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ENCLOSURE 3

**AREVA 86-9168766-001 – CR-3 EPU INADVERTENT
ENGINEERED SAFEGUARDS ACTUATION (IESA) ANALYSIS
SUMMARY**



CALCULATION SUMMARY SHEET (CSS)

Document No. 86 - 9168766 - 001 Safety Related: Yes No

Title CR-3 EPU Inadvertent Engineered Safeguards Actuation (IESA) Analysis Summary

PURPOSE AND SUMMARY OF RESULTS:

The purpose of this document is to summarize the thermal hydraulic evaluation of the Inadvertent ECCS actuation event for CR-3 at the EPU power level (Reference [5]). This evaluation only considers the thermal hydraulic behavior of ECCS actuation. The reactivity effects associated with changing the core boron concentration are not modeled.

The inadvertent ECCS actuation event was analyzed for CR-3 (Reference [5]). All acceptance criteria were met using an operator action time of 10 minutes to isolate HPI. In particular, the pressurizer did not become liquid solid and liquid did not pass through the PORV. Therefore, the inadvertent ECCS actuation event does not evolve into a more limiting condition. Additional results are described in Section 4.0.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:

CODE/VERSION/REV	CODE/VERSION/REV
None	

THE DOCUMENT CONTAINS ASSUMPTIONS THAT SHALL BE VERIFIED PRIOR TO USE

YES
 NO

Controlled Document



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CR-3 EPU Inadvertent Engineered Safeguards Actuation (IESA) Analysis Summary

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1.0 ANALYTICAL METHODOLOGY

The thermal-hydraulic analysis of the Inadvertent Engineered Safeguards Actuation (IESA) analyses at the CR-3 EPU power level was performed using the RELAP5/MOD2-B&W (R5/M2-B&W) computer program (Reference [1]). The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, moderator temperature changes, and changes in boron concentration. The RCS model provides for heat transfer from the core, transport of the coolant to the SGs, and heat transfer to the SGs. The secondary model includes a detailed depiction of the main steam system, including steam relief to the atmosphere through the MSSVs and simulation of the TSVs. The secondary model also includes the delivery of feedwater, both main and emergency, to the SGs.

The R5/M2-B&W code has been approved by the NRC for use in non-LOCA safety analyses (Reference [2]). The approved methodology in Reference [2] as well as the standard review plan in Reference [3] will be used as guidance for performing the IESA event. However, the IESA event is not a part of the plant design basis for CR-3.

The analysis performed in Reference [5] considers only the thermal hydraulic effects of ES actuation. In other words, the reactivity effects associated with changes in the core boron concentration are not evaluated.



CR-3 EPU Inadvertent Engineered Safeguards Actuation (IESA) Analysis Summary

2.0 ASSUMPTIONS

2.1 Unverified Assumptions

There are no unverified assumptions associated with the analysis.

2.2 Justified Assumptions

There are no justified assumptions associated with the analysis.

2.3 Modeling Simplifications

There are no modeling simplifications associated with the analysis.



CR-3 EPU Inadvertent Engineered Safeguards Actuation (IESA) Analysis Summary

3.0 DESIGN INPUTS

3.1 Event Description

Reference [3] describes the inadvertent operation of ECCS event (i.e. IESA event) and the Chemical and Volume Control System (CVCS) malfunction (Makeup and Purification system in the CR-3 plant design) that increases Reactor Coolant System (RCS) inventory. For the CR-3 design, the IESA event is a more limiting RCS inventory increase event than a malfunction in the Makeup and Purification System. In a B&W designed plant, the IESA event would result in starting the Engineered Safeguards (ES) selected makeup pump and alignment of the operating makeup pump such that both pumps deliver flow from the Borated Water Storage Tank (BWST) to the four cold legs at high RCS pressures. On the other hand, a malfunction in the Makeup System could open the makeup control valves, but with only one HPI pump. In the B&W design, neither event would lead to an automatic reactor trip; therefore the event with the higher HPI flow is more limiting. The IESA event allows for operation of two HPI pumps, and is consequently the more limiting event.

The IESA event is postulated to occur as a result of a spurious signal resulting in automatic ES initiation. As a result of the IESA, the idle ES selected High Pressure Injection (HPI) pump starts, the valves in each HPI line open, the valves in the lines to the Borated Water Storage Tank (BWST) open, the normal makeup and seal injection isolation valves close, the valve in each HPI cross-tie line opens, and the HPI pump recirculation valves close. Letdown is also assumed to be isolated which maximizes the RCS inventory. The reactor coolant inventory begins to increase, which causes an insurge of reactor coolant into the pressurizer. As the pressurizer level increases, the system pressure also increases due to compressing the steam bubble within the pressurizer. The pressurizer spray is actuated in an attempt to reduce the pressure increase. The actions of the pressurizer spray may be insufficient to prevent the pressurizer pressure from increasing.

If the ES actuation is not terminated in a timely manner, the system pressure continues to increase to the Reactor Protection System (RPS) High RCS Pressure trip setpoint. When power to the CRDMs is interrupted following reactor trip, a signal is sent to trip the turbine. Should the turbine bypass system be unavailable, the secondary side pressure rapidly increases until the Main Steam Safety Valves (MSSVs) lift, allowing for the continued removal of heat through the Steam Generators (SGs). Main feedwater is assumed to be rapidly reduced by the Integrated Control System (ICS) following reactor trip. The actuation of both trains of ES will cause an actuation of Emergency Feedwater (EFW) at the start of the event. However, EFW will not be needed to the SGs unless the SG level falls below the forced circulation control setpoint.

Shortly after reactor trip, the heat removal capability of the secondary side is sufficient to reduce the reactor coolant temperature, which results in a pressurizer level decrease due to contraction of the RCS. If HPI flow continues, the RCS pressure and pressurizer level will begin to increase again. The RCS pressure will increase to the PORV setpoint (if operable) and the PORV will attempt to maintain the RCS pressure. Should the PORV not be available or effective, the pressure continues to increase until the Pressurizer Code Safety Valves (PCSVs) lift. The IESA analysis will determine whether these valves have sufficient relieving capacity to prevent the system pressure from exceeding the accident acceptance criterion.



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The PCSV or PORV continue to cycle until the ES actuation is bypassed and the HPI injection is terminated. At this time the reactor coolant pressure will begin to decrease and the pressurizer level will stabilize. The IESA event analysis is terminated at this time.

3.2 Applicable Acceptance Criteria

The IESA analysis will demonstrate compliance with the following acceptance criteria from Reference [3].

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values in accordance with the ASME Boiler and Pressure Vessel Code.
2. Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs based on acceptable correlations.
3. An Anticipated Operational Occurrence (AOO) should not generate a more serious plant condition without other faults occurring independently.

The design pressure for the RCS is 2500 psig, giving an acceptance criterion of 2750 psig for RCS pressure.

In the IESA event, the secondary side pressure increases due to the turbine trip that occurs after reactor trip. Since the reactor has already been tripped, the peak secondary side pressure results are bounded by the results of the Turbine Trip event, where the turbine is tripped before the reactor trips. Therefore, acceptable results for the Turbine Trip transient ensure that the peak secondary side pressure in the IESA event will also be acceptable.

With respect to DNBR, the acceptance criterion is not analyzed specifically. In the IESA event, the RCS pressure is continuously increasing with continuous forced circulation. The core inlet temperature stays the same or slightly decreases due to the cooling effect of the HPI. With negative moderator feedback, the core power could gradually increase, but the power increase would be relatively slow and the high flux trip and high RCS pressure trips ensure that the thermal power remains below 112% of the rated thermal power. Therefore, the IESA event will not challenge the DNBR acceptance criterion and is bounded by the loss of coolant flow events.

There are no analyzed dose consequences associated with the IESA event, since the fission product barrier remains intact. The secondary release is bounded by the Steam Generator Tube Rupture (SGTR) event.

The limiting acceptance criterion for the IESA event is to ensure that the IESA event does not generate a more serious plant condition. In particular, the IESA event is analyzed to prevent a liquid-solid pressurizer and subsequent liquid relief through the Pilot Operated Relief Valve (PORV). If the PORV opens while the pressurizer is liquid-solid, then the PORV is assumed to fail open resulting in a Small Break Loss of Coolant Accident (SBLOCA). **The analysis objective is to show that the pressurizer does not become liquid-solid before the operator can terminate the transient, usually at about 10 minutes (or longer) after the event begins.**



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3.3 Core and Reactor Coolant Pump (RCP) Power

The IESA event is analyzed at an initial core power level of 3026.1 MWt. This corresponds to the rated EPU power level of 3014 MWt plus the maximum heat balance uncertainty of 0.4%.

Consistent with Section A.3.2.2 of Reference [2], the decay heat used for the IESA event is 1.0 times the ANS 1971 decay heat standard plus the B&W heavy actinides. This combination of decay heat curve and multiplier has been shown to produce more energy than the ANS-1979 decay heat curve plus 2σ uncertainty for a wide variation in fuel enrichment and burnup.

During normal operation, the four RCPs produce 20.96 MWt of heat. The net heat removed from makeup and letdown is 4.84 MWt. Therefore, the net heat added by the RCPs minus the net heat removed from makeup and letdown is $(20.96 - 4.84)/4 = 4.03$ MWt per RCP. This value is rounded up to 4.1 MWt per pump to give a total net heat addition of 16.4 MWt.

The steady-state run assumes letdown and makeup is operating, therefore the net heat added is 16.4 MWt. If the secondary heat load indicates insufficient RCP heat, the RCP energy is supplemented by use of passive heat structures in the cold leg piping to achieve the desired heat.

In the transient run, letdown is assumed to be isolated. Therefore, the heat added by the passive heat structure is increased to model the 20.96 MWt added by the RCPs.

3.4 Reactor Coolant System (RCS) Flow Rate

Consistent with Section A.3.1.4 of Reference [2], the initial RCS flow rate is set to the minimum thermal design flow, which is 374,880 gpm for the CR-3 EPU with replacement Steam Generators (SG). This flow rate is applicable for up to 5% SG tube plugging. The RCPs are modeled to remain on for the entire IESA event.

The IESA event is not an overheating or an overcooling event. Therefore, the core bypass flow does not have a significant effect on the transient progression. A core bypass flow of 7.5% is modeled.

3.5 RCS Pressure

An initial pressure of 2170 psia at the hot leg pressure tap is modeled for the IESA analyses.

3.6 RCS Temperature

An initial T_{ave} of 582 °F is modeled for the IESA analyses.



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3.7 Pressurizer Level

Since the IESA transient results in an increase in RCS pressure and pressurizer level, Section A.3.1.6 of Reference [2] requires that the initial pressurizer level be biased high to account for uncertainty. The nominal HFP level is 220 inches and the measurement uncertainty is +18.8 inches. The IESA event conservatively assumes a starting pressurizer level of 240 inches.

3.8 Pressurizer Spray

The IESA event is analyzed with and without pressurizer spray. The pressurizer spray is regulated by a control system that opens the spray valve when the pressure at the hot leg tap increases to 2205 psig. Once open, the spray valve remains open until the pressure decreases to 2155 psig. The design capacity of the pressurizer spray is 190 gpm when all four RCPs are operating. A stroke time of 0.0 second will be used for the pressurizer spray valve to maximize the impact of the pressurizer spray when modeled.

3.9 Pressurizer Heaters

The pressurizer heaters are considered in the IESA transient. The bank powers and setpoints are summarized in Table 3-1.

Table 3-1: Pressurizer Heater Power and Setpoints

Bank	kW	On	Off	Comments
A	126	< 2169.7 psia	≥ 2169.7 psia	Bank A turns on as required to maintain 2155 psig setpoint
B	126	< 2169.7 psia	≥ 2169.7 psia	Bank B turns on as required to maintain 2155 psig setpoint
C	504	< 2169.7 psia	≥ 2169.7 psia	Bank C turns on as required to maintain 2155 psig setpoint
D	378	2134.7 psia	2154.7 psia	Corresponds to 2120 psig and 2140 psig
E	504	2119.7 psia	2139.7 psia	Corresponds to 2105 psig and 2125 psig

3.10 Pilot Operated Relief Valve (PORV)

The IESA event is analyzed with and without the PORV. The PORV is modeled to open at 2450 psig as measured in the hot leg. Once open, the PORV remains full open until the hot leg pressure decreases to 2380 psig. In order to maximize the insurge of liquid into the pressurizer, the PORV is modeled with a 0.0 second stroke time.

The IESA analysis includes a sensitivity study on the PORV capacity. On the low end, the analysis considers the PORV inoperable. These results bound a case with a PORV at the minimum required capacity of 100,000 lbm/hr at 2300 psig, which is equivalent to 106,480 lbm/hr at 2450 psig. On the high end, the nominal PORV capacity of 165,806 lbm/hr at 2450 psig is considered. A case with a PORV capacity that is ~10% less than the nominal capacity was also evaluated and found to have essentially the same results as a case with the nominal PORV capacity.



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3.11 Pressurizer Code Safety Valves (PCSV)

PCSVs will be modeled for all IESA analyses. The nominal setpoint is 2500 psig. A conservative setpoint drift of 3% is modeled. As mentioned in Reference [2], Section A.3.2.3, no accumulation will be modeled because all B&W designed plants have safety valves with steam-to-seat internals (i.e. no liquid loop seals). Reference [2] also states that, in the EPRI/C-E test of Dresser and Crosby pressurizer safety valves with steam-to-seat internals, no accumulation was observed. The blowdown to be modeled is 4% of the nominal setpoint.

The saturated steam discharge rate through each of the two PCSVs is modeled as 317,973 lbm/hr/valve at 2750 psig, for a total relief capacity of 635,946 lbm/hr.

3.12 Main Feedwater (MFW)

The main feedwater temperature is modeled as 460 °F. The initial MFW flow is varied as necessary during the steady-state initialization to achieve the desired T_{ave} .

Following reactor trip, the ICS will begin a rapid feedwater reduction. For conservatism, the MFW is assumed to ramp from full flow to no flow in 3.2 seconds. This is consistent with the reduction in MFW modeled in the LOFW accident.

3.13 Emergency Feedwater (EFW) and Emergency Feedwater Initiation and Control (EFIC)

EFW is initiated at the start of the event when ES is actuated. The LOFW accident determined that for the EPU, the EFW delay time must be 40 seconds or less and the minimum EFW flow rate must be at least 660 gpm. The IESA event also models a required EFW flow rate of at least 660 gpm. For additional conservatism, a EFW delay time of 60 seconds is modeled in the IESA event.

The IESA event will use an EFW temperature of 140 °F. The normal temperature of EFT-2 is ≤ 110 °F. Once EFW is initiated, EFW pump recirculation will increase the temperature of the EFW. A temperature rise of 10°F bounds the temperature rise expected after 3 hours of pump recirculation. The EFW that reaches the SGs is further increased by 9°F by the EFW pumps. This gives a total temperature of $110 + 10 + 9 = 129$ °F. Using 140 °F adds additional conservatism.

The IESA event models the EFW SG level control setpoint with RCPs running, which is 24 inches on the Startup Range or 31.94 inches above the lower tubesheet.

3.14 Main Steam Turbine Header Pressure

The initial turbine header pressure is set to achieve the targeted T_{ave} . The IESA event results are not sensitive to the starting turbine header pressure.



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3.15 Main Steam Isolation

Turbine trip is initiated coincident with reactor trip. The Turbine Stop Valves (TSVs) are closed using a stroke time of 200 ms, which represents a nominal closure time.

3.16 Main Steam Safety Valves (MSSVs)

The MSSVs will be modeled for the IESA event. Consistent with Reference [2] Section A.3.2.3, the MSSVs are modeled to lift to 70% of the full area at the nominal setpoint plus the maximum lift tolerance (modeled as 3% for the IESA event). The open valve area increases linearly with pressure until the full open area is reached at 3% accumulation (106% of the nominal lift setpoint). Once the MSSV is full open, the valve will blow down at the full open area until the valve reseats at the blow down pressure (5% for the IESA event). Should the inlet pressure decrease before the valve reaches full open, the valve flow area will linearly decrease as a function of inlet pressure until the valve reseats at the blowdown pressure.

CR-3 has 8 MSSVs for each SG. Seven of the MSSVs on each SG have a capacity of 845,759 lbm/hr at 1159.2 psia. The remaining MSSV on each SG has a capacity of 583,574 lbm/hr at 1159.2 psia. The nominal lift setpoints are shown below.

Nominal Lift Setpoint Psig	Number per SG
1050	2
1070	2
1090	2
1100	2 ⁽¹⁾

(1) The 1 MSSV with a capacity of 583,574 lbm/hr has a nominal setpoint of 1100 psig.

3.17 Steam Generator (SG) Inventory

The IESA event is not an overheating or an overcooling event. Therefore, the initial SG inventory does not have a significant effect on the transient results. The IESA event uses an initial SG inventory that corresponds to < 70% on the operating range (OR).

3.18 SG Tube Plugging

The effects of tube plugging on heat transfer does not have a significant effect on the IESA event. However, modeling tube plugging decreases the available RCS volume and result in a faster increase in the pressurizer level. Consequently, the IESA event models 5% SG tube plugging.



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3.19 Core Kinetics Parameters

The emphasis of this analysis is to quantify the thermal-hydraulic response of the IESA event without consideration of reactivity contributions due to boron and other sources. For this reason, parameters typical of beginning-of-cycle (BOC) are chosen to minimize reactivity feedback. In particular, the Doppler Coefficient is $-1.30E-5 \Delta k/k/^\circ F$, the MTC is $0.0E-4 \Delta k/k/^\circ F$, and the β_{eff} is 0.0070. The IESA event is not sensitive to the selection of the prompt neutron generation time. A value of $24.8 E-6$ seconds is used.

The initial RCS boron concentration and the BWST boron concentration are set to 0 ppm to remove the reactivity effects of boron. The boron worth modeled is not pertinent to the results since the core boron concentration will not change.

The Shutdown Margin (SDM) for EPU is 1.3 % $\Delta k/k$. The control rod scram worth is determined based on the initial average fuel temperature and the initial average core moderator temperature for the case of interest plus the required shutdown margin. The scram curve is shown in Table 3-2 for a control rod drop time of 1.4 seconds to 2/3 insertion.

Table 3-2: Scram Curve

Time After Reactor Trip sec	Reactivity Insertion %
0.0	0.0
0.2	0.58
0.3	0.99
0.4	1.83
0.6	5.29
0.8	12.33
1.0	21.41
1.2	33.09
1.4	50.75
1.6	72.96
1.8	91.30
2.0	99.26
2.2	99.99
2.3	100.00

3.20 Engineered Safeguards Actuation System (ESAS) and High Pressure Injection (HPI)

The IESA event begins with actuation of both ES trains of HPI. For conservatism, no delay time is modeled. The HPI flow modeled in the IESA event is a maximum HPI flow for two HPI pumps. The flow is shown in Table 3-3.



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Table 3-3: HPI Flow Rate

RCS Pressure psia	Total Flow Rate gpm
15	904.30
615	815.90
1215	712.10
2215	462.75
2315	433.86
2415	404.19
2515	373.30

The HPI flow is provided by the Borated Water Storage Tank (BWST), which has a minimum temperature of 40 °F and a maximum temperature of 100 °F. When determining the HPI flow rate in lbm/sec, the IESA analysis uses the minimum BWST temperature to maximize the flow rate. However, the source volume for the HPI is set to the maximum BWST temperature to maximize the expansion of the RCS fluid. This conservative approach ensures that the IESA event is bounding for any BWST temperature in the allowed range.

The BWST boron concentration is ignored in the IESA analysis to isolate the thermal hydraulic effects of inadvertent ECCS actuation.

3.21 Reactor Protection System (RPS)

For the IESA analysis, the RPS High RCS Pressure Trip function initiates reactor trip. The IESA analysis uses a setpoint of 2400 psia at the hot leg tap, which is based on a nominal allowable setpoint of 2355 psig, +20.75 psi for uncertainty, and 9.55 psi of margin. The response time for the High RCS Pressure trip is modeled as 0.610 seconds.

3.22 Integrated Control System (ICS)

Section A.5 of Reference [2] explains that if a control system aggravates the response to a postulated accident, the control system is modeled. If the control system mitigates the response to a postulated transient, the control system is not modeled.

The IESA analysis does not model the ICS. Without boron effects, the IESA event does not result in a significant change in the core power (Figure 4-1). The cooling effects of the HPI does lead to a small decrease in T_{ave} , but the change is only ~ 1°F prior to reactor trip (Figure 4-8). Therefore, the ICS would not make a significant change to the transient behavior prior to the reactor trip. If the ICS were modeled to try to maintain T_{ave} , the behavior would result in a slightly earlier reactor trip.

Following the reactor trip, the IESA analysis models the MFW being ramped to zero flow over 3.2 seconds. This results in more expansion of the primary side inventory than would occur if the ICS were modeled to perform a more realistic ramp to 3% of the full power flow within 6 seconds.



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Similarly, the Turbine Bypass Valves (TBV) are not modeled. This requires the secondary side pressure to reach the MSSV setpoint before steam can be relieved. Consequently, the primary side temperatures are higher resulting in a higher RCS pressure and pressurizer level.

3.23 Offsite Power

Consistent with Reference [3], loss of offsite power is not considered for the IESA event.

3.24 Single Failure

The only safety related equipment modeled in the IESA event (other than HPI which is the event initiator) is EFW. The EFW flow modeled (660 gpm) is consistent with the LOFW transient, which assumes the failure of one train of EFIC.

3.25 Operator Action

The objection of the IESA analysis is to determine the time at which the operators must isolate HPI in order to prevent filling the pressurizer with liquid.

3.26 Input Summary

Table 3-4 summarizes the inputs and boundary conditions used in the IESA analysis.



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Table 3-4: Summary of Inputs for IESA

Parameter	Value
RCS Conditions	
Core Power, MWt	3026.1
Decay Heat	1.0*ANS71 plus B&W Actinides
Total RCP Power, MWt	20.96
Net Heat from RCPs, Letdown, and Makeup, MWt	16.4
Average RCS Temperature, °F	582
Initial RCS Pressure, psia	2170 ¹
Total RCS Flow Rate, gpm	374,880
Pressurizer	
Initial Indicated Pressurizer Level, in	240
Pressurizer Spray Setpoints, psig	2205 (on) ¹ / 2155 (off) ¹
Pressurizer Spray Capacity, gpm	190
Pressurizer Spray Valve Stroke Time, s	0.0
Pressurizer Heater Power and Setpoints	See Table 3-1
PORV Setpoints, psig	2450 (open) ¹ / 2380 (close) ¹
Minimum Required PORV Capacity, lbm/hr	106,480 @ 2450 psig
Nominal PORV Capacity, lbm/hr	165,806 @ 2450 psig
PCSV Setpoints, psig	2500 * (1 + 0.03) (open) 2500 * (1 - 0.04) (close)
Total PCSV Capacity, lbm/hr	2 * 317,973 @ 2750 psig
Secondary Side	
MFW Temperature, °F	460 °F
MFW Coastdown Time, s	3.2 (linear)
Tube Plugging, %	5
Initial SG Level, %OR	< 70
EFW Temperature, °F	140
EFW Minimum Required Flow, gpm	660
EFW Delay Time, sec	60
EFW SG Level Control Setpoint, in above lower tube sheet	31.94
Turbine Trip Delay Time, s	0.0
TSV Stroke Time, s	0.2
Number of MSSVs per SG	8
MSSV Nominal Setpoint / Capacity	Section 3.16
MSSV Setpoint Tolerance	+3%
MSSV Accumulation	+3%
MSSV Blowdown	-5%
Core Kinetics Parameters	
Doppler Temperature Coefficient ($\Delta k/k/^\circ F$)	-1.30 E-5
Moderator Temperature Coefficient ($\Delta k/k/^\circ F$)	0.0 E-4
Prompt Neutron Generation Time, (μs)	24.8
Effective Delayed Neutron Fraction	0.0070
Shutdown Margin, % $\Delta k/k$	1.300
Scram Curve	Table 3-2
Initial RCS Boron Concentration, ppm	0

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CR-3 EPU Inadvertent Engineered Safeguards Actuation (IESA) Analysis Summary

Parameter	Value
ESAS / HPI	
HPI Flow	Table 3-3
BWST Temperature, °F	$\geq 40 / \leq 100$
BWST Boron Concentration, ppm	0
RPS	
RPS High RCS Pressure Trip Setpoint, psig	2400 ¹
RPS High RCS Pressure Response Time, s	0.610
Miscellaneous	
Offsite Power	Available
Single Failures	One Train of EFIC
Operator Actions	Isolate HPI

Notes:

1) As measured at the hot leg pressure tap.



CR-3 EPU Inadvertent Engineered Safeguards Actuation (IESA) Analysis Summary

4.0 SUMMARY OF RESULTS

The thermal hydraulic effects of an inadvertent ECCS actuation were evaluated (Reference [5]). The reactivity effects from changes in core boron concentration are not considered. The results from Reference [5] are summarized below:

4.1 Sensitivity Studies

The first 12 minutes of the inadvertent ECCS actuation transient was evaluated with no operator action modeled. The purpose of these evaluations was to determine the time at which the pressurizer would become liquid solid and/or when liquid would be passed out of the PORV if operator action is not taken.

Sensitivity studies were performed assuming the following three scenarios:

- PORV Modeled, Pressurizer Spray Modeled
- PORV Unavailable, Pressurizer Spray Modeled
- PORV Unavailable, Pressurizer Spray Unavailable

The sequence of events for the three sensitivity cases are shown in Table 4-1.

The peak RCS pressure and peak secondary side pressure for all three cases were essentially the same. The capacities of the pressurizer code safety valves and the main steam safety valves are more than sufficient to ensure that 110% of the design pressure is not exceeded on the primary or secondary side for an inadvertent ECCS actuation transient.

In all three cases, the pressurizer level did not reach liquid solid conditions during the first 600 seconds of the transient. The pressurizer level at 600 seconds, as well as the time to reach a liquid solid pressurizer, indicate that the case with the PORV and pressurizer spray modeled is the limiting case. In other words, modeling the PORV and the pressurizer spray results in a faster increase in the pressurizer level and is conservative for determining acceptable operator action times.

NRC Regulatory Issue Summary (RIS) 2005-29 (Reference [4]) documents NRC staff concerns that licensees are taking credit for non-safety grade pressure-operated relief valves for mitigation of the inadvertent ECCS actuation event. The sensitivity studies performed here demonstrate that for the CR-3 plant design, modeling the PORV does not reduce the severity of the event. Modeling the PORV results in a faster pressurizer level increase and does not provide any significant benefit to lowering the peak RCS pressure that occurs during the event. Therefore, modeling the behavior of the PORV for the inadvertent ECCS actuation ensures that a conservative evaluation of the operator action time is performed. Consequently, the NRC guidelines relative to the PORV operation not being credited to mitigate the event, presented in RIS 2005-29, are adequately considered.

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Document No. 86-9168766-001

CR-3 EPU Inadvertent Engineered Safeguards Actuation (IESA) Analysis Summary

Table 4-1: Sequence of Events and Key Results for Inadvertent ECCS Actuation with No Operator Action

Parameter	PORV, Spray (TFHQ)	No PORV, Spray (UQKU)	No PORV, No Spray (TNAN)
HPI Begins, Letdown Isolated, s	0.0	0.0	0.0
EFIC Initiates EFW Based on ES Signal, s	1.0	1.0	1.0
Pressurizer Spray Flow Starts, s	22.1	22.1	N/A
EFW Available to SG if Level Reaches Control Setpoint, s	61.0	61.0	61.0
RPS High RCS Pressure Trip Setpoint Reached, s	231.0	231.0	91.8
Control Rods Begin to Drop, Turbine Trip (Turbine Stop Valves Begin to Close), MFW Reduction Begins, s	231.6	231.6	92.4
Turbine Stop Valves Close, s	231.8	231.8	92.6
PORV Opens, s	232.0	N/A	N/A
Pressurizer Code Safety Valves Open, s	234.8	234.5	95.8
Peak RCS Pressure Reached, s	234.8	234.5	95.8
Peak SG Pressure Reached, s	236.8	236.9	97.7
Pressurizer Spray Off, s	243.3	243.7	N/A
Pressurizer Heaters Turn On, s	243.3	243.7	105.7
Pressurizer Heaters Turn Off, s	350.0	350.0	210.0
Pressurizer Spray Flow Starts, s	357.6	357.0	N/A
EFW Flow Needed to Maintain SG at Control Setpoint, s	390.0	390.0	245.0
PORV Opens for Second Time, s	507.0	N/A	N/A
Pressurizer Code Safety Valves Open for Second Time, s	N/A	558.0	350.0
Pressurizer Becomes Liquid Solid, s	660.0	690.0	> 720
Evaluation Ended, s	720	720	720
Peak RCS Pressure, psia	2692.2	2687.9	2681.3
Peak SG Pressure, psia	1148.9	1149.2	1150.1
Peak Steam Line Pressure, psia	1140.1	1140.3	1141.1
Pressurizer Level at 600 seconds, ft	38.7	38.3	34.8
Pressurizer Level at 600 seconds, in	464.4	459.6	417.6

Note that the top of the pressurizer is 41.9 ft (502.8 in).



4.2 Inadvertent ECCS Event with HPI Isolation at 10 Minutes

The inadvertent ECCS Actuation event was evaluated with operator action to isolate HPI at 10 minutes. Based on the sensitivity studies described in Section 4.1, the behavior of the PORV and the pressurizer spray was modeled. The key results are plotted in Figures 4-1 through 4-23.

The sequence of events for the first 10 minutes of the transient is identical to the case described in Table 4-1 with the PORV and pressurizer spray modeled. Following HPI isolation at 10 minutes, the RCS pressure begins to decrease. The pressurizer level increases until ~740 seconds and then remains constant for the remainder of the analyzed transient. The maximum pressurizer level reached is 40.6 ft (487.2 inches), which is below the top of the pressurizer (41.9 ft or 502.8 inches). At the end of 20 minutes, the pressurizer level is stable and the RCS pressure has decreased to the point that pressurizer spray is off and pressurizer heaters are acting to try and maintain the RCS pressure at the control setpoint. Therefore, the event is considered terminated.

The transient results show that isolating HPI at 10 minutes will prevent the pressurizer from becoming liquid solid. Furthermore, once HPI is isolated the RCS pressure begins to decrease and the PORV will not open. Since the RCS pressure is below the PORV setpoint and the pressurizer is not liquid solid, liquid will not pass through the PORV and the IESA event will not develop into a more serious event.

As demonstrated in Table 4-1, the peak RCS pressure occurs shortly after reactor trip and is < 2700 psia. This is significantly less than 110% of the RCS design pressure ($1.10 * 2500 = 2750$ psig).

The secondary side pressure increases due to the turbine trip that occurs after reactor trip. Since the reactor has already been tripped, the peak secondary side pressure results are bounded by the results of the Turbine Trip event, where the turbine is tripped before the reactor trips. Therefore, acceptable results for the Turbine Trip transient ensure that the peak secondary side pressure for the inadvertent ECCS event will also be acceptable. A comparison of the maximum steam line pressure from the inadvertent ECCS event results in Table 4-1 (1141.1 psia) to the peak steam line pressure in the Turbine Trip event (1152.99 psia) confirms that the Turbine Trip event bounds the inadvertent ECCS event with respect to secondary side pressure consequences.

With respect to DNBR, the acceptance criterion is not analyzed specifically. Continuous forced circulation is maintained throughout the event. The RCS pressure is at or above the initial value prior to reactor trip (Figure 4-2). The core inlet temperature stays the same or slightly decreases prior to reactor trip due to the cooling effect of the HPI (Figure 4-8). The reactor power does not change significantly prior to reactor trip (Figure 4-1). The combined behavior of the flow, pressure, temperature and power suggest that DNBR would be the same or increasing during the inadvertent ECCS actuation event before and after reactor trip. Therefore, the inadvertent ECCS event is bounded by the loss of coolant flow events with respect to DNBR.

Based on the above discussion, all acceptance criteria are met for the TH evaluation of the inadvertent ECCS event summarized in this document.

Figure 4-1 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

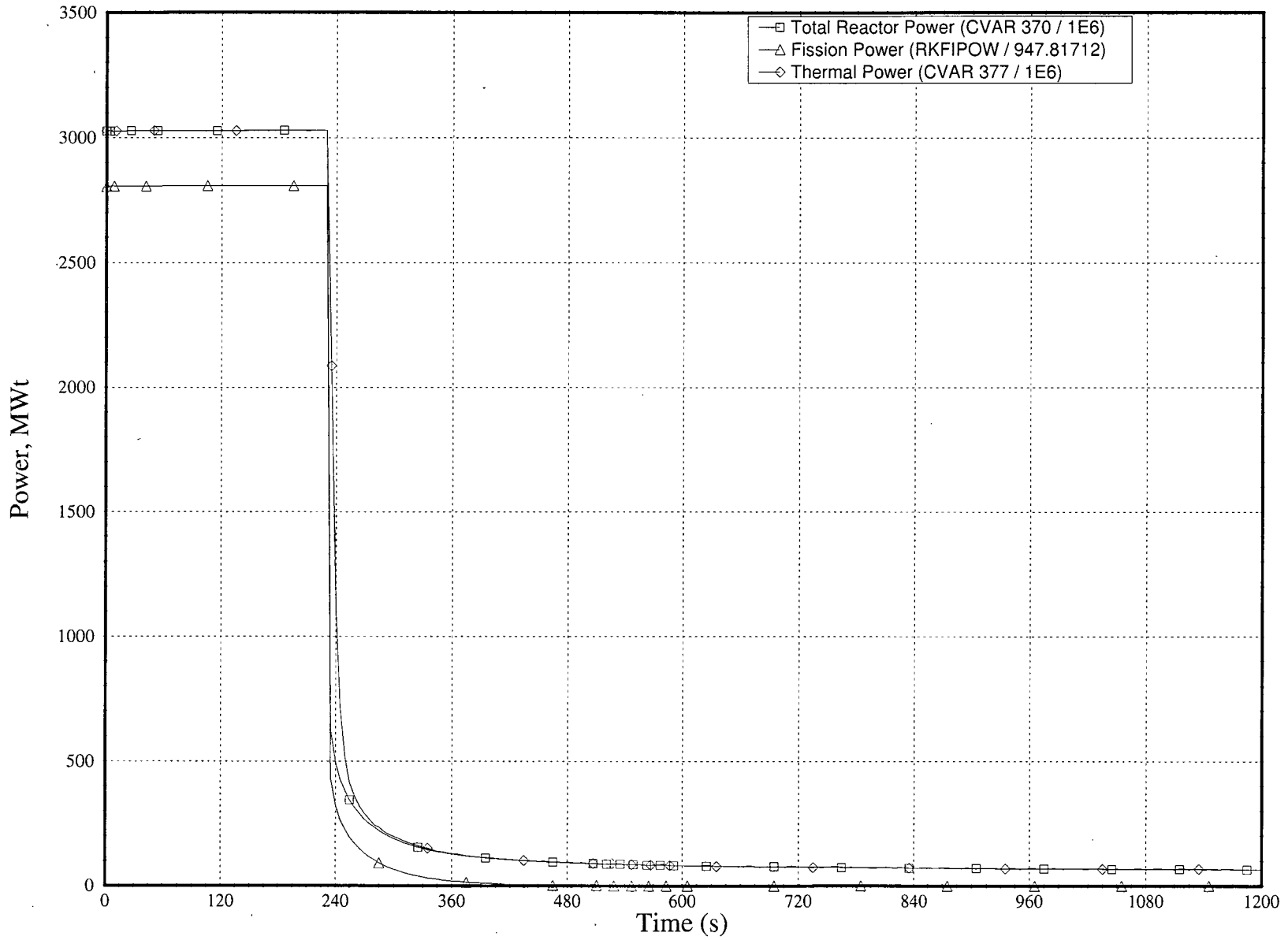


Figure 4-2 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

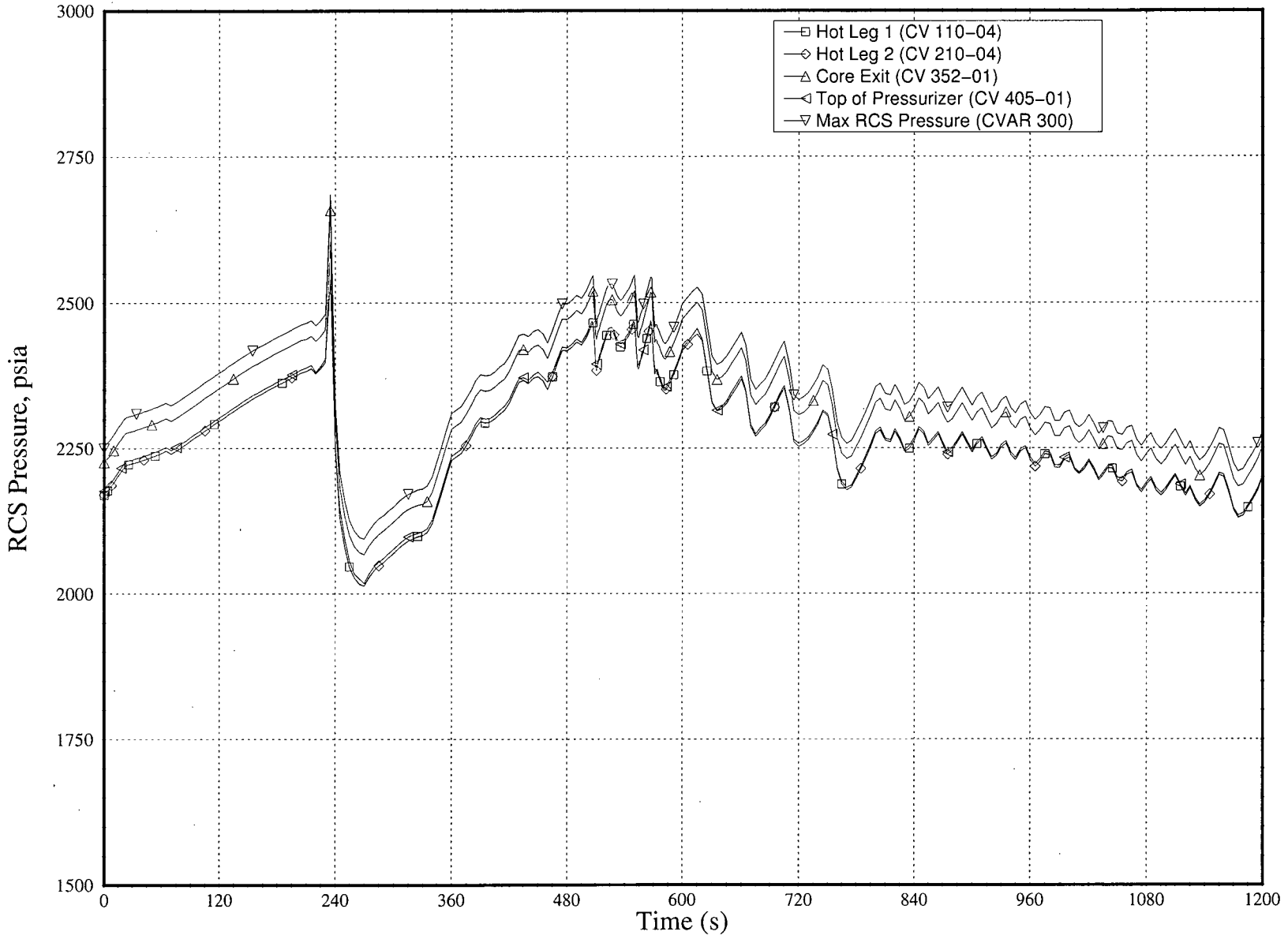


Figure 4-3 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

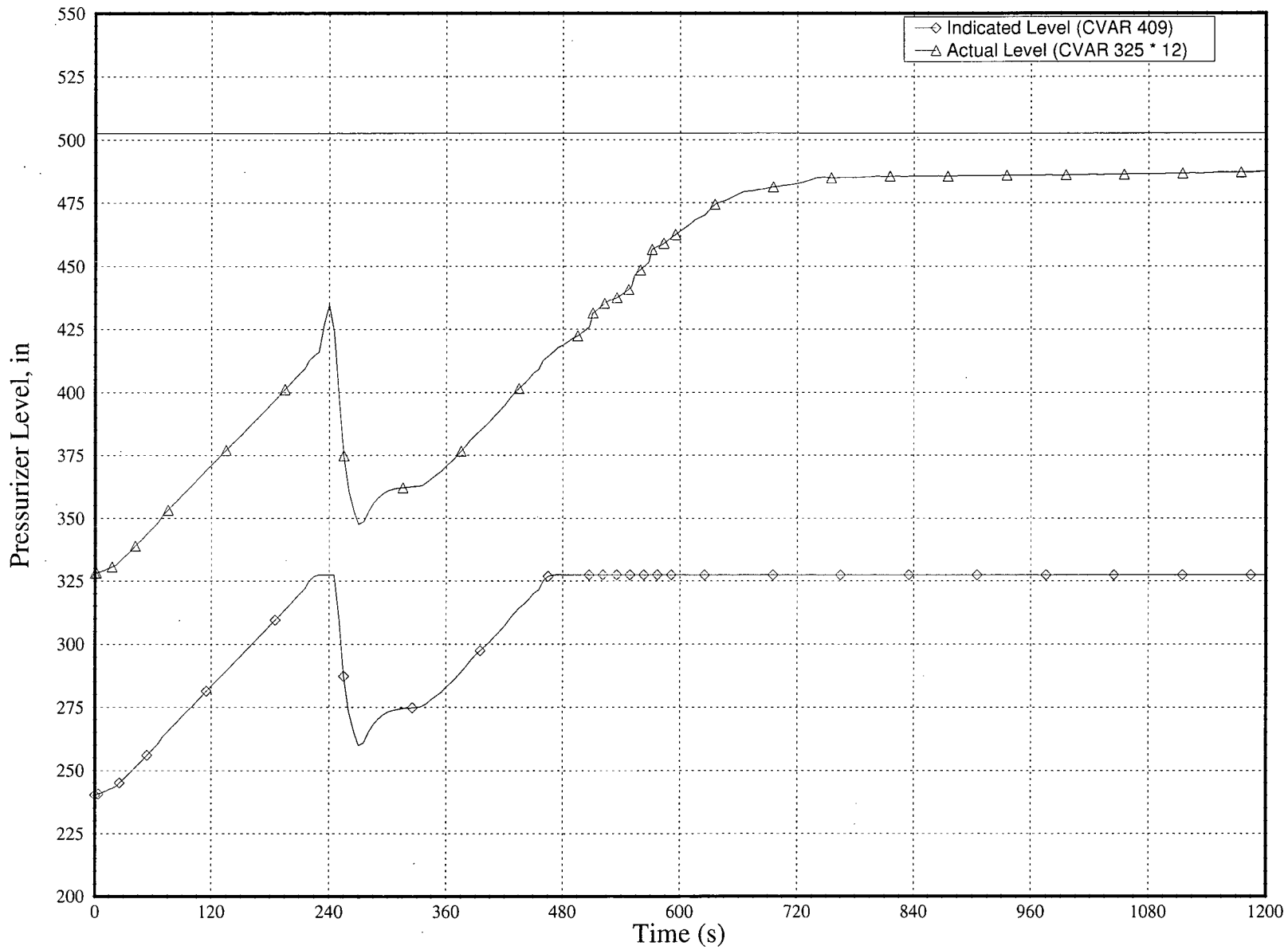


Figure 4-4 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

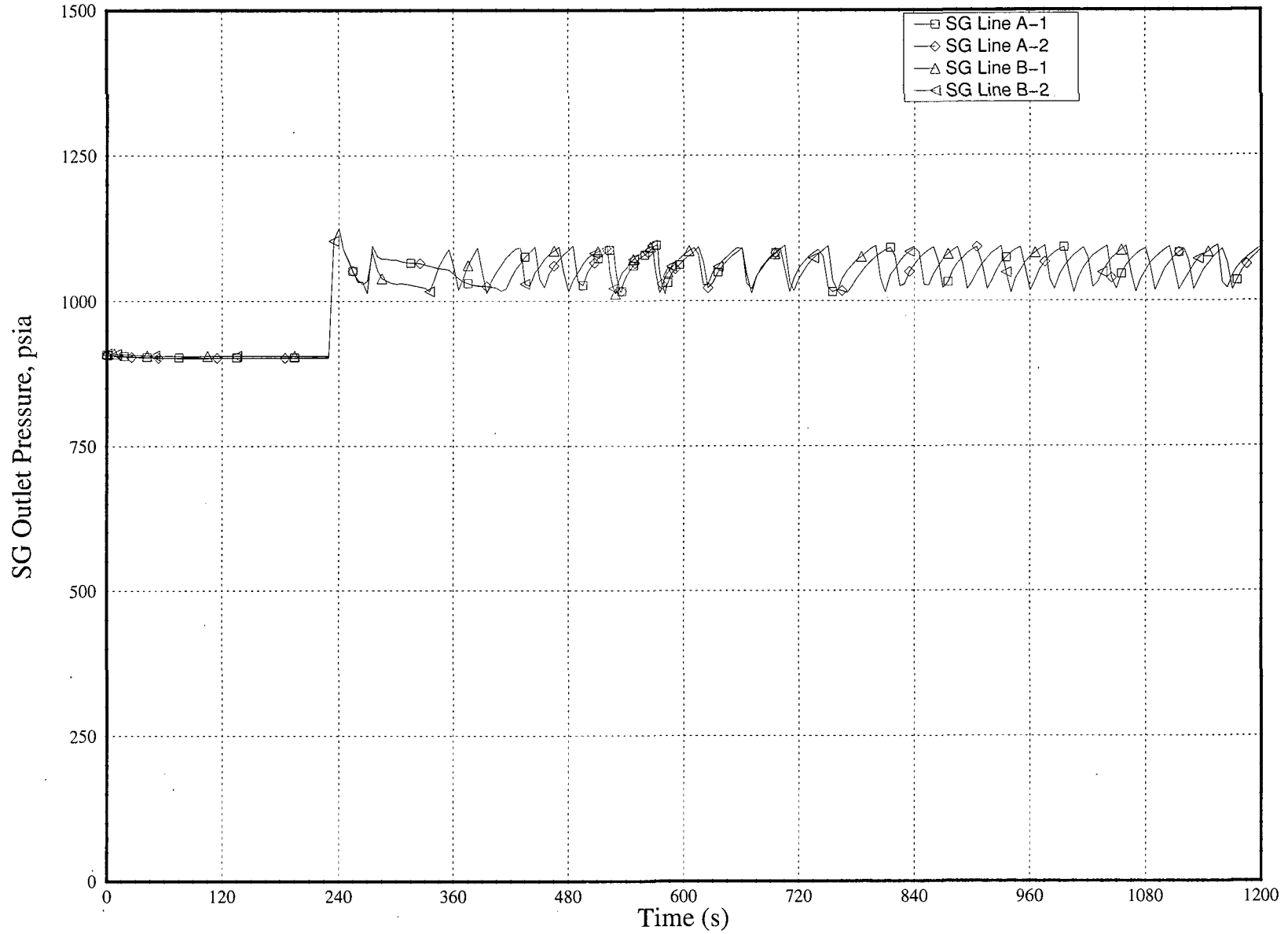


Figure 4-5 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

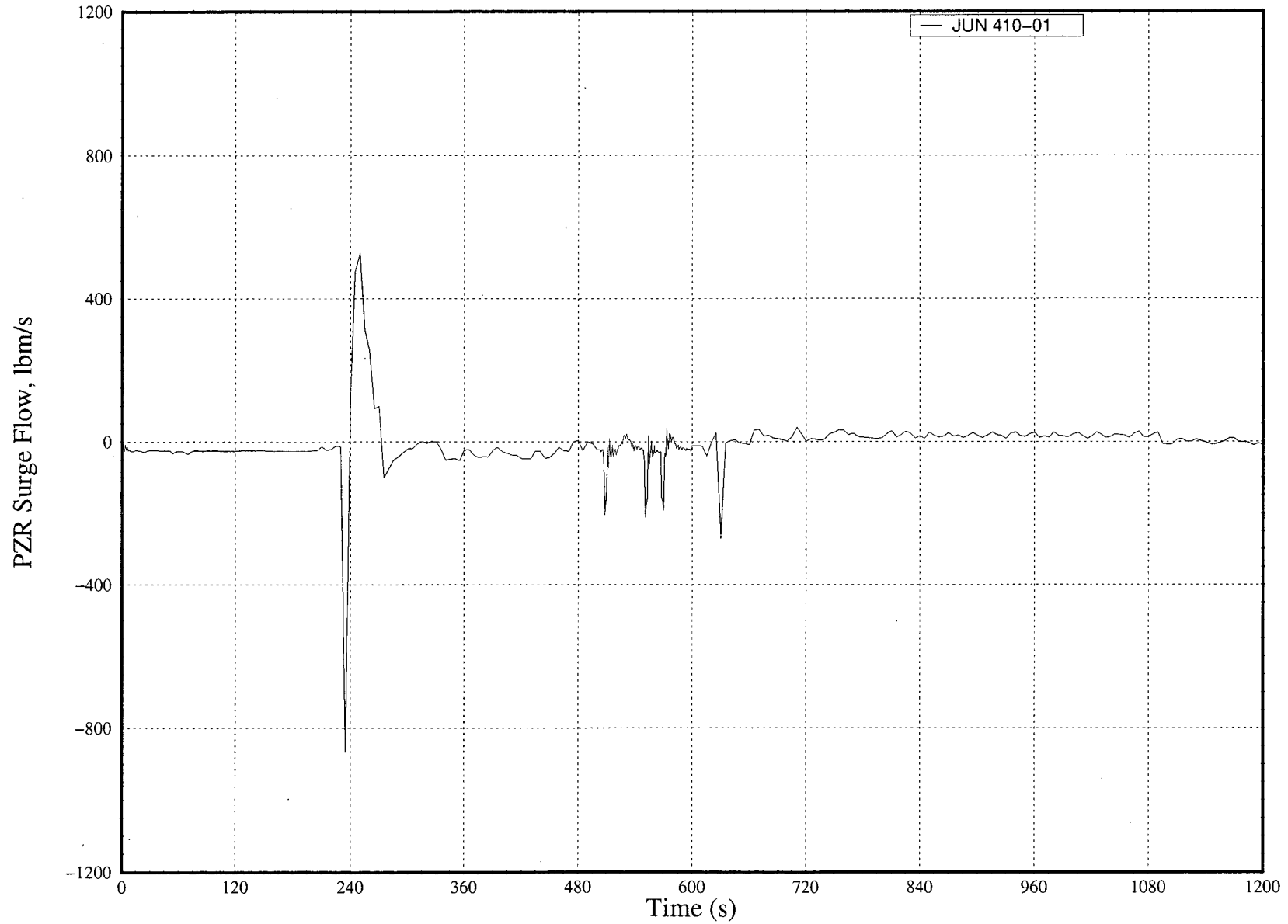


Figure 4-6 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

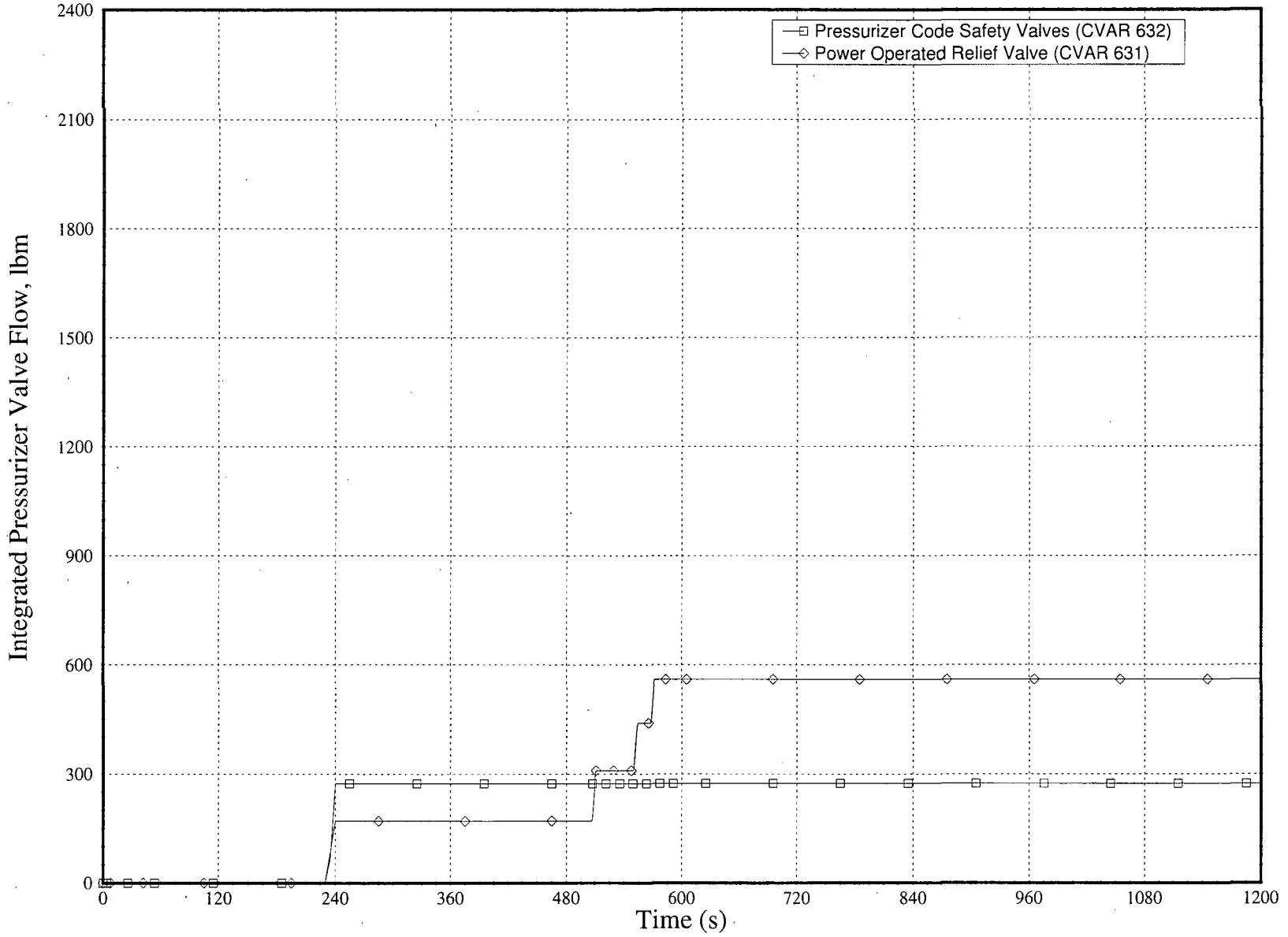


Figure 4-7 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

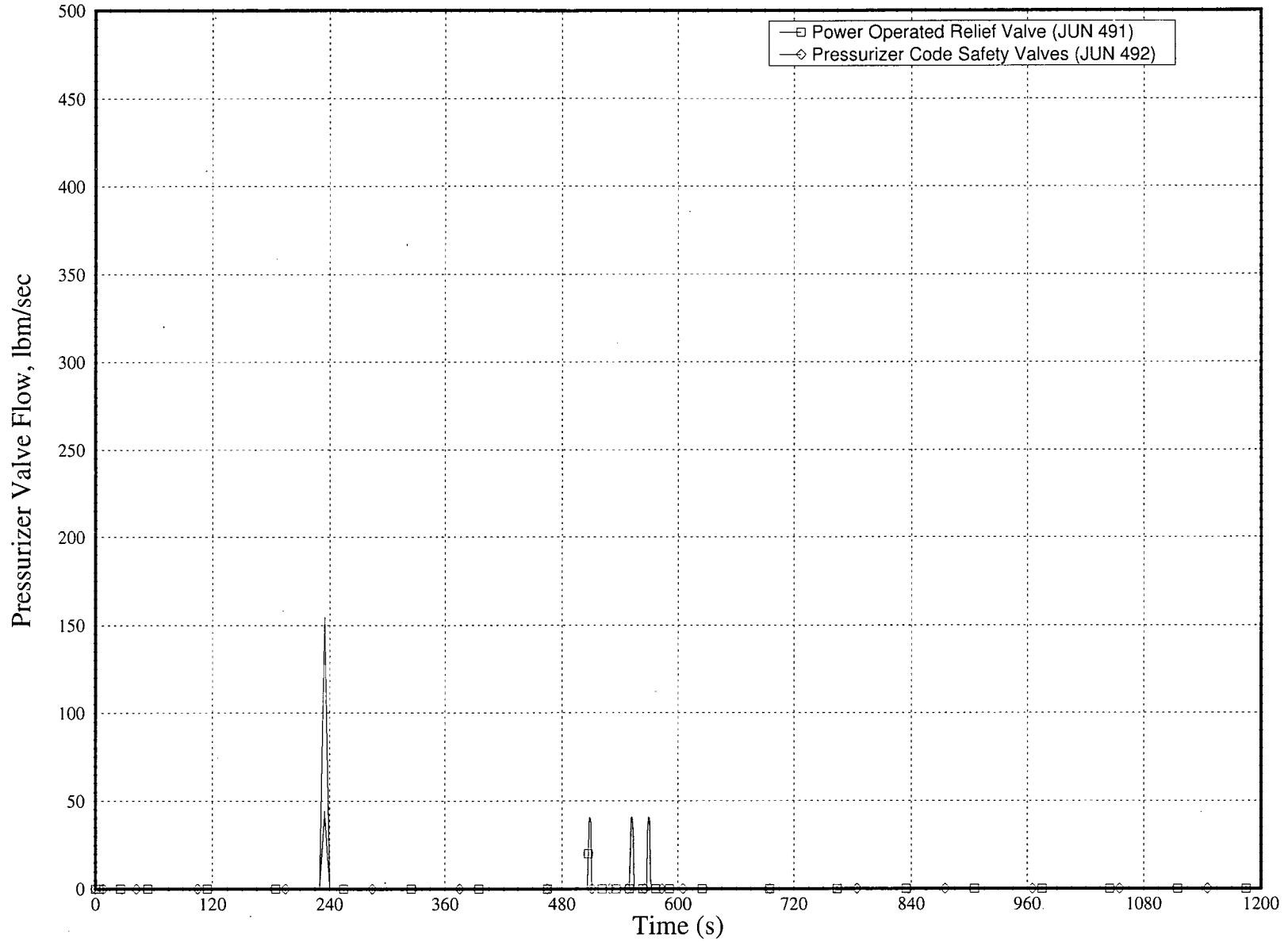


Figure 4-8 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

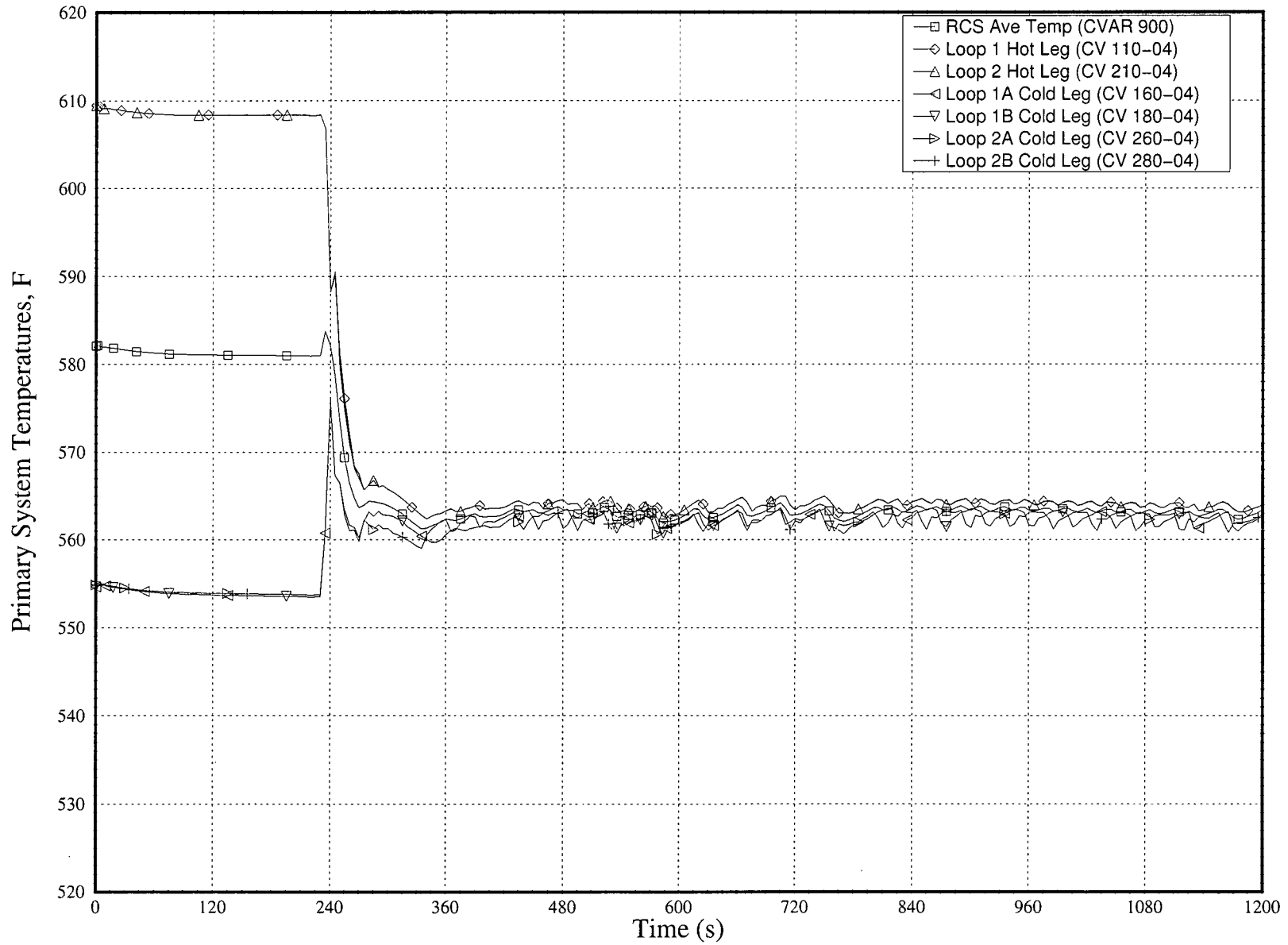


Figure 4-9 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

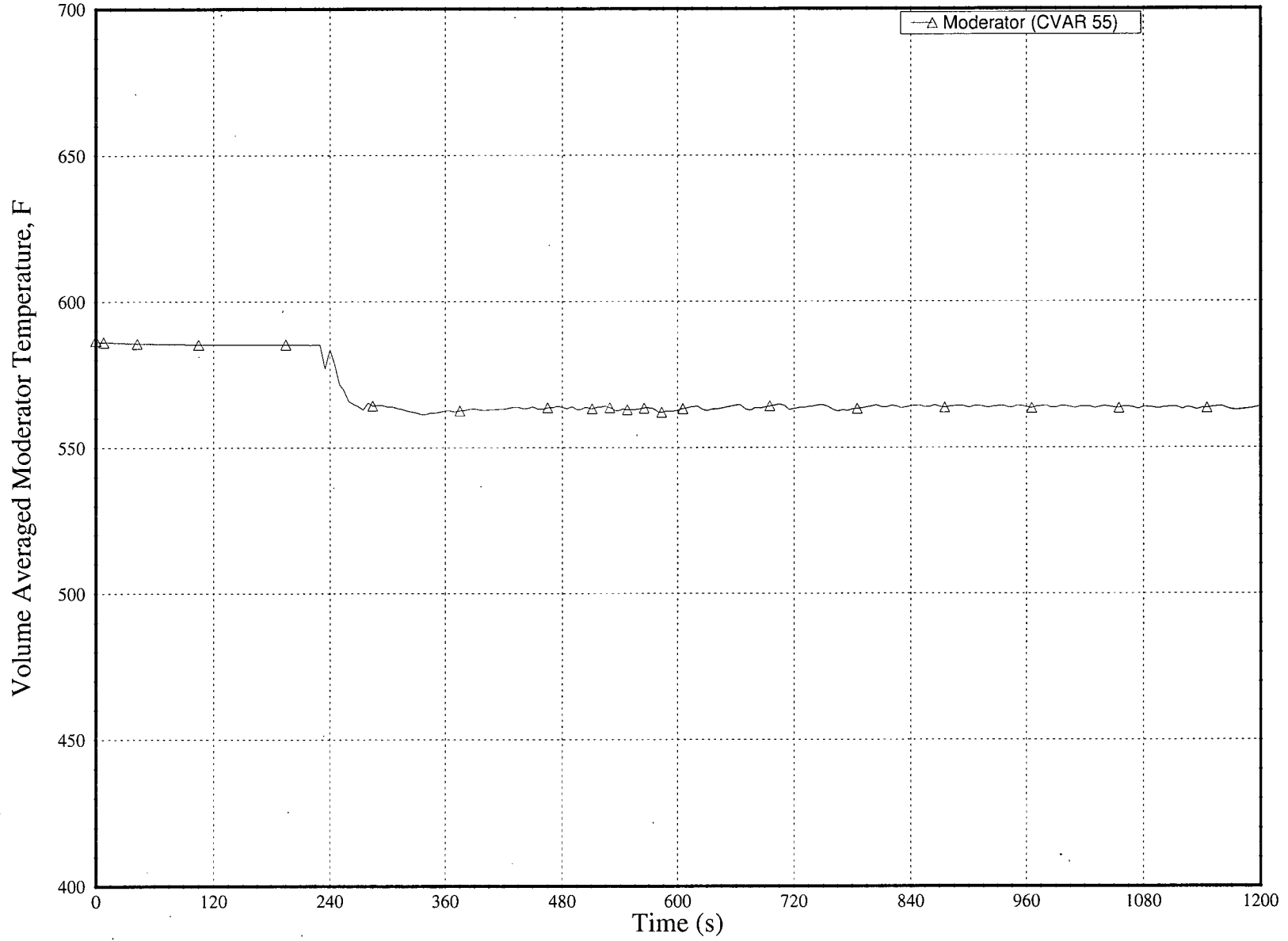


Figure 4-10 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

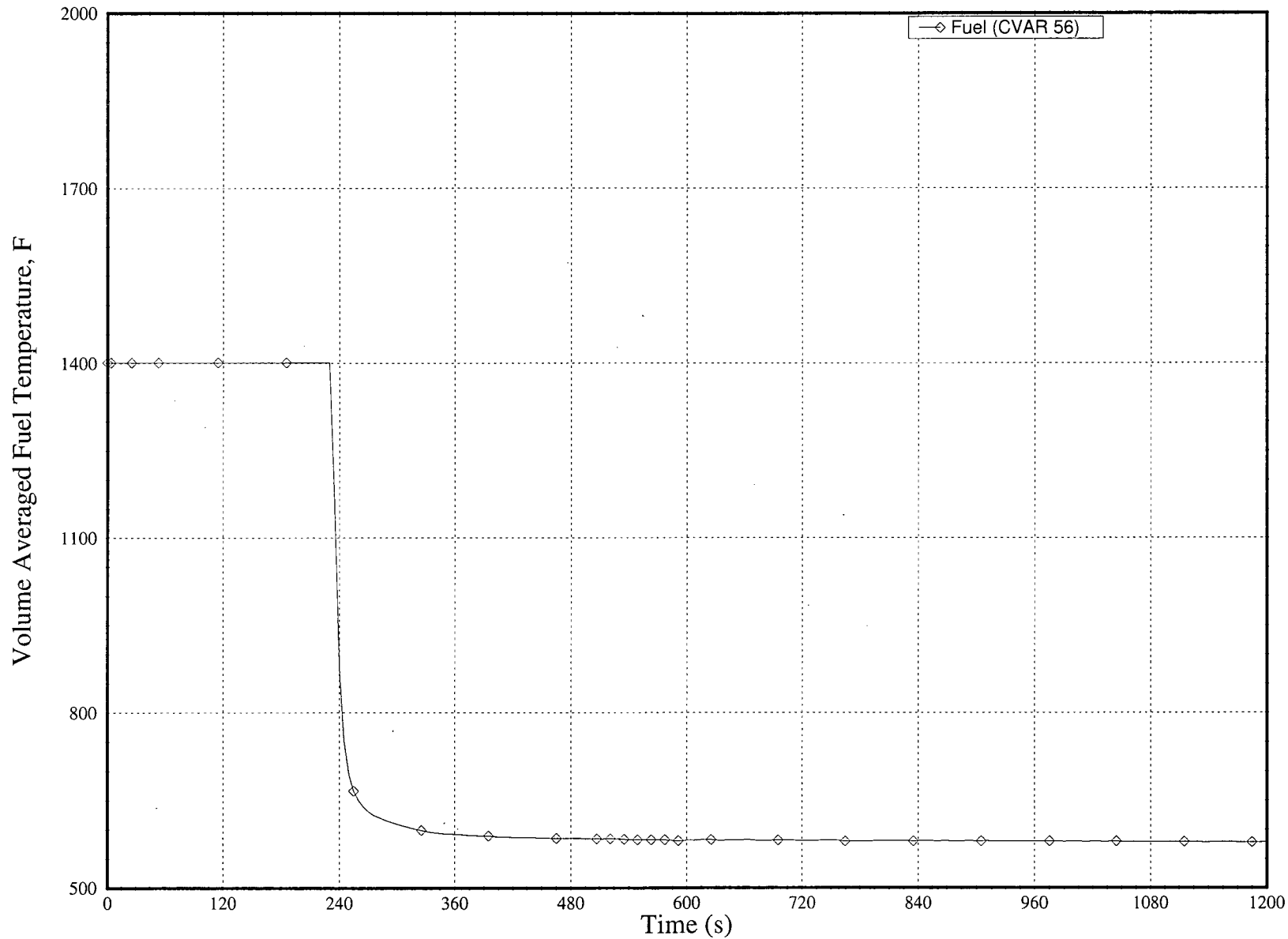


Figure 4-11 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

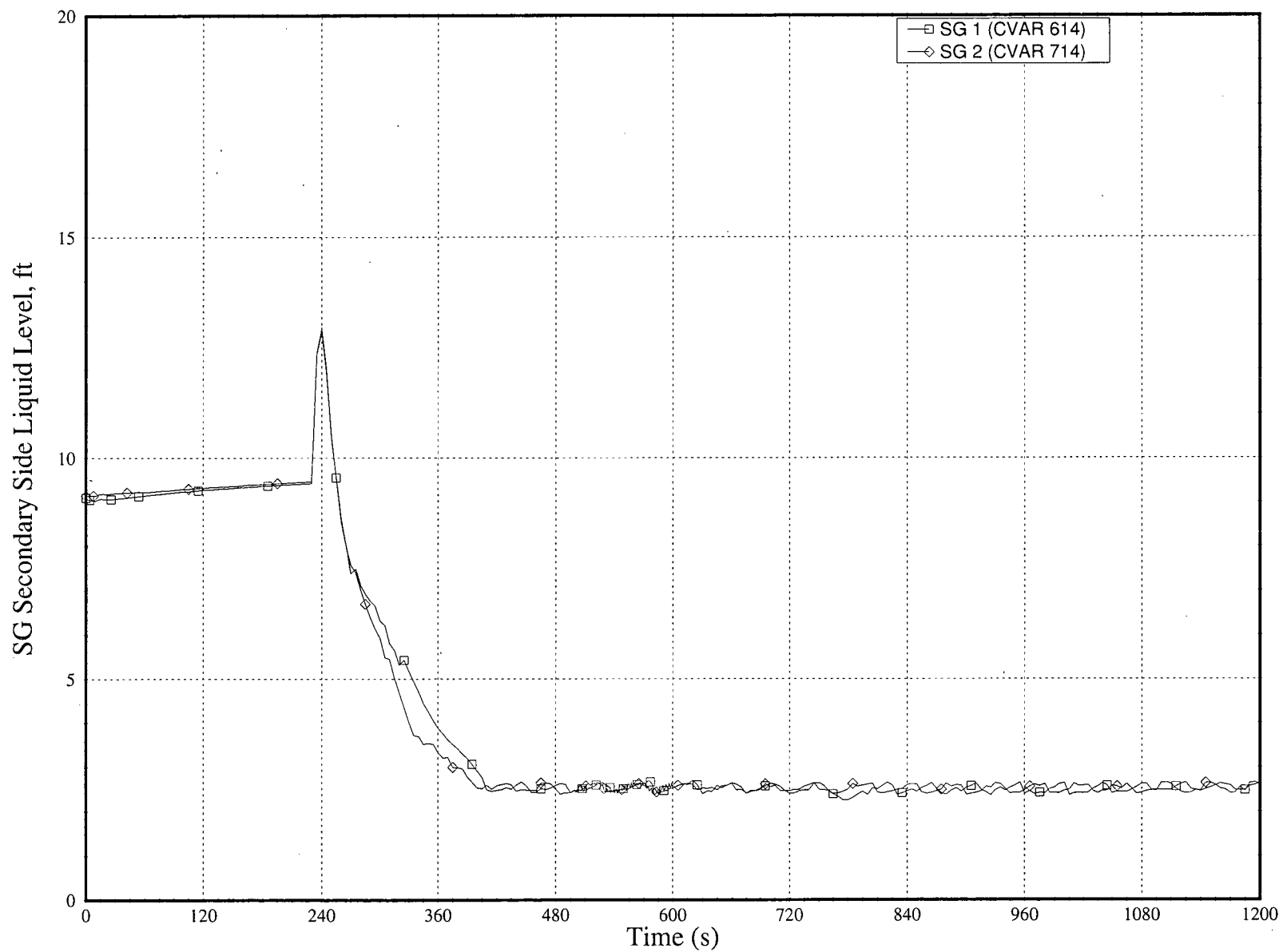


Figure 4-12 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

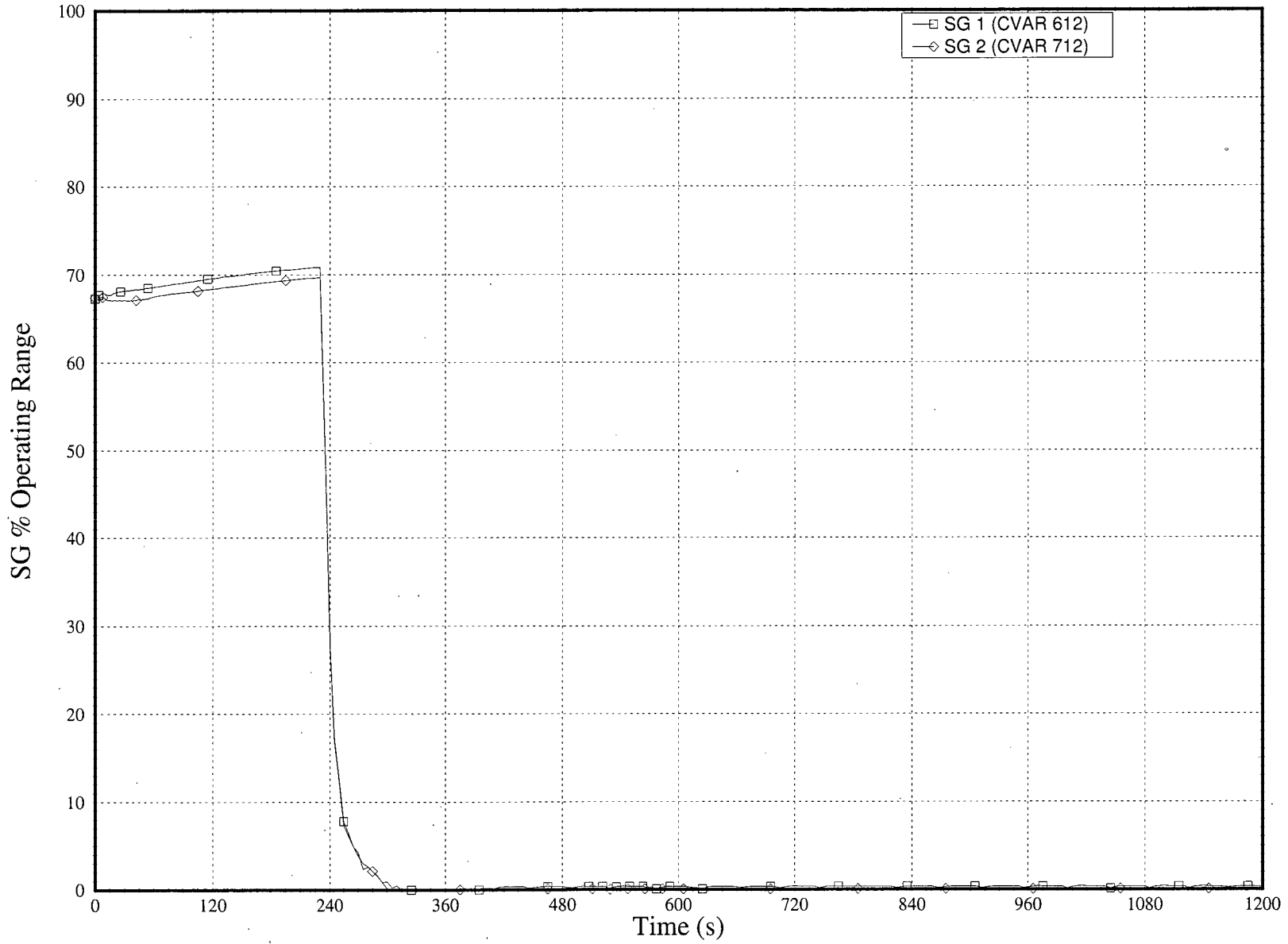


Figure 4-13 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

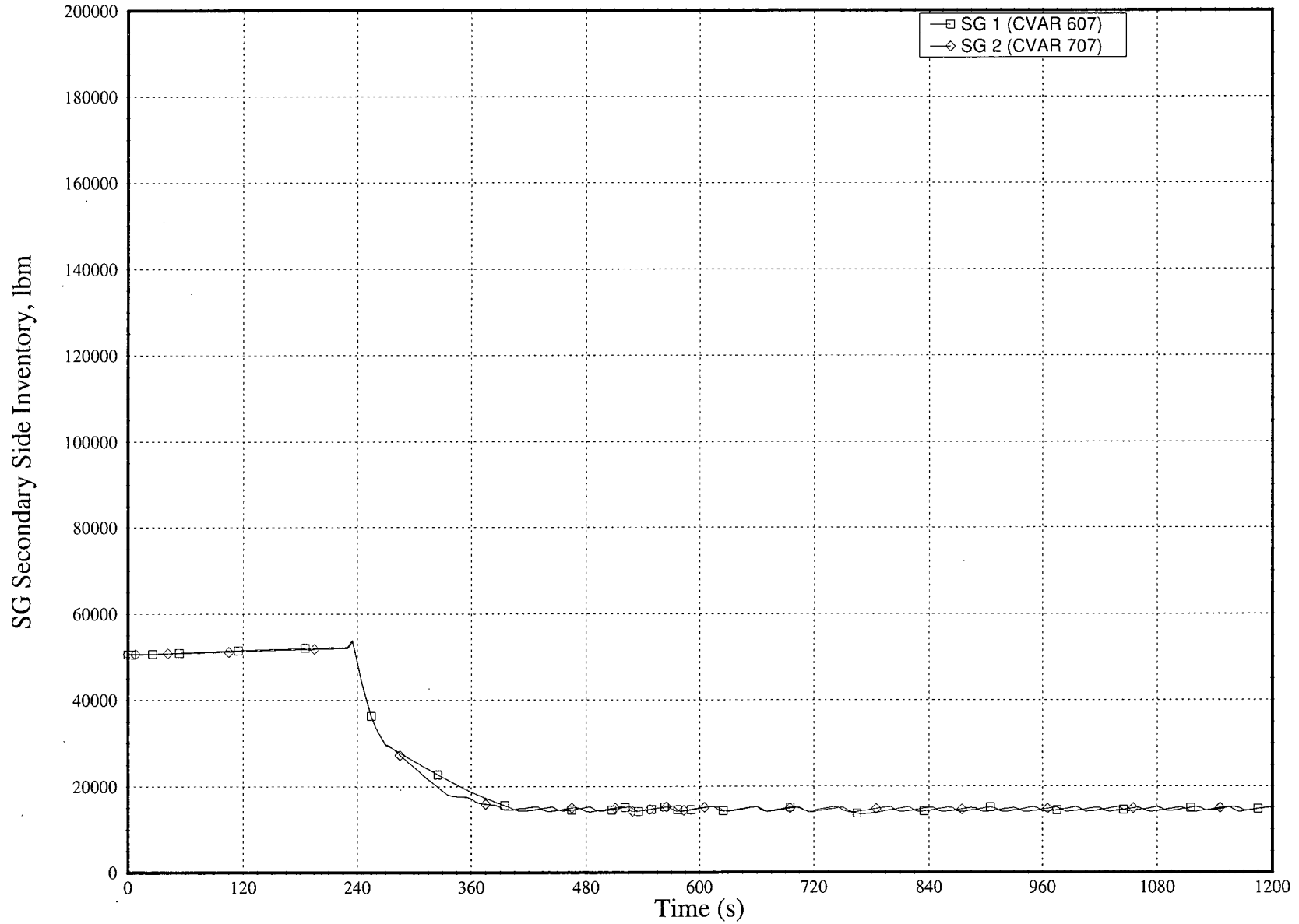


Figure 4-14 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

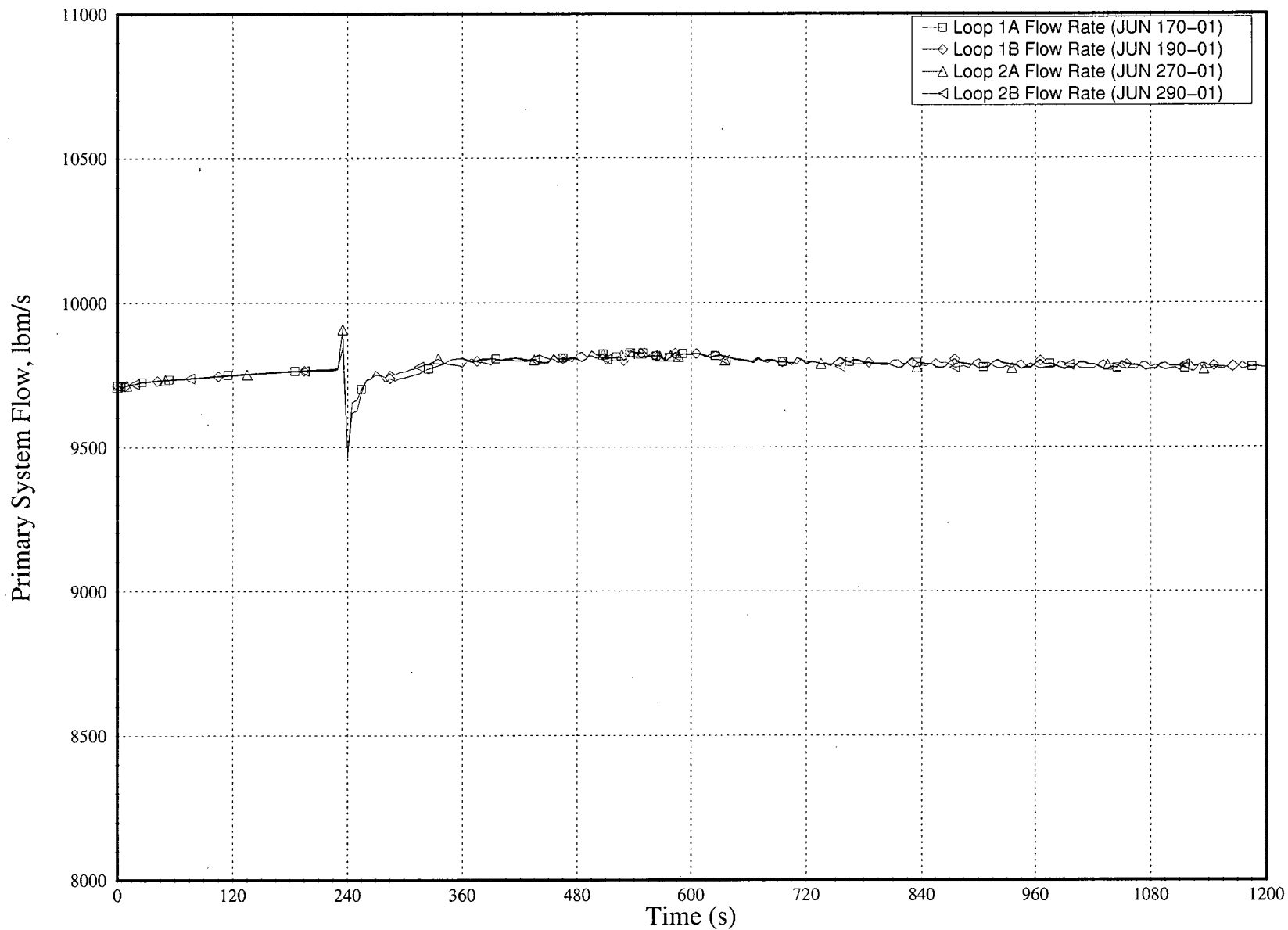


Figure 4-15 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

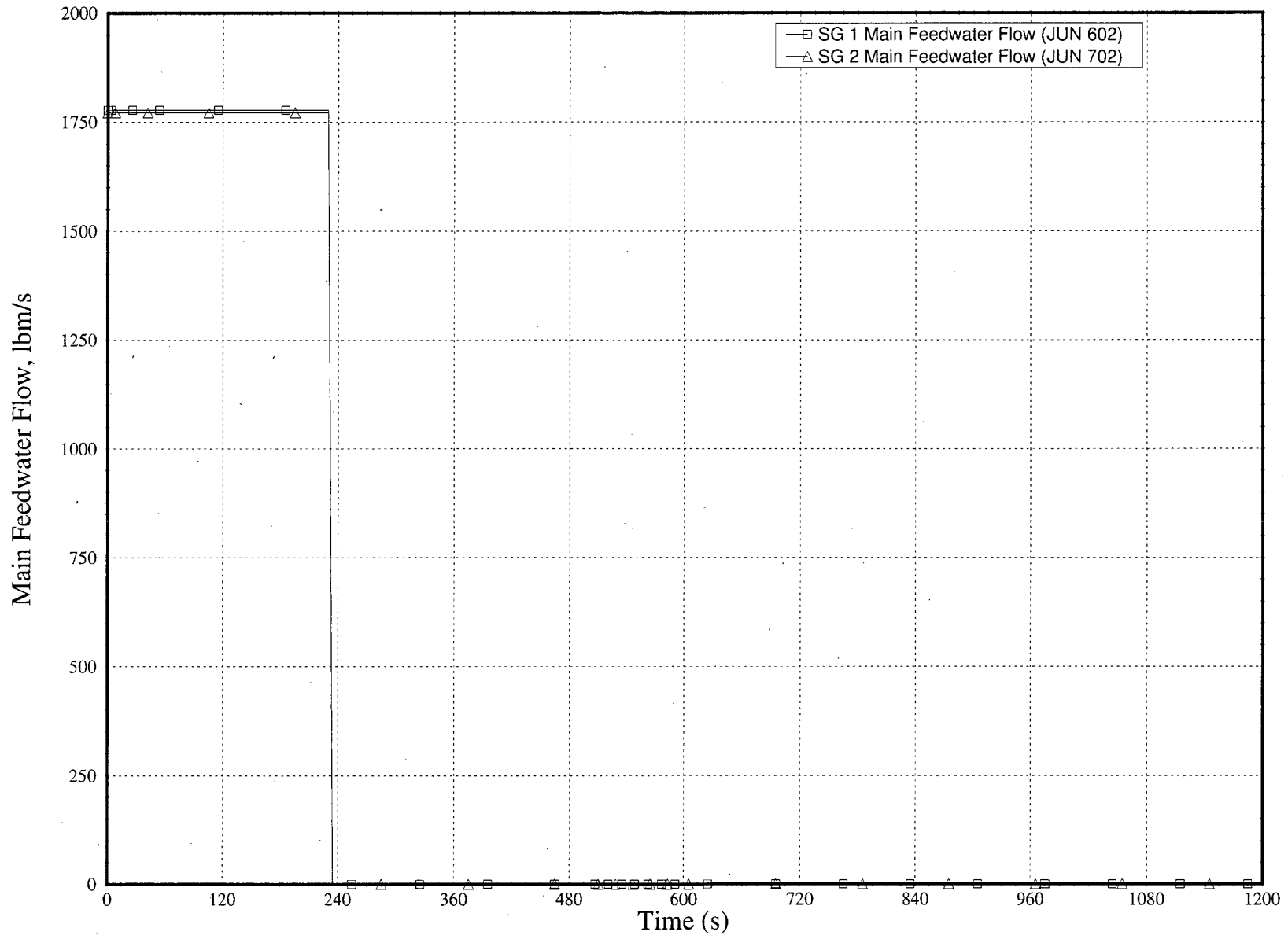


Figure 4-16 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

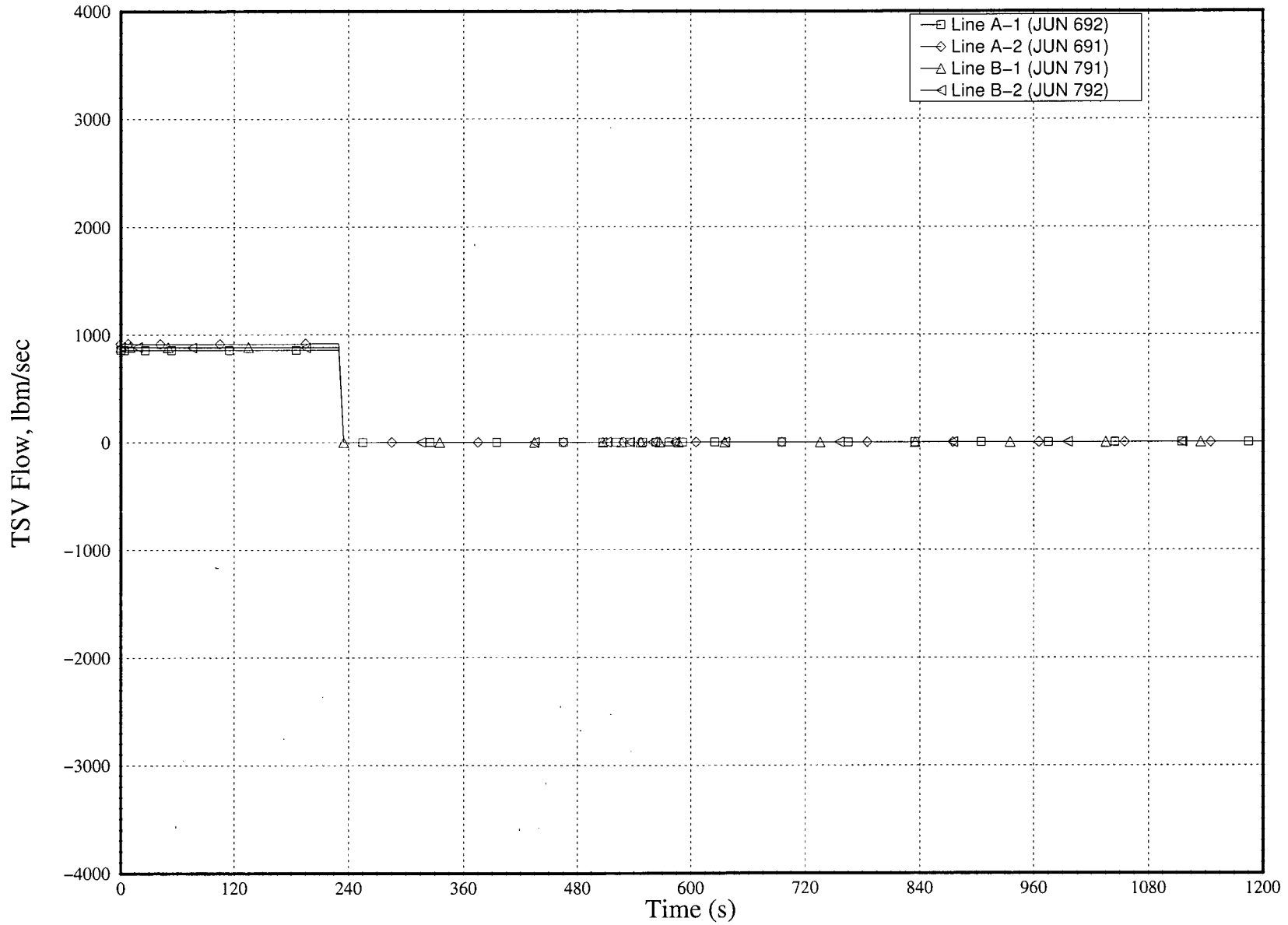


Figure 4-17 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

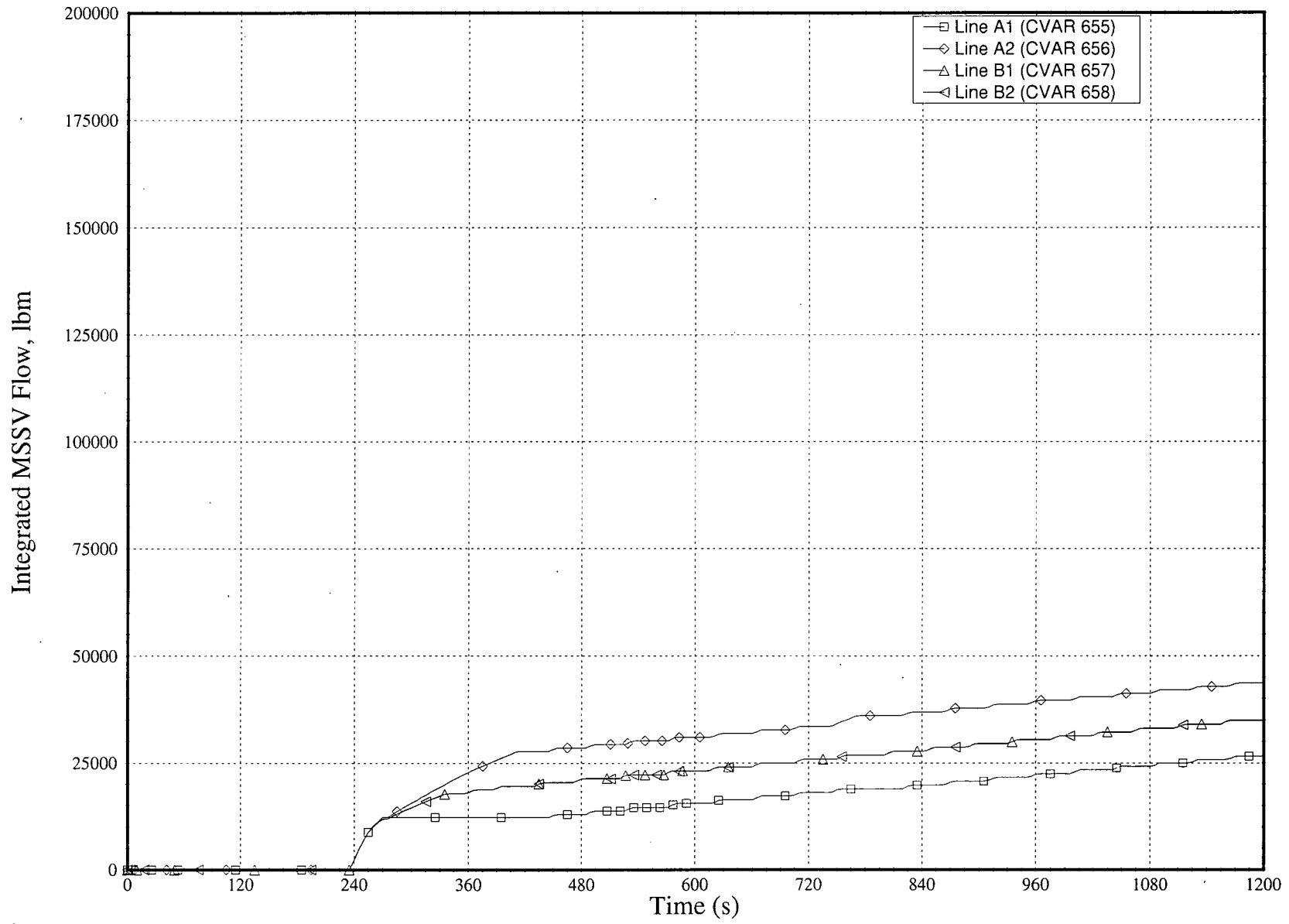


Figure 4-18 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

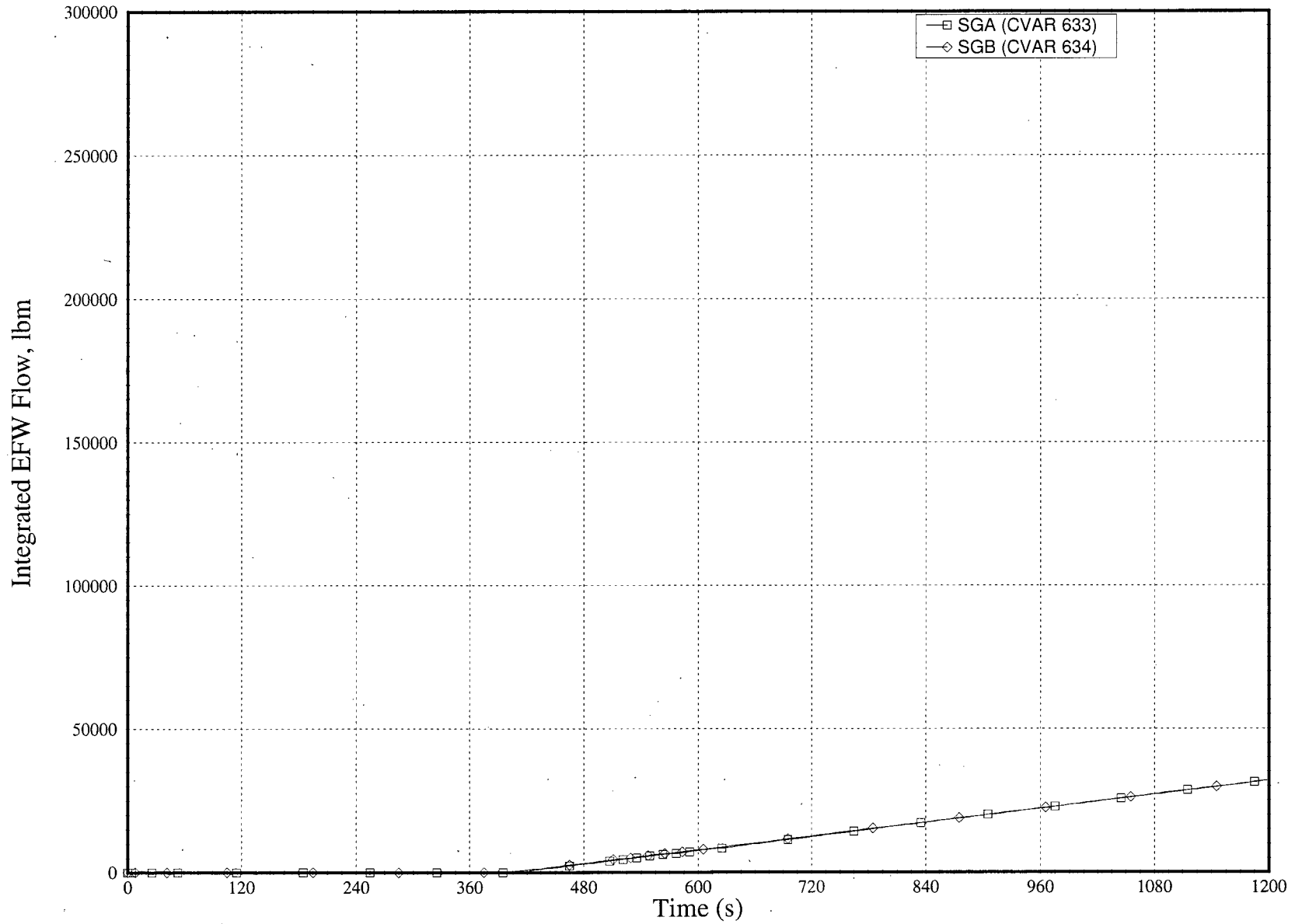


Figure 4-19 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

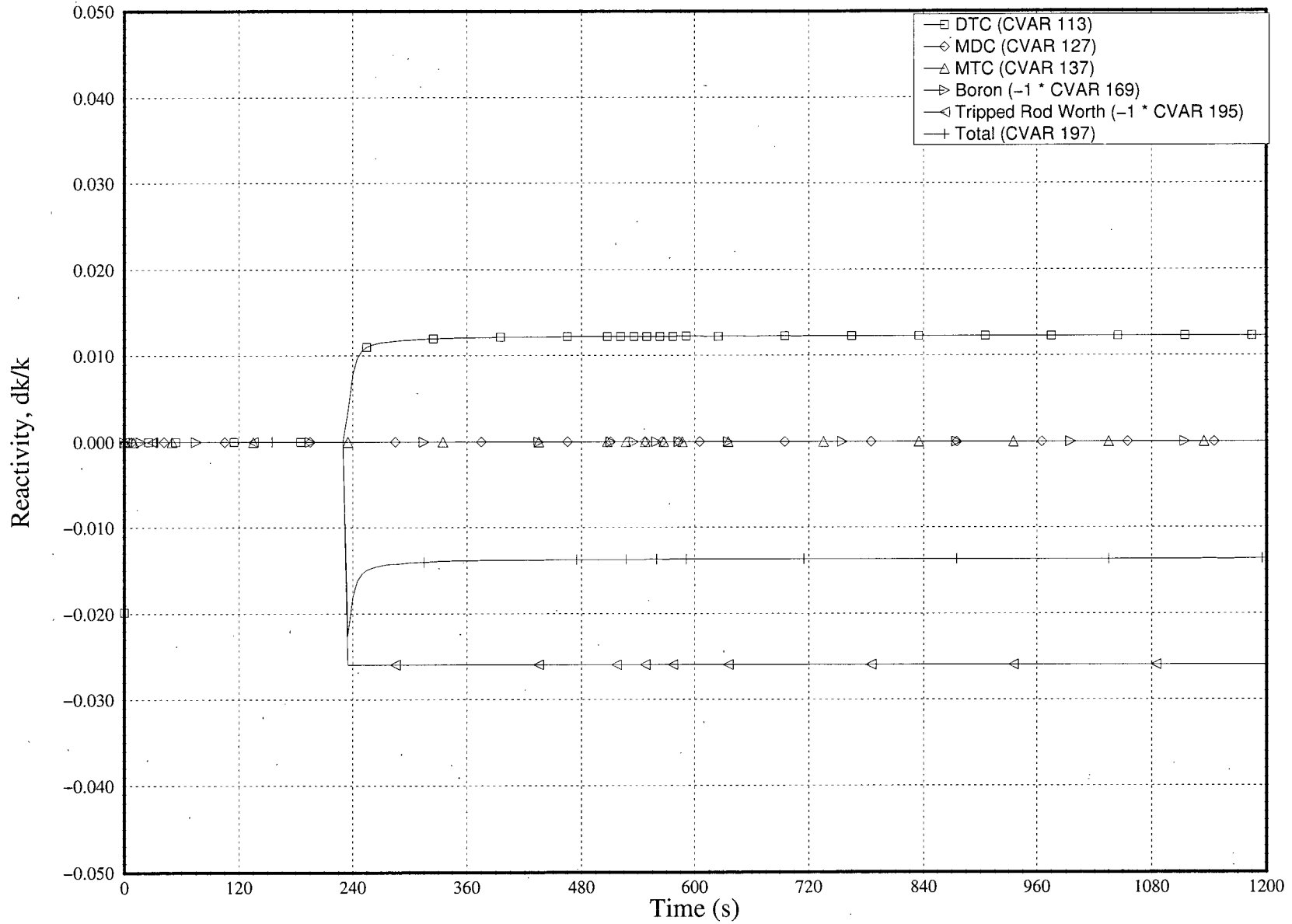


Figure 4-20 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

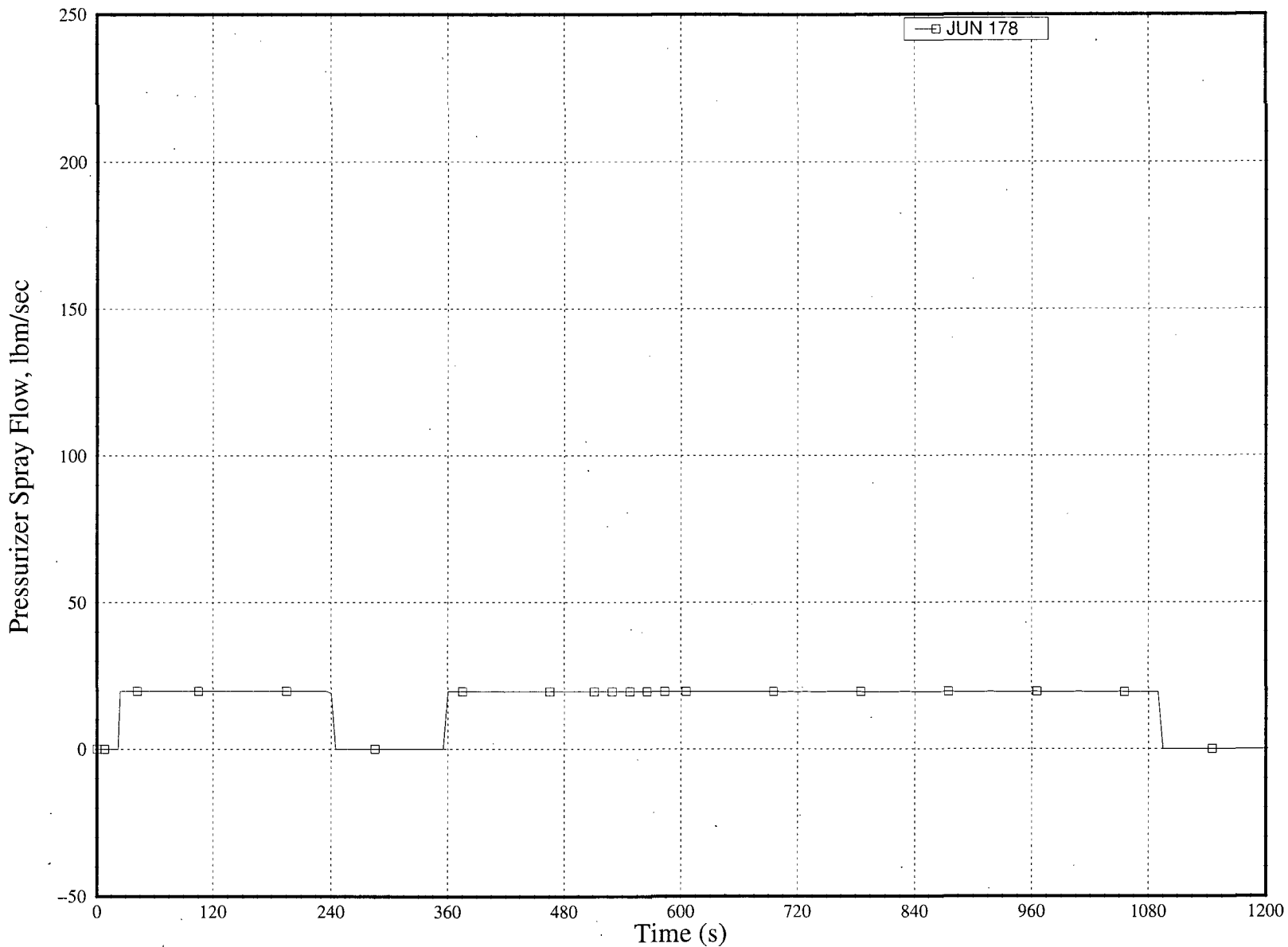


Figure 4-21 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

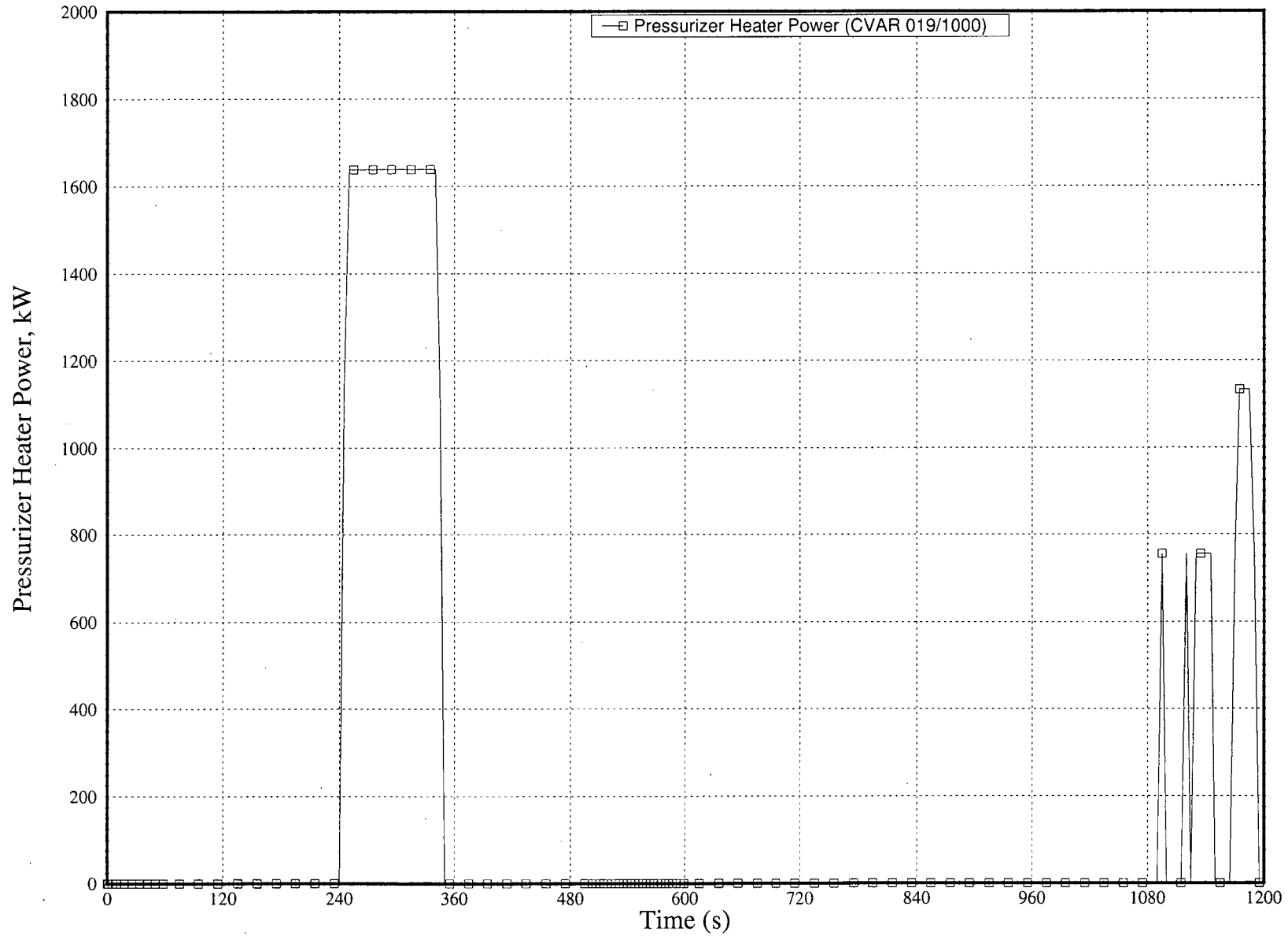


Figure 4-22 CR-3 EPU TH Analysis of Inadvertent ECCS
HPI Isolation at 10 Minutes

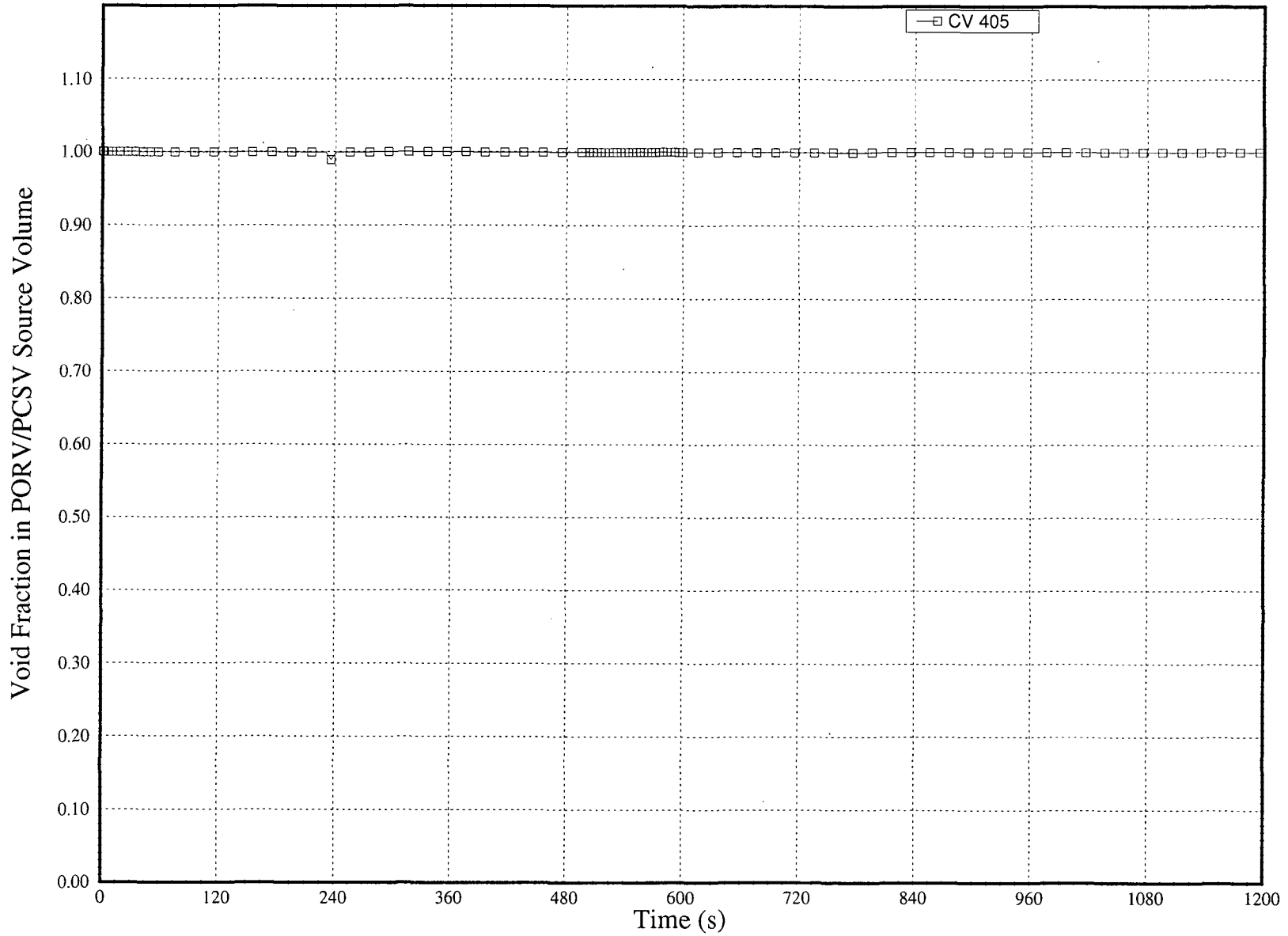
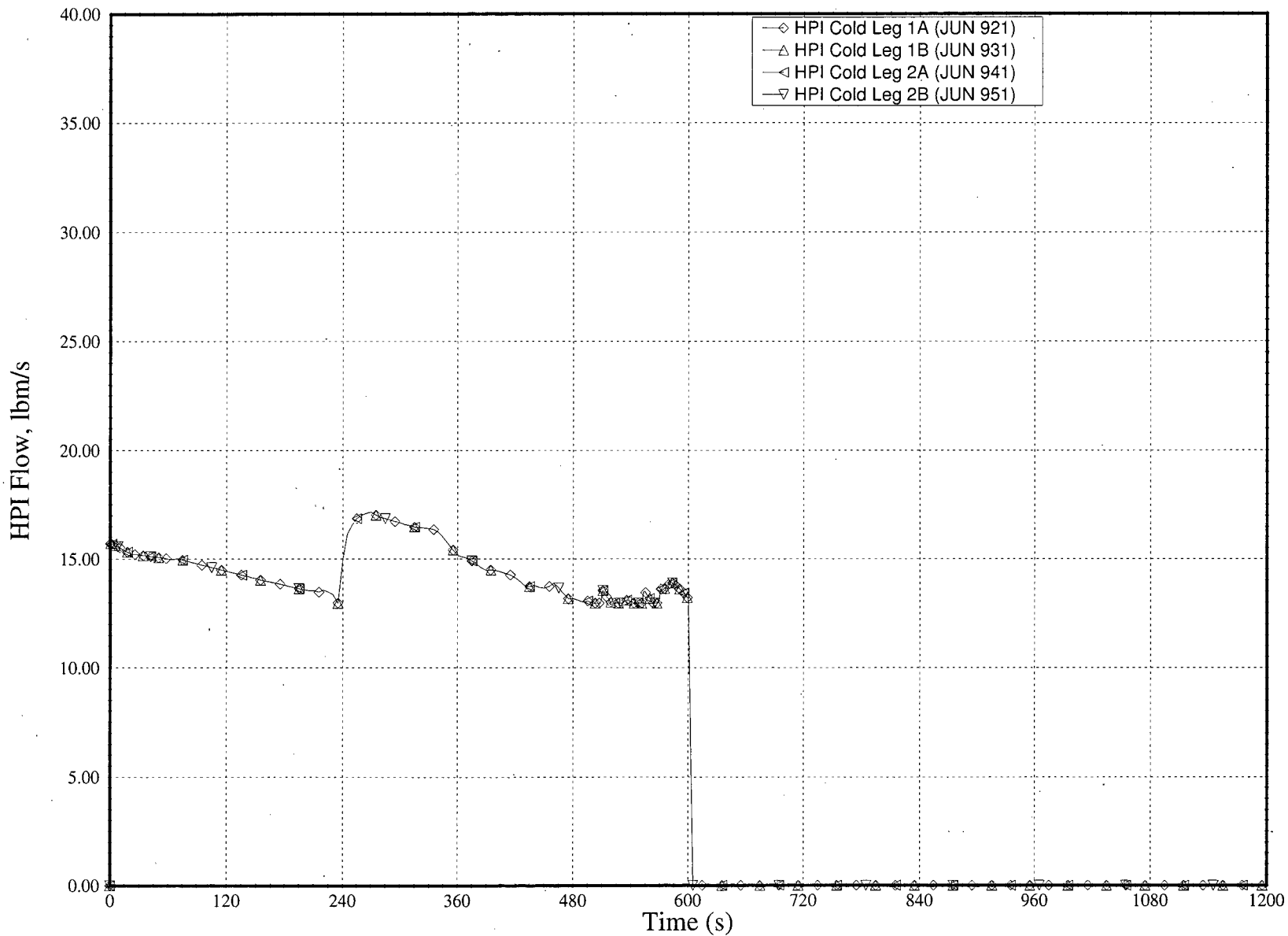


Figure 4-23 CR-3 EPU TH Analysis of Inadvertent ECCS

HPI Isolation at 10 Minutes





5.0 REFERENCES

1. AREVA Document 43-10164PA-06, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."
2. AREVA Document 43-10193PA-00, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurizer Water Reactors."
3. NRC Document NUREG 0800 Section 15.5.1 – 15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory," Revision 2, March 2007.
4. NRC Document RIS 2005-29, "Anticipated Transients that Could Develop into More Serious Events," Dec. 14, 2005.
5. AREVA Document 32-9166984-001, "CR-3 EPU Inadvertent Operation of ECCS Event Analyses." |

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

ENCLOSURE 4

**MARKUP PAGES OF ATTACHMENT 1 OF CR-3 EPU LAR
#309, REVISION 0**

- Credit the new LPI System HLI line as the primary method of boron precipitation mitigation during a LOCA, thereby eliminating the need for the two active methods currently credited and allowing deletion of CR-3 OL License Condition 2.C.11;
- Credit the use of the LPI System cross-tie line in the mitigation of a core flood tank (CFT) line break accident;
- Credit the use of soluble boron in the spent fuel pool to preclude spent fuel pool criticality accidents as allowed by 10 CFR 50.68(b)(4); and
- Revise the RCS pressure acceptance criterion for feedwater line break (FWLB) accident consistent with the criterion in NUREG-0800, "Standard Review Plan" (SRP), Section 15.2.8, "Feedwater System Pipe Break Inside And Outside Containment (PWR)," Revision 2 (March 2007).

- Approve for use an RCS pressure acceptance criterion for the Rod Ejection Accident (REA) consistent with the criterion in SRP, Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Revision 3 (March 2007).

3.0 BACKGROUND

CR-3 proposes to operate the reactor at 3014 MWt, an approximate 15.5% increase from the current authorized OL reactor core power level. CR-3 has defined this power increase as an EPU consistent with the guidance of the Office of Nuclear Reactor Regulation's Review Standard (RS)-001, "Review Standard for Extended Power Uprates," Revision 0.

Florida Power Corporation (FPC) has evaluated the proposed EPU for the applicable systems, structures, components, and safety analyses at CR-3 in accordance with the guidance of RS-001. The results of these evaluations are described in the CR-3 EPU TR (Attachments 5 and 7). The EPU TR provides the technical details that support the requested OL and ITS changes and, in concert with the other attachments, provide a comprehensive evaluation of the effects of the proposed EPU on CR-3 plant operation.

4.0 JUSTIFICATION FOR THE CHANGE

The acceptability for each proposed OL and ITS change is addressed in Tables 1 and 2 in Section 2.0 above. Additional details and justification are provided below for the additional licensing basis items. The EPU TR summarizes the evaluations performed to assure acceptable unit operation at EPU conditions and is therefore referenced throughout this section as additional technical justification for the EPU related changes. Additional detail is also provided for the plant modifications in EPU TR Appendix E, "Major Plant Modifications."

4.1 ICCMS, ADVs and FCS

CR-3 requests approval to credit the use the ICCMS and the FCS function of the ADVs to assure adequate core cooling during a SBLOCA at EPU conditions. The FCS is assumed in the SBLOCA analyses to open both ADVs to depressurize the secondary plant and ultimately the RCS which will allow ECCS to provide adequate core cooling in the event of a single failure of an HPI train as summarized in EPU TR Section 2.8.5.6.3, "ECCS and Loss of Coolant

revise

4.5 Revise RCS Pressure Acceptance Criterion for FWLB Accident

CR-3 requests approval to ~~of~~ the RCS pressure acceptance criterion for the feedwater line break (FWLB) accident to be consistent with the criterion in SRP, Section 15.2.8, Revision 2.

The FWLB is considered a limiting fault event per the CR-3 FSAR, Section 14.2.2.9.2. The FSAR acceptance criteria for this limiting fault include an RCS pressure criterion of 110% of design pressure (2750 psig). For EPU conditions, CR-3 requests revising the FWLB accident acceptance criterion to reflect an RCS pressure limit of 120% of design pressure (3000 psig). This is consistent with acceptance criteria specified in SRP, Section 15.2.8 for the FWLB event. EPU TR Section 2.8.5.2.4, "Feedwater System Pipe Breaks Inside and Outside Containment," provides further discussion related to revising the FWLB accident RCS pressure acceptance criterion and the associated plant impact as a result of EPU.

4.6 Conclusion

7

The CR-3 EPU has been evaluated for impact on the plant and the OL. With implementation of plant modifications described in the EPU TR, analyses demonstrate that the plant will maintain the capability for safe operation and the mitigation of postulated accident scenarios at EPU conditions. The proposed Technical Specification changes will assure that the plant and its SSC remain within acceptable limits at all times. Procedure changes and training will be implemented prior to plant startup under EPU conditions and will assure that plant personnel are capable of operating the plant and responding to abnormal or emergency conditions.

Insert New 4.6

Insert New 4.6

4.6 Revise RCS Pressure Acceptance Criterion for REA

CR-3 requests approval to use an RCS pressure acceptance criterion of 3200 psig for the CR-3 REA as a method of ensuring maximum rod worth during an REA will not lead to further coolant boundary rupture, the applicable CR-3 design criterion. Currently, the CR-3 FSAR does not include an explicit RCS pressure criterion for the REA. Rather, the licensing basis acceptance criteria for the REA include a fuel enthalpy of 280 cal/gm and the amount of thermal energy release necessary to cause reactor pressure vessel deformation; 3.37×10^8 cal. For EPU conditions, CR-3 requests the use of an REA acceptance criterion of 3200 psig, which is less than the value that would result in stresses that exceed the Babcock & Wilcox (B&W) "Service Limit C" as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and meets the related acceptance criterion specified in SRP, Section 15.4.8 for the REA event. Maintaining RCS peak pressure during an REA less than that required to exceed the ASME Service Level C stress limits will ensure the thermal energy release necessary to deform the RPV (i.e., local vessel yielding) is not reached. The peak RCS pressure to preserve the Service Level C stress limit is conservatively considered to be 3200 psig for B&W plants, including CR-3. Previous comprehensive assessments and reviews confirming the structural adequacy of the reactor coolant pressure boundary (RCPB) during a postulated ATWS event determined that a majority of RCPB components could withstand RCS pressures up to 3750 psig using ASME Emergency (Service Level C) stress levels with the limiting components identified as the CR-3 RCPs. Analysis results indicated approximately 6% of the RCP pressure boundary area exceeds the primary membrane Level C stress limit when subjected to an internal pressure of 3500 psig. However, this overstress condition was determined to cause limited localized distortion and subsequent analysis indicated that it would not affect the Operability of the RCP. Based on these evaluations, 3200 psig is considered an acceptable RCS pressure limit for the CR-3 REA.

Adding an RCS pressure acceptance criterion of 3200 psig is considered acceptable since the REA is considered an American Nuclear Society (ANS) Condition IV event. ANS Condition IV events are defined as those events not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Since an REA is a highly unlikely event, adding an RCS pressure acceptance criterion that complies with the SRP acceptance criteria, as required by 10 CFR 50.34(h), does not represent a significant risk to the health and safety of the public. Additionally, adding an RCS pressure acceptance criterion of 3200 psig will continue to ensure the thermal energy release necessary to deform the RPV is not reached thereby preserving the applicable CR-3 design criterion of ensuring maximum rod worth during a REA will not lead to further coolant boundary rupture.

of resulting in a limited amount of fuel cladding failure, but the analyses conclude that the plant remains within the acceptance criteria of 10 CFR 50.46. Large break LOCA scenarios also satisfy the criteria of 10 CFR 50.46 under EPU conditions, and the large break LOCA remains the most limiting accident at EPU conditions. LOCA analyses indicate a small reduction in margin (< 5%) related to maximum local clad oxidation and hydrogen generation at EPU conditions and a slight improvement in peak clad temperature margin based on minor changes in LOCA design inputs and modifications that will enhance the plant response capabilities to LOCAs and improve the method for boron precipitation mitigation. The non-LOCA events identified in the CR-3 Final Safety Analysis Report were shown to meet existing acceptance criteria.

, with the exception of the feedwater line break (FWLB) and rod ejection accidents,

← Insert NSHC-3

The containment building response to mass and energy releases was evaluated under EPU conditions. The evaluations indicated that temperature and pressure limits were met.

No plant changes associated with the EPU reduce the degree of component or system redundancy. The small break LOCA response will require two High Pressure Injection (HPI) pumps injecting. In the event of inadequate HPI and loss sub-cooling margin (SCM), secondary depressurization will be achieved via a new ICCMS and FCS, which includes automatic actuation of the ADVs, thereby assuring that the reactor core receives the necessary ECCS flow to minimize core damage and satisfy the requirements of 10 CFR 50.46. The features required to automatically open ADVs during a small break LOCA are incorporated into proposed TS 3.3.19, "Inadequate Core Cooling Monitoring System (ICCMS) Instrumentation," and TS 3.3.20, "Inadequate Core Cooling Monitoring System (ICCMS)," and TS 3.7.20, "Fast Cooldown System (FCS)." The ICCMS and FCS are configured and supported such that a single failure will not prevent completion of the ECCS safety function.

To support this enhancement, operators will be provided indication of subcooling margin and HPI System flow adequacy to ensure actuation of the FCS. A new safety related display system will be available to determine when insufficient subcooling margin is available and HPI System flow is inadequate. Additional modifications will, upon indication of loss of subcooling margin, automatically trip the RCPs and raise steam generator secondary side level to the inadequate subcooling margin level. These automatic actions replace the current actions performed by the operator, thus reducing the reliance on manual operator action for event mitigation.

Operator training programs will be revised in accordance with the industry standard systematic approach to training process and appropriate training will be provided on all plant modifications, administrative/technical requirement changes, Technical Specification revisions, and procedure revisions. The CR-3 simulator will be updated and tested in sufficient time to provide effective reinforcement of procedure and plant physical changes as well as build proficiency with required manual operator actions.

Insert NSHC-3

The FWLB and rod ejection accidents were shown to meet the acceptance criteria specified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

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ENCLOSURE 5

**REVISED PAGES OF ATTACHMENT 1 OF CR-3 EPU
LAR #309, REVISION 0**

- Credit the new LPI System HLI line as the primary method of boron precipitation mitigation during a LOCA, thereby eliminating the need for the two active methods currently credited and allowing deletion of CR-3 OL License Condition 2.C.11;
- Credit the use of the LPI System cross-tie line in the mitigation of a core flood tank (CFT) line break accident;
- Credit the use of soluble boron in the spent fuel pool to preclude spent fuel pool criticality accidents as allowed by 10 CFR 50.68(b)(4); and
- Revise the RCS pressure acceptance criterion for feedwater line break (FWLB) accident consistent with the criterion in NUREG-0800, "Standard Review Plan" (SRP), Section 15.2.8, "Feedwater System Pipe Break Inside And Outside Containment (PWR)," Revision 2 (March 2007).
- Approve for use an RCS pressure acceptance criterion for the Rod Ejection Accident (REA) to be consistent with the criterion in SRP, Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Revision 3 (March 2007).

3.0 BACKGROUND

CR-3 proposes to operate the reactor at 3014 MWt, an approximate 15.5% increase from the current authorized OL reactor core power level. CR-3 has defined this power increase as an EPU consistent with the guidance of the Office of Nuclear Reactor Regulation's Review Standard (RS)-001, "Review Standard for Extended Power Uprates," Revision 0.

Florida Power Corporation (FPC) has evaluated the proposed EPU for the applicable systems, structures, components, and safety analyses at CR-3 in accordance with the guidance of RS-001. The results of these evaluations are described in the CR-3 EPU TR (Attachments 5 and 7). The EPU TR provides the technical details that support the requested OL and ITS changes and, in concert with the other attachments, provide a comprehensive evaluation of the effects of the proposed EPU on CR-3 plant operation.

4.0 JUSTIFICATION FOR THE CHANGE

The acceptability for each proposed OL and ITS change is addressed in Tables 1 and 2 in Section 2.0 above. Additional details and justification are provided below for the additional licensing basis items. The EPU TR summarizes the evaluations performed to assure acceptable unit operation at EPU conditions and is therefore referenced throughout this section as additional technical justification for the EPU related changes. Additional detail is also provided for the plant modifications in EPU TR Appendix E, "Major Plant Modifications."

4.1 ICCMS, ADVs and FCS

CR-3 requests approval to credit the use the ICCMS and the FCS function of the ADVs to assure adequate core cooling during a SBLOCA at EPU conditions. The FCS is assumed in the SBLOCA analyses to open both ADVs to depressurize the secondary plant and ultimately the RCS which will allow ECCS to provide adequate core cooling in the event of a single failure of an HPI train as summarized in EPU TR Section 2.8.5.6.3, "ECCS and Loss of Coolant Accidents," and TR Appendix E. A new ICCMS will provide instrumentation to automatically initiate the FCS and provide indication to ensure the ICCMS actions are performed.

The ICCMS will also automatically trip the RCPs and send a signal to the Emergency Feedwater Initiation and Control (EFIC) System which will raise OTSG secondary side level to the inadequate subcooling margin (ISCM) level to support certain spectrum of LOCAs as summarized in EPU TR Section 2.8.5.6.3. The automatic actions to trip the RCPs and adjust the EFIC System to the ISCM level setting replace the current actions performed by the operator, thus reducing the reliance on manual operator action for event mitigation.

The FCS function of the ADVs will assure that CR-3 is capable of mitigating the effects of a SBLOCA at EPU conditions. Preliminary analysis indicated that for the range of SBLOCA events, the HPI System flow, using single failure considerations, is not sufficient for core cooling in the early stages of an accident. Additionally, the initial pressure in the RCS remains too high for the CFTs to provide adequate flow to the core. Inadequate HPI System flow would result in elevated peak clad temperatures and oxidation levels prior to achieving RCS conditions where sufficient core cooling can be provided.

The FCS will automatically actuate safety-related ADVs to rapidly reduce secondary pressure in the Main Steam System which increases primary-to-secondary heat transfer thereby decreasing temperature and pressure in the RCS. The intent of this plant modification is to rapidly reduce the RCS pressure to below the CFT discharge pressure which would increase ECCS flow to the reactor vessel and maintain the fuel within limits. LOCA analyses at EPU conditions assume both HPI pumps with adequate flow or one HPI pump and two ADVs are available to ensure sufficient core cooling.

The FCS will initiate within 10 minutes of the onset of sustained loss of sub-cooling margin and inadequate HPI flow. By rapidly reducing and controlling the secondary pressure in the OTSG, the RCS pressure has been analytically demonstrated to decrease sufficiently to allow timely addition of CFT liquid and additional flow from the HPI System to begin injecting into the core. The analysis performed for this event shows that the results are acceptable and all 10 CFR 50.46 acceptance criteria are satisfied. Enclosure 2 of EPU TR Appendix E provides specific information related to the ADVs and FCS plant modification.

The ICCMS monitors specific parameters; HPI System flow, RCS pressure, and core exit thermocouples (CETs) to determine core degrees of subcooling and to determine if a loss of

subcooling margin (SCM) exists. Additionally, the total HPI flow is compared to a generated curve of minimum HPI flow versus RCS pressure to determine inadequate HPI flow.

Following a sustained loss of SCM and inadequate HPI flow, ICCMS initiates FCS within 10 minutes which automatically opens both ADVs to ensure sufficient core cooling during a SBLOCA. Additionally, upon a sustained loss of SCM, the ICCMS will automatically trip the RCPs within one minute and within 10 minutes send a signal to the Emergency Feedwater Initiation and Control (EFIC) System which will raise OTSG secondary side level to the ISCM level. These automatic actions replace the current actions performed by the operator, thus reducing the reliance on manual operator action for event mitigation. Enclosure 3 of EPU TR Appendix E provides specific information related to the ICCMS plant modification.

ITS 3.3.17, "Post Accident Monitoring (PAM) Instrumentation," will be revised to include an additional instrument function (HPI Flow Margin) which, along with existing PAM instruments, will provide operators with indication necessary to ensure FCS initiates when required. The new ICCMS will provide redundant displays consistent with the criteria of Regulatory Guide 1.97. The ICCMS computes HPI flow margin, degrees of subcooling, and will provide operators with a fully qualified, safety related method of monitoring these parameters and determining inadequate HPI flow and a loss of subcooling margin, thereby ensuring initiation of the FCS in a timely manner.

Refer to Attachment 4, "Improved Technical Specification Bases Changes (Markup)," for further basis information related to revisions to ITS 3.3.17.

ITS 3.3.19, "Inadequate Core Cooling Monitoring System (ICCMS) Instrumentation," and ITS 3.3.20, "Inadequate Core Cooling Monitoring System (ICCMS)," will be added to address the requirements related to the automatic initiation of FCS, RCP trip, and OTSG ISCM level setpoint adjustment.

Regarding proposed ITS 3.3.19 and ITS 3.3.20:

- Applicability of the FCS instrument functions and logic is consistent with current licensed power level (2609 MWt) to support the FCS function of the ADVs. The Applicability is considered acceptable since with power at or below 2609 MWt, the ECCS can provide sufficient core cooling during a LOCA assuming a single failure of one HPI subsystem without the need for the FCS function of the ADVs.
- Applicability of the RCP Trip and OTSG ISCM Level Setpoint Actuation instrument functions and logic is consistent with the Applicability of ITS 3.3.17. These automatic functions are intended to reduce the reliance on manual operator action and therefore, the Applicability is considered acceptable since it is equivalent to the Applicability of the current indication requirements necessary to perform these actions manually.

- Appropriate ITS ACTIONS are provided to address inoperability the ICCMS instrumentation and logic. The proposed Required Actions and Completion Times are consistent with the current licensing basis ACTIONS, which rely on PAM instrumentation for credited manual operator actions for these same and similar functions. ITS 3.3.17, which provide PAM instrument requirements needed to support current credited manual operator actions, will continue to require a plant shutdown if required PAM instrumentation is not available to perform these manual actions. Also, proposed ITS 3.3.17, which provides an additional PAM instrument requirement to support manual actuation of FCS, will require a power reduction to a level where the FCS function of the ADVs is no longer required if the PAM instrument is not available to manually actuate the FCS.
 - With one or more required channels or logic trains inoperable, the inoperable equipment must be restored to OPERABLE status within 30 days. The 30 day Completion Time takes into account the provisions to support manual actuation of the FCS, RCP trip, and OTSG ISCM Setpoint adjustment. The 30 day Completion Time is also considered acceptable based on maintaining FCS actuation capability and the low probability of an event requiring the ICCMS during this time period.
 - For the RCP Trip and OTSG ISCM Level Setpoint Actuation instrument functions and logic, if an associated channel or logic train is not restored to OPERABLE status within the allowed Completion Time, action must be initiated in accordance with Specification 5.7.2, "Special Reports," to submit a special report to the NRC within 14 days. The special report will discuss the preplanned alternate method of monitoring, the cause of the inoperability, and provide a schedule for restoring the ICCMS instruments and logic to OPERABLE status. This proposed Required Action is considered acceptable since the CR-3 current licensing basis credits manual operator action for RCP trip and OTSG ISCM Setpoint adjustment. ITS 3.3.17, which provide PAM instrument requirements needed to support the current credited manual operator actions, will continue to require a plant shutdown if appropriate PAM instrumentation is not available to perform these manual actions.
 - For the FCS Actuation instrument functions and logic, if an associated channel or logic is not restored to OPERABLE status within the allowed Completion Time or automatic FCS actuation capability is not maintained, the FCS function of the ADVs will be immediately declared inoperable and action taken in accordance with the requirements of ITS 3.7.20.

- Appropriate SRs are provided to ensure the ICCMS instruments and logic trains are capable of performing their intended safety function. CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION tests are provided to verify the functionality of the ICCMS instrumentation and automatic actuation logic. Additional automatic actuation logic CHANNEL FUNCTIONAL TESTS are added to verify RCP breaker actuation and OTSG ISCM level setpoint actuation every refueling interval since these end devices have no other Technical Specification testing requirements. ADV actuation testing is performed in accordance with surveillances specified in ITS 3.7.20.

Refer to Attachment 4, "Improved Technical Specification Bases Changes (Markup)," for further basis information related to proposed ITS 3.3.19 and ITS 3.3.20.

ITS 3.7.20, "Fast Cooldown System (FCS)," will be added to address the requirements for the FCS function of the safety related ADVs.

Regarding proposed ITS 3.7.20:

- Applicability of proposed ITS 3.7.20 is consistent with current licensed power level (2609 MWt) to support the FCS function of the ADVs. The Applicability is considered acceptable since with power at or below 2609 MWt, the ECCS can provide sufficient core cooling during a LOCA assuming a single failure of one HPI subsystem without the need for the FCS function of the ADVs.
- Appropriate ITS ACTIONS are provided in ITS 3.7.20 to address inoperability of the FCS function of the ADVs and the impact on SBLOCA event mitigation.
 - With the FCS inoperable due to inoperable backup air supply, 7 days is provided to restore the inoperable backup air supply to OPERABLE status. The 7 day Completion Time is consistent with Completion Times in NUREG-1430 for similar levels of system degradation and is acceptable based on multiple sources of air that can provide the ADVs.
 - With the FCS function inoperable for reasons other than backup air supply, 72 hours is provided to restore the FCS to OPERABLE status provided both HPI subsystems are OPERABLE.
 - The 72 hour Completion Time is consistent with a single failure Completion Time of one ECCS train inoperable per ITS 3.5.2, "ECCS-Operating," and is considered a reasonable time to repair the inoperable FCS components.
 - If either HPI subsystems is determined to be inoperable or the FCS is not restored with the required Completion Time, one hour is provided to reduce power to at or below 2609 MWt. The Required Action to reduce

power to the pre-EPU license power level provides an appropriate compensatory measure since below this power the FCS is not required to mitigate a SBLOCA; one HPI train is capable of mitigating the SBLOCA. The one hour Completion Time will ensure prompt action is taken to reduce power where FCS is no longer required for ECCS to perform its function during a SBLOCA and is consistent with the one hour provided in LCO 3.0.3 for loss of safety function conditions.

- Appropriate SRs are provided to ensure the ADVs and FCS are capable of performing their intended safety function in the event of a SBLOCA. The periodic verification of the Backup Air System pressure and volume is added to ensure the ADVs can maintain at least four hours of operation. A CHANNEL CALIBRATION test is provided to verify the functionality of the FCS automatic actuation circuitry and calibration of the FCS OTSG modulating controller. Two SRs are added to verify the capability and capacity of the FCS backup battery system. An additional SR is added to verify that on actual or simulated actuation signal, the ADVs will open and cycle as required.

Refer to Attachment 4, "Improved Technical Specification Bases Changes (Markup)," for further basis information related to proposed ITS 3.7.20.

4.2 Boron Precipitation Mitigation

CR-3 requests approval to credit the LPI System HLI line to prevent boron precipitation in the reactor vessel during design basis accidents at EPU conditions thereby eliminating the need for the current active boron precipitation mitigation methodologies required in the CR-3 OL.

Existing boron precipitation mitigation methodologies at CR-3 consist of two approved, active techniques that provide sufficient liquid movement through the reactor core to preclude boric acid from precipitating out of solution and potentially clogging flow passages in the fuel or settling on the fuel rods causing a decrease in heat transfer. The most likely occurrence of boron precipitation is during the recirculation phase of an accident when the level in the reactor vessel is below the top of the fuel and boiling is in process. The current licensing and design bases methodologies to preclude this condition are: the Dump-to-Sump of RCS fluid from the hot leg to the Reactor Building sump using the Decay Heat Removal (DHR) System drop line; and hot leg injection via the auxiliary pressurizer spray line. These methodologies allow for additional flow through the reactor core that would preclude the precipitation or plating out of the boron. As stated in the Crystal River Unit 3 Issuance of Exemption from the Requirements of 10 CFR Part 50, Appendix K, Section I.D.1, dated October 29, 1998, the NRC acknowledged that the reactor vessel vent valves may not be effective in preventing boron precipitation for specific break locations. Due to specific vulnerabilities associated with these active methods, a single failure criterion exemption was granted to CR-3. Additionally, evaluation has shown that these methods cannot be expanded to provide sufficient flow at EPU conditions to adequately preclude boron precipitation.

The current exemption related to boron precipitation mitigation methods is associated with the active failure of a single Engineered Safeguards motor control center (MCC) that supplies electrical power to both active methods of preventing boron precipitation. At EPU conditions, prevention of boron precipitation during LOCAs will be accomplished through the use of the LPI System HLI line and CR-3 will no longer require the existing single failure criterion exemption. Existing procedural guidance requires that the boron precipitation mitigation function occur after a certain boron concentration in the core is reached. The procedure directs that boron concentration be sampled after the ECCS pump suction swap to the sump is complete and adequate core cooling is not occurring. The currently credited method of determining boron concentration is the Post Accident Sampling System Boronometer. Should the boronometer be unavailable, manual sampling and analysis is also credited.

The new HLI actuation will not be based on core boron concentration and precludes the need for continued use of the boronometer. CR-3 will discontinue reliance on the boronometer once use of the new HLI line is approved and implemented.

The new HLI line provides a flow path for boron precipitation control, thus assuring compliance with the long term core cooling acceptance criterion by providing a flow path to inject a portion of LPI System flow into the DHR System drop line. The HLI line is normally closed, but will be opened by the operators during the transition to the sump recirculation, which will be prior to the occurrence of significant boron concentration in the core. The HLI line is hydraulically designed to provide sufficient flow to prevent boron precipitation for the entire spectrum of LOCA break sizes. For large break LOCAs, the HLI flow exceeds the core boiloff shortly after its initiation.

The excess HLI flow that is not boiled off by the core decay heat dilutes the core boron concentration via reverse core flow prior to the core reaching concentrations that could precipitate. For SBLOCAs, the RCS pressure could be above the LPI pump shutoff head or in the range where the HLI flow does not match core boiloff. However, at these elevated RCS pressures, the solubility limit is above the maximum boron concentration that the core could achieve. The HLI flow will increase as RCS pressure decreases such that the flow matches the core boiloff rate and provide a boron dilution flow prior to reaching the solubility limit. The HLI line design meets the single failure criterion of CR-3 FSAR Criterion 1.4.38. Therefore, the existing single failure criterion exemption is no longer necessary and may be removed from the CR-3 license basis.

EPU TR Section 2.8.5.6.3 provides additional discussion related to boron precipitation mitigation. Enclosure 1 of the CR-3 EPU TR Appendix E provides specific information related to the LPI System cross-tie line modification, including the HLI line design.

ITS Section 3.4.13, "RCS PIV Leakage," will be revised to address the new series check valves in the HLI line used to provide pressure isolation between high and low pressure systems. These valves are smaller than the current PIVs required in LCO 3.4.13, and as such, the guidance in NUREG 1430 will be incorporated for the LCO and surveillance acceptance criteria related to

leakage through the valves. Refer to Attachment 4, "Improved Technical Specification Bases Changes (Markup)," for further information related to proposed changes to ITS 3.4.13.

4.3 Core Flood Line Break Mitigation

CR-3 requests approval to credit the use of the LPI System cross-tie line in the mitigation of a CFT line break accident.

During a CFT line break accident coincident with a LOOP assuming one emergency diesel generator (EDG) fails to start, one ECCS train associated with the failed EDG will not be available for coolant injection. The remaining LPI subsystem, powered by the operating EDG, may be unavailable for coolant injection if aligned to the broken CFT line. As a consequence, only one HPI subsystem and one CFT will be available to provide ECCS flow into the RCS. At pre-EPU conditions, the intact CFT flow and HPI flow are adequate to meet the 10CFR50.46 acceptance criteria. However, at EPU conditions, the increased core decay heat power increases the core boil off rates, slowing the RCS depressurization rate and slightly reducing the ECCS flow. As a consequence, one HPI subsystem and the inventory of one CFT are not adequate to mitigate a CFT line break accident. The LPI System cross-tie line modification ensures the availability of some LPI flow for the larger CFT line breaks by providing a passive LPI crossflow path and the CFT line break analyses at EPU conditions continue to demonstrate compliance with 10 CFR 50.46. EPU TR Section 2.8.5.6.3 provides additional discussion related to CFT line break mitigation. Enclosure 1 of the CR-3 EPU TR Appendix E provides specific information related to the LPI System cross-tie line modification.

ITS Section 3.5.2, "Emergency Core Cooling Systems (ECCS) – Operating," and associated Bases will be revised to add ECCS OPERABILITY requirements associated with the LPI cross-tie line, including an SR to ensure the proper LPI cross-tie line valve lineup. Refer to Attachment 4, "Improved Technical Specification Bases Changes (Markup)," for further information related to proposed changes to ITS 3.5.2.

4.4 Credit Soluble Boron in Spent Fuel Pool

CR-3 requests approval to credit the use of soluble boron in the spent fuel pool to preclude spent fuel pool criticality accidents as allowed by 10 CFR 50.68(b)(4).

At pre-EPU conditions, the spent fuel storage criticality analysis does not credit soluble boron when fuel assemblies are loaded in a specific pattern. The reactivity of the fuel racks alone is adequate to preserve the assumptions of the pre-EPU criticality analysis. However, at EPU conditions spent fuel storage criticality analyses rely on crediting sufficient boron concentrations.

EPU TR Section 2.8.6.2, "Spent Fuel Storage," provides additional discussion related to the impact on the spent fuel storage criticality analysis as a result of EPU.

ITS Section 3.7.14, "Spent Fuel Pool Boron Concentration," will be revised to be applicable whenever fuel assemblies are stored in the spent fuel pool and ITS 4.3.1, "Criticality," will be

revised to reflect the k_{eff} criteria of 10 CFR 50.68(b)(4). Refer to Attachment 4, "Improved Technical Specification Bases Changes (Markup)," for further basis information related to proposed changes to ITS 3.7.14.

4.5 Revise RCS Pressure Acceptance Criterion for FWLB Accident

CR-3 requests approval to revise the RCS pressure acceptance criterion for the feedwater line break (FWLB) accident to be consistent with the criterion in SRP, Section 15.2.8, Revision 2.

The FWLB is considered a limiting fault event per the CR-3 FSAR, Section 14.2.2.9.2. The FSAR acceptance criteria for this limiting fault include an RCS pressure criterion of 110% of design pressure (2750 psig). For EPU conditions, CR-3 requests revising the FWLB accident acceptance criterion to reflect an RCS pressure limit of 120% of design pressure (3000 psig). This is consistent with acceptance criteria specified in SRP, Section 15.2.8 for the FWLB event. EPU TR Section 2.8.5.2.4, "Feedwater System Pipe Breaks Inside and Outside Containment," provides further discussion related to revising the FWLB accident RCS pressure acceptance criterion and the associated plant impact as a result of EPU.

4.6 Revise RCS Pressure Acceptance Criterion for REA

CR-3 requests approval to use an RCS pressure acceptance criterion of 3200 psig for the CR-3 REA as a method of ensuring maximum rod worth during an REA will not lead to further coolant boundary rupture, the applicable CR-3 design criterion. Currently, the CR-3 FSAR does not include an explicit RCS pressure criterion for the REA. Rather, the licensing basis acceptance criteria for the REA include a fuel enthalpy of 280 cal/gm and the amount of thermal energy release necessary to cause reactor pressure vessel deformation; 3.37×10^8 cal. For EPU conditions, CR-3 requests the use of an REA acceptance criterion of 3200 psig, which is less than the value that would result in stresses that exceed the Babcock & Wilcox (B&W) "Service Limit C" as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and meets the related acceptance criterion specified in SRP, Section 15.4.8 for the REA event. Maintaining RCS peak pressure during an REA less than that required to exceed the ASME Service Level C stress limits will ensure the thermal energy release necessary to deform the RPV (i.e., local vessel yielding) is not reached. The peak RCS pressure to preserve the Service Level C stress limit is conservatively considered to be 3200 psig for B&W plants, including CR-3. Previous comprehensive assessments and reviews confirming the structural adequacy of the reactor coolant pressure boundary (RCPB) during a postulated ATWS event determined that a majority of RCPB components could withstand RCS pressures up to 3750 psig using ASME Emergency (Service Level C) stress levels with the limiting components identified as the CR-3 RCPs. Analysis results indicated approximately 6% of the RCP pressure boundary area exceeds the primary membrane Level C stress limit when subjected to an internal pressure of 3500 psig. However, this overstress condition was determined to cause limited localized distortion and subsequent analysis indicated that it would not affect the Operability of

the RCP. Based on these evaluations, 3200 psig is considered an acceptable RCS pressure limit for the CR-3 REA.

Adding an RCS pressure acceptance criterion of 3200 psig is considered acceptable since the REA is considered an American Nuclear Society (ANS) Condition IV event. ANS Condition IV events are defined as those events not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Since an REA is a highly unlikely event, adding an RCS pressure acceptance criterion that complies with the SRP acceptance criteria, as required by 10 CFR 50.34(h), does not represent a significant risk to the health and safety of the public. Additionally, adding an RCS pressure acceptance criterion of 3200 psig will continue to ensure the thermal energy release necessary to deform the RPV is not reached thereby preserving the applicable CR-3 design criterion of ensuring maximum rod worth during a REA will not lead to further coolant boundary rupture.

4.7 Conclusion

The CR-3 EPU has been evaluated for impact on the plant and the OL. With implementation of plant modifications described in the EPU TR, analyses demonstrate that the plant will maintain the capability for safe operation and the mitigation of postulated accident scenarios at EPU conditions. The proposed Technical Specification changes will assure that the plant and its SSC remain within acceptable limits at all times. Procedure changes and training will be implemented prior to plant startup under EPU conditions and will assure that plant personnel are capable of operating the plant and responding to abnormal or emergency conditions.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Florida Power Corporation (FPC) has evaluated the proposed License Amendment Request (LAR) against the criteria of 10 CFR 50.92(c) to determine if any significant hazards consideration is involved. FPC has concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92(c) criteria is satisfied.

- (1) *Does not involve a significant increase in the probability or consequences of an accident previously evaluated?*

The impacts of the proposed Extended Power Uprate (EPU) on plant systems, structures, and components (SSCs) were reviewed with respect to SSC design capability, and it was determined that following completion of plant changes to support the EPU, no SSC would exceed its design conditions or limits. Evaluations supporting those conclusions were performed and demonstrate that the equipment reliability and structural integrity will not be adversely affected by EPU. Control system studies demonstrated that plant response to operational transients under EPU conditions does not significantly increase reactor trip frequency, so there will be no significant increase in the frequency of SSC

challenges caused by a reactor trip. The EPU does not create new failure modes for existing SSCs and eliminates the need for a single failure exemption currently in the Crystal River Unit 3 (CR-3) licensing basis for boron precipitation mitigation. A new potential inter-system loss-of-coolant accident (LOCA) mechanism is created by the installation of the HLI flow path, but the probability of an inter-system LOCA occurring has not significantly increased. Additionally, new pressure isolation valves installed in series between reactor coolant system (RCS) high and low pressure piping will minimize the likelihood of an inter-system LOCA. ASME Boiler and Pressure Vessel Code requirements for procurement, installation, and testing of the new pressure isolation valves will be followed. Also, a new potential steam line break mechanism is created by the inadvertent opening of both Atmospheric Dump Valves (ADVs) simultaneously. A new Inadequate Core Cooling Monitoring System (ICCMS) and Fast Cooldown System (FCS) are being installed which will automatically open both ADVs to support a small break LOCA in the event of a single failure in the Emergency Core Cooling System (ECCS). Modifications to ADVs and procurement, installation, testing and operation of the ICCMS and FCS will ensure that the potential inadvertent opening of both ADVs is minimized. Also, any adverse consequences of inadvertent opening both ADVs are bounded by the consequences of the main steam line break accident.

The fission product barriers; fuel cladding, RCS pressure boundary, and the containment building, remain fully capable of performing their design functions. The spectrum of previously analyzed postulated accidents and transients was evaluated, and effects on the fuel, the RCS pressure boundary, and the containment were determined. Specific accident scenarios (small break LOCA, locked reactor coolant pump rotor, and rod ejection accident) have been determined to potentially cause cladding rupture under EPU conditions in limited amounts, but the quantity of the failures and the consequences are bounded by the large break LOCA analysis. The fuel remains within the acceptance criteria of 10 CFR 50.46, the RCS pressure will not increase for normal operation as a result of EPU, and accident conditions remain within the ASME Boiler and Pressure Vessel Code limits as well as SSC design limits. Analysis has also confirmed that during the worst case accident (large break LOCA), the containment building remains within its design limit. These analyses were performed and demonstrate that existing RCS pressure boundary and containment limits are met and the effects on the fuel are such that dose consequences meet existing criteria at EPU conditions.

With the exception of the steam generator tube rupture (SGTR) accident, the EPU analyses have been performed using conservative methodologies, as specified in Regulatory Guide 1.183, "Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors." The SGTR analyses were performed using current licensing basis methodologies. Safety margins have been evaluated and margin has been retained to ensure that the analyses adequately bound the postulated limiting event scenarios. These analyses indicate increased doses for certain analyzed accidents.

Various factors contribute to these increases. Several actions have been taken to limit the increased consequences. Modifications to the Low Pressure Injection (LPI) and ADV systems are being made to ensure that the consequences of previously evaluated accidents are not significantly increased. Specifically, these modifications will enhance the plant response capabilities to a small break LOCA and improve the method for boron precipitation mitigation. The proposed amendment reduces the maximum allowed RCS specific activity. The limits on specific activity ensure that the doses remain within the regulatory limits during analyzed transients and accidents. The maximum allowed operating containment pressure is being reduced from 3.0 psig to 1.5 psig; thereby ensuring that maximum peak containment internal pressure does not exceed limits in the event of a design basis accident.

The revised accident analyses demonstrate that the plant site and the dose-mitigating Engineered Safety Features remain acceptable with respect to the radiological consequences of postulated DBAs since the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR Part 50.67 "Alternative source term.". Therefore, the consequences of analyzed accidents are not significantly impacted by the proposed EPU.

Based on the above, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

- (2) *Does not create the possibility of a new or different kind of accident from any accident previously evaluated?*

Equipment that could be affected by EPU has been evaluated. No new operating mode, or accident scenario was identified. The LPI, HLI, and ADV systems will be modified and an ICCMS and FCS will be installed to better respond to accident and non-accident conditions. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. The limiting accident remains the large break LOCA and analysis results are acceptable under EPU conditions. EPU uses developed technology and applies it within capabilities of existing or modified plant safety-related equipment in accordance with the regulatory criteria (including NRC approved codes, standards and methods). Modifications to existing SSCs and installation of new SSCs are designed in accordance with regulatory criteria to minimize equipment failures. Potential equipment failures of new or modified SSCs have been evaluated and postulated failures are equivalent or bounded by existing equipment failures or effects of these equipment failures are bounded by previously evaluated accidents. No new accidents or event precursors have been identified.

The Technical Specification (TS) revisions required to implement EPU continue to assure that the plant is operated within the limits established for safe operation of the plant.

Additionally, the limits in the TS reflect the initial conditions for the safety analyses performed to demonstrate the plant can mitigate the effects of accidents and ensure public safety by maintaining offsite doses within the limits in 10 CFR 50.67. The revisions have been assessed and it was determined that the proposed change will not introduce a different accident than that previously evaluated.

Based on the above, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) *Does not involve a significant reduction in a margin of safety?*

Structural evaluations performed at EPU conditions demonstrated that calculated loads on affected SSCs after modification, if necessary, remain within their design allowable for all design basis event categories. ASME Code requirements continue to be met.

Fuel performance evaluations were performed using parameter values appropriate for a reload core operating at EPU conditions. Those evaluations demonstrate that fuel performance acceptance criteria continue to be met. Core reload evaluation processes ensure that the planned fuel load in the first reactor core to be operated at the increased power level, will meet applicable regulatory criteria.

LOCA and non-LOCA safety analyses were performed under EPU conditions. ECCS performance was shown to meet the criteria of 10 CFR 50.46. Small break LOCA, locked rotor, and rod ejection scenarios may, under EPU conditions, have some potential of resulting in a limited amount of fuel cladding failure, but the analyses conclude that the plant remains within the acceptance criteria of 10 CFR 50.46. Large break LOCA scenarios also satisfy the criteria of 10 CFR 50.46 under EPU conditions, and the large break LOCA remains the most limiting accident at EPU conditions. LOCA analyses indicate a small reduction in margin (< 5%) related to maximum local clad oxidation and hydrogen generation at EPU conditions and a slight improvement in peak clad temperature margin based on minor changes in LOCA design inputs and modifications that will enhance the plant response capabilities to LOCAs and improve the method for boron precipitation mitigation. The non-LOCA events identified in the CR-3 Final Safety Analysis Report, with the exception of the feedwater line break (FWLB) and rod ejection accidents, were shown to meet existing acceptance criteria. The FWLB and rod ejection accidents were shown to meet the acceptance criteria specified in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

The containment building response to mass and energy releases was evaluated under EPU conditions. The evaluations indicated that temperature and pressure limits were met.

No plant changes associated with the EPU reduce the degree of component or system redundancy. The small break LOCA response will require two High Pressure Injection

(HPI) pumps injecting. In the event of inadequate HPI and loss sub-cooling margin (SCM), secondary depressurization will be achieved via a new ICCMS and FCS, which includes automatic actuation of the ADVs, thereby assuring that the reactor core receives the necessary ECCS flow to minimize core damage and satisfy the requirements of 10 CFR 50.46. The features required to automatically open ADVs during a small break LOCA are incorporated into proposed TS 3.3.19, "Inadequate Core Cooling Monitoring System (ICCMS) Instrumentation," and TS 3.3.20, "Inadequate Core Cooling Monitoring System (ICCMS)," and TS 3.7.20, "Fast Cooldown System (FCS)." The ICCMS and FCS are configured and supported such that a single failure will not prevent completion of the ECCS safety function.

To support this enhancement, operators will be provided indication of subcooling margin and HPI System flow adequacy to ensure actuation of the FCS. A new safety related display system will be available to determine when insufficient subcooling margin is available and HPI System flow is inadequate. Additional modifications will, upon indication of loss of subcooling margin, automatically trip the RCPs and raise steam generator secondary side level to the inadequate subcooling margin level. These automatic actions replace the current actions performed by the operator, thus reducing the reliance on manual operator action for event mitigation.

Operator training programs will be revised in accordance with the industry standard systematic approach to training process and appropriate training will be provided on all plant modifications, administrative/technical requirement changes, Technical Specification revisions, and procedure revisions. The CR-3 simulator will be updated and tested in sufficient time to provide effective reinforcement of procedure and plant physical changes as well as build proficiency with required manual operator actions. Based on the above, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, CR-3 concludes that the proposed LAR presents a no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

6.0 ENVIRONMENTAL IMPACT EVALUATION

The environmental considerations evaluation is contained in Attachment 9, "Supplemental Environmental Report Extended Power Uprate." It concludes that the EPU will not result in a significant change in non-radiological impacts on land use, water use, waste discharges, terrestrial and aquatic biota, transmission facilities, or social and economic factors, and will have no non-radiological impacts other than those evaluated in the Supplemental Environmental Report. The Supplemental Environmental Report further concludes that the EPU will not introduce any new radiological release pathways, will not result in a significant increase in occupational or public radiation exposures, and will not result in significant additional fuel cycle

environmental impacts.

Therefore, the proposed amendment does not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite nor does it involve a significant increase in individual or cumulative occupational radiation exposure.

7.0 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

CR-3 has determined that the proposed changes do not require any exemptions or relief from regulatory requirements and do not adversely affect conformance with any regulatory requirements differently than described in the CR-3 Final Safety Analysis Report. The exemption to 10 CFR 50 Appendix K, Item I.D.1, requirements for single failure considerations is being deleted due to a modification to the plant that will adequately provide the boron precipitation mitigation function and is designed to remain functional even with a single failure condition.

This LAR will not reduce the effectiveness of the safety related systems, structures, or components and will not require the plant to operate outside of analyzed limits. Therefore, based on the considerations discussed above:

- 1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;
- 2) Such activities will be conducted in compliance with the Commission's regulations; and
- 3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.