 2575 psig, is for the HPI line pinch break. The matchup times for all cases are determined assuming recirculation from the RBES. With only one HPI pump available, the BWST will be drained and recirculation from the RBES will be established before the HPI flow can matchup with the core decay heat. With two HPI pumps available, with suction from the BWST, the BWST will also be drained before the HPI flow will absorb core decay heat. The greater HPI flow rate when in recirculation (as opposed to taking suction from the BWST) is necessary to match decay heat in either scenario. The rated CR-3 core power level is 2544 MWt. A conservative power level of 1.02 times 2568 MWt will be used to determine the HPI/core decay heat matchup time.

<u>One HPI Case from RBES-Pinch Break</u>

The fluid density at 140°F and 135 psig is 61.4 lbm/ft^3. The HPI flow into the RCS at the PSV lift setpoint for the pinch breaks is 119.4 gpm from Table 2. The resulting mass flow rate is:

$$\begin{aligned} W_{hpi} &= (119.4 \text{ gpm})*(61.4 \text{ lbm/ft}^3)/(7.4805 \text{ gal/ft}^3)/(60 \text{ sec/min}) \\ &= 16.33 \text{ lbm/sec} \end{aligned}$$

The energy absorption of the HPI fluid is:

$$\begin{aligned} Q_{hpi} &= W_{hpi} * \Delta h \\ Q_{hpi} &= (16.33 \text{ lbm/sec})*(1083.2-108.3 \text{ Btu/lbm}) \\ &= 15{,}924 \text{ Btu/sec} \end{aligned}$$

The equivalent decay heat fraction, based on an initial core power level of 2568 MWt, is:

$$Q_{hpi}/Q_{core} = (15{,}924\ \text{Btu/sec})/(1.02*2568\ \text{MWt} * 948\ \text{Btu/sec/MWt})$$
$$= 0.00641$$

From Reference 6, this corresponds to 35 hours using 120 percent of the 1971 ANS 5.1 decay heat standard. Using realistic decay heat, 90 percent of the 1971 standard, this time is reduced to about 14.2 hours.

Two HPI Case from RBES-Pinch Break

The fluid density at 140°F and 135 psig is 61.4 lbm/ft^3. The HPI flow at the PSV lift setpoint is 283.8 gpm from Table 4. The resulting mass flow rate is:

$$W_{hpi} = (282.5\ \text{gpm})*(61.4\ \text{lbm/ft}^3)/(7.4805\ \text{gal/ft}^3)/(60\ \text{sec/min})$$
$$= 38.65\ \text{lbm/sec}$$

The energy absorption of the HPI fluid is:

$$Q_{hpi} = W_{hpi} * \Delta h$$
$$Q_{hpi} = (38.65\ \text{lbm/sec})*(1083.2\text{-}108.3\ \text{Btu/lbm})$$
$$= 37{,}676\ \text{Btu/sec}$$

The equivalent decay heat fraction, based on an initial core power level of 2568 MWt, is:

$$Q_{hpi}/Q_{core} = (37{,}676\ \text{Btu/sec})/(1.02*2568\ \text{MWt} * 948\ \text{Btu/sec/MWt})$$
$$= 0.015173$$

From Reference 6, this corresponds to 1.62 hours using 120 percent of the 1971 ANS 5.1 decay heat standard. Using realistic decay heat, 90 percent of the 1971 standard, this time is reduced to less than one hour.

A summary of the HPI/core decay heat matchup times are provided in Table 7.

Table 7. HPI/Core Decay Heat Matchup Time
(EFW Mission Time)

No. of HPI Pumps	Decay Heat Matchup Time[1]		EFW Mission Time[2]	
	Realistic	Appendix K	Realistic	Appendix K
1	14.2 hours	35.0 hours	18.84 hours	35.0 hours
2	< 1.0 hours	1.62 hours	2.7 hours	2.7 hours

1) Decay heat matchup time is based on HPI flow in the recirculation mode.
2) The EFW mission time is whichever is the longer between the decay heat matchup time or BWST drain time.

4.5 Conclusion

The minimum time that the BWST can be drained for these specific scenarios and break sizes has been determined. With one train of ECCS equipment (one HPI and one RB spray pump), the BWST can be drained in 2.2 hours for the limiting break size of 0.04 ft^2. With two complete trains of ECCS equipment (two HPI and two RB spray pumps), the minimum drain time is 2.7 hours for the limiting 0.01 ft^2 break. If one RB spray pump is secured, leaving two HPI pumps and one RB spray pump operating, this time can be extended to roughly three hours.

In the LOBB solution set, the limiting operational time limit to crosstie the EFW trains was believed to be when CC cooling must be established. The earliest that this can be accomplished is estimated to be one hour into the transient. The BWST can be drained in 2.2 hours. Based on this calculation, the loading of the CC chiller is still limiting relative to the time dependency to crosstie the EFW trains as compared to emptying the BWST. In this single failure, only one OTSG ADV is available. With a restricted relief capacity, the RCS can not be cooled to the 500 psig RCS pressure EFW/LPI interlock setpoint before CC cooling needs to be established.

For the EFW mission time, with one HPI pump, the BWST will be drained before the HPI flow into the RCS is sufficient to remove the core decay heat via break/HPI/PSV cooling. Accounting for a higher enthalpy of the injection fluid during the recirculation mode, or piggy-back operation, the time for the HPI to matchup with the core decay heat is 35 hours using conservative decay heat levels. With realistic core decay heat, this time is significantly reduced to 14.2 hours, but the BWST will not be emptied. Therefore, under realistic decay heat

assumptions, the EFW mission time is when the BWST is drained, or 18.84 hours post-LOCA.

For the EFW mission time during the EFP-2 single failure with two operating HPI pumps available, the energy absorption of the HPI fluid can match the core decay heat earlier in the transient than with one HPI pump. However, due to the lower HPI flow rate when supplied from the BWST, the energy absorption of HPI fluid is still not sufficient to match the core decay energy before the BWST is emptied using conservative, Appendix K assumptions. Once recirculation from the RBES is established, the increased HPI flow rate is sufficient to match core decay heat. Therefore, for the break sizes that require EFW for mitigation with a single failure of EFP-2, the EFW mission time will be when the BWST is emptied, or 2.7 hours.

5.0 EFP-2 Operation

In developing the solution sets (Figures 2 through 4) and during the reviews of the CR-3 design bases, potential operational limitations were identified for the turbine-driven EFW pump, EFP-2. Specifically, the CR-3 Improved Technical Specifications and bases for EFP-2 define the operability of the pump in terms of a low OTSG pressure (< 200 psig) and the capability to provide accident analysis flow rates. The required accident analysis flow rate is defined early in the transient, i.e., within the first couple of minutes when the OTSG pressure is near the main steam safety valves (MSSVs) lift pressure. By the time OTSG pressure has been reduced below 200 psig for SBLOCAs that require EFW for mitigation, core decay heat has decreased substantially, and the high EFW flow rates are no longer necessary.

The LOBA and LOBB single failures require long term steam generator cooling using EFP-2. The potential exists for the steam pressure to decrease below 200 psig. This can be a result of operator actions to cooldown the RCS using the OTSG ADVs, or due to the operation of EFP-2 (which draws steam from the OTSGs) with the RCS and OTSGs coupled or decoupled. Without good coupling--primary to secondary heat transfer--supplying steam to provide motive power to EFP-2 may cause the OTSG pressure to decrease without a commensurate reduction in RCS pressure. In either case, if the operability of EFP-2 is challenged before it can be demonstrated that HPI/break/PSV cooling is sufficient to remove the core decay heat, inadequate core cooling may result.

In order to address these possibilities, two tasks were undertaken: (1) information from the EFP-2 pump and turbine vendors was solicited and (2) "proof of principle" analyses were performed using RELAP5 to ascertain the long-term operating conditions for EFP-2, given how the RCS and OTSGs interact for these very small SBLOCAs. The following sections describe, for the

single failures, the information that was received from the turbine and pump vendors and the results of the RELAP5 calculations.

LOBA Single Failure

For the LOBA scenario, EFP-1 is unavailable and EFP-2 alone must provide the required EFW. The issue for this single failure is the operability of EFP-2 at low steam pressures. The low OTSG pressure can be a result of an operator-initiated cooldown, decoupling of the RCS and OTSGs while still providing motive steam to drive EFP-2, or simply providing motive steam to drive EFP-2.

LOBB Single Failure

For the LOBB scenario, both EFP-1 and EFP-2 will initially be operating. However, EFP-2 must be crosstied to the 'A'-powered EFW control valves to provide the EFW before EFP-1 is secured. EFP-1 can not operate concurrent with either the CC cooling system operating, or the LPI pump when the RB sprays are operating, since this would overload EDG-1A. EFP-1 could also be tripped due to actuation of the 500 psig EFW/LPI interlock during an operator-initiated cooldown.

The issues related to the LOBB single failure are determining to how long it takes to crosstie EFP-2 such that flow can be controlled through the 'A'-powered EFW control valves, and whether flow from EFP-1 can be preserved until that time. The specific issues are: (1) loss of EFP-1 to initiate CC cooling or draining the BWST and initiation of recirculation from the RBES, (2) operator initiated cooldown of the RCS below the 500 psig EFW/LPI interlock setpoint, and (3) EFP-2 operability at low steam pressures.

Section 4.0 addressed the loss of EFP-1 due to emptying the BWST, and concluded that having to start the control complex cooling system at one hour will occur before the BWST is drained. EFP-2 operability at low steam pressures and the RCS response to these SBLOCAs are discussed below.

When the EFW/LPI interlock is actuated, EFP-1 is tripped. The interlock is automatically actuated when the RCS pressure has decreased to 500 psig, concurrent with a LOOP. The interlock may be actuated because the break is large enough to pass all of the steam generated in the core or by an operator initiated cooldown of the RCS using the ADVs. For break sizes that are large enough to depressurize the RCS to the interlock setpoint, EFW will not be required for mitigation because the break area is large enough to pass all of the core steam production. For the smaller break sizes, the main issue is operation and operability of EFP-2 over the mission time.

In this single failure only one OTSG ADV is available. With a restricted relief capacity, the RCS can not be cooled to the 500 psig RCS pressure EFW/LPI interlock setpoint before CC cooling needs to be established.

EFP-2 Operability at Low Steam Pressures

FPC and FTI contacted the EFP-2 pump vendor (Ingersoll-Rand now Ingersoll-Dresser) and turbine vendor (Terry Turbine-now Dresser Rand) to ensure a thorough review.

FPC and FTI were advised by the Terry Turbine division of Dresser-Rand that there have been no changes to the operation of the Terry Turbine since the original purchase specification (Reference 17). The purchase specification for the EFP-2 pump and driver called for a capability to provide EFW to the steam generators for heat removal and cooldown over the NSS steam pressure operating range from the steam generator design pressure of 1050 psig down to about 20 psig. This steam pressure is low enough to cool the RCS to the decay heat removal system cut-in temperature of 280 °F.

Recent testing performed by Ingersoll-Dresser on a pump similar to the turbine-driven pump at CR-3 confirms operation of EFP-2 down to a turbine inlet pressure (assuming that the OTSG pressure and turbine inlet pressure are equal) of 20 psig will support a turbine speed of 1080 rpm. Under these conditions, EFP-2 can provide total pump flow rates (injection plus recirculation) between 50 and 320 gpm, Reference 18.

FTI also performed an evaluation of the emergency feed water cavitating venturis that have been installed in the EFW lines to assess the effect on EFP-2 at low pressure (speeds). The characteristic response of the cavitating venturis is that the flow will be reduced with reduced inlet pressure, Figure 13. The flow limiting characteristics of the cavitating venturis, combined with the EFP-2 head capacity curve at 1080 rpm (Figure 14, Reference 19), less recirculation flow, yields the approximate maximum flow rate when operating at 1080 rpm. With the EFW valves to the steam generators full-open, the total EFP-2 pump flow is about 280 gpm

When the required OTSG level setpoint has been reached and EFIC has reduced the flow to both OTSGs to zero, the pump recirculation flow will be about 50 gpm at a pump speed of 1080 rpm. This satisfies the Ingersoll-Dresser recommendation and permits operation of EFP-2 at 1080 rpm with a steam generator supply pressure of about 20 psig. This also allows the RCS to be cooled to 280°F, the lower end of Mode 3.

Figure 15 provides the EFW flow necessary to match core decay heat. This figure includes realistic and 10CFR50 Appendix K decay heat contributions. As can be seen in this figure, 230 gpm (280 gpm pump flow less 50 gpm for pump recirculation) is sufficient to remove core decay heat after about 70 minutes after reactor trip based on realistic decay heat. Using Appendix K decay heat, 230 gpm will absorb core decay heat at 3 hours. It is also assumed that the OTSGs have been filled to the loss of subcooling margin setpoint (LSCM) setpoint.

With the LOBA and LOBB single failures, only one HPI pump is operating. The mission time for EFW is about 35 hours, and by the time the OTSG pressure can be reduced sufficiently for EFP-2 to be challenged by low steam pressures, 200 gpm of EFW flow will be more than sufficient to remove core decay heat. Based on these discussions, EFP-2 can perform the intended safety function at OTSG pressures as low as 20 psig.

SBLOCA RELAP5 Analyses

Core cooling during a LOCA is assured by various safety-related systems and power supplies that supply ECCS inventory makeup to offset the break discharge and provide secondary side heat removal as needed to depressurize the RCS and augment the ECCS inflow rates. The key safety systems required to ensure successful LOCA mitigation are the emergency diesel generators, HPI pumps and the valves used to align the required flow paths, core flood tanks, LPI pumps and the valves used for flow path alignment, service water flow for the control complex coolers, building spray pumps and building coolers for containment pressure control, and emergency feedwater pumps and flow control necessary for steam generator heat removal.

These systems are used to mitigate any postulated break size from holes that exceed the normal makeup system capacity up a double-ended guillotine rupture of the hot leg pipe. The successful mitigation of these events involves continuous removal of the core heat by liquid natural circulation, boiling, or forced convection cooling. The decay heat energy that is removed from the core must be either rejected to the steam generators or discharged out of the break. The steam generators are the heat sink for normal operation and are the dominant heat sink for the smaller break sizes.

If the break is sufficiently small, the HPI system replenishes the liquid lost out of the break and the RCS remains in single phase natural circulation. The EFW flow provides a continuous heat removal path when the primary system remains in natural circulation. As the break size increases or if the HPI RCS inflows are severely degraded, the HPI system will not be able to match the leak discharge and liquid natural circulation will be interrupted. The loss of circulation disrupts the continuous steam generator heat removal mechanism. Without circulation, the core will begin to boil or boil more violently.

If the break is unable to accommodate all of the core energy and the steam generator heat removal is not reestablished by either hot leg spillover or high-elevation boiler-condenser mode of cooling, the RCS will repressurize. If the break is very small, the break energy discharge will not match the core decay heat until the RCS reaches the pressurizer safety valve lift pressure. Lifting of the pressurizer code safety valves increases the energy discharge from the system. The energy removal capability exceeds the core decay heat rate, and the safety valves reseat after the RCS depressurizes. Cyclic pressurizer safety valve flow will continue until the core decay heat is removed without the safety valve energy discharge. The cyclic behavior will stop when the core decay heat declines with time or the reestablishment of steam generator heat removal. The reestablishment of steam generator heat removal may be in the form of a high-elevation or pool boiler-condenser process. With EFW available, the OTSG heat removal ensures that the RCS can be depressurized to improve the HPI flow delivery to the core, and ensure that adequate core cooling is provided.

Mitigation of the SBLOCA transient requires at least one HPI pump, one LPI pump, and one EFW pump with a sufficient supply of EFW liquid. There is a real limit to the available EFW liquid source at the site and there may also be a realistic limit to the conditions and specific time interval in which the turbine-driven EFW pump can be shown to be operable. These inventory requirements (Section 8.0) and mission times (Section 4.0) are functions of break size, break location, break type, and the HPI flow profile. The longest operability interval is determined by an HPI line pinch break (pinched on both the RCS and HPI pump side). The most restrictive HPI pump-side pinch is one that is "unrecognizable" at a pressurizer safety valve lift pressure of 2575 psig. The pinched line is unrecognizable when the pinched line is the highest flow line and it is less than 50 gpm (indicated) above the next highest flowing HPI line (Section 7.0).

After the BWST empties, the LPI pump is aligned to the sump and the LPI pump provides the suction for the HPI pump. At 2575 psig with only one HPI pump operable, the lowest inflow from the sum of the three lower flow lines is 119.4 gpm, Table 2. From Reference 5, the actual individual HPI line flows at this pressure are: the pinched line is 120.2 gpm, second highest line is 41.7 gpm, and 41.7 and 36.0 gpm for the lower two lines. The instrument uncertainty is subtracted from the pinched line and added to the next highest flowing line to give a net difference of 50 gpm. This value corresponds to the revised HPI isolation criterion discussed in Section 7.0. If 119.4 gpm reaches the core and the RCS has repressurized to 2575 psig, this flow (when totally boiled-off from an inlet temperature of 140°F) will match the core decay heat at 35 hours using 1.2 times 1971 ANS 5.1 1971 fission product decay heat with B&W heavy actinides.

The RCS evolution includes loss of EFW, with a loss of OTSG heat removal after the level boils down to below the natural circulation setpoint or below the reactor coolant pump (RCP) spillover elevation. The loss of circulation and reduced HPI flow will result in core boiling that passes steam through the reactor vessel internal vent valves and to the break.

At 2575 psig, an 0.00247 ft^2 RCS-side HPI line break will discharge a saturated steam mass flow rate that is equivalent to the HPI flow from the sump (The HPI flow matches the core decay heat at 35 hours). Breaks smaller than these sizes will cycle the pressurizer safety valves until the decay heat matches the break energy discharge. After 35 hours, adequate core cooling is assured because the amount of HPI flow that is available for core cooling can not be totally boiled-off by the core decay heat. The long-term cooling configuration without EFW is HPI through the pressurizer safety valves until the decay heat decreases and can be matched by the break energy discharge, or offsite power is returned and EFW restored.

To demonstrate the RCS and OTSG response to these break areas, a set of representative RELAP5 calculations were performed (Reference 11). The first case to be described corresponds to a 0.002 ft^2 break in the CLPD region. In this case, single-phase natural circulation was maintained and the HPI flow was throttled to maintain a 50°F subcooling margin. EFP-2 was allowed to continually operate and no operator induced cooldown was modeled.

The steam flow required to drive EFP-2 as a function of the steam pressure was modeled in the RELAP5 calculations. As can be seen in the pressure response, Figure 16, the steam drawn from the OTSGs to supply motive force to EFP-2, was more than sufficient to supply enough EFW to remove the core energy plus depressurize the OTSGs. The typical steam flow to drive the turbine-driven pump is approximately 10 lbm/sec at steam pressures above 280 psig. Since the RCS and OTSG remained coupled, this steam flow can be translated into an equivalent break area. At 1000 psia, a steam flow rate of 10 lbm/sec is roughly equivalent to a 0.005 ft^2 break. This area coupled with a small break in the RCS piping is more than adequate to depressurize both the OTSGs and the RCS several hours into the transient.

At the end of the calculation, at 40 hours, the RCS is in stable single-phase natural circulation, Figure 17, and is 50°F subcooled, Figures 18 and 19. The OTSG pressure is about 235 psig and is decreasing at approximately 4.0 psi per hour. These results are based on an adiabatic steam piping boundary, SG isolation, and Appendix K core decay heat. From these results, as long as the RCS and OTSGs remain coupled, there will be sufficient steam production to drive EFP-2 and the operators will not have to "manage" EFP-2 by cycling the steam admission valves.

In the second case, with a break size of 0.0014 ft^2, the steam admission valves to EFP-2 were closed and EFW was stopped at 185 psig OTSG pressure. Since the RCS was in single-phase natural circulation, both the RCS and OTSG pressure began to increase. Once the SG pressure reached the MSSV lift setpoint, the steam admission valves to EFP-2 were opened, and EFW was started. The RCS and OTSG pressure response is included on Figure 20.

The mission time for EFW is 35 hours for these single failures and HPI flow rates. In both cases described above, the RCS was adequately cooled and the OTSG pressure did not decrease below approximately 185 psig. Smaller break sizes will remain in natural circulation, and the expected results would be similar. For larger break sizes, natural circulation may be interrupted. Therefore, a third case was investigated. This case, was for a slightly larger break size. 0.0035 ft^2.

Significantly larger break sizes will evolve to EFW spray or pool boiler-condenser cooling, and OTSG pressure will remain near the lift pressure for the MSSVs. Smaller break sizes can remain in natural circulation as described above. In this case, the OTSG pressure, Figure 21, does decrease to approximately 135 psig in approximately 13 hours because the steam flow required to drive EFP-2 exceeds the boil-off rate. The RCS and OTSG are loosely coupled, and the RCS remains saturated. The break/HPI cooling, with some OTSG heat removal, is sufficient to cool the RCS, i.e., the HPI and break flow rates are equal. If EFW is lost due to insufficient steam pressure to drive EFP-2, the RCS will repressurize. However, in this case, the break is large enough at this point in the transient to limit the amount that the RCS can repressurize. Based on the HPI flow rates in recirculation mode (Table 2), a break area of 0.0035 ft^2, and at this time into the event, the maximum pressure that the RCS can reach is approximately 2400 psig. The mission time for EFW at this pressure is significantly less than if the RCS pressure is at the PSV lift setpoint due to the higher HPI flow rate at 2400 psig. In fact, the HPI flow is high enough in this case to refill the RCS and re-establish two-phase natural circulation.

The results presented above should be considered representative of a range of break sizes and the results considered to be general representations of the system interactions. It can be concluded from the above discussions, that as long as the RCS and OTSGs remain coupled, little or no operator action to "manage" EFP-2 is necessary. However, if the RCS and OTSGs become decoupled or if the OTSG pressure decreases below 200 psig due to steam leakage, operator action may be necessary to preserve operability of EFP-2.

6.0 EFP-2 Failure: Cooldown Restrictions

The EFP-2 single failure solution set relies on EFW via EFP-1 for a relatively short period of time during a SBLOCA to remove energy from the RCS. The mission time is relatively short (less than 2 hours or when the BWST is emptied, whichever is more limiting) compared with the other single failures because both EDGs are available, and two HPI pumps are operating. Potential challenges to maintaining EFP-1 available are having to stop EFP-1 to load a decay heat pump on EDG-1A to maintain HPI injection capability during the recirculation cooling from the RBES, or an operator-initiated cooldown of the RCS below the 500 psig EFW/LPI interlock setpoint. This section will determine if, using the available steaming capacity, the RCS can be cooled below the 500 psig EFW/LPI interlock setpoint.

In design basis analyses only safety-grade equipment is typically credited. For OTSG pressure control on the B&W-designed plants, the only safety-grade equipment available are the MSSVs. The MSSVs are for overpressure protection of the OTSGs and not for cooling down the RCS. Therefore, in the design base accident analyses, no operator-initiated cooldown of the RCS below the MSSVs lift pressure is assumed.

Plant emergency operating procedures (EOPs), however, direct the operator to cool the plant down within the specific limits identified in the pressure temperature limits report (Reference 20). With a LOOP, the ADVs are used. For the EFP-2 single failure, it needs to be determined if the RCS can be cooled below the 500 psig EFW/LPI interlock setpoint before EFW is no longer needed.

6.1 RCS Cooldown

To assess the effect of cooldown of the RCS toward the 500 psig EFW/LPI interlock setpoint, a RELAP5 analysis was performed. The RELAP5 model of the CR-3 plant was the same model used to assess the reduced HPI flow delivery rates prior to startup of Cycle 11 (Reference 21). This model was modified to use best estimate core decay heat (90 percent of the 1971 ANS 5.1 decay heat standard) and both ADVs. The ADVs were limited to approximately 80 percent of valve wide open capacity (301,246 lbm/hr at 540°F and 948 psig), consistent with the actual physical limitations placed on the valves, which is reflected in the main steam enhanced design bases document (Reference 22). Using a more realistic decay heat is conservative for this analysis since, for a given break size, the RCS pressure will be lower--because of the lower decay heat--and the effectiveness of EFW in cooling the RCS and reducing the system temperature will increase.

A case simulating a failure of EFP-2 and LOOP, Reference 11, was run for a 0.0025 ft^2 break in the CLPD region. In this case, two HPI pumps are available and operating. At 20 minutes (1200 seconds) into the transient, both ADVs were opened, and a forced cooldown of the RCS was imposed by decreasing the OTSG pressure at a rate consistent with a 100 F/hr cooldown. The 20-minute action time was judged to be a representative time based on typical operator actions required to mitigate a SBLOCA.

In the RELAP5 analysis, the required operator actions to make the transition from forced to natural circulation and to maximize HPI injection to the RCS was completed by 20 minutes into the transient, and the RCS cooldown was initiated. Single-phase natural circulation was maintained, and at 66.7 minutes (4000 seconds), the RCS was approximately 50°F subcooled, the minimum subcooling margin required by the EOPs. Based on the OTSG pressure response and throttling the HPI flow, it was projected that the 100 F/hr cooldown rate was fast enough that the RCS pressure would decrease below the 500 psig EFW/LPI interlock setpoint. Therefore, the HPI flow was throttled to maintain a 50°F subcooling margin, consistent with the EOPs, and the cooldown rate was reduced to 50 F/hr. These cooldown rates are conservatively in excess of those expected during a post-LOCA cooldown with adequate subcooling. At the time HPI was throttled, the RCS pressure was at approximately 1085 psig and the hot leg temperature was about 508°F (50°F subcooled).

At 2.5 hours, EFW was stopped, simulating a switch to recirculation from the RBES. The OTSG levels were high enough to sustain natural circulation, and the cooldown rate was not affected. At the end of the analysis, 2.8 hours, the RCS was approximately 585 psig and 437°F (50°F subcooled). The OTSG pressure was roughly 285 psig. Figure 22 provides the RCS and OTSG pressure and response. Figure 23 contains the RCS temperature response for this transient.

For larger break sizes, natural circulation may be interrupted and the RCS and OTSGs will decouple. The OTSG pressure will continue to decrease with essentially no heat removal from the RCS. These larger break sizes, which require EFW, will evolve to an equilibrium pressure in the RCS in which the core steam production and break flow are equal, and the 500 psig RCS pressure EFW/LPI interlock will not be actuated. For the break sizes that are large enough to depressurize the RCS to the interlock setpoint, EFW will not be required for mitigation because the break area is large enough to pass all of the core steam production.

6.2 Conclusions

In the cases analyzed, the RCS did not reach the 500 psig EFW/LPI interlock setpoint before HPI alone could remove core decay heat. However, had a uniform 100 F/hr cooldown been assumed with throttling HPI flow to maintain subcooling margin, the setpoint would have been reached. The RCS pressure is a function of HPI flow, break size, core decay heat, and OTSG steam venting capability. Therefore, specifying a fixed cooldown limit is difficult. A more logical guidance, when both HPI pumps are available, would be to start the RCS cooldown at a reasonable time, maintain the cooldown rate within technical specification limits, and limit the OTSG pressure to a value above the 500 psig RCS pressure EFW/LPI interlock setpoint until an assured long-term source of EFW is available.

Initiating the RCS cooldown should follow the emergency operating procedures for mitigating a SBLOCA with a LOOP and loss of SCM. The limit on OTSG pressure will depend on the hot leg/core exit subcooling margin and the core temperature difference. The case analyzed remained in single-phase natural circulation with approximately a 15°F temperature change across the core. However, a slightly larger break size or higher core decay levels could evolve into two-phase natural circulation and a different core ΔT. RCS subcooling and the temperature difference across the core are dependent on many parameters, and it is difficult to specify a bounding cooldown limit. Therefore, to assure in all cases that the interlock is not actuated by an operator-initiated cooldown, it would be conservative to limit the OTSG pressure for a period of time until break/HPI cooling is adequate to remove core decay heat or an assured long-term source of EFW is available. For break sizes that are large enough to depressurize the RCS to the interlock setpoint, EFW will not be required for mitigation because the break area is large enough to pass all of the core steam production.

7.0 Isolation of HPI Line Breaks

The current HPI line isolation criterion contained in the CR-3 EOPs specifies that using the low-range flow instruments, if only one injection line indicates flow greater than 75 gpm above the lowest line, the highest line should be isolated. This is a one-time check that is performed at approximately 20 minutes after loss of SCM or actuation of the ESAS. The decision to isolate a broken line should also be made after all of the HPI lines are fully-opened, with normal makeup isolated. If normal makeup cannot be isolated, or if an HPI injection valve in the broken line is failed open, additional actions may be required. In any case, the isolation criterion will identify the line that needs to be isolated, and as long as

OTSG cooling is preserved, sufficient HPI flow will reach the RCS assuring adequate core cooling.

This current criterion was based on one operating HPI pump and EFW always available. The criterion accounted for the limiting HPI pump in terms of head and flow, the CR-3 makeup system hydraulics model and flow uncertainties, and the limiting break location. With two HPI pumps operating and EFW always available, the criterion was still valid.

With two HPI pumps operating and EFP-1 available for a period of time, the RCS is in a safe and coolable state, but the existing isolation criterion may not be met when the operators reach the point in the EOPs that require isolation of a broken line. If EFW is lost later in the transient, the break may not be large enough to pass all of the steam produced by boiling in the core. The RCS pressure can increase, and the available HPI flow for core cooling will be reduced. If the broken line is not isolated, there will not be adequate HPI to cool the core. Since the existing HPI line isolation criterion is <u>not</u> periodically reassessed throughout the duration of the transient, a new criterion must be established when two HPI pumps are operating and EFW is not available. This new criterion should be applicable for one and two operating HPI pumps and with or without EFW. The criterion should also be functional for other applicable single failures, e.g. an injection line failing to open or a makeup valve failing to close, and should not invalidate previous SBLOCA analyses.

In the process of developing a revised HPI isolation criterion for CR-3, a concern was identified that was associated with the existing HPI isolation criterion. With the worst-case stack up of instrumentation uncertainty, particularly at low flow rates in the individual HPI lines, the potential exists for the operator to take a non-conservative action (i.e., failure to isolate) for a full HPI line break, given a single failure of a vital DC channel or EDG. With new instrument uncertainties, the current 75 gpm isolation criterion was shown to be acceptable, provided that EFW was available.

An alternative HPI line isolation criterion was developed to more effectively discriminate a broken HPI line, with or without EFW. New instrumentation uncertainties were also developed (References 7 and 8). These new uncertainties were used to establish the revised HPI isolation criterion.

7.1 Breaks Considered for the Isolation Criterion

As a part of this effort to develop alternative HPI line isolation criterion, a broad range of breaks and single failures were considered. The major break classifications are: the classical SBLOCA, i.e. breaks in the CLPD piping, a core

flood line break, a full HPI line break, and the HPI line pinch break. Each of these breaks are discussed below.

- Cold Leg Pump Discharge Break - For the worst-case break in the bottom of the RCS cold leg piping, the smallest amount of HPI flow will be delivered to the core when the broken cold leg is supplied from the HPI line with the lowest hydraulic resistance (in the first 20 minutes of the accident). At CR-3, this corresponds to the A1 line which is the HPI nozzle fed with normal makeup. The HPI flow delivered to the core is conservatively calculated as the summation of flow through the three low-flow HPI lines. The minimum HPI flows for a CLPD break are given in Tables 2 and 4 for one- and two-HPI pumps, respectively. The flow rates were established using a PIPF-PC model of the CR-3 HPI system.

- Core Flood Line Break - A worst-case break of a core flood line at CR-3 will result in loss of ECCS injection capability from one core flood tank and one train of LPI. If a single failure disables the other train of LPI (e.g., loss of an EDG), then the core can only be cooled in three ways: (1) one core flood tank will passively inject through the intact core flood line, (2) the operating HPI pump will inject through two HPI nozzles, initially, and then through four lines based on the timing of operator actions, with suction from the BWST, and (3) the operating HPI pump(s) will inject through all four HPI nozzles with suction from the RB sump via the operating LPI pump(s). The HPI flow delivered to the core for this accident is calculated as the summation of flow through two HPI lines initially, and through all four lines thereafter consistent with the timing of operator actions.

- Full HPI Line Break - For full HPI line breaks that meet the HPI line isolation criterion early during the transient (i.e., within the first 20 minutes), a break in the A1 line is assumed, which is the lowest hydraulic resistance HPI line prior to isolating normal makeup. For breaks that are isolated later during accidents (e.g., single failure of EFP-2 with RCS repressurization and when HPI suction is switched to the RB sump), a break in the B1 line is assumed, which is the lowest hydraulic resistance HPI line after normal makeup is isolated. The HPI flow delivered to the core is conservatively calculated as the summation of flow through the three intact HPI lines.

- HPI Line Pinch Break - An HPI line pinch break is a break for which the cross-sectional area of the broken HPI line becomes constricted due to jet impingement and/or pipe whip loads. For breaks that meet the HPI line isolation criterion early during the transient (i.e., within the first 20 minutes), a spectrum of HPI line pinch breaks in the A1 line are assessed since the A1 line represents the lowest hydraulic resistance HPI line prior to isolating normal makeup. These HPI line pinch breaks are assessed at an RCS pressure of 1100 psig when judging the adequacy of operator actions. At 20

minutes into the event, EFW will be available and for this break size, the RCS pressure will approach the OTSG pressure. At higher RCS pressures, the flow difference between the broken and intact lines increases, and it will be easier for the operators to identify the affected line.

For breaks that are isolated later during the transient (e.g., single failure of EFP-2 with RCS repressurization), a spectrum of HPI line pinch breaks in the B1 line are assessed since the B1 line represents the lowest hydraulic resistance HPI line after normal makeup is isolated. This class of HPI line pinch break is limiting in terms of EFW mission time and is evaluated at an RCS pressure of 2575 psig, which corresponds to the error-adjusted lift setpoint of the pressurizer code safety valves because the HPI flow to the RCS is minimized.

The HPI flow delivered to the core for all types of HPI line pinch breaks is conservatively calculated as the summation of flow through the three intact HPI lines. The calculated HPI flows for a small-effective area HPI line pinch break are given in Table 2 for an identifiable pinch at 2575 psig. HPI flow rates for other pinch-pressures are given in Table 3. These flow rates are based on a revised isolation criterion of 50 gpm between the highest and next-highest flowing HPI lines and the new flow measurement uncertainties.

7.2 Single Failures

A variety of single failures were considered to assure the adequacy of the HPI isolation criterion. Each of the pertinent single failures, the consequences of the single failures relative to HPI system operability, and the relevant breaks listed above that are considered for the single failures are discussed as follows:

- Loss of One Vital DC Channel - A single failure of one vital DC channel, up to and including a full battery (LOBA or LOBB), is postulated. The failure of one vital DC channel will result in the loss of one EDG. The loss of one EDG will result in the loss of one HPI pump and the failure of two HPI injection valves in the closed position until the operator can cross-connect power to the operating EDG and open the valves. This single failure is undesirable because (1) it results in only one HPI pump being available for core cooling, and (2) it results in reduced HPI flow capability until a manual operator action is taken to open two of the HPI valves. The single failure of one vital DC channel is assessed for all of the breaks.

- Failure of One HPI Valve in the Open Position - A single failure of one HPI valve in the open position is postulated in the B1 line since this is the HPI line with the lowest hydraulic resistance after normal makeup is isolated. Failure of an HPI valve in the open position is undesirable for those breaks that

require isolation (i.e., full HPI line break and HPI line pinch break). With respect to HPI flow delivery, the full HPI line break is always worse than the HPI line pinch break. Failure of the valve in the open position is undesirable because (1) it reduces the HPI flow delivery to the core, and (2) it prevents termination of the HPI flow diversion. As such, a single failure of MUV-25 (i.e., HPI valve in the B1 line) in the open position is assessed for a full HPI line break in the B1 line (Section 7.4).

- Failure of One HPI Valve in the Closed Position - A single failure of one HPI valve in the closed position is postulated in both the A1 or in the B1 lines since these are the lowest hydraulic resistance HPI lines before and after normal makeup is isolated. Failure of an HPI valve in the closed position is undesirable because (1) it reduces the HPI flow delivery to the core, and (2) it has the potential to mislead the operators when the HPI line isolation criterion is applied. With this in mind, a single failure of MUV-24 or MUV-25 is assessed for all of the breaks listed above.

- Inability to Isolate Normal Makeup - A single failure that results in normal makeup not being isolated is postulated. Failure to isolate normal makeup is undesirable because (1) it reduces the HPI flow delivery to the core for breaks associated with the A1 line, (2) it has the potential to mislead the operators when the HPI line isolation criterion is applied, and (3) closure of the HPI valve (MUV-24) in the A1 line will not terminate the HPI flow diversion. With this in mind, a single failure that results in normal makeup not being isolated was assessed and is discussed further in Section 7.4.

- Inability to Isolate Seal Injection - A single failure that results in RC pump seal injection not being isolated is postulated. Failure to isolate RC pump seal injection is undesirable primarily because it reduces the HPI flow delivery to the core. HPI flow diversion through the RC pump seal injection pathway is, however, limited by the hydraulic line losses in the RC pump seal injection pathway. Previous hydraulic analyses have shown that even without isolating the RC pump seal injection pathway, the HPI flow delivery with two HPI pumps operating is adequate to cool the core. As such, a single failure that results in the RC pump seal injection not being isolated does not require further evaluation.

- Failure of EFP-2 - In the current CR-3 HPI design, an interdependency exists between the HPI and EFW systems through the EFW/LPI interlock. Historically, safety analyses have assumed that at least one train of EFW is available during the mitigation of all SBLOCAs. For CR-3, all EFW may be lost if a single failure of the turbine-driven EFW pump is postulated, and either the RCS pressure decreases below the EFW/LPI interlock actuation setpoint, or the BWST becomes depleted such that the HPI pump must be aligned to the RB sump via the LPI pump. If the RCS pressure decreases

below the EFW/LPI interlock actuation setpoint before the BWST is depleted without any operator action to depressurize the OTSGs, it can be shown that EFW is no longer required because break/HPI cooling is adequate to remove core decay heat. If the HPI pumps must be aligned to the RB sump when the BWST is depleted, then all EFW can be lost. If all EFW is lost and the RCS break size is very small, then the RCS may repressurize up to the lift pressure of the pressurizer code safety valves.

A single failure of EFP-2 is undesirable because (1) it appreciably reduces the HPI flow delivery to the core if RCS repressurization occurs, and (2) the effects of the failure will not be apparent early in the transient. Based on these considerations, the operators will need to periodically check the adequacy of the HPI flow distribution to assure that a broken HPI line, if present, is isolated.

For a single failure of EFP-2, the HPI line isolation criterion is evaluated for a CLPD break, full HPI line break, and an HPI line pinch break. The core flood line break need not be considered because the break area is large enough that EFW is not required for core cooling.

7.3 Plant Configuration

When evaluating the breaks and single failures described above, four major plant configurations were considered. These plant configurations are based on an assumed single failure of one vital DC channel and an assumed set of operator actions (e.g., open all four HPI valves at 10 minutes). Note that every plant configuration does not have to be assessed for every break/single failure combination. Rather, only the relevant plant configurations need to be considered.

The four major plant configurations are summarized as follows:

- Configuration A - Configuration A represents the 0-10 minute time period following ESAS action or loss of SCM, and assumes that HPI suction is from the BWST, two HPI injection valves are open, normal makeup is open, RC pump seal injection is open, and normal letdown is open.

- Configuration B - Configuration B represents the 10-20 minute time period, and assumes that HPI suction is from the BWST, four HPI injection valves are open, normal makeup is open, RC pump seal injection is open, and normal letdown is open.

- Configuration C - Configuration C represents the post-20 minute time period, and assumes that HPI suction is from the BWST, four HPI injection valves

are open, normal makeup is closed, RC pump seal injection is closed, and normal letdown is isolated.

- Configuration D - Configuration D represents the long-term operation after an accident, and assumes that HPI suction is in recirculation mode, piggy-back, from the RB sump (via the LPI pumps), four HPI injection valves are open, normal makeup is closed, RC pump seal injection is closed, and normal letdown is isolated.

7.4 Single Failure to Isolate an HPI Line Break or Isolate Normal Makeup

In developing an HPI isolation criterion, several single failures, in addition to those presented in Figures 2 through 4, are considered. Of the single failures described in Section 7.2, there are two cases where the HPI flow to a broken line can not be isolated: failure to isolate an HPI injection valve and failure to isolate the normal makeup isolation valve. These single failures represent unique challenges to the plant operators in order to mitigate a SBLOCA.

In LOCA mitigation, there are two mechanisms to remove core decay energy, i.e., the OTSGs or break/HPI cooling. As long as EFW is available, the RCS pressure will remain low enough that sufficient HPI flow will be available to remove core decay heat. In these specific cases, two HPI pumps are operating and both EFW pumps are available. However, if one of these mechanisms are lost, like EFW for instance, operator action may be required to preserve (maximize) HPI flow to the core. Specific descriptions of these single failures are presented below:

Failure to Isolate an HPI Line Break

Failure to isolate HPI flow to the broken HPI line necessitates continuous OTSG cooling to maintain reduced RCS pressure and assure adequate safety injection flow. Two trains of ECCS and EFW are immediately available to provide adequate RCS and core cooling. EFP-2 will effectively act as a steam dump, since it is powered by steam from the generators, while providing EFW flow to maintain OTSG level at the required setpoint. EFP-1 will also be available until the ECCS must be reconfigured to support recirculation, piggy-back operation. At this point, EDG-1A load management can be implemented (SW/RW pumps placed in "pull-to-lock"), and EFP-1 automatic trip on LPI actuation will be defeated. This essentially assures continued EFW availability to support OTSG cooling until RCS pressure and temperature reaches decay heat removal cut-in conditions (Temperature < 300°F, Pressure < 284 psig). Once this condition is achieved, then one ECCS train can be configured for decay heat removal while the other train provides RCS makeup.

An additional strategy consists of splitting the MUP discharge header. This is discussed further in the text to follow.

If off-site power is restored, then the main condenser can be used in conjunction with the turbine bypass valves to facilitate a cooldown to decay heat cut-in conditions. Once this condition is achieved, then one ECCS train can be configured for decay heat removal while the other train provides RCS makeup.

Failure to Isolate Normal Makeup

As with a single failure to isolate an HPI line break, both trains of ECCS and EFW are immediately available. If normal makeup cannot be isolated, application of the isolation criterion will still identify the broken HPI line. Even if the normal makeup isolation valve (MUV-27) is not closed, as long as EFW is available, sufficient HPI flow will be available to cool the core. In this scenario, the isolation criterion would identify the broken HPI line and would reduce the HPI flow to the broken line, but it would not terminate the HPI flow diversion.

If EFW cannot be preserved, the operator can also split the MUP discharge header. This configuration results in one HPI pump feeding two intact injection lines while the other train HPI pump supplies flow to one intact line and the failed HPI line. Operator actions needed to accomplish this include power restoration to the motor operated cross-connect MU valve (MUV-3 or MUV-9) to enable a control room operator to close the valve. If power is not available and the radiation levels are tolerable, then an operator would have to close one of the manual cross-connect valves (MUV-4 or MUV-8) in the field.

7.5 New HPI Line Isolation Criterion

The above breaks, single failures, and plant configurations were evaluated to develop a revised isolation criterion. Based on the cases evaluated in Reference 23, a revised isolation criterion can be developed that provides sufficient HPI flow to the core for the various breaks and single failures listed above to demonstrate that core cooling will be assured. The new criterion is as follows:

At any time in an unthrottled condition, IF the highest-reading HPI line indicates flow > 50 gpm higher than the next highest-reading HPI line, THEN isolate the high flow HPI line.

This new criterion should be applied when all four injection lines are fully-open, with normal makeup isolated. If normal makeup cannot be isolated, or if an HPI

injection line valve in the broken line is failed open, additional actions may be required. In any case, the isolation criterion will identify the line that needs to be isolated. If subcooling margin has been maintained or restored before the criterion is met, and the HPI flow has been throttled, no isolation action should be taken for SBLOCA mitigation. If subcooling margin is subsequently lost, the operators should maximize the HPI flow before taking any action to isolate a broken line.

The new criterion takes advantage of the latest CR-3 HPI hydraulics model and new flow measurement uncertainties, References 7 and 8. The new criterion is logically similar to the existing criterion, but is much simpler. The current criterion is biased to the lowest flowing lines which introduces significant uncertainties. The new criterion is biased to only the two high flowing lines, and there should be less flow measurement uncertainty. The new criterion would not result in the isolation of an intact line or fail to identify a line that is required to be isolated.

The revised criterion does require that the operators periodically monitor the HPI flow splits to ensure that, for specific HPI line pinch areas, a broken line will be isolated if warranted.

8.0 EFW Inventory Requirements

Based on the solution sets, EFW is required for a period of time until break/HPI cooling is sufficient to remove the core energy. Since EFW is required, an assessment was made of the inventory of emergency feedwater required until break/HPI cooling is adequate to cool the core.

This assessment was based upon the following assumptions and boundary conditions:

- Initial Core Power of 1.02 * 2568 MWt with 10CFR50 Appendix K Decay Heat

  The 1.02 factor is required by Appendix K. The initial core power level of 2568 MWt is higher than the current licensed power level of 2544 MWt. The one percent difference is conservative for a planned future power upgrade. The decay heat is required by 10CFR50 for LOCA design bases accident analyses, i.e. 120 percent of ANS 5.1 1971 using infinite irradiation.

- Over the period of interest, EFW fills the OTSGs to 95 percent on the operate range.

  Consistent with EOP guidance, EFW will be used to fill the OTSGs to the loss of subcooling margin setpoint following a SBLOCA. The required inventory will include filling the OTSGs with subcooled liquid plus the balance to remove core decay heat. It is conservative to assume that the OTSGs are initially empty and, in addition to removing core decay heat, EFW must also fill both OTSGs.

- EFW must remove excess core decay heat until break/HPI/PSV cooling at 2575 psig is adequate to match core decay heat

  A pressure of 2575 psig assumes that EFW is lost and the RCS will instantaneously repressurize to the pressurizer safety valves. At this pressure, the HPI flow, heated to saturated steam, must match the core decay heat. This is valid because, before the core can uncover, the hot leg piping will be empty and steam will be available to pass to the pressurizer. The lift setpoint of the pressurizer safety valves is 2500 psig. An additional three percent for valve accumulation is added for conservatism. The pressure at the top of the pressurizer will not be significantly different than that at the HPI injection nozzles, i.e. the RCS is saturated and partially void of liquid. Therefore, the same pressure can be used at both locations.

- EFW temperature will be 120°F.

- Primary metal and cooling of the fuel and cladding will not be accounted for in the calculation.

  Over the relatively long time of interest, the potential heat addition from these sources will not significantly change the results.

- Heat loss to the ambient will be ignored.

- The required EFW inventory is calculated by integrating the core decay heat over time with an end time based on the mission time for EFW as calculated in Section 4.0.

During a SBLOCA, the core decay heat is removed through the break, via the reactor vessel internals vent valves, to the OTSGs, if natural circulation is maintained or by boiler-condenser cooling, and by heating the HPI fluid. It is difficult to determine the contribution of each potential source of energy removal because of the wide variations in RCS conditions that could exist. For instance,

if the RCS stays or becomes subcooled, the operators are given guidance to throttle the HPI flow to manage RCS subcooling margin. The amount of HPI flow is not known, and therefore it is conservative to not credit the potential for energy absorption. Similarly, all possible break sizes, types, and locations also need to be considered. Since the fluid conditions at the break can change, a conservative assumption must be made to bound the break energy removal capability.

Since not all break sizes require emergency feedwater for the same period of time, a series of calculations for different break areas were performed. The results are presented in Figure 24. The data presented in this figure are developed in Reference 11. These calculations provide an approximate value of the required EFW flow for various break sizes. A conservative assumption that the break (at critical flow conditions) will pass saturated liquid at 600 psia. For the break sizes that require EFW for mitigation, the expected RCS pressure will be greater than 600 psia. Below this pressure, other equipment is available to provide injection flow, i.e., the core flood tanks or LPI if the RCS pressure is low enough. The calculation integrates the core decay heat and then subtracts the integrated break energy. The balance must be removed by the OTSGs. Once the energy removed through the break exceeds the core decay heat, the required EFW inventory to remove core decay heat for a given break size is known. The amount of EFW needed to fill both OTSGs to the loss of SCM setpoint must be added to the integrated value to determine the total inventory requirements.

The liquid volume to fill both OTSGs to the loss of SCM setpoint, 95 percent on the operate range, is calculated below using information from Reference 27:

Loss of SCM setpoint level = {(0.95)*(292 inches)+(102 inches)}/12 in/ft
= 31.62 ft

The OTSG cross sectional area, from References 24 and 25, is

OTSG Area = (43.4955 + 23.3308)*1.0152
= 67.84 ft^2

The total inventory in both OTSGs is:

OTSG inventory = (67.84 ft^2)*(31.62 ft)*(2)*(7.4805 gal/ft^3)
= 32092.9 gallons

As seen in Figure 24, break sizes above approximately 0.04 ft^2 effectively do not need EFW to mitigate the transient and the inventory required is what is necessary to fill both OTSGs. At the opposite end of the curve, a "zero" break area, requires approximately 339,000 gallons of condensate over 35 hours after

reactor trip. This is based on the mission time calculated when one HPI pump is available, in the recirculation mode, with the limiting pinch break HPI flow rates in Table 2, Section 4.0.

The 339,000 gallon inventory required over the 35 hour EFW mission time is available on site. The total capacity of the EFW tank is 184,000 gallons, with a 150,000 gallon minimum capacity, as required by the Technical Specifications. In addition, water in the condensate storage tank (200,000 gallon capacity) and in the condenser hotwells (200,000 gallon combined capacity) is also available.

In developing the isolation criterion contained in Section 7.0, additional single failures are considered. Specifically, if the single failure is one in which an HPI line fails to close or the makeup isolation valve fails to close, the mission time for EFW can be quite long. For these cases, Figure 25 was developed. This figure provides the total EFW inventory requirements as a function of time for a "zero" break area using a conservative core decay heat value. In effect, the integrated core power based on the initial core power level (1.02 times 2568 MWt) is multiplied by the integrated full power seconds (FPS) taken from Reference 6 at several times. No credit is taken for the energy removed by the break flow or to heat the HPI fluid. It is assumed that EFW must absorb all of the decay heat, so at each of the times, the integrated core decay energy is divided by the enthalpy rise of the EFW fluid, approximately 1100 Btu/lbm, and the result is converted to a volume. Assuming a realistic core decay heat, 90 percent of the 1971 ANS 5.1 standard, the required inventory can be reduced by approximately 30 percent.

If credit is taken to split the HPI pump discharge header, then EFW mission time and EFW inventory requirements will be of the same magnitude as for the other worse-case single failures evaluated, i.e., 35 hours and 339,000 gallons.

9.0 References

*1. Florida Power Corporation Interoffice Communication, NOE 97-0584, Startup Team Solution Set - Safety Assessment (Revision 1), 5/23/97.

2. Small Break Loss-of-Coolant Accident Analysis for B&W 177-FA Lowered-Loop Plants in Response to NUREG-0737, Item II.K.3.31, Babcock & Wilcox, BAW-1976A, Lynchburg, Virginia, May 1989.

3. FTI Document 51-1245866-00, Reevaluation of HPI Requirements During Small Break LOCAs, April 1996.

*4. Florida Power Corporation Nuclear Operations Engineering, Crystal River Unit 3, Enhanced Design Basis Document for the Decay Heat System, System Code: DDH, Tab 6/3, Revision 6, 9/27/96.

*5. FPC Calculation M-97-0026, Revision 0, HPI Pump Flows for SBLOCA.

6. FTI Document 32-1258134-00, Decay Heat for LOCA Analysis, 9/26/96.

*7. FPC Calculation I89-0037, Revision 5, Make-Up/HPI Flow LOOP Accuracy (Low Range), 6/10/97.

*8. Florida Power Corporation Interoffice Communication, NOE 97-0230, CR-3 HPI Isolation Criteria, 4/12/97. (See Also Suspected Design Bases Issue Review, PC 97-1505.)

*9. FPC Calculation I-91-0012, BWST Level Accuracy, Revision 3, 11/20/96.

*10. FPC Calculation M-93-0017, Revision 2, Borated Water Storage Tank (DHT-1) Volume, 2/14/97.

11. FTI Document 32-1266136-00, CR-3 Startup Team SBLOCA Analyses.

*12. FPC Calculation H-97-0001, Revision 0, Control Complex Transient Temperature Model.

*13. FPC Calculation M-95-0016, Revision 1, BWST Swapover and Minimum Allowable Level Evaluation, 3/4/96.

*14. Florida Power Corporation Nuclear Operations Engineering, Crystal River Unit 3, Enhanced Design Basis Document for the Reactor Building Spray System, System Code: BS, Tab 6/4, Revision 5, Dated 3/3/97.

15. FTI Document 32-1266137-00, CR-3 RB Analysis with 2 Fan Coolers.

*16. FPC Interoffice Correspondence, PROG96-0003, BS and LPI Actuation Points, S.K. Balliet to K.R. Campbell, June 28, 1996.

*17. EFP-2 Purchase Specification: FPC RO 2931, Addendum B, FPC-322-013, PR-1374, November 18, 1969.

*18. Ingersoll-Dresser Letter Paul J. Kasztejna (Supervising Design Engineer Ingersoll-Dresser Pump Co.) to Mark Liebmann (Florida Power Corporation), dated May 7, 1997.

*19. Ingersoll-Rand, Pump Curve 082A, 9.25-Inch Impeller, Part Number 4X9N3A/3B, May 7, 1997.

20. Methods of Compliance With Fracture Toughness and Operational Requirements of 10CFR50, Appendix G, Babcock and Wilcox, Lynchburg, Virginia, BAW-10046A, Rev. 2, June 1986.

21. FTI Document 32-1244465-00, RELAP5 CR-3 SBLOCA Spectrum.

*22. Florida Power Corporation Nuclear Operations Engineering, Crystal River Unit 3, Enhanced Design Basis Document for the Main Steam System, System Code: MS, Tab 6/10, Revision 7, 3/21/97.

23. FTI Document 51-1266161-00, CR-3 Revised HPI Line Isolation Criterion.

24. B&W Document 32-1159004-00, Task AS-4 Operator Actions to Reestablish Natural Circulation, 2/20/87.

25. BWNT Document 32-1229132-00, Oconee R5 LBLOCA EM, 3/8/94.

26. B&W Document 32-1158464-01, ANS 5.1 (1979) Decay Heat Curve Fit, 9/30/86.

27. BWNT Document 51-1212232-01, Key Elevations for All Plants, 3/24/94.

* The documents marked with an asterisk are maintained and controlled by Florida Power Corporation. Per FTI procedures, use of these references are allowed in safety-grade calculations with the approval of the cognizant unit manager or contract manager. The signature below authorizes the use of these documents for input to this evaluation.

BL Boman for JJ Cudlin
(Unit Manager/Contract Manager)

6/12/97
(Date)

FIGURE 1.

SOLUTION SET FLOW PATH

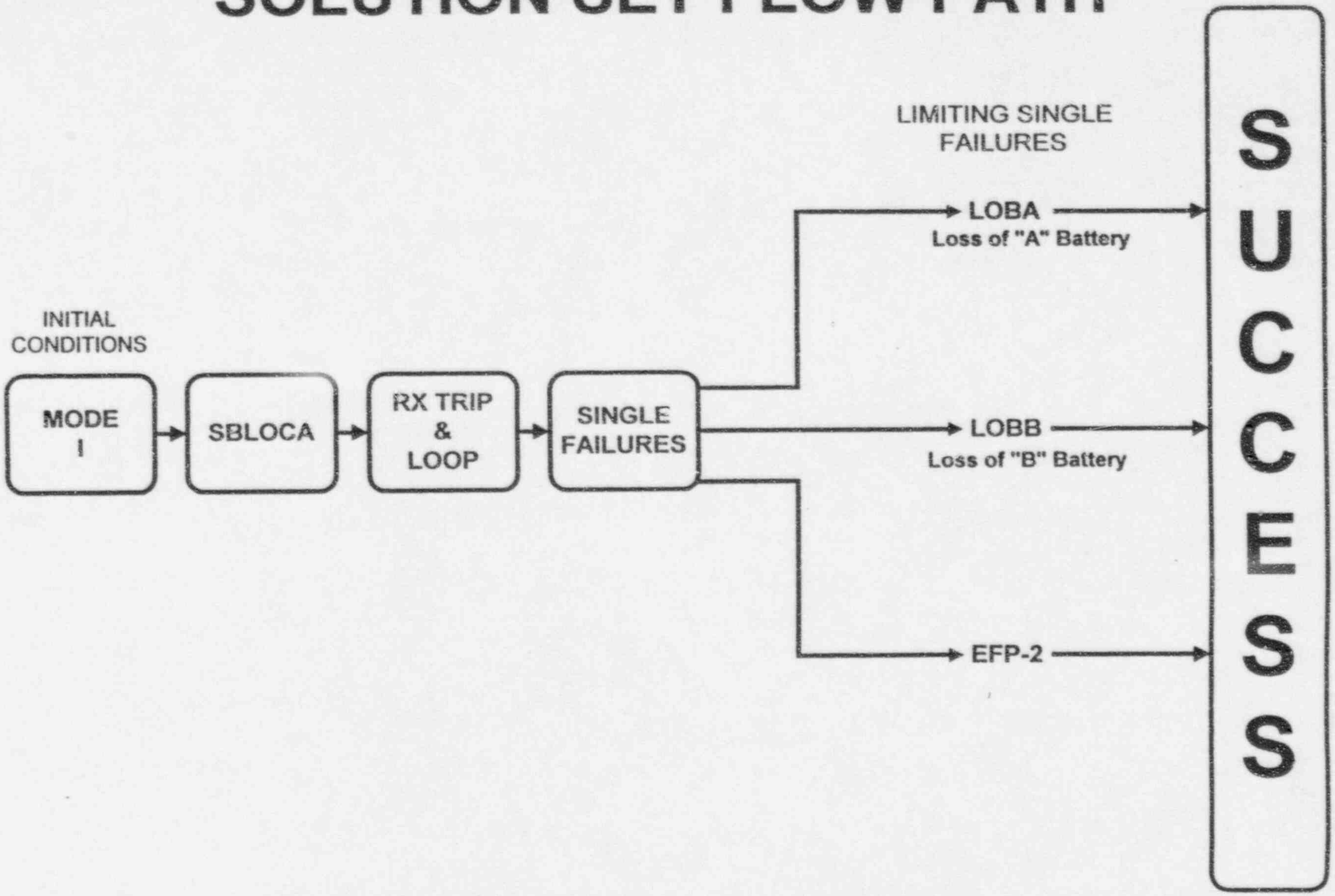

FIGURE 2. LOBA Single Failure Solution Set

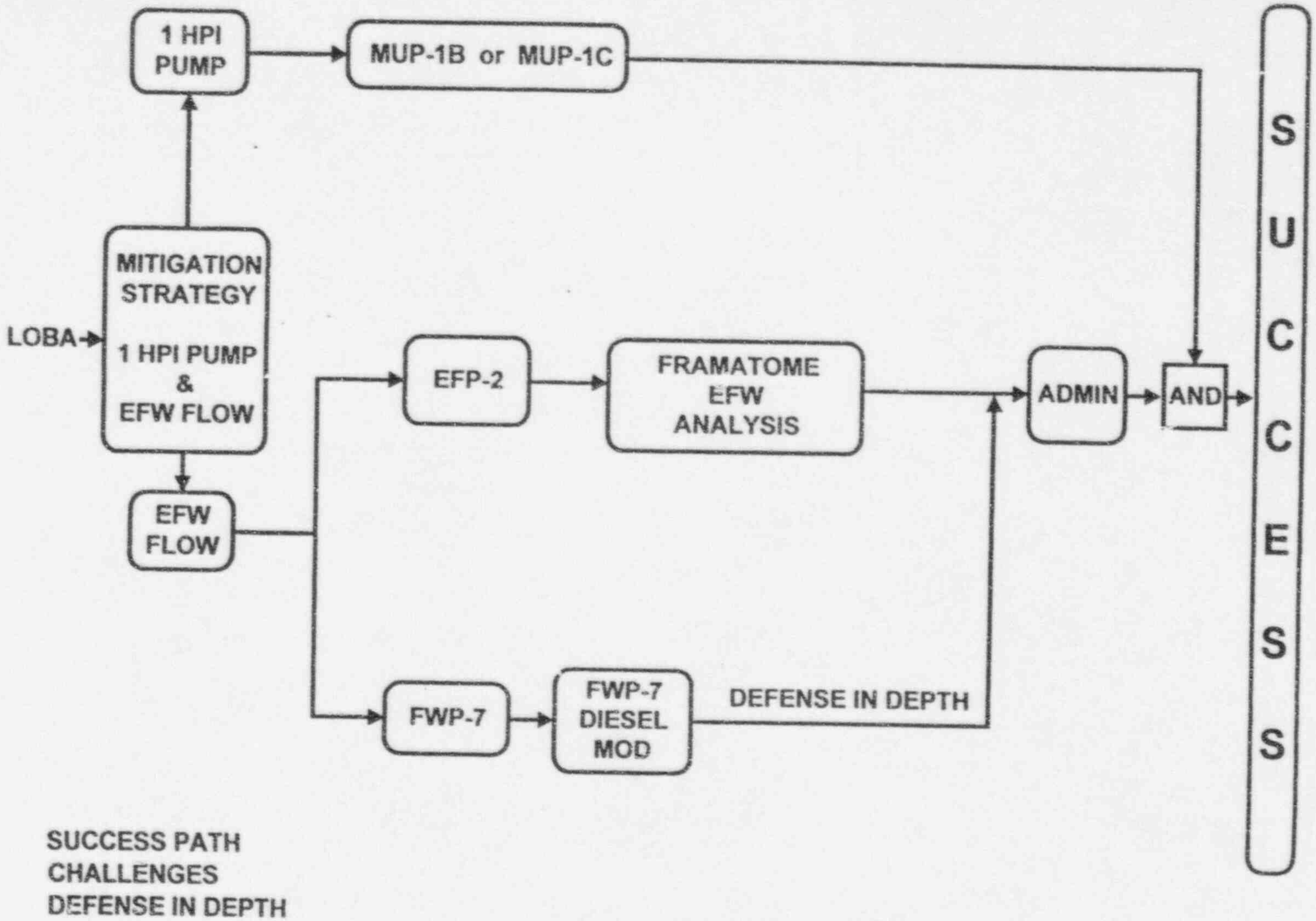

FIGURE 3. LOBB Single Failure Solution Set

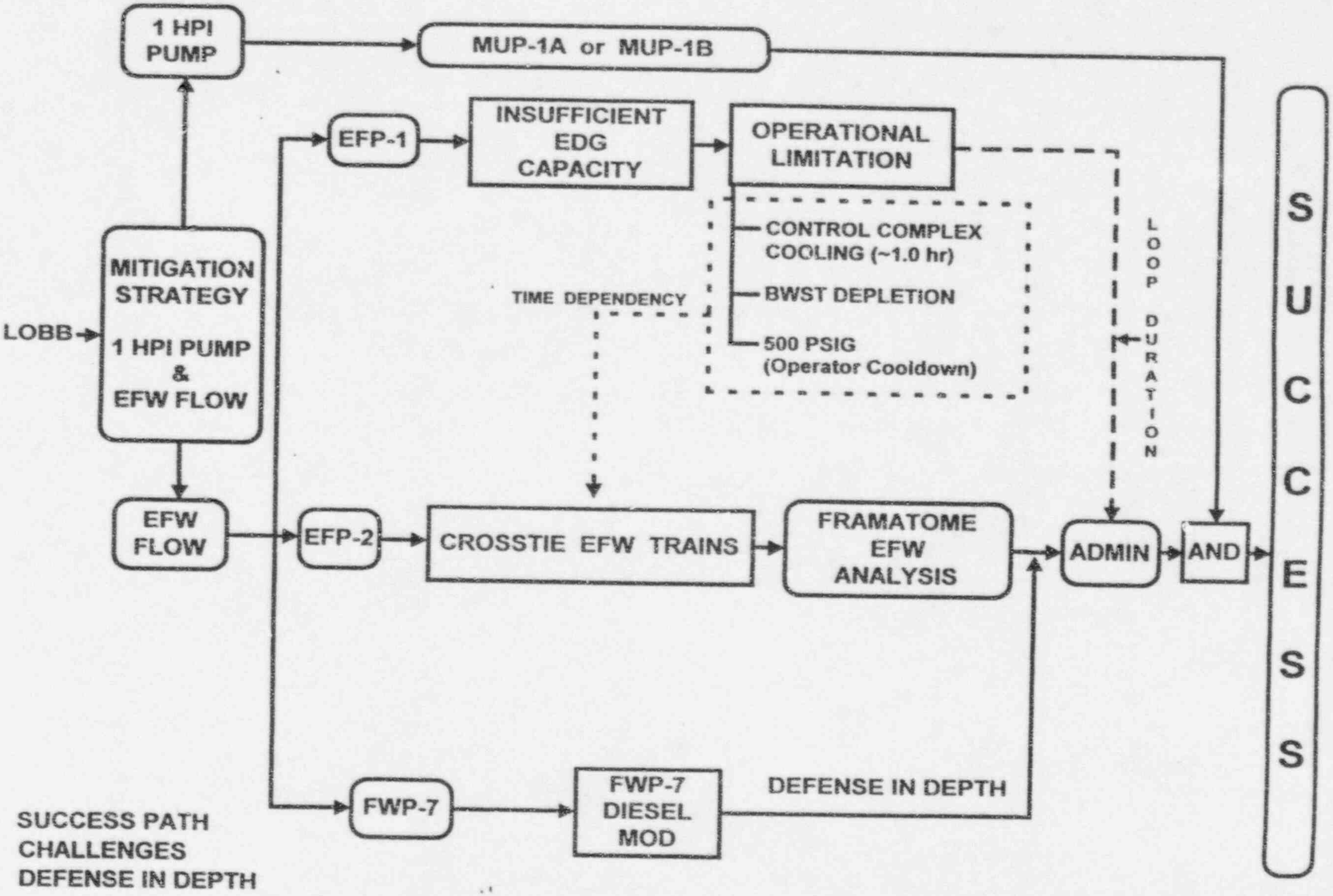

FIGURE 4. EFP-2 Single Failure Solution Set

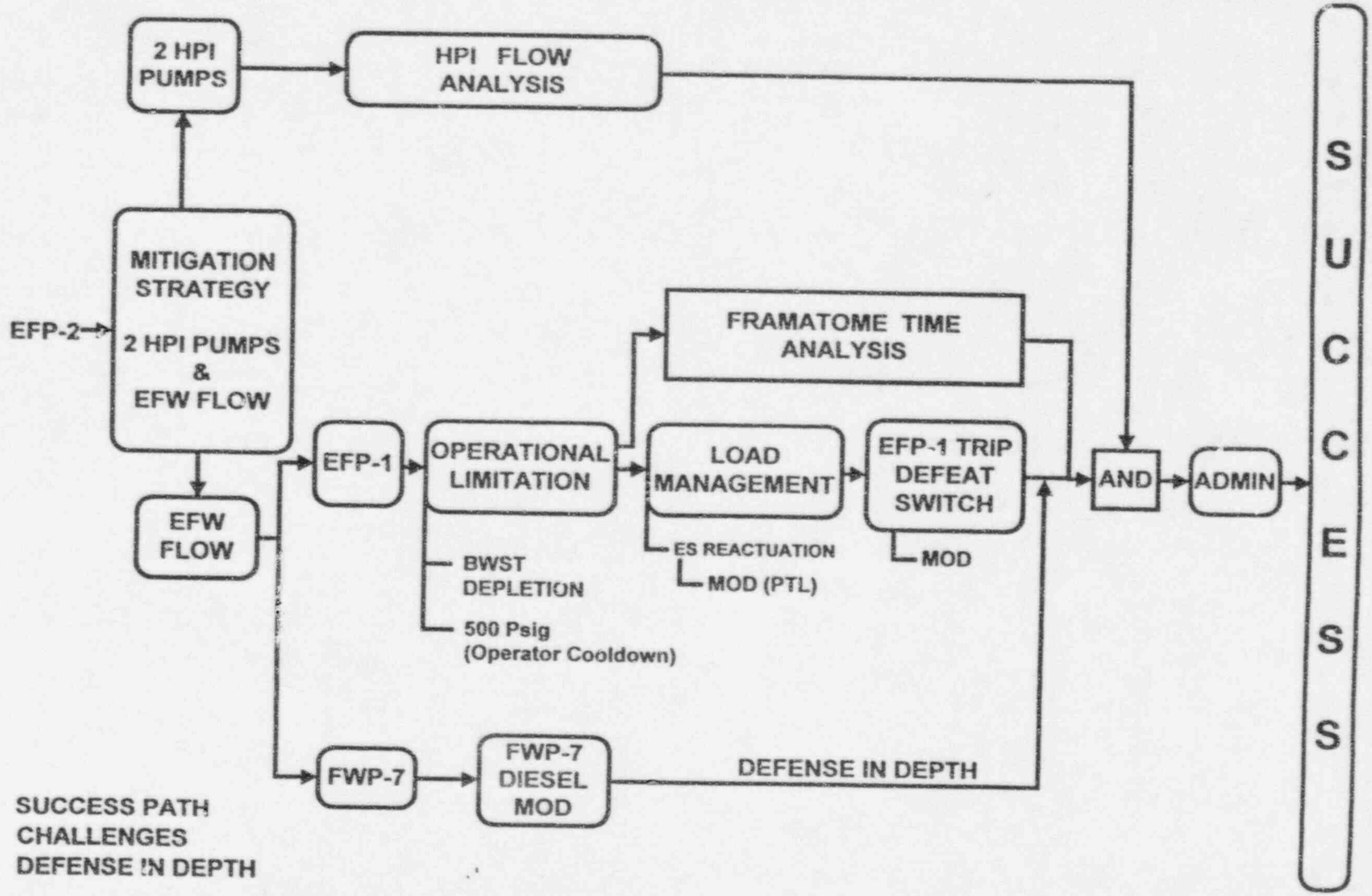

FIGURE 5. CR-3 0.0025 ft^2 Break - Pressure Versus Time

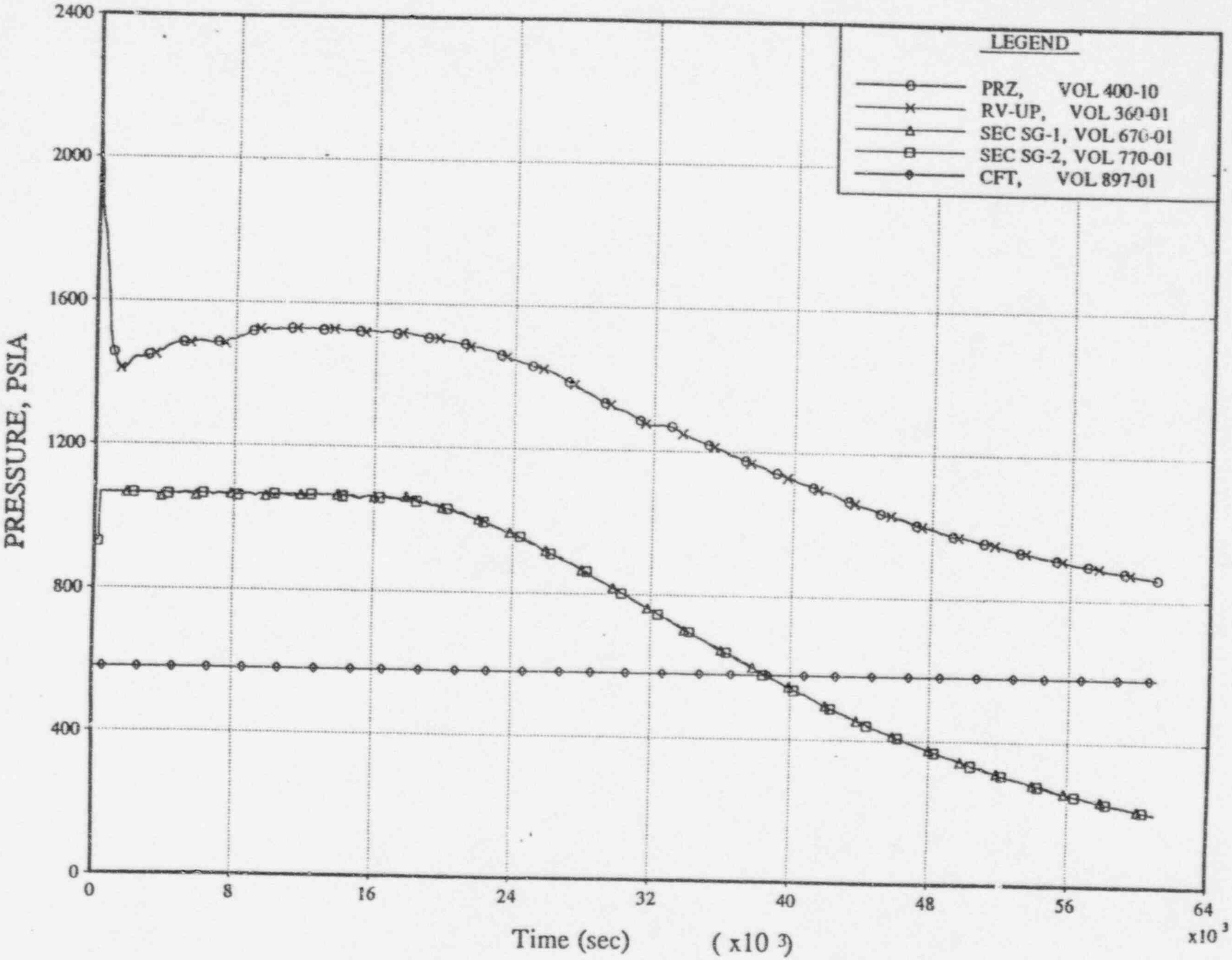

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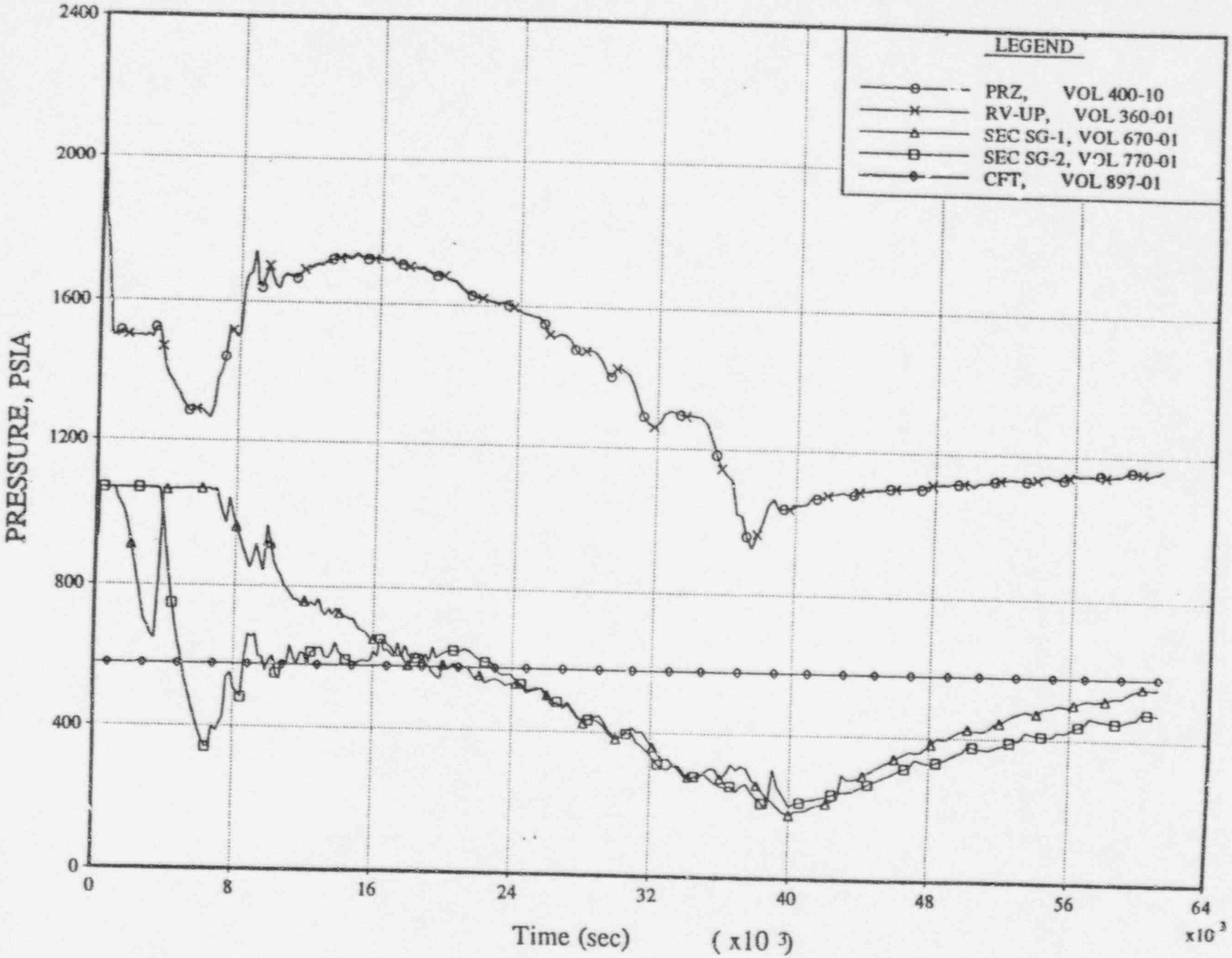
FIGURE 6. CR-3 0.003 ft^2 Break - Pressure Versus Time



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51-1266138-01

FIGURE 7. CR-3 0.0035 ft^2 Break - Pressure Versus Time

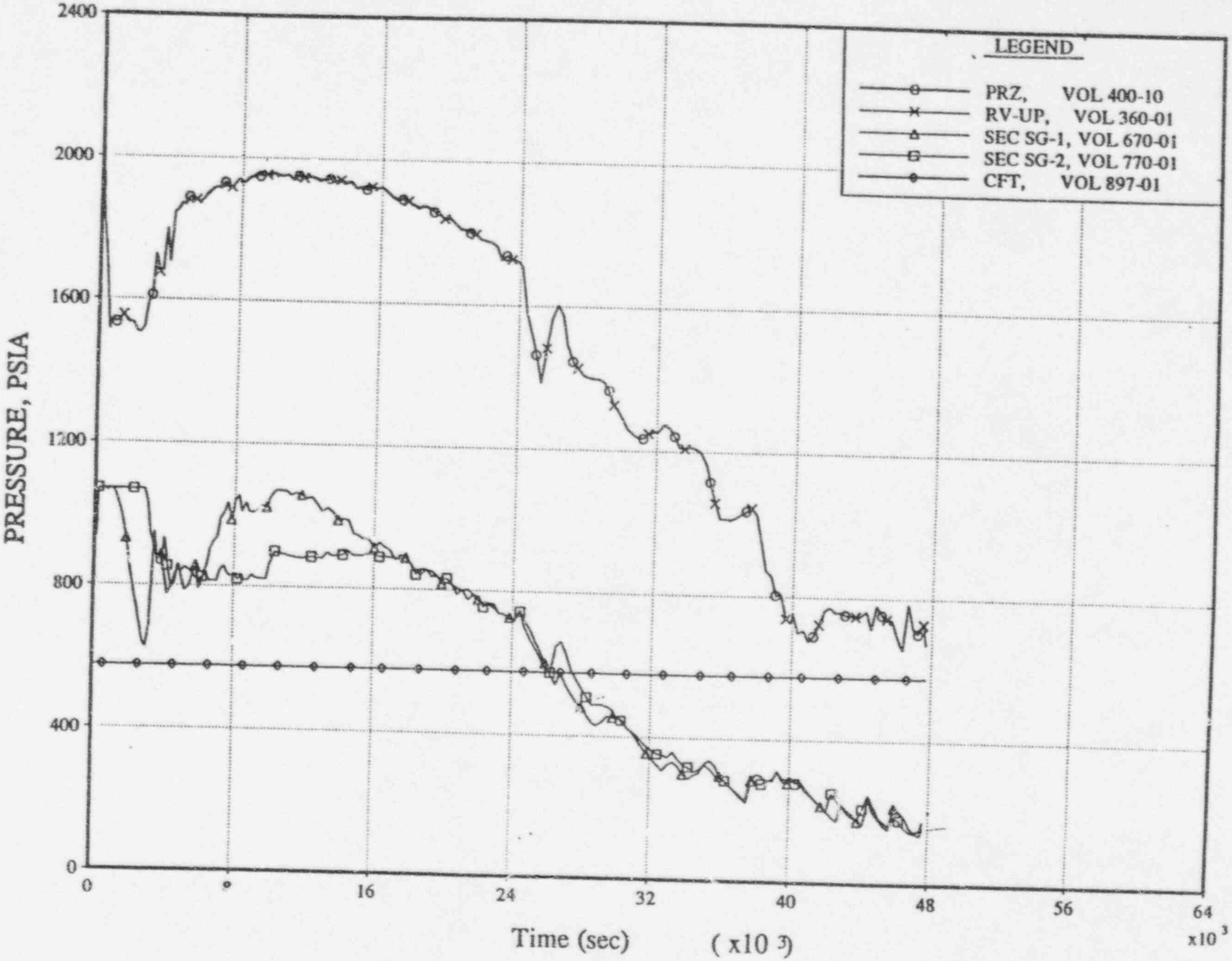

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FIGURE 8. RCS and Secondary Pressures, 0.04 ft^2 Break at RC Pump Discharge

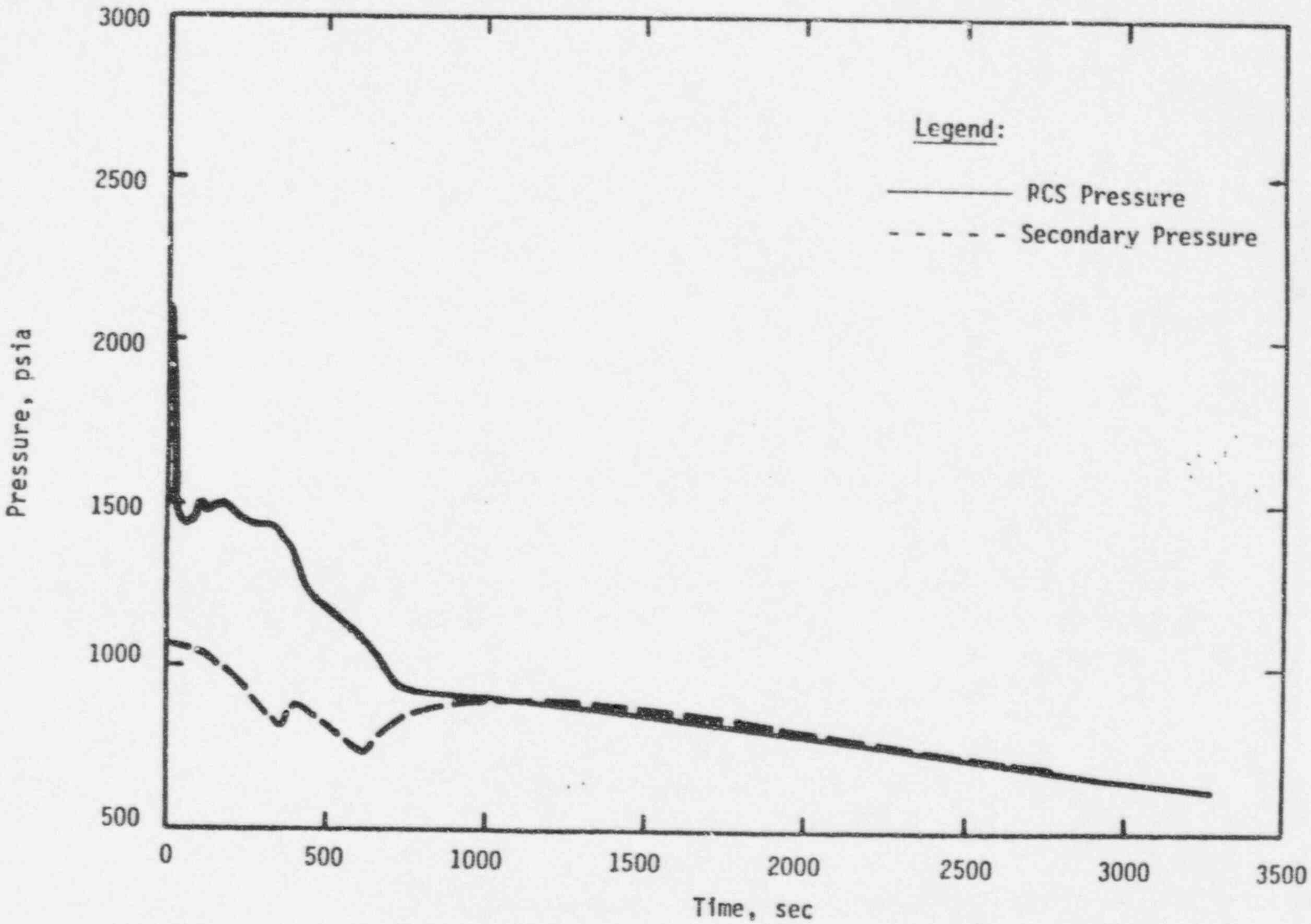
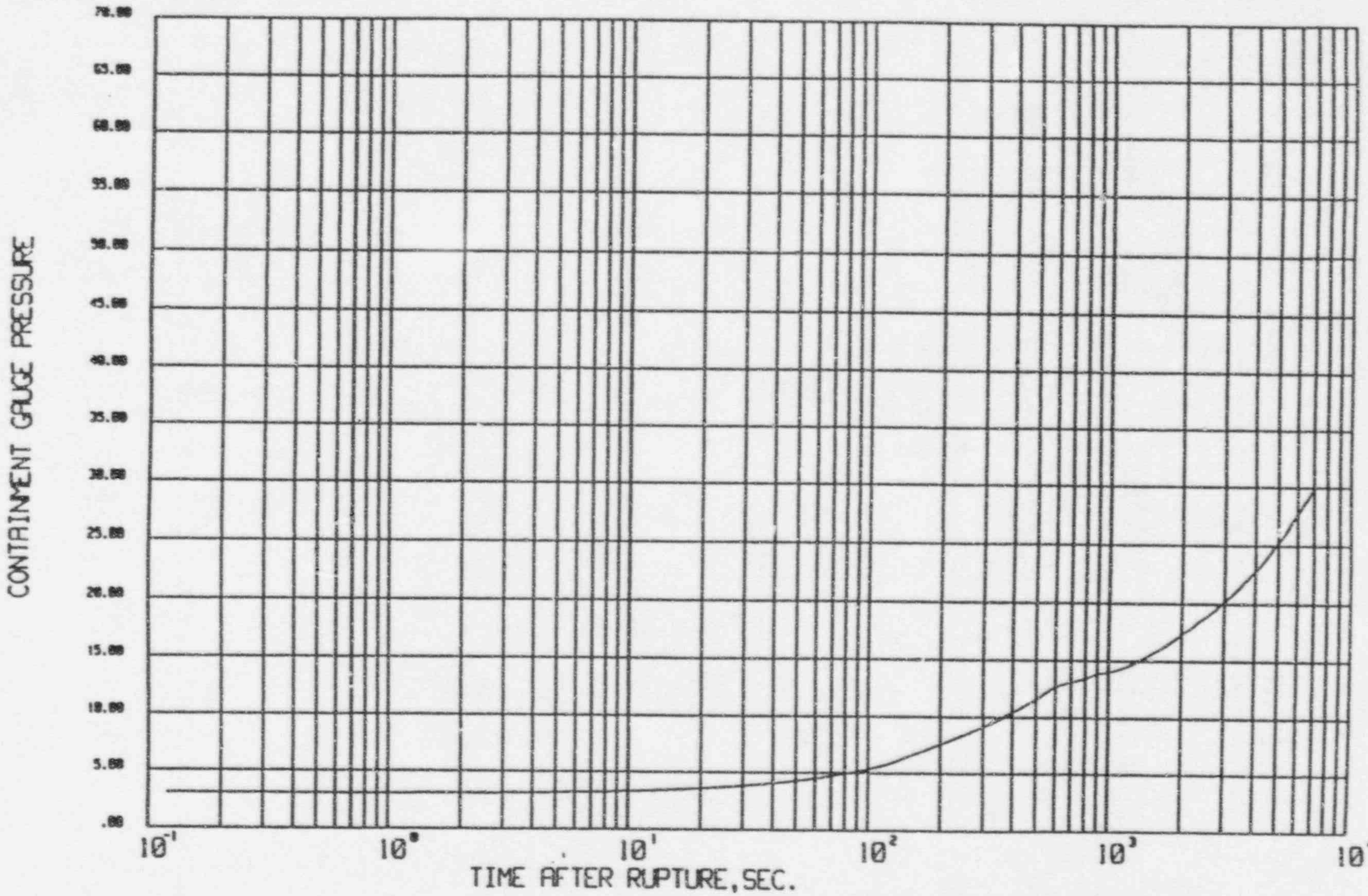


FIGURE 9. CR-3 Maximum RB Pressure for 0.01 ft^2 CLPD Break - One Cooler



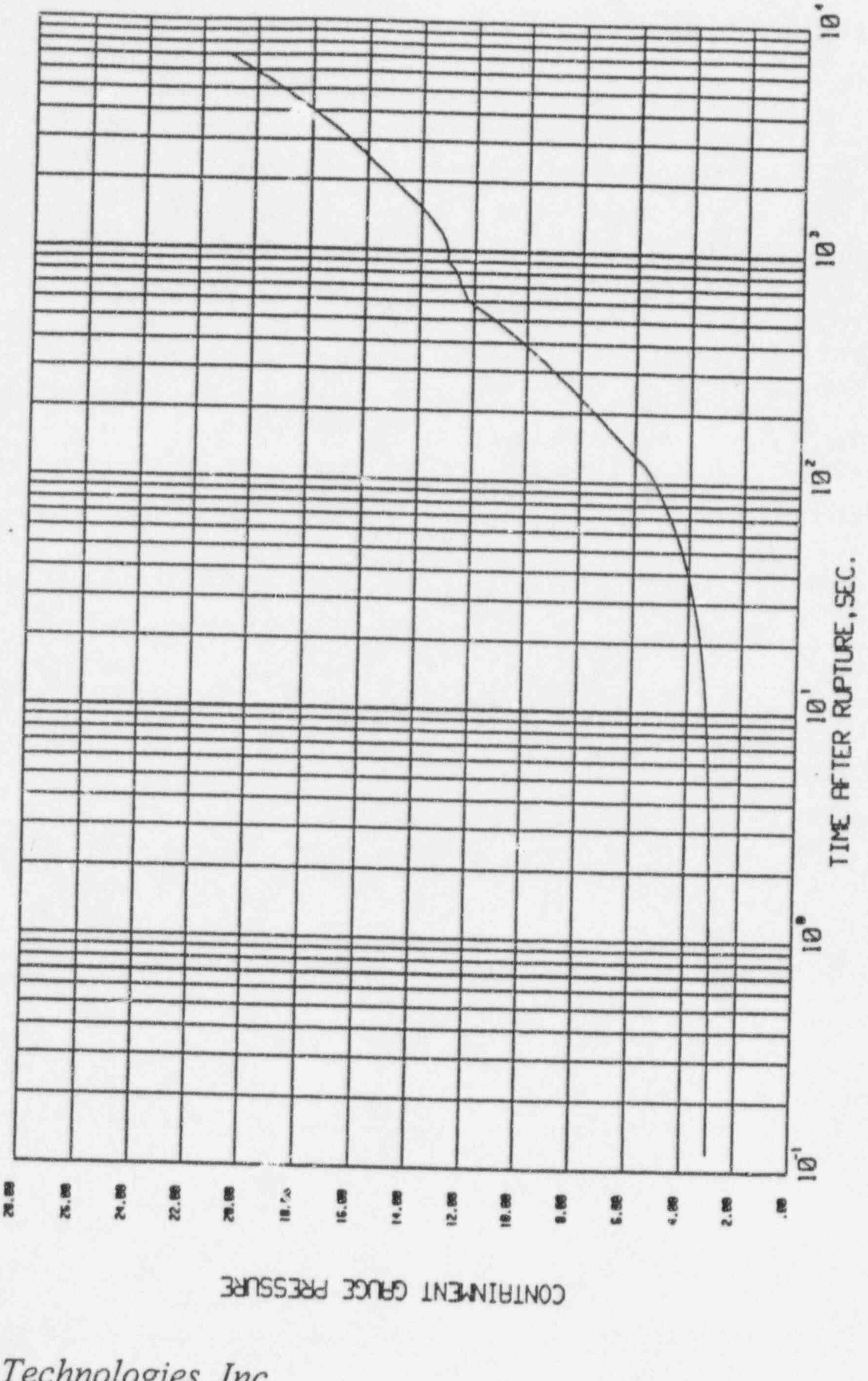


FIGURE 10. CR-3 Maximum RB Pressure for 0.01 ft^2 CLPD Break - Two Coolers

FIGURE 11. CR -3 Maximum RB Pressure for 0.04 ft^2 CLPD Break - One Cooler

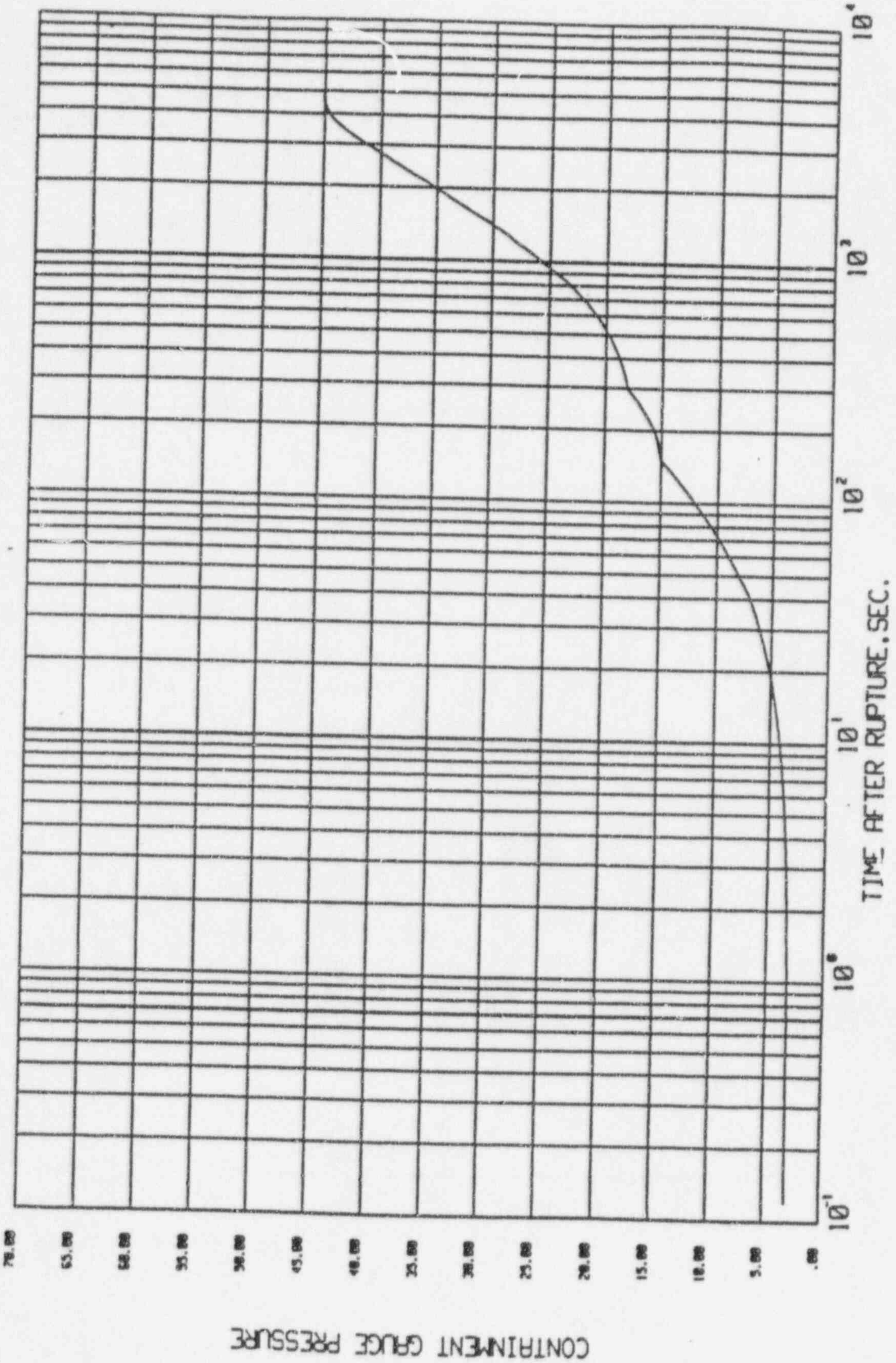


FIGURE 12. CR-3 Maximum RB Pressure for 0.04 ft^2 CLPD Break - Two Coolers

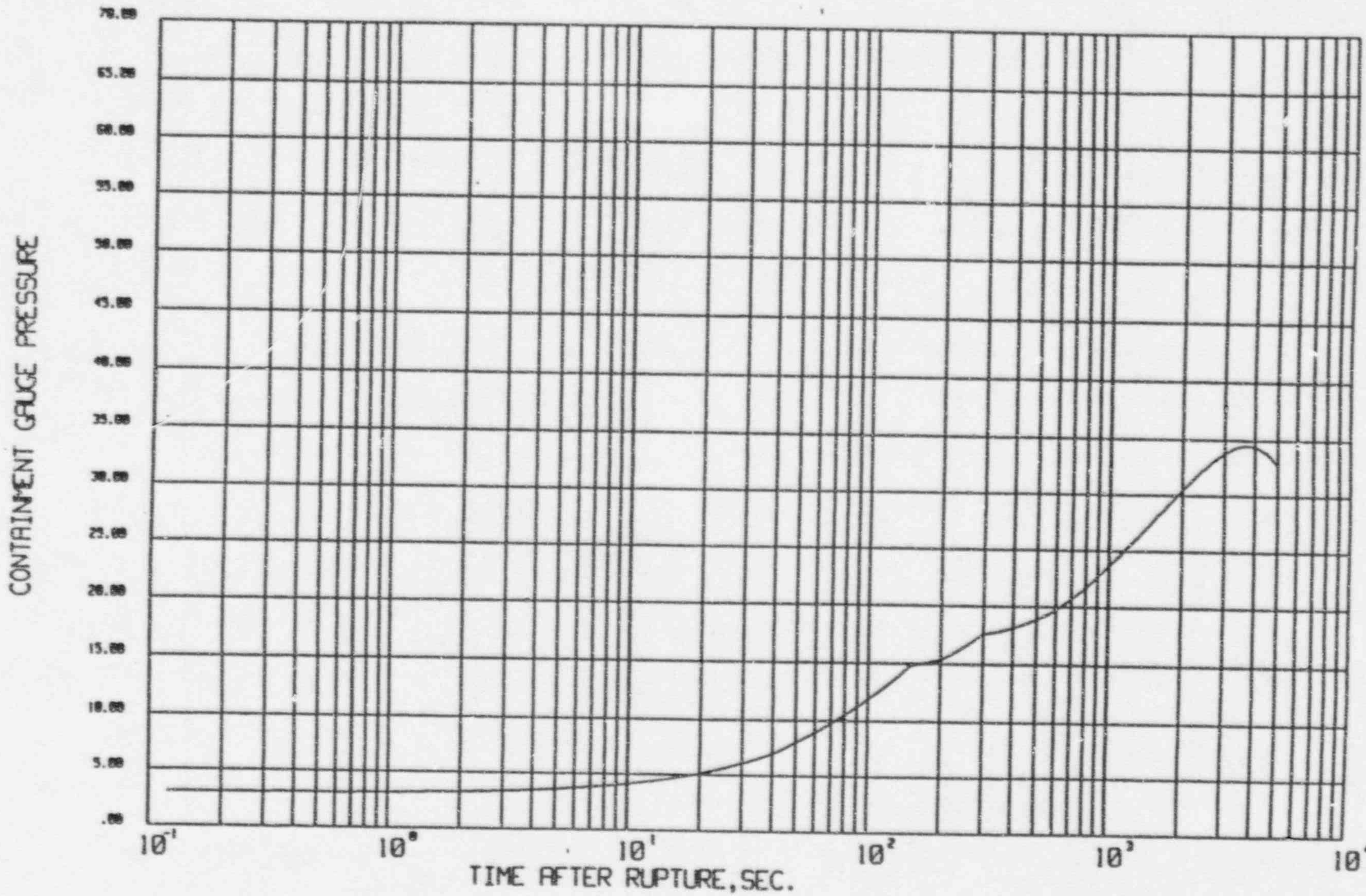


FIGURE 13. EFW Cavitation Flow Versus Inlet Pressure

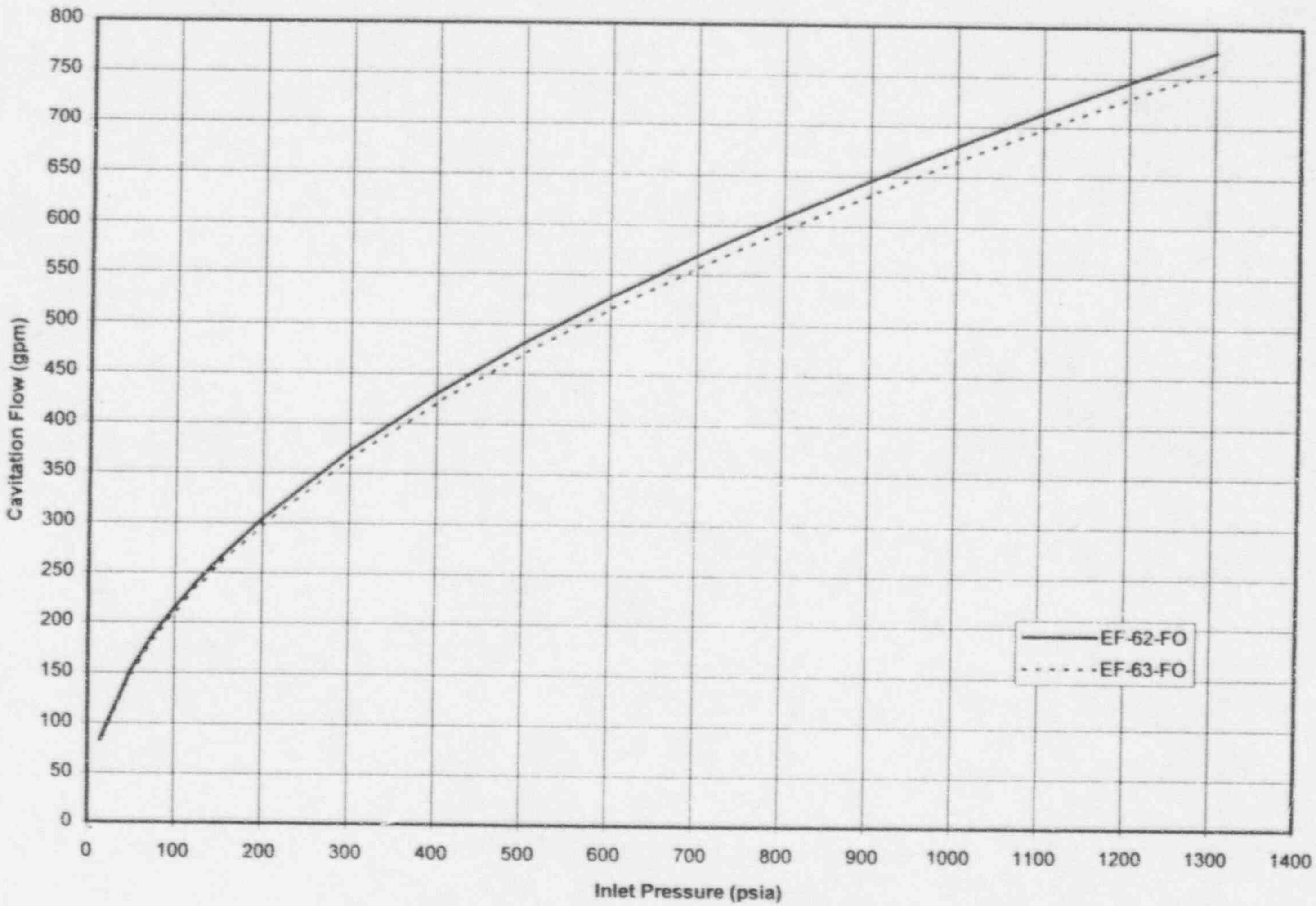


FIGURE 14. Florida Power Corp. Feed Pump Characteristic Curve for 1080 RPM

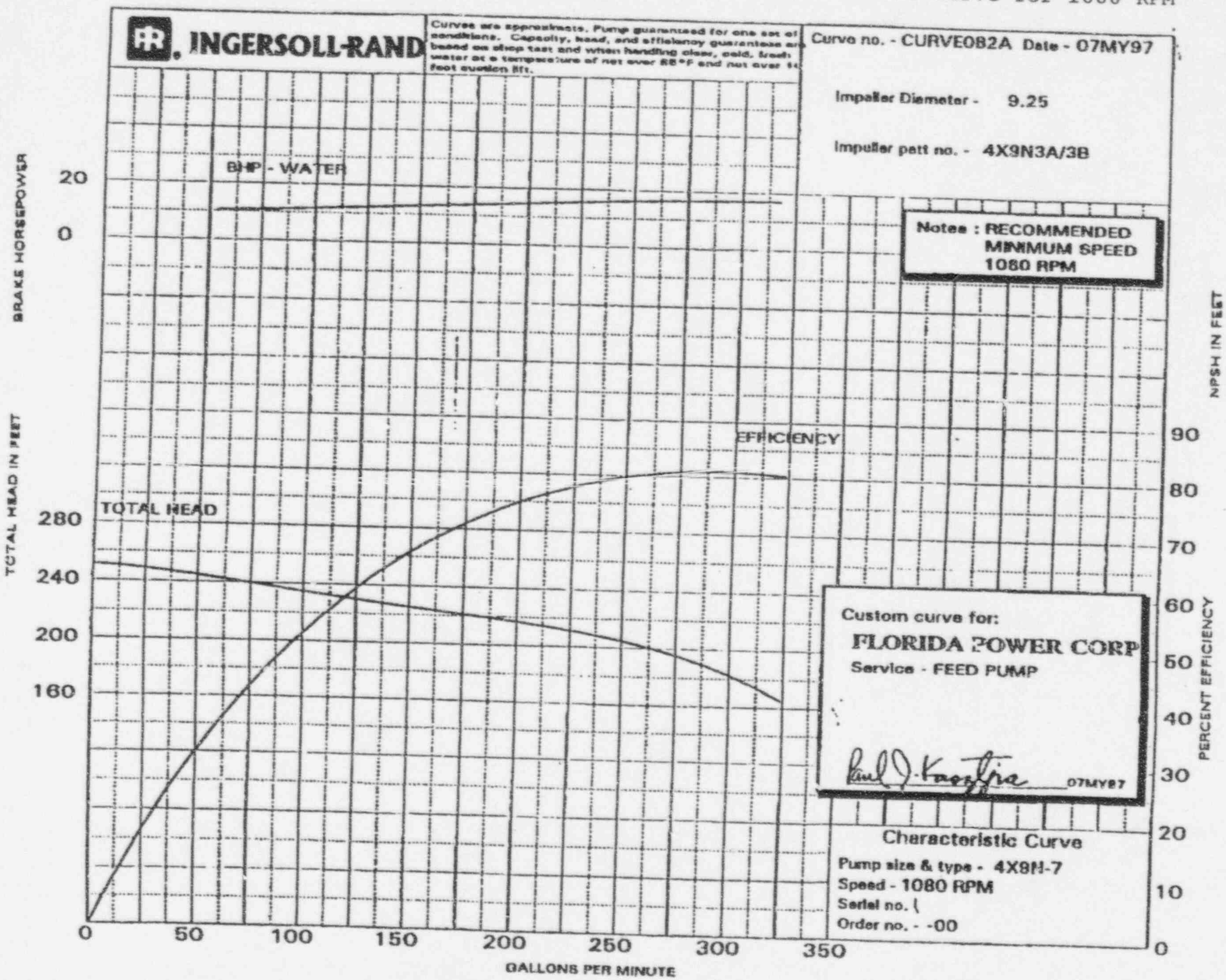


FIGURE 15. EFW Flow to Match Core Decay Heat

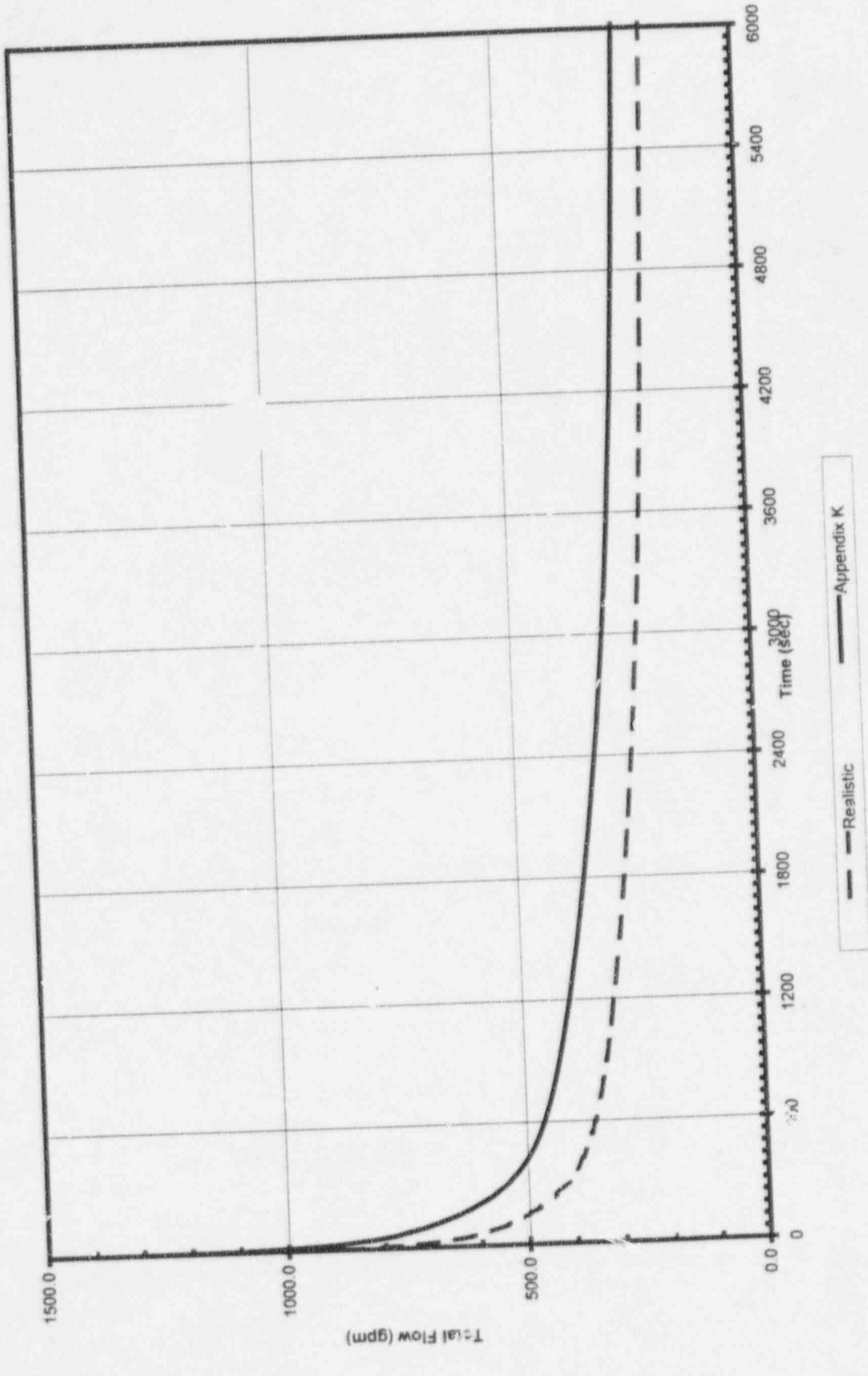


FIGURE 16. CR-3 0.002 ft^2 Break - Pressure Versus Time

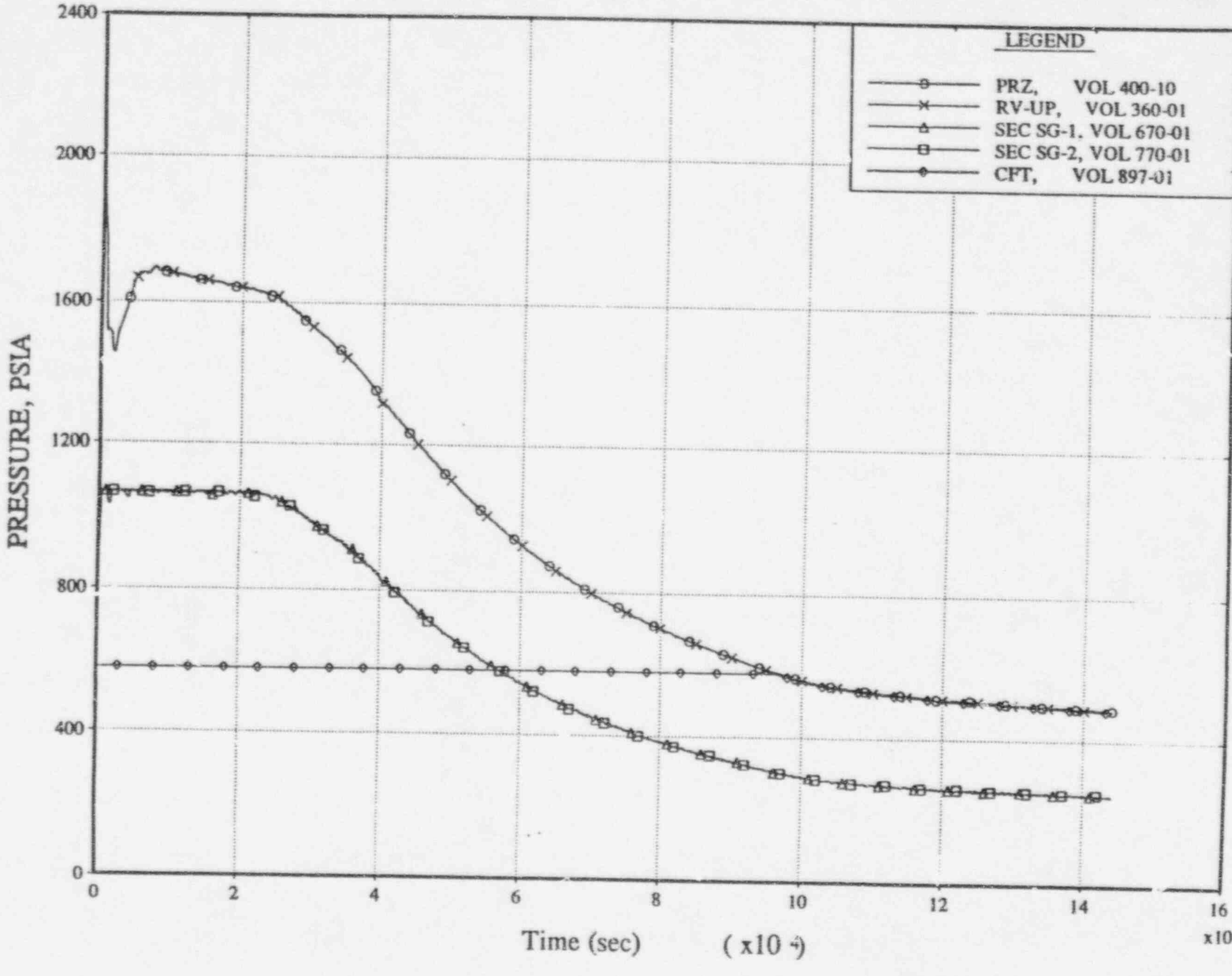


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FIGURE 17. CR-3 0.002 ft^2 Break - Mass Flow Versus Time

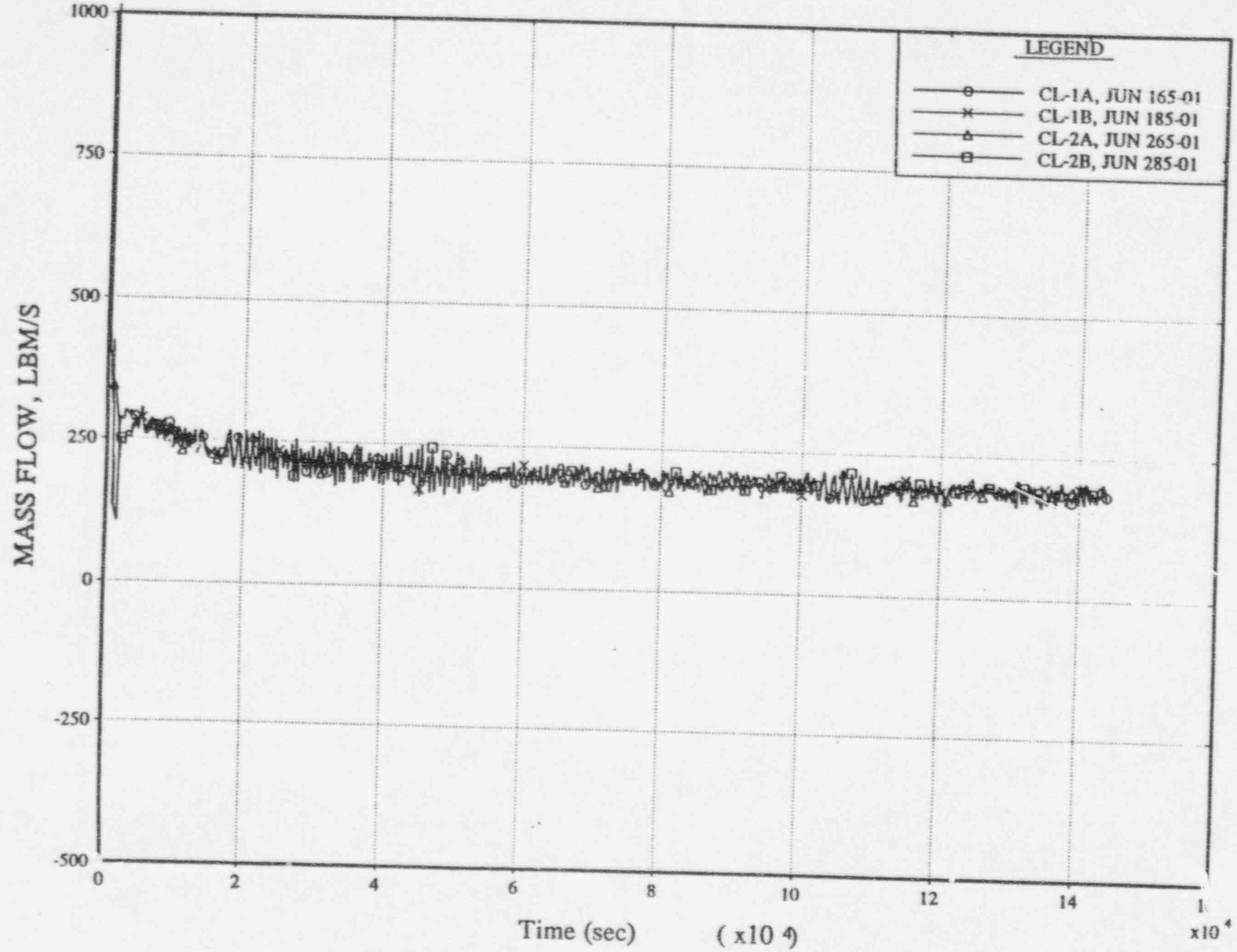

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FIGURE 18. CR-3 0.002 ft^2 Break - Temperature (CVs 175/275) Versus Time

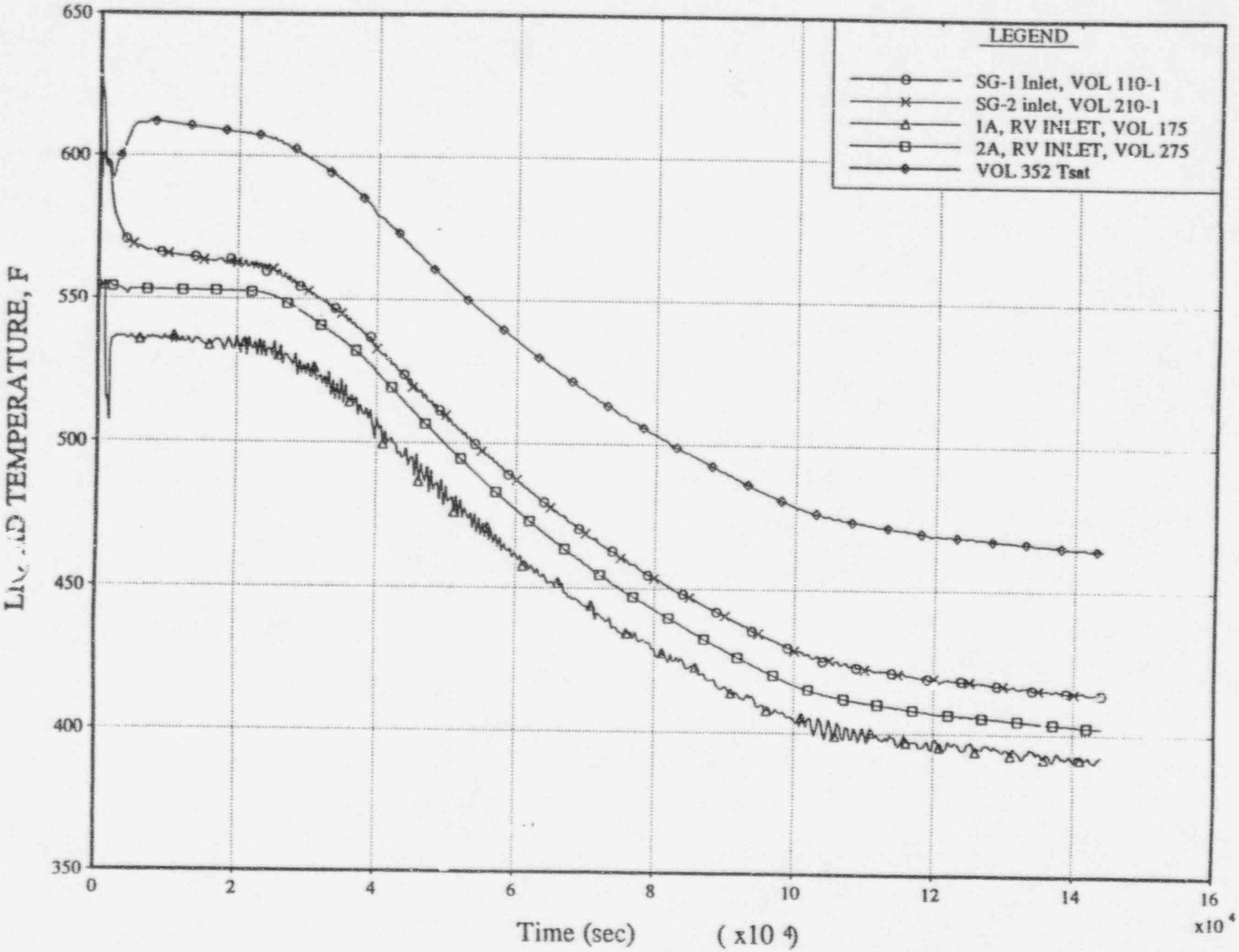

FIGURE 19. CR-3 0.002 ft^2 Break - Temperature (CVs 195/295) Versus Time

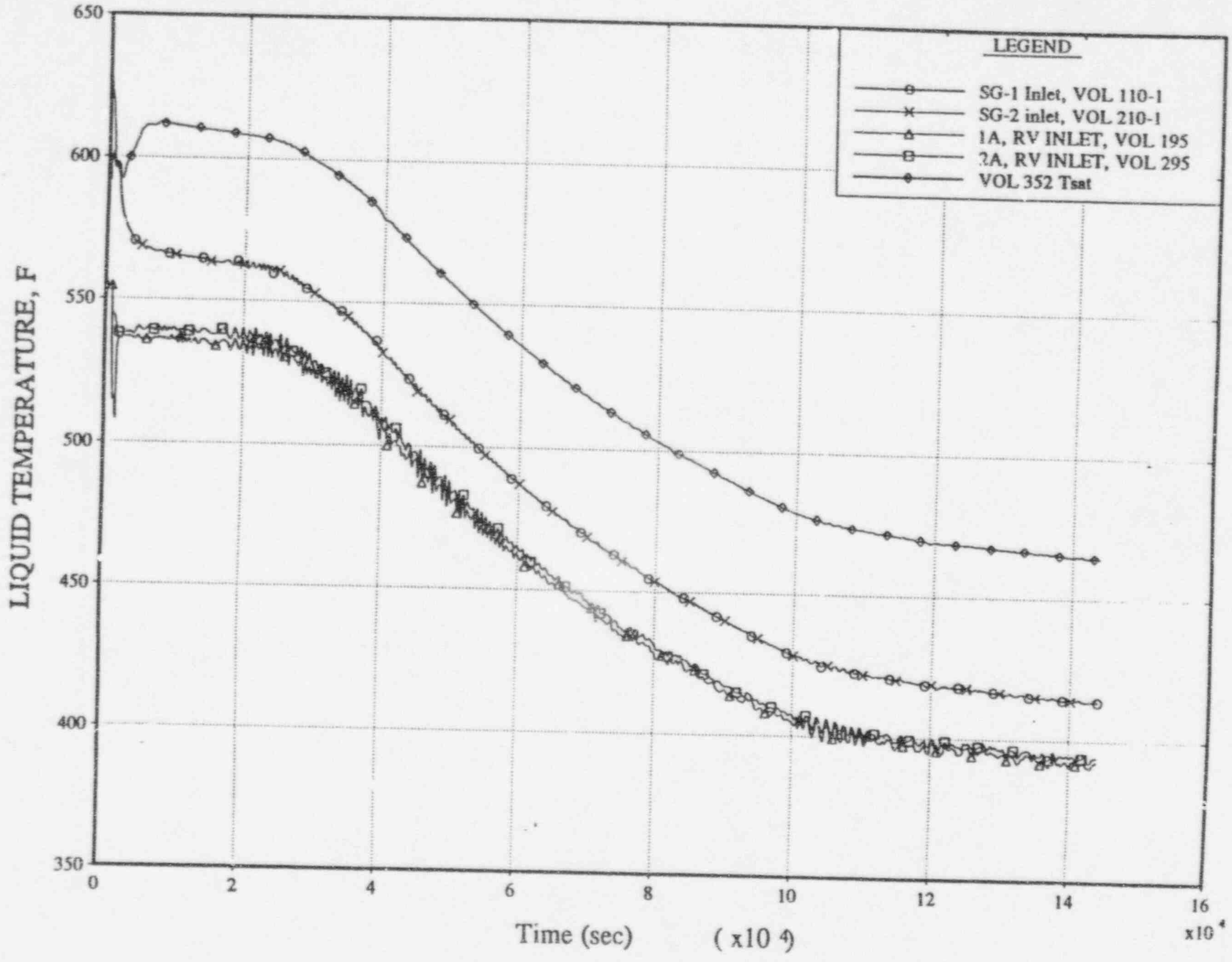

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FIGURE 20. CR-3 0.0014 ft^2 Break - Pressure Versus Time

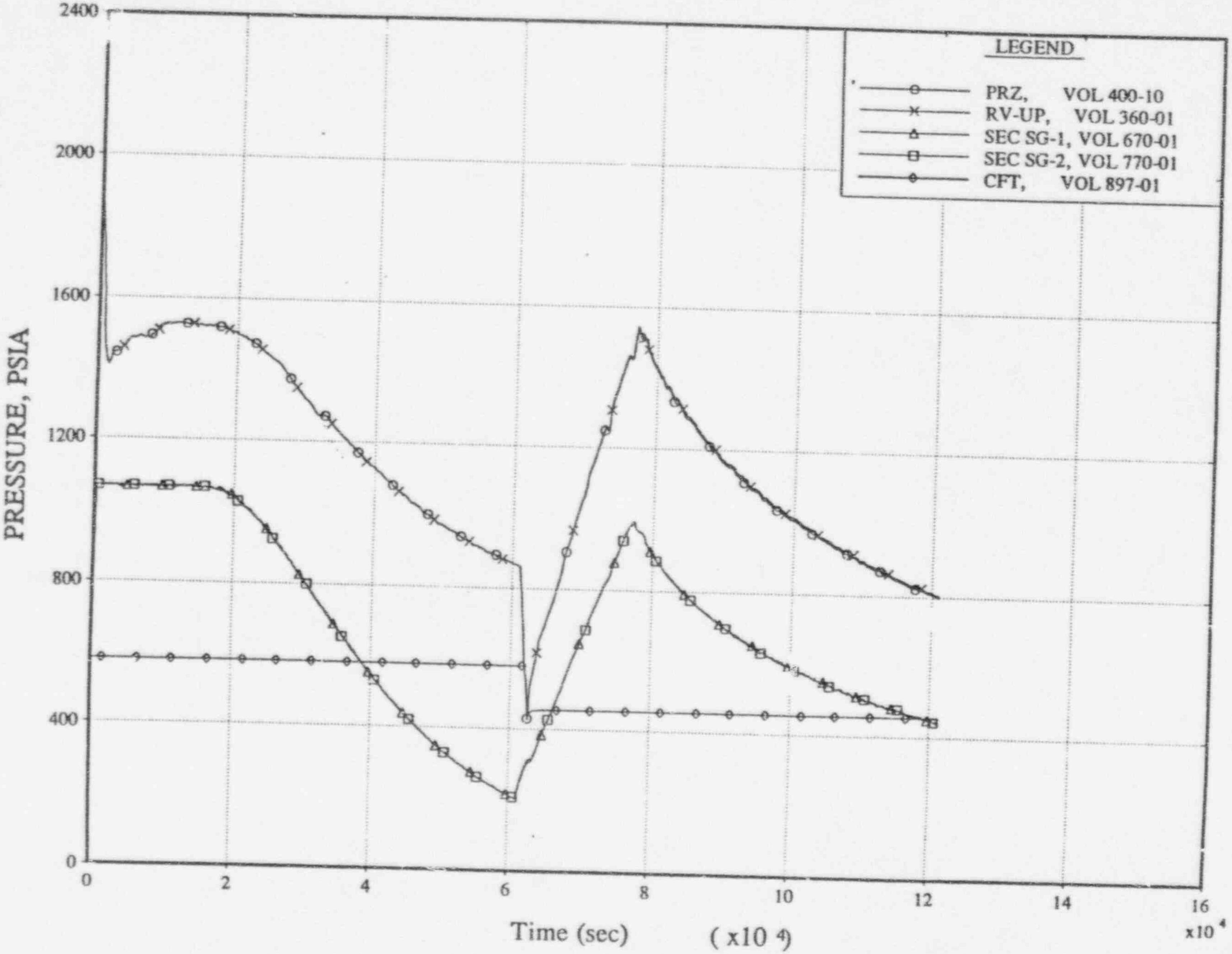

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FIGURE 21. CR-3 0.0035 ft^2 Break - Pressure Versus Time

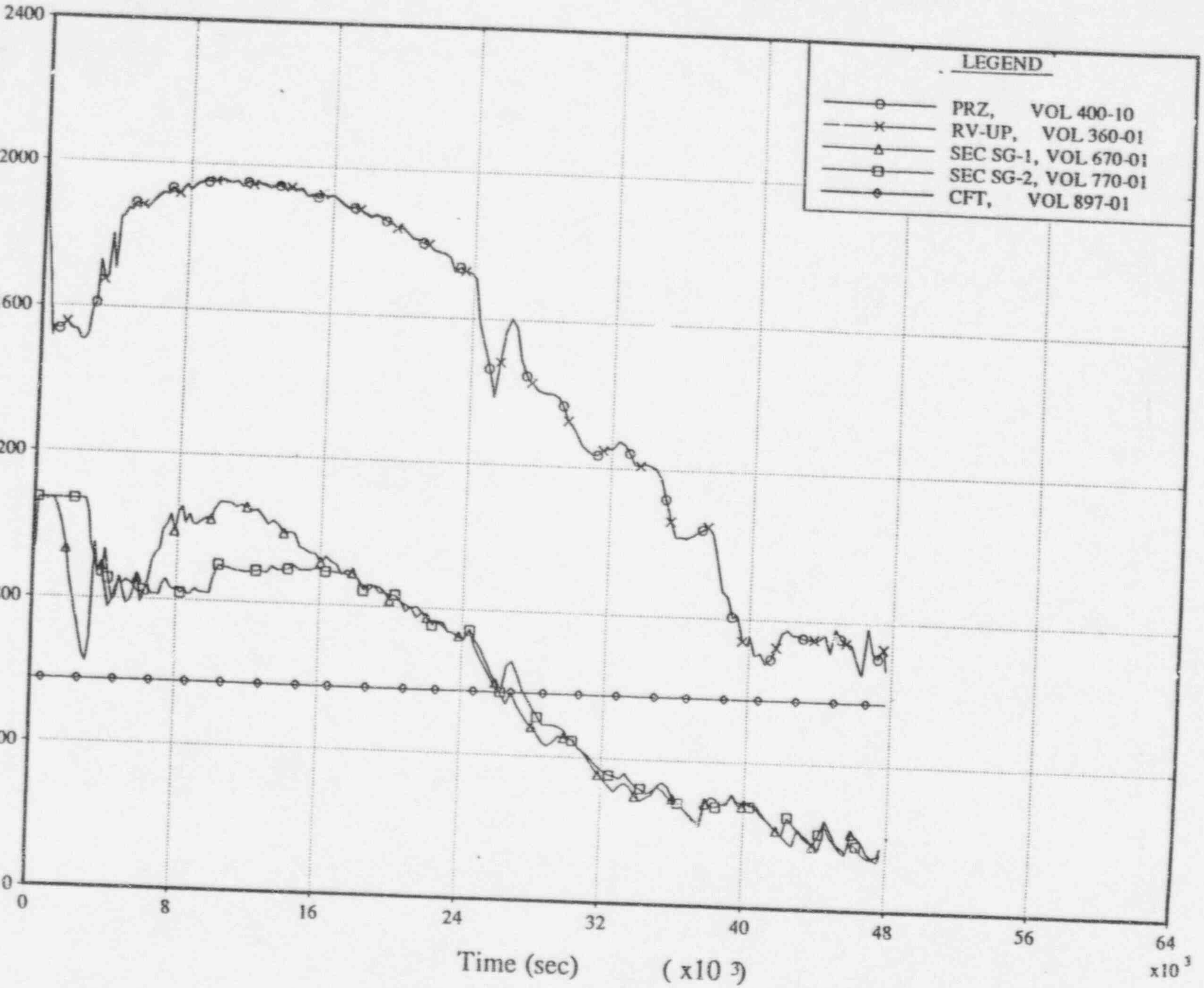

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FIGURE 22. CR-3 0.0025 ft^2 Break (2 HPI Pumps) - Pressure Versus Time

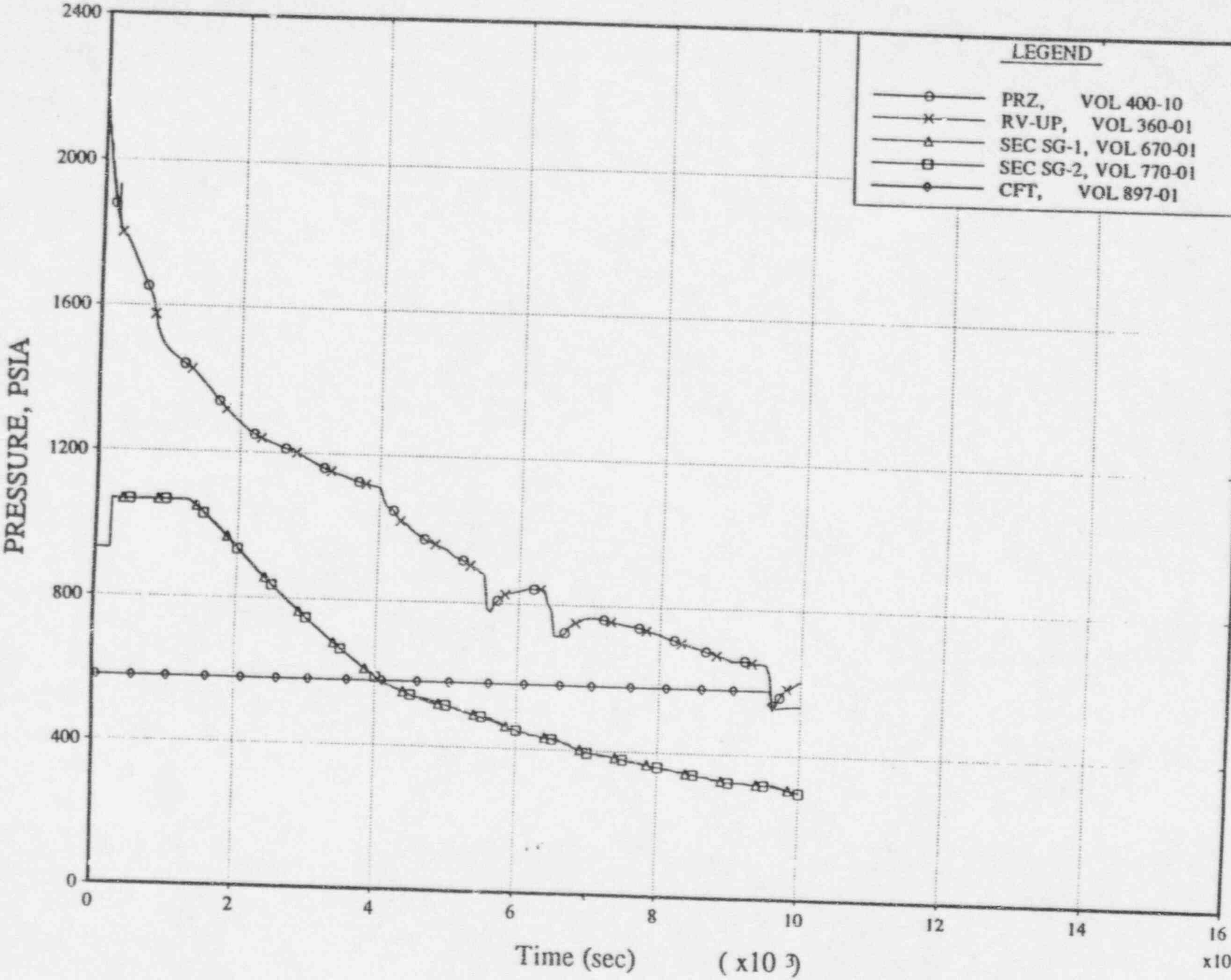

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FIGURE 23. CR-3 0.0025 ft^2 Break (2 HPI Pumps) - Temperature Versus Time

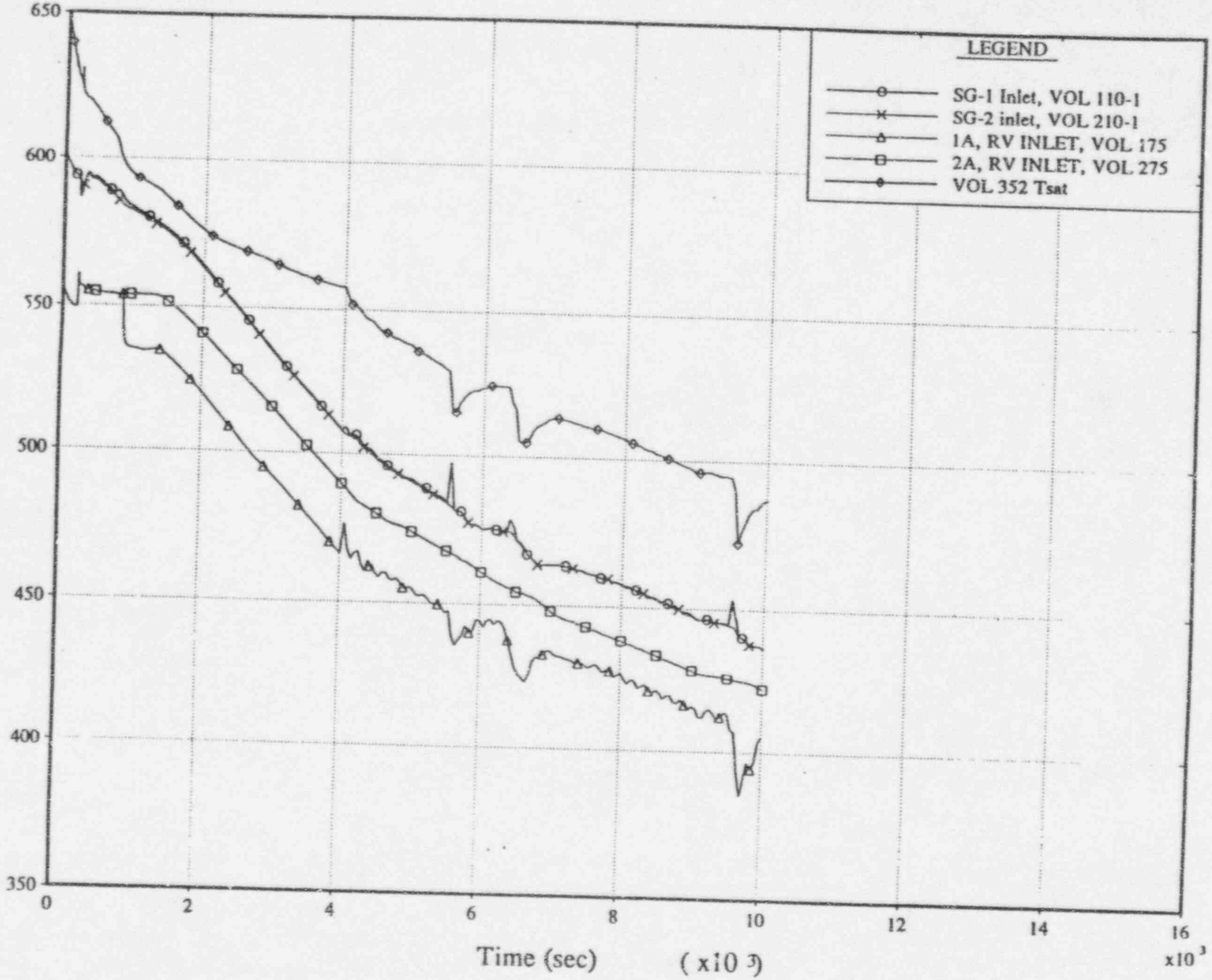


FIGURE 24. Total EFW Flow Required Post-SBLOCA in CLPD Region

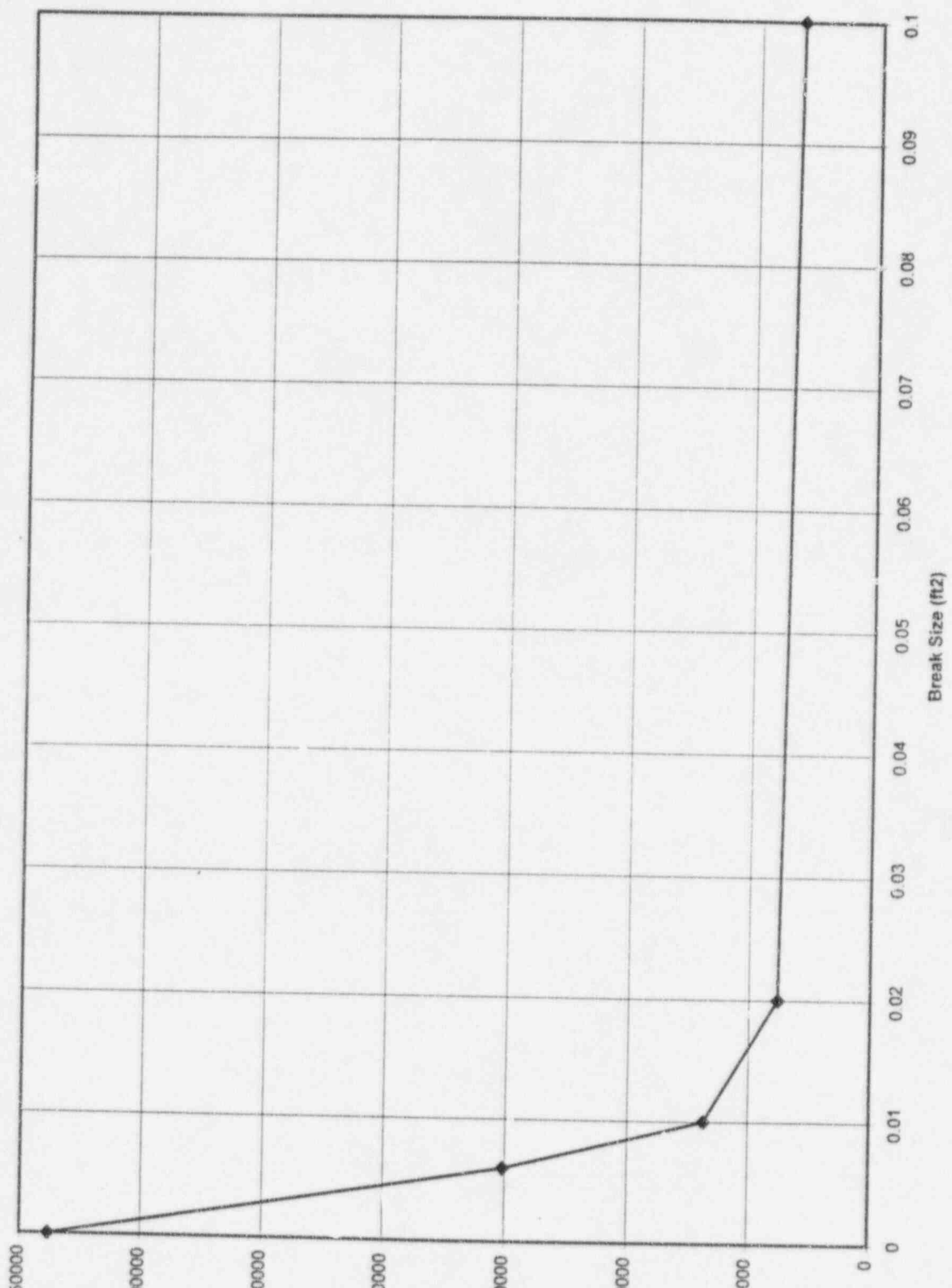
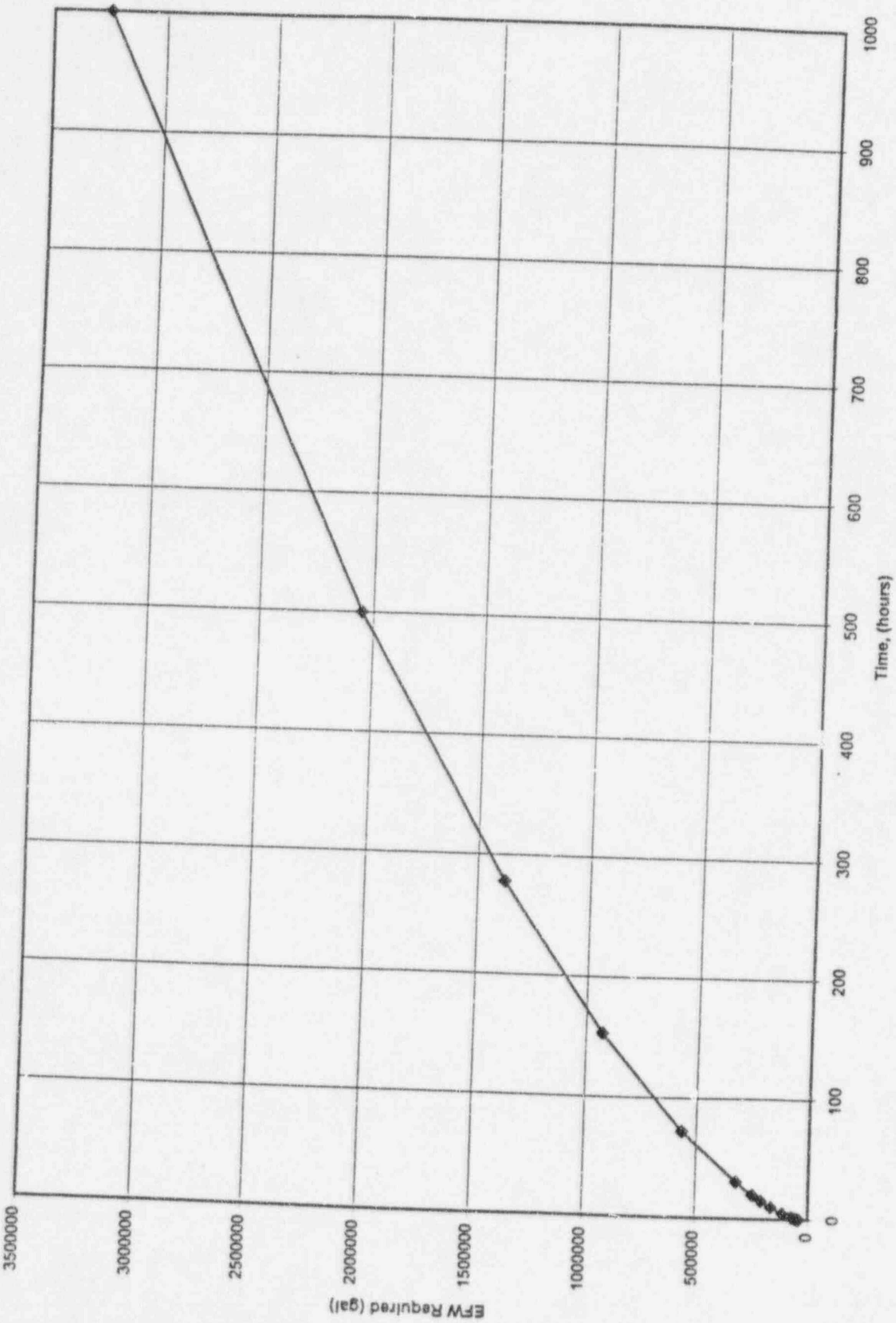


FIGURE 25. Total EFW Flow Required for Core Decay Heat Removal for the "Zero" Break SBLOCA



FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT E

ASSESSMENT OF LIMITED USE OF RELAP5

FRAMATOME
TECHNOLOGIES

Integrated Nuclear Services

June 11, 1997
INS-97-2323

Mr. F. X. Sullivan (NA1E)
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Crystal River Energy Complex
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Crystal River, FL 34428-6708

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J. C. Seals/OF53
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J. J. Cudlin/OF53
T. G. Stack/OF54
B. L. Brooks/OF54
Record Center NSS7/T1.2

Attention: Mr. D. F. Kunsemiller (SA2D)
Mr. D. Rice

Subject: Assessment of Limited Use of RELAP5

Reference: FPC Contract NPM010AD, WA73, Task 4100929 - Safety Analysis Support

Dear Mr. Sullivan:

At the request of Mr. D. F. Kunsemiller and Mr. D. Rice, FTI is providing an assessment of limited use of RELAP5. The assessment is attached.

The assessment finds that FPC has satisfied NRC criteria for reliance on BAW-10192P and finds RELAP5 as the licensing basis code to confirm EFW requirements for long-term cooling.

This work was performed under the referenced FPC Work Authorization. Should you have any questions, please give me a call at 804/832-2574.

Very truly yours,

L. M. Lesniak
Customer Service Manager

LML/bcc

Attachment

c: R. J. Finnin/OF57
R. W. Knoll/FPC (NT02)

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ASSESSMENT OF LIMITED USE OF RELAP5

I. Background

FTI performed engineering evaluations based upon RELAP5/MOD2 analyses to support the mission times for and operability of the turbine-driven emergency feedwater pump (EFP-2) during certain small break LOCA transients. These evaluations were used to demonstrate the RCS pressure and temperature responses and their relationship to EFP-2 operability. The RELAP5 analyses were performed with FTI's NRC-approved version of RELAP5/MOD2, and, with two exceptions, were done with the ECCS evaluation model (EM) as approved in BAW-10192-P. The EM adequately determines the transient evolution; however, its primary purpose is to calculate a conservative peak cladding temperature (PCT) following a postulated LOCA. The EM models and methods are developed to predict core uncovering and resulting PCTs for transients with typical end times of less than 2 hours. The EFP operability analyses examine very small breaks for durations of several days with abundant core cooling. Given that these analyses were not specifically performed to address core cooling, two adjustments were made to the EM input deck to facilitate code execution. These changes were (1) increases in the user requested maximum time step and (2) a switch from equilibrium to non-equilibrium control volumes in the heated core region. The approved evaluation model requires a small time step and equilibrium core volumes for SBLOCA core uncovering and PCT analyses. These changes will not affect the RCS or secondary-side pressure or temperature evolution as demonstrated in previous EM sensitivity studies.

II. NRC Topical Report BAW-10192-P

The NRC documented its review of the subject topical report (dated February 1994) along with Framatome's May 6, 1996; October 11, 1996; and January 7, 1997 responses to NRC requests for additional information in a February 16, 1997 letter to Framatome.[1] On the basis of its review, the NRC concluded that BAW-10192 (hereinafter, "the Topical") is acceptable for referencing in licensing applications in the analysis of LOCA accidents for once-through steam generator plants. The NRC also noted that when the report is referenced in a license application (as exists in the subject case), it will not repeat its review of matters described in the report and found acceptable.

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[1] See letter from James E. Lyons, Acting Chief, Reactor Systems Branch, NRC to J.H. Taylor, Manager, Licensing Services, Framatome Technologies Inc.; "Acceptance for Referencing of Topical Report BAW-10192-P, 'Loss-of-Coolant Accident Evaluation Model For Once-Through Steam Generator Plants' (TAC No. M89400)".

Enclosure 1 of the NRC's February 18, 1997, letter states, in part, that use of the Topical methodology for reference in licensing applications involving large and small break LOCA analysis for B&W plants is acceptable, subject to eleven conditions. Accordingly, justifying use of the Topical in licensee actions by comparing Code use to the eleven conditions is necessary for plant specific utilization of the Topical. In that regard, this text addresses those eleven conditions and demonstrates that use of the Topical in support of this license amendment is acceptable.

III. <u>NRC Conditions for Licensee Use of BAW-10192-P</u>

FTI provides below, its evaluation of the eleven conditions that must be satisfied for a licensee to justify use of the Topical:

1. *The LOCA methodology should include any NRC restrictions placed on the individual codes used in the evaluation model.*

   <u>Response</u>

   FPC (with Framatome being the implementer of the Code) has satisfied all NRC restrictions placed on the use of RELAP5 as defined in the evaluation model presented by Framatome to the NRC in its letters dated February 1994 and as supplemented in correspondence dated May 6, 1996 except for the two input options listed in Item I and described in detail in Item III.2 below.

2. *The guidelines, code options, and prescribed input specified in Tables 9-1 and 9-2 in both Volume I and Volume II of BAW-10192P should be used in LBLOCA and SBLOCA evaluation model applications, respectively.*

   <u>Response</u>

   Given that these analyses were demonstration cases used for proof of principle, and that no core uncovering would be predicted for the break sizes analyzed, an increase in the maximum time step is acceptable so long as the code convergence is maintained. The increased time step size is required to reduce the clock time required for analyses that have end times in excess of 30 reactor hours. The code convergence was maintained by keeping the time step increase within the confines of sensitivity studies and benchmark analyses used to validate the RELAP5/MOD2 code for EM application. The maximum time step used in these analyses was increased from the typical EM analysis value of 20 milliseconds per time step to 50 or 100 milliseconds per time step in these analyses. EM sensitivity studies performed in Reference 1 evaluated the variations on results with time step sizes ranging between 10 and 50 milliseconds. The RCS pressure behavior was nearly identical for all of the cases. Typical MIST SBLOCA benchmark analyses (Ref. 2) successfully used 200 to 250 millisecond time steps. Therefore, the time steps used for these SBLOCA cases are well within justified ranges for acceptable analytical convergence of the RCS evolution and its relationship to EFP-2 operability.

The switch from equilibrium to non-equilibrium in the core was made to facilitate code execution during long-term quasi-steady RCS pressure periods. Previous EM analyses for HPI line break cases have encountered execution difficulties with code water property searches during these periods. The difficulty has been traced to the generation of vapor in the core by the subcooled boiling wall heat transfer model and subsequent condensation by the high interphasic heat transfer coefficients that are forced by the application of the equilibrium control volume option. The code failures can be overcome by reductions in the maximum time step; however, this simply increases the run times without altering the calculational results. Abundant fuel pin cooling is predicted during subcooled or saturated nucleate boiling, and the non-equilibrium switch will not alter this conclusion or affect the RCS pressure behavior. The non-equilibrium volumes can alter the state conditions used for the EM heat transfer package at very high void fractions in the post-CHF regime. The transition and film boiling heat transfer correlations included in this package were formulated based on thermo-dynamic equilibrium conditions. Use of non-equilibrium control volumes has been shown to be reasonable in sensitivity studies, but its use is not totally consistent with the heat transfer package formulation. The applicability of core heat transfer package using the non-equilibrium formulation for the EFP-2 operability cases is supported by these previous studies and even more so by the fact that the post-CHF heat transfer regimes will not be predicted during any of these analyses since there is no core uncovering. If core uncovering and heatup were to be predicted, the equilibrium formulation must be used in all control volumes that predict a transition or film boiling regime.

3. *The limiting linear heat rate for LOCA limits is determined by the power level and the product of the axial and radial peaking factors. An appropriate axial peaking factor for use in determining LOCA limits is one that is representative of the fuel and core design and that may occur over the core lifetime. The radial peaking factor is then set to obtain the limiting linear heat rate. For this demonstration, calculations were performed with axial peak of 1.7. The general approach is acceptable for demonstrating the LOCA limits methodology. However, as future fuel or core designs evolve, the basic approaches that were used to establish these conclusions may change. FTI must revalidate the acceptability of the evaluation model peaking methods if: (1) significant changes are found in the core elevation at which the minimum core LOCA margin is predicted or (2) the core maneuvering analyses radial and axial peaks that approach the LOCA LHR limits differ appreciably from those used to demonstrate Appendix K compliance.*

   Response

   The analyses performed did not relate to core kw/ft limits, so this restriction does not apply.

4. *The mechanistic ECCS bypass model is acceptable for cold leg transition (0.75 ft^2 to 2.0 ft^2) and hot leg break calculations. The nonmechanistic ECCS bypass model must be used in the large cold leg break ($\geq$ 2.0 ft^2) methodology since the demonstration calculations and sensitivities were run with this model.*

Response

For SBLOCA analyses, the ECCS bypass model is typically not executed. ECCS bypass is a phenomenon applicable to blowdown during a large break of the reactor coolant piping in the reactor coolant pump leg. Therefore, this restriction does not apply to the analyses referenced in this report.

5. *Time-in-life LOCA limits must be determined with, or shown to be bounded by a specific application of the NRC-approval evaluation model.*

Response

No LOCA limits were calculated; therefore, the restriction does not apply.

6. *LOCA limits for three pump operation must be established for each class of plants by application of the methodology described in this report. An acceptable approach is to demonstrate that three pump operation is bounded by four pump LHR limits.*

Response

No LOCA limits were calculated, therefore; the restriction does not apply.

7. *The limiting ECCS configuration, including minimum versus maximum ECCS must be determined for each plant or class of plants using this methodology.*

Response

This restriction applies to analyses of large break LOCAs. The use of minimum flow rates is *per se* the limiting case for SBLOCA analyses. In the analyses done to support the evaluations in this document limiting ECCS flow rates were assumed.

8. *For the small break model, the hot channel radial peaking factor to be used should correspond to that of the hottest rod in the core, and not to the radial peaking factor of the 12 hottest bundles.*

Response

The hot channel radial peaking factor used corresponded to the hottest rod in the core. Since no core heatup calculations were performed, this restriction is not relevant.

9. *The constant discharge coefficient model (discharge coefficient = 1.0) referred to as the "High or Low Break Voiding Normalized Value," should be used for all small break analyses. The model which changes the discharge coefficient as a function of void fraction, i.e., the "Intermediate Break Voiding Normalized Value", should not be used unless the transient is analyzed with both discharge models and the intermediate void method produces the more conservative results.*

Response

The constant discharge coefficient model (discharge coefficient = 1.0) referred in the BAW-10192P analysis as the "High or Low Break Voiding Normalized Value," was used for the SBLOCA analyses performed to support the evaluations in this document.

10. *For a specific application of the FTI small break LOCA methodology, the break size which yields the local maximum PCT must be identified. In light of the different possible behaviors of the local maximum, FTI should justify its choice of break sizes in each application to assure that either there is no local maximum or the size yielding the maximum local PCT has been found. Break sizes down to 0.01 ft^2 should be considered.*

Response

The analyses done to support the evaluations reported in this document were not done to predict limiting PCT consequences for SBLOCA, therefore the requirement does not apply. The break sizes analyzed were smaller than those specified in the requirement.

11. *B&W-designed plants have internal reactor vessel vent valves (RVVVs) that provide a path for core steam venting directly to the cold legs. The BWNT LOCA evaluation model credits the RVVV steam flow with the loop steam venting for LBLOCA analyses. The possibility exists for a cold leg pump suction seal to clear during blowdown and then reform during reflood before the evaluation model analyses predict average core quench. Since the REFLOOD3B code cannot predict this reformation of the loop seal, FTI is required to run the RELAP5/MOD2-B&W system model until the whole core quench, to confirm that the loop seal does not reform. This documentation should be performed at least once for each plant type (raised loop and lowered loop) and be judged applicable for all LBLOCA break sizes.*

Response

This restriction applies to post-blowdown RCS behavior following a large break LOCA. None of the cases analyzed to support the evaluations in this document were for break sizes sufficient to produce formation of a cold leg pump suction seal. Therefore, the restriction does not apply.

IV. Interface of RELAP5 Use and CRAFT/THETA (FSAR Chapter 14)

FPC has evaluated whether the use of RELAP5 for SBLOCA analyses is compatible with the use of CRAFT/THETA in the CR-3 FSAR Chapter 14 accident analyses. In sum, FPC concludes that no interface problems exist as a result of the use of the two codes because (1) the RELAP5 code has been reviewed and accepted by the NRC for application to SBLOCA transients, and (2) the analyses performed for this application do not supplant or alter any of the SBLOCA analyses reported in Chapter 14 of the CR3 FSAR.

V. Licensing Basis

Based on the above discussions, FPC has satisfied the NRC's criteria for reliance on BAW-10192-P in this license amendment. Accordingly, for purposes of SBLOCA analyses to confirm EFW requirements for long-term cooling, FPC considers the licensing basis code to be RELAP5 (BAW-10192-P) as approved by the NRC in its letter dated February 18, 1997.

References

1. BWNT Document 32-1224876-00, "EM SBLOCA Sensitivity Studies I," March 1, 1994.

2. Klingenfus, J.A. & Parece, M.V., "Multiloop Integral System Test (MIST): Final Report," NUREG/CR-5395, EPRI/NP-6480, BAW-2078 (Vol. 10), December 1989.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT F

SUPPORTING INFORMATION

ATTACHMENT F

SUPPORTING INFORMATION

As discussed in the cover letter, Florida Power Corporation (FPC) is addressing the issues related to small break loss of coolant accident (SBLOCA) mitigation for Crystal River Unit 3 (CR-3) through a combination of Technical Specification revisions, Emergency Operating Procedure (EOP) revisions, and plant modifications. This integrated approach ensures that necessary accident mitigation systems are available and maximizes defense in depth. To support NRC's review of this submittal, the following listed information is provided in this attachment.

Table	Title*
1	NRC Identified Unreviewed Safety Questions (USQ)
2	Modifications (MOD)
3	Operator Actions (OA)
4	Summary of Planned FSAR Changes
5	Related LERs (LER)

*Letters in parentheses indicate cross-reference designations used for item references to other tables in this attachment.

1. NRC Identified Unreviewed Safety Questions

This table lists those issues that have been previously identified by the NRC as containing Unreviewed Safety Questions (USQ). The resolution of these NRC Identified USQs along with references to other tables in this attachment are provided.

2. Modifications

This table provides a listing of the modifications to be performed during this outage related to SBLOCA mitigation issues. These modifications are currently in various stages of implementation and, therefore, the information provided in this table is draft. Implementation of these modifications will support the Technical Specification/Bases changes in Attachment C, resolve issues identified as USQs, and resolve outstanding issues related to SBLOCA mitigation.

3. Operator Actions

To support the NRC's review of the TSCRN 210, two tables are provided to identify certain operator actions required for the mitigation of SBLOCAs as presented in Attachment B. However, the EOPs affected by this TSCRN are in various stages of revision and, therefore, the information provided in these tables are draft. Table 3A provides a complete list of operator actions relied upon in the first 20 minutes for these SBLOCA scenarios. Table 3B identifies the additional operator actions, beyond those

currently in the operating procedures, required after the first 20 minutes of these SBLOCA scenarios.

Not all of the operator actions listed in this table would be required for all SBLOCAs. As indicated in the table, some of these actions would only be required to be implemented under certain low probability scenarios. Additionally, some of these actions are considered to be "defense in depth" and are not considered in design basis mitigation analyses.

FPC has been able to reduce the number of operator actions required in the first 20 minutes of these SBLOCA scenarios relative to the previous requirements. Specifically, one operator (OA2) will address more than one previously required operator action. These operator actions in conjunction with the proposed Technical Specification changes and plant modifications have been taken in an effort to minimize the operator burden during the response to a SBLOCA.

FPC requests specific NRC review of the operator actions presented in Tables 3A and 3B as an integral part of the amendment review.

4. **Summary of Planned FSAR Changes**

This table is a summary of planned changes to the CR-3 FSAR based on the modifications and changes to the SBLOCA analysis as provided in this submittal. The information in this table is draft and based on the most recent information available for the modifications scheduled to be completed this outage to support the SBLOCA analysis. The finalized changes to the FSAR will be completed via the FSAR change program. As stated in the cover letter, these changes will be addressed prior to restart.

5. **Related LERs**

This table lists those events related to this submittal that have been previously reported to the NRC as voluntary reports and unanalyzed conditions. The descriptions provide a reference to previously reported issues and refer to resolutions contained within this TSCRN.

Table 1
NRC Identified Unreviewed Safety Questions

<table>
<tr><th>USQ</th><th colspan="2">USQ Description</th><th>References</th><th>Resolution</th></tr>
<tr><td rowspan="3">1</td><td rowspan="3"><u>EDG Loading (ASV-204 Mod)</u>
The implementation of MAR 96-04-12-01 (removal of EFIC automatic open signal for ASV-204) prevented the automatic start of EFP-2 on a failure of the B train vital DC power, thus preventing a NPSH concern for EFP-1. However, EFP-1 would need to pump more water and thus, if there were a concurrent LOOP, would represent a larger load on the A EDG.
[EEI 96-12-02 - first example]</td><td>A. This modification increased the potential loading of the A EDG such that the design load limit of 3500 kW (TS Basis 3.8.1; FSAR 8.2.3) would be exceeded for short periods of one to three seconds during certain EDG block loadings.</td><td rowspan="3">EEI 96-12-02

LER1</td><td rowspan="3"><u>Modifications</u>
MOD1, MOD5, MOD6, MOD7, MOD8, MOD9, MOD10

<u>TS/Bases Changes</u>

Parts 2, 3</td></tr>
<tr><td>B. This modification increased the automatically connected accident load at the one-minute interval to 3159 kW in excess of the minimum test load specified by TS SR 3.8.1.11. The TS Basis stated that the minimum load of 3100 kW provides margin above the predicted worst-case automatically connected accident load at one minute.</td></tr>
<tr><td>C. This modification increased the motor-driven EFW pump load to 666 kW which exceeded the TS SR Basis 3.8.1.8 statement that the largest single post-accident load (that the A EDG would have to reject) was 616 kW.</td></tr>
</table>

Table 1
NRC Identified Unreviewed Safety Questions

USQ	USQ Description		References	Resolution
2	<u>EDG Loading (EOP-13 Revision)</u> The procedure change for EOP-13 increased the motor-driven EFW pump post-accident load from 666 kW to 713 kW by directing operators to take manual control and increase EFW flow. The resulting EDG load was greater than that calculated to support MAR 96-04-12-01 and, therefore, the same three USQs as discussed above were introduced. [EEI 96-12-02, second example]	A. Design load limit of 3500 kW (TS Basis 3.8.1; FSAR 8.2.3). B. Automatically connected accident load at one-minute exceed 3100 kW (TS SR 3.8.1.11) C. Largest post-accident load for reject (TS SR Basis 3.8.1.8)	EEI 96-12-02 LER1	<u>Modifications</u> MOD5, MOD6, MOD7, MOD8, MOD9 <u>TS/Bases Changes</u> Parts 2,3
3	<u>EDG Loading (OP-402 Revision)</u> The procedure change for OP-402 increased the post-accident HPI pump load on the A EDG by 75 kW and on the B EDG by 86 kW by allowing operators to ES select the swing B HPI pump to either EDG. The post-accident load of the B HPI pump (691 kW) exceeded the TS Basis value of the largest single post-accident load (616 kW). [EEI 96-12-02, third example]		EEI 96-12-02 LER1	<u>Modifications</u> MOD7, MOD8, MOD9 <u>TS/Bases Changes</u> Part 3

Table 1
NRC Identified Unreviewed Safety Questions

USQ	USQ Description	References	Resolution
4	EFW NPSH TMAR T87-10-09-01 and MAR 96-04-12-01 (removal of EFIC automatic open signal for ASV-204) increased the probability of a malfunction of equipment important to safety (damage to EFP-2 due to insufficient NPSH). Additionally, the plant design basis relied upon EFP-2 to share the EFW flow with EFP-1 (EFV-12 crosstie) in order to maintain the A EDG within its loading limits. The safety evaluations failed to identify this potential USQ. [EEI 96-19-03]	EEI 96-19-03 LER2, LER3	Modifications MOD1, MOD2, MOD3 MOD4 TS/Bases Changes Part 1
5	ASV-204 Automatic Open Signal Removal Removal of the automatic open signal from ASV-204 disabled one of the two automatic steam supplies to EFP-2. This reduced the reliability of EFP-2, which in turn, increased the probability of a failure of EFP-2. This was an increase in the probability of occurrence or malfunction of equipment previously evaluated in the FSAR. [EEI 96-19-06]	EEI 96-19-06 LER 1, LER2	Modifications MOD1 TS/Bases Changes Part 1

Table 1
NRC Identified Unreviewed Safety Questions

USQ	USQ Description	References	Resolution
6	Operator Actions Added and revised certain operator actions associated with the mitigation of small break LOCA [97-06-01]	NRC Inspection Report 50-302/97-06	FPC's response to the Violation has not been prepared at this time. Those operator actions addressed by the Violation are included in this submittal and FPC requests NRC review of these operator actions as an integral part of the amendment review.

Table 2
Modifications

MOD	MAR	Subject	Description	Reference
1	96-11-01-01	ASV-204 EFIC Auto opening reinstallation	Restores the automatic opening of ASV-204, the steam admission valve to EFP-2, on an "A" EFIC actuation. This will restore the load sharing capability of the Emergency Feedwater System for the LOCA concurrent with LOOP and loss of EDG-1B in order to reduce the load on EDG-1A	USQ1, USQ4, USQ5, LER1, LER2, LER3, LER4
2	96-10-02-01	Emergency Feedwater Cavitating Venturis	Installs passive flow restricting devices on the discharge side of both EFP-1 and EFP-2. This will prevent excessive pump flow resulting in possible failure mechanisms of runout or inadequate NPSH available.	USQ4, LER2, LER3, LER4
3	96-10-10-01 96-10-10-02 96-10-10-03	EFV-12 Valve Mods, MOV Installation, Conduit Supports	Replace valve EFV-12 on the cross-tie piping between EFW train A and train B, a manual operated gate valve, with a motor operated gate valve. This will facilitate operator action to open this valve remotely and route discharge of EFP-2 through the cross-tie piping to the OTSGs.	USQ4, LER2, LER3, LER4
4	97-01-04-01	EFP-2 Flow Indications	Installs flow indication from the cavitating venturis installed downstream of EFP-2. This control room indication of EFP-2 flow rate will be powered from the opposite train ('A' side) to provide flow indication should a 'B' side failure disable its flow indication. This will provide feedback to the operator of flow from EFP-2 when EFP-1 needs to be secured for EDG load management.	USQ4, LER2, LER3, LER4
5	97-04-01-01	EFP-1 500 psig Trip Defeat Switch	Installs a control switch to allow operator action to defeat the automatic trip of EFP-1 (500 psig RCS pressure). Defeating this trip will allow EFP-1 operation during a SBLOCA. This switch will allow continued EFP-1 operation when DHP-1A starts on a 500 psig actuation, after EDG-1A load management by operator action.	USQ1, USQ2, LER1, LER4
6	97-04-02-01	RW/SW Pumps Pull-To-Lock Switches	Replaces existing control switches with a Pull-To-Lock switch on Nuclear Service and Decay Heat Seawater pumps RWP-2A and -2B and Nuclear Services Closed Cycle Cooling pumps SWP-1A and -1B. This will prevent automatic restart of these pumps on subsequent Engineered Safeguards actuation signal facilitating EDG-1A load management.	USQ1, USQ2, LER1, LER4

Table 2
Modifications

MOD	MAR	Subject	Description	Reference
7	96-12-17-01	EDG Small Load Reduction Modifications, DOP 2A/2B	This modification will remove the auto-start function from both nonsafety control circuits of the Flush Water Pumps. This will prevent them from auto-loading onto the EDGs.	USQ1, USQ2, USQ3, LER1, LER4
8	96-10-05-01	Diesel Power Uprate Project	Implements modifications to increase the service ratings of the EDGs. (1) The combustion air flow rate will be increased by replacing nozzle rings in turbochargers with larger ones, and (2) combustion air intercoolers will be replaced with a dual pass intercooler.	USQ1, USQ2, USQ3, LER1, LER4
9	96-03-12-01 & associated FCN's	Emergency Diesel Generator Indication Upgrade	Installs more accurate power meters (kW indication) for EDGs-1A and -1B. Accuracy was further improved by changes to CT/PT's. EDGs can be loaded higher because of improved instrument accuracy.	USQ1, USQ2, USQ3, LER1, LER4
10	96-06-02-01	EFIC Integral Windup Reset	Installs windup reset on integral controller on the EFIC system. This will provide for faster response of EFW for control of flow to the OTSGs. This reduces EFW flow and consequential EDG-1A loading upon initiation.	USQ1, LER1, LER4
11	97-03-01-01	Standby Generator for FWP-7	Installs a new diesel generator (not safety-related) to provide an alternate backup power supply for FWP-7.	
12	97-02-17-01	MUV-27 HPI Autoclosure	Changes the Engineered Safeguards automatic actuation logic for the normal Makeup supply valve MUV-27 to add automatic closure upon receipt of a diverse containment isolation signal (which also initiates HPI). The purpose of the modification is to aid in HPI flow balancing actions in the event of a broken HPI line. MUV-27 must be closed to help ensure accurate HPI flow indication.	

Table 3A
Operator Actions Less Than 20 Minutes

O A	Operator Action	Assumed in Design Analysis	Time	Basis	Reference
1	Trip all running RCPs	Yes	< 2 minutes	Required for loss of subcooling margin based on voiding condition of reactor coolant.	NRC letter to FPC dated 5/29/86 (Generic Letter 86-05) refers to B&W Owners Group (BWOG) studies which concluded that compliance with 10 CFR 50.46 is achieved if operator action to trip RCPs is taken within 2 minutes.

Table 3A

Operator Actions Less Than 20 Minutes

OA	Operator Action	Assumed in Design Analysis	Time	Basis	Reference
2	If Subcooling Margin (SCM) is lost and ES has not actuated, initiate manual HPI and Reactor Building Isolation and Cooling (RBIC) - isolates letdown (USQ6) - initiates HPI flow - isolates normal makeup (USQ6) (contingency actions are provided in OA 4 within 20 minutes if power is not available) - isolates RCP seal control bleed off valves - actuates EFIC - initiates Emergency RB cooling	Yes (isolation of letdown and makeup were previously identified as separate actions)	< 10 minutes	Required for loss of subcooling margin (precedes automatic initiation)	NRC letter to FPC dated 7/6/79 (SER for Order dated 5/16/79 based on TMI-2 Accident) recognizes CR-3 revision to Emergency Procedure EP-106, which defines operator action in response to a spectrum of break sizes. States EP-106 was "judged to provide adequate guidance to the operators to cope with small break LOCA." EP-106 (currently EOP-03, "Loss of Subcooling Margin") contained guidance to initiate HPI and ensure adequate HPI flow.

Table 3A

Operator Actions Less Than 20 Minutes

OA	Operator Action	Assumed in Design Analysis	Time	Basis	Reference
3	Ensure all four HPI injection valves are open - switch power supply for affected injection valves by manipulating switches in control room	Yes	< 10 minutes	Required only for loss of 1 train of Class 1E power	NRC letter to FPC dated 5/29/79 "Permanent Solution to SBLOCA Issue" recognizes operator action to turn associated transfer switc[illegible] open affected HPI valves by 10 minutes.

Table 3A
Operator Actions Less Than 20 Minutes

OA	Operator Action	Assumed in Design Analysis	Time	Basis	Reference
4	Isolate RCP seal injection (USQ6) (As a contingency action, if power is lost to MUV-27 (normal makeup) and MUV-18 (RCP seal injection), transfer to an energized bus and close valves)	Yes	< 20 minutes	Required to maximize HPI flow to reactor	FPC letter to NRC dated 2/28/79, answers a previous question of whether or not it was necessary to isolate any flow paths in the makeup system after a LOCA. FPC refers to RCP seal injection and normal makeup and refers to a Gilbert Associates report that concludes adequate HPI flow is achieved without these lines isolated. NRC letter to licensees with B&W designed systems (Generic Letter 86-05) dated 5/29/86 states the cooling water sources supporting the RCP with the potential of being isolated are seal injection, seal bleedoff, component cooling water to seal line coolers, and component cooling water to RCP motors and oil coolers. The need to isolate RCP Seal Injection was discovered in 1995 to be necessary due to discovery that operators relied on non-Reg Guide 1.97 instrumentation to measure this flow when determining HPI pump runout flow limits (see LER 95-026). Seal injection isolation was also determined necessary during Refuel 10 in 1996 upon discovery that worst case instrument error may result in inadequate HPI flow (see LER 96-006).

Table 3A

Operator Actions Less Than 20 Minutes

OA	Operator Action	Assumed in Design Analysis	Time	Basis	Reference
5	Ensure adequate HPI flow (USQ6) (isolate a broken injection line using new isolation criteria)	Yes	< 20 minutes	Required only for break in HPI line	FPC letter to NRC dated 10/27/89 states HPI must be successfully balanced to support SBLOCA mitigation as described in various B&W topical reports accepted by NRC. Subsequent FPC letter dated 10/31/89 states that mitigation strategy employed from the late 1970's through reviews done in response to NUREG 0737 relied on balancing HPI flow for breaks in HPI injection lines. These letters relate to LER 89-037, issued in November 1989 reporting a design basis condition in which instrumentation used for balancing HPI flow was inadequate. NRC letter dated 12/20/89 confirmed verbal concurrence to resume power operation with the HPI instrumentation problems. One condition was operator action for HPI flow balancing. NRC letter dated 2/17/95 from Gary Holahan to Ed Jacks (BWOG Operator Support Committee) states staff has completed its review of BWOG response to NUREG 0737 Item I.C.1 regarding EOP Guidelines and is finalizing an SER on the topic. Balancing HPI flows was a part of the ATOG/TBD guidelines incorporated into FPC procedures. FPC issued LER 96-007 on 3/15/96 to report another design basis condition involving HPI flow instrumentation. The flow deficiencies described therein were addressed by revised SBLOCA analyses provided by Framatome Technologies in April 1996 which required isolation of the affected HPI line versus balancing. Most recent FTI analyses have provided new isolation criteria.

Table 3A
Operator Actions Less Than 20 Minutes

OA	Operator Action	Assumed in Design Analysis	Time	Basis	Reference
6	Ensure adequate EFW flow (USQ6) (EFIC was initiated in OA2; therefore, ensuring EFW flow is a confirmation step only) This step manually raises OTSG levels to the Inadequate Subcooling Margin, ISCM level	Yes	< 20 minutes	Raise OTSG levels to ISCM setpoint (95%)	B&W (Taylor) letter to NRC (Baer) dated 5/1/78 provides topical report 10104, "B&W's ECCS Evaluation Model," which notes operator action is necessary during early stages of the accident to mitigate consequences and meet 10 CFR 50.46. Auxiliary feedwater is assumed to be available. NRC letter to FPC dated 7/6/79 provides a SER for actions taken in response to Commission Order dated 5/16/79. The SER states that a generic review of B&W analyses entitled "Evaluation of Transient Behavior and Small RCS Breaks in the 177 Fuel Assembly Plant" resulted in a principle finding that reconfirms SBLOCA analyses demonstrate a combination of heat removal by the steam generator and the HPI system combined with operator action to ensure adequate core cooling. These results are applicable to CR-3 considering the ability to manually start the redundant EFW pumps and HPI pumps from the control room, assuming failure of automatic EFW actuation. NRC letter to FPC dated 8/30/85 provides a SER for NUREG 0737 Item II.K.3.30, "SBLOCA Methods." Section III 5.a of the SER states "the timing of operator action to raise the secondary system water level to 95% was found not to be critical."

Table 3B

New Operator Actions After 20 Minutes

OA	Operator Action	Failure Scenario	Cycle 11 Only	Basis
7	If 'B' DC power is lost, crosstie EFP-2 to A train (EFV-12) AND Secure EFP-1	LOBB	Yes	EFP-1 can only provide flow for a specific time period, then EFP-2 must be aligned.
8	If only EFP-2 is supplying feedwater to the OTSG, then there will be no operator initiated RCS cooldown whether or not offsite power is available. If other sources of feedwater are available, cooldown may be initiated. (Mitigation strategy includes operation of diesel backed FWP-7 as a Defense in Depth action)	LOBA	No	**Defense in depth** for possible decoupling of the OTSGs. Use of FWP-7 provides additional resources available to operators during a LOOP
		LOBB	No	
9	If EFP-2 is not operating when in a LOOP condition with inadequate subcooling, limit cooldown prior to the EFP-1/LPI Interlock	EFP-2	Yes	If EFP-2 is not available, steps must be taken to ensure EFP-1 operates as long as needed.
10	Periodically re-evaluate HPI line break criteria on RCS repressurization	LOBA	No	Required for specific HPI line pinch areas to ensure a broken line will be isolated if warranted
		LOBB	No	
		EFP-2	No	

Table 3B
New Operator Actions After 20 Minutes

OA	Operator Action	Failure Scenario	Cycle 11 Only	Basis
11	Manage operation of EFP-2 by closing ASV-5 and ASV-204 on low OTSG pressure(Cycle EFW).	LOBA LOBB	No Yes	For a LOBA, or a LOBB (to manage EDG load), EFP-1 would be secured and EFW flow would rely on EFW-2. EFP-2 would be cycled due to operational limitations on low OTSG pressures.
12	Put EFIC in manual permissive AND Close EFW block valves	LOBB	Yes	Required to prevent cycling of the limited duty motors on the EFW block valves. This action may be included in the EOPs for both trains of EFW.
13	Manage EDG load in order to extend EFP-1 operation by - • Shutdown SWP-1A & RWP-2A after verifying redundant pumps are operating and placing switches in Pull-to-Lock to prevent reactuation of pumps (EDG loading) • Place EFP-1 Trip Defeat Switch in defeat position to prevent automatic trip of EFP-1 on RCS pressure of 500 psig	EFP-2	Yes	**Defense in Depth action** for postulated single failure of the loss of EFP-2. These actions extend the time EFP-1 is available for OTSG cooling.

Table 4
Summary of Planned FSAR Changes

Section	Section Title	Summary of Planned Changes
6.0	Engineered Safeguards	(1) Add a discussion that certain size small break LOCAs require EFW to maintain primary to secondary cooling via the OTSG until reactor core decay heat can be removed solely by HPI and flow out the break.
6.1	Emergency Core Cooling System	(1) Add discussion that EFW is needed for some sizes of SBLOCA and clarify current discussion about break sizes. (2) Update discussions of RCS and HPI line breaks based on FPC's Safety Assessment. (3) In 6.1.2.1.1 add item d. to state that the normal makeup supply valve (MUV-27) is closed to facilitate accurate HPI flow indication for HPI flow. Also, add reference to the normal makeup supply valve (MUV-27) as being supplied by either of two channels of ES electrical buses. (4) Table 5-9, item no. 9, add "(A/B)" to MUV-27 and add function that diverse containment isolation signal isolates normal makeup from HPI line.
7.1	Protection Systems	In 7.1.3.2.3, delete reference to "Flush Water Pumps."
7.2	Control Systems	(1) In 7.2.4.2, reflect that trip module in "A" cabinet starts turbine-driven EFP-2. Add discussion that starting of both EFW pumps on "A" EFIC actuation is necessary to assure that EFP-2 will operate with failure of "B" DC system and loss of offsite power. EFP-2 is relied upon to share the EFW load with EFP-1 to decrease load on EDG-1A. Also, EFP-2 operation is necessary in SBLOCA with LOOP and loss of Battery "B". (2) Revise 7.2.4.1, EFIC "Design Bases" for flow rate control of EFW from 600 gpm to 550 gpm, and delete discussion of flow rate control of EFW when OTSG pressure is less than 600 psig. (3) Add discussion to 7.2.4.2 that cavitating venturis are to choke flow at approx. 750 gpm to avoid high flow problems that could occur if control valves fail open. (4) In Table 7-8A, add EFP-2, with 1 channel. (5) In 7.2.4.1, in number 3. add: "Manual control may be required under certain circumstances such as low decay heat rates and the transition between low and high range level instruments."
7.3	Instrumentation	(1) From section 7.3.5.2, in Table 7-12, add EFP-2 Flow for Cross-tie, Type and Category: D, 2.

Table 4
Summary of Planned FSAR Changes

Section	Section Title	Summary of Planned Changes
8.2	Electrical System Design	(1) In 8.2.3.1.3 revise 2000 hour rating from 2851-3000 kW to 2851-3200 kW, and revise 200 hour rating from 3001-3250 kW to 3201-3400 kW. Change the stated seven-day required volume of storage tanks from "maximum continuous rating of 2850 kW" to "200 hour rating of 3400 kW". Change the stated required volume of day tanks. Add discussion that EFP-2 will be automatically started by opening of ASV-204 and share the EFW load thereby reducing EDG loads, and that this is needed to keep EDG load within 2000 hour rating. (2) Update discussion of loads based on EDG load testing; (3) In Tables 8-1 and 8-2 delete Flush Water Pump DOP-2A as an auto connected load on EDG-1A. (4) Section 8.2.2.4, revise last sentence of first paragraph as follows: The other bus section, designated Reactor Auxiliary Bus 3, provides power for the Auxiliary Feedwater Pump (FWP-7) through transformer MTTR-5 or a Standby Diesel Generator (MTDG-1).
9.5	Cooling Water Systems	(1) PRELIMINARY: In 9.5.2.1.1 add discussion of the new "Pull-To-Lock" switches which replace the existing control switches for Nuclear Service and Decay Heat Seawater System (RW) pumps RWP-2A and RWP-2B, and Nuclear Services Closed Cycle Cooling System (SW) pumps SWP-1A and SWP-1B, and that this will allow the operator to block restart of SWP-1A and RWP-2A as part of EDG load management required as part of SBLOCA mitigation.

Table 4
Summary of Planned FSAR Changes

Section	Section Title	Summary of Planned Changes
10.5	Emergency Feedwater System	(1) In 10.5.1 add discussion of cavitating venturis EF-62-FO and EF-63-FO as being designed to provide 750 gpm flow. (2) In 10.5.2, discuss how recirculation lines provide EFW pump protection against low flow damage and cavitating venturis provide protection against high flow damage. Add discussion to 10.5.2 that the turbine driven EFP-2 is independent of AC power and starts by opening of either ASV-5 or ASV-204 when activated by EFIC. Also, add discussion of dependencies: (a) Starting of both EFW pumps on "A" train EFIC actuation is necessary to assure that EFP-2 will operate on failure of ES "B" DC system or EDG-1B coincident with loss of power, EFP-2 will be relied upon to share the EFW load with EFP-1 to decrease the load on EDG-1A. (b) Operation of EFP-2 is necessary in a small break LOCA with a LOOP and loss of "B" battery as the RCS cooling source to reduce RCS pressure from 500 psig to 200 psig where LPI cooling is effective. (3) Add section 10.5.2.7, "EFW Cavitating Venturis" to discuss details about the cavitating venturis. The venturis, installed in the discharge from each pump, are designed to choke EFW flow into the OTSGs at 750 gpm and protect against high flow damage due to pump runout and low NPSH. The venturis are a passive protection feature in the event any control valves fail open. The venturis protect the OTSG tubes from flow induced vibration problems. Add discussion that EF-62-FT across cavitating venturi EF-62-FO provides EFP-2 flow measurement for EFV-12 cross-tie operation. (4) In 10.5.2.3, clarify that both ASV-5 and ASV-204 open upon actuation from EFIC. (5) In section 10.5.3, add ASV-204 as receiving power from Battery 'A'. Add discussion that EFP-2 will supply flow through cross-tie and utilize 'A' train instruments an control valves to monitor and control flow to the OTSGs. (6) Add new section, "EFW System Cross-Tie", discussing the cross connect piping between discharge of EFP-1 and EFP-2, the valves in that piping, and conditions when cross-tie operation is required. (7) Table 10-1, add EFV-12, as a 6-inch Atwood & Morril gate valve.

Table 4
Summary of Planned FSAR Changes

Section	Section Title	Summary of Planned Changes
10.6	Auxiliary Feedwater	(1) In section 10.6.2, add to discussion of power sources for AFW pump motor that on a LOOP, the motor is fed from MTDG-1, a non-safety related diesel generator, not automatically started but under manual control from the MCB. (2) In section 10.6.5 add that a non-safety Standby Diesel Generator (MTDG-1) is capable to be lined up to the 4160V Reactor Auxiliary Bus 3 which will enable FWP-7 to operate in a LOOP.
14.2	Standby Safeguards Analysis	Section 14.2.2.5, "Loss of Coolant Accident", revise and add additional details to the discussion of SBLOCA based on revised analyses performed by Framatome for different sizes and locations of breaks.

Table 5
Related LERs

LER	LER # (Event Date)	Title	Description	Reference	Resolution
1	96-020 (9/10/96)	Unreviewed Safety Questions Concerning Diesel Generator Loading Caused by Interpretation of Regulatory Requirements other than Prescribed.	An USQ was identified which was associated with MAR 96-04-12-01 (ASV-204 EFIC Auto Open Removal). This modification contributed to increased in loading of the A EDG. It was discovered that, contrary to information contained in the TS, EDG loading analyses indicated the worst case maximum automatically connected accident load at one-minute (3100 kW) would be exceeded, maximum EDG design rating of 3500 kW loading would be exceeded for up to 3 seconds during three of the six EDG block loading sequences, and the single largest rejected load was greater than previously identified.	USQ1, USQ2, USQ3, USQ5	Modifications MOD1, MOD5, MOD6, MOD7, MOD8, MOD9, MOD10 TS/Bases Changes Parts 1, 2, 3

Table 5
Related LERs

LER	LER # (Event Date)	Title	Description	Reference	Resolution
2	96-024 (10/11/96)	Plant Modification Creates Unanalyzed Condition Regarding Emergency Feedwater Availability	During a review of a EDG loading calculation, it was determined that the calculation assumed EFP-2 was running when EFP-1 received an ES automatic trip signal at a RCS pressure of 500 psig for LPI actuation. MAR 96-12-04-01 (ASV-204 EFIC Auto Open Removal) had removed the automatic start from EFIC A system to prevent runout and NPSH concerns with EFP-2 during certain accident conditions when its flow control valves would fail open. EFP-2 would remain available but require operator action to crosstie EFP-2 flow to the EFP-1 flow path control valves which would have control power available during a loss of DC power train B scenario.	USQ4, USQ5	<u>Modifications</u> MOD1, MOD2, MOD3, MOD4 <u>TS/Bases Changes</u> Part 1

Table 5
Related LERs

LER	LER # (Event Date)	Title	Description	Reference	Resolution
3	97-001 (1/28/97)	Ineffective Change Management Results in Unrecognized NPSH Issue Affecting Emergency Feedwater Availability	FPC had not explicitly reported a condition that existed prior to May 1996 involving inadequate NPSH affecting EFP-2. In addition to an EDG load management concern, the postulated loss of B DC power single failure coincident with a SBLOCA and LOOP, could have resulted in two situations in which EFW may not have been available to perform its intended functions. These include a design feature which trips EFP-1 at a RCS pressure of 500 psig, and a point in time at which EFP-1 would need to be secured in order to load the LPI pump onto the EDG in order to provide adequate NPSH to the HPI.	USQ4	<u>Modifications</u> MOD1, MOD2, MOD3, MOD4 <u>TS/Bases Changes</u> Part 1

Table 5
Related LERs

LER	LER # (Event Date)	Title	Description	Reference	Resolution
4	97-005 (2/14/97)	Unanalyzed Condition Regarding Small Break LOCA Mitigation Caused by Misunderstanding Reliance on Emergency Feedwater	A team studying EDG capacity and EFW system dependency issues discovered an unanalyzed condition involving a SBLOCA in the reactor coolant cold leg discharge piping line coincident with a LOOP and single failure of EFP-2. For a certain range of small breaks, the RCS may repressurize. For an unisolable cold leg pump discharge line break or an isolable broken HPI line that may not be identified, if EFW is lost early in the transient, there may be inadequate HPI flow to the core, even with two HPI pumps operating. With a combination of insufficient HPI injection and no EFW, 10 CFR 50.46 requirements may not be met.		Modifications MOD1, MOD2, MOD3, MOD4, MOD5, MOD6, MOD7, MOD8, MOD9, MOD10 TS/Bases Changes Parts 1, 2

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT G

LIST OF ACRONYMS AND ABBREVIATIONS USED

ATTACHMENT G

LIST OF ACRONYMS AND ABBREVIATIONS USED

ADV ---------------- Atmospheric Dump Valves
AFW ---------------- Auxiliary Feedwater
ATOG---------------- Abnormal Transient Operating Guidelines
B&W ---------------- Babcock and Wilcox
BWOG --------------- Babcock and Wilcox Owners Group
BWST---------------- Borated Water Storage Tank
CLPD ---------------- Cold Leg Pump Discharge
CFR ----------------- Code of Federal Regulations
CR-3 ---------------- Crystal River Unit 3
CREVS -------------- Control Room Emergency Ventilation System
DC ------------------ Decay Heat Closed Cycle Cooling Water System
DC Electrical ------- Direct Current Electrical
DHP ----------------- Decay Heat Pump
DHR ---------------- Decay Heat Removal
ECCS ---------------- Emergency Core Cooling System
EDG ----------------- Emergency Diesel Generator
EEI ------------------ Escalation Enforcement Item
EFIC ----------------- Emergency Feedwater Initiation and Control System
EFP ------------------ Emergency Feedwater Pump
EFV ------------------ Emergency Feedwater Valve
EFW ----------------- Emergency Feedwater
EOP ------------------ Emergency Operating Procedure
ESAS ----------------- Engineered Safeguards Actuation System
F --------------------- Fahrenheit
FPC ------------------ Florida Power Corporation
FSAR ---------------- Final Safety Analysis Report
FTI ------------------- Framatome Technologies Incorporated (formerly B&W)
FWP ------------------ Auxiliary Feedwater Pump
gal ------------------- gallon
gpm ------------------ gallons per minute
HPI ------------------- High Pressure Injection
kW -------------------- kilowatts
LCO ------------------ Limiting Condition for Operation
LER ------------------ Licensee Event Report
LOBA ---------------- Loss of Battery A
LOBB ---------------- Loss of Battery B
LOCA ---------------- Loss of Coolant Accident
LOOP ---------------- Loss of Offsite Power
LPI ------------------- Low Pressure Injection
MCAP --------------- Management Corrective Action Plan
MSSV --------------- Main Steam Safety Valve

MUP ----------------- Makeup Pump
MWt------------------ Megawatt (Thermal)
NPSH ---------------- Net Positive Suction Head
NRC------------------ Nuclear Regulatory Commission
OTSG ---------------- Once Through Steam Generator
PCT ------------------ Peak Clad Temperature
psi -------------------- pounds per square inch
psig ------------------- pounds per square inch gauge
PSV ------------------ Pressurizer Safety Valve
RBCU---------------- Reactor Building Cooling Units
RCP ------------------ Reactor Coolant Pump
RCS ------------------ Reactor Coolant System
RW ------------------- Nuclear Services Seawater System
RWP ----------------- Nuclear Services Seawater Pump
SBLOCA ------------ Small Break Loss of Coolant Accident
SER ------------------ Safety Evaluation Report
SR -------------------- Surveillance Requirement
SW-------------------- Nuclear Services Closed Cycle Cooling System
SWP------------------ Nuclear Services Closed Cycle Cooling Pump
TBD ------------------ Technical Bases Document
TMI ------------------ Three Mile Island
TS -------------------- Technical Specification
TSCRN -------------- Technical Specification Change Request Notice
USQ ------------------ Unreviewed Safety Question