FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

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ATTACHMENT E

TECHNICAL REPORT ANP-3052:
CR-3 EPU FEEDWATER LINE BREAK ANALYSIS WITH FAILURE OF FIRST SAFETY GRADE TRIP, REVISION 2

ANP-3052
Revision 2

CR-3 EPU Feedwater Line Break Analysis with Failure of First Safety Grade Trip

June 2012

AREVA NP Inc.

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Record of Revision

Revision No.	Pages/Sections/Paragraphs Changed	Brief Description / Change Authorization
000	All	Initial Release
001	All	The FWLB analysis with the first safety grade trip failed was re-evaluated using the model and limiting conditions from Reference [3]. Revision 001 is a complete revision.
002	Page 10	Added discussion on the rod worth and break location
	Page 11	Added Figure 3-1
	Table 3-1	Added critical flow models used

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Nomenclature

Acronym	Definition
BOC	Beginning of Cycle
CR-3	Crystal River Unit 3
DSS	Diverse Scram System
EFIC	Emergency Feedwater Initiation and Control
EFW	Emergency Feedwater
EPU	Extended Power Uprate
FWLB	Feedwater Line Break
LAR	Licensing Amendment Request
LOOP	Loss of Offsite Power
MFW	Main Feedwater
MSSV	Main Steam Safety Valves
NRC	Nuclear Regulatory Commission
PORV	Pilot Operated Relief Valve
PSV	Pressurizer Safety Valve
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RPS	Reactor Protection System
SG	Steam Generator
Tave	Average RCS Temperature
TSV	Turbine Stop Valves

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) Reactor Systems Branch staff requested an additional analysis be performed to support the review of the Crystal River Unit 3 (CR-3) extended power uprate (EPU) licensing amendment request (LAR). In particular, the staff requested an analysis of the feedwater line break (FWLB) transient assuming that the first safety grade reactor protection system (RPS) trip function fails to trip the reactor. This report documents the results of the requested analysis.

The analysis documented in this report assumes that the RPS high reactor coolant system (RCS) pressure trip function fails to trip the reactor. This analysis models the non-safety grade diverse scram system (DSS) trip, which inserts the regulating control rod banks upon reaching the DSS high RCS pressure trip setpoint. The peak RCS pressure during the FWLB transient is reported and compared to an acceptance criterion of 120% of the reactor coolant pressure boundary (RCPB) design pressure (1.20 * 2500 = 3000 psig).

The evaluations of the FWLB transient with the first safety grade trip failed are based on the model and conclusions in Reference [3]. The model refinements described in Section 4.1 of Reference [3] and the limiting initial conditions described in Section 4.12 of Reference [3] are included in the model for evaluating the first safety grade trip being failed.

2.0 ANALYTICAL METHODOLOGY

The thermal-hydraulic analysis of the FWLB transient at the CR-3 EPU power level with the first safety grade trip failed is performed using the RELAP5/MOD2-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 steam generators (SG), and heat transfer to the steam generators. The secondary model includes a detailed depiction of the main steam system, including steam relief to the atmosphere through the main steam safety valves (MSSVs) and simulation of the turbine stop valves (TSVs). The secondary model also includes the delivery of feedwater, both main and emergency, to the steam generators.

The RELAP5/MOD2-B&W code has been approved by the NRC for use in non-LOCA safety analyses (Reference [2]). The analysis documented in this report is consistent with Reference [2] with two exceptions:

1) The first safety grade trip is not credited. Instead, this analysis credits the next available trip, which is the non-safety grade DSS trip on high RCS pressure.
2) Reference [2] was not used to define the initial conditions for the transient. The initial conditions are based on the FWLB sensitivity studies documented in Reference [3]. The conditions in Reference [3] that result in the highest peak RCS pressure during a FWLB transient are:
    a. Nominal RCS pressure (2170 psia hot leg pressure)
    b. Maximum RCS flow (398,850 gpm)
    c. 100.4 %FP
    d. Beginning of Cycle (BOC)
    e. 585 °F Average RCS Temperature (Tave)
    f. 940 psia turbine header pressure
    g. 80 %OR SG level
    h. 290 inch indicated pressurizer level

3.0 ANALYSIS INPUTS

The FWLB analysis with the first safety grade trip failed is based the FWLB sensitivity analyses in Reference [3]. Section 3.0 of Reference [3] describes how the sensitivity analyses compare to the FWLB analyses without pressurizer spray performed to support Section 2.8.5.2.4 of the CR-3 EPU LAR.

The key input changed to model the first safety grade trip being failed includes:

1. The RPS high RCS pressure trip function is disabled.

2. The DSS high RCS pressure trip function is modeled. The DSS trip setpoint is modeled as 2465 psia. The DSS high RCS pressure trip setpoint includes margin to bound possible changes in the containment pressure during a FWLB. The DSS trip delay time is modeled as 1.23 seconds. Finally, the rod worth available to the DSS system is modeled as only 2.0 %Δk/k, since the DSS system only inserts the regulating control rod groups. The rod worth is based on the regulating control rod groups having a worth of 2.2 %Δk/k at the rod insertion limits. The worth is then reduced by 10% (0.2 %Δk/k) for uncertainty. An explicit allowance for the worth of a stuck regulating control rod is not considered.

3. Section 4.10 of Reference [3] contains a sensitivity study on the break location. The failed first trip analysis was performed at the three limiting break locations from that study, to ensure that the limiting break location is captured. The break locations considered are shown in Figure 3-1. The peak RCS pressures for the three break locations ranged from 2911.61 psia to 2915.39 psia. The limiting break location is a 1.418 ft^2 break in one of the side branches of steam generator B at the junction connecting the main feedwater line to the side branches.

Table 3-1 summarizes all of the key input to the FWLB analysis with the first safety grade trip failed, including the changes highlighted above.

The peak RCS pressure is reached shortly after reactor trip before any active safety systems such as emergency feedwater (EFW) are credited to mitigate the results. Therefore, there are no single failure assumptions that would result in a more limiting peak RCS pressure. However, the FWLB overpressure protection analysis with the first safety grade trip failed assumes the same single failure assumption as the FWLB analysis performed to support Section 2.8.5.2.4 of the CR-3 EPU LAR. The single failure assumption modeled is the failure of one train of Emergency Feedwater Initiation and Control (EFIC) such that EFW flow is not initiated automatically in one train. Consequently, only one of the two EFW pumps is assumed available to provide flow to the SGs. This single failure assumption produces a conservative long-term transient response.

Figure 3-1: Feedwater Line Break Locations

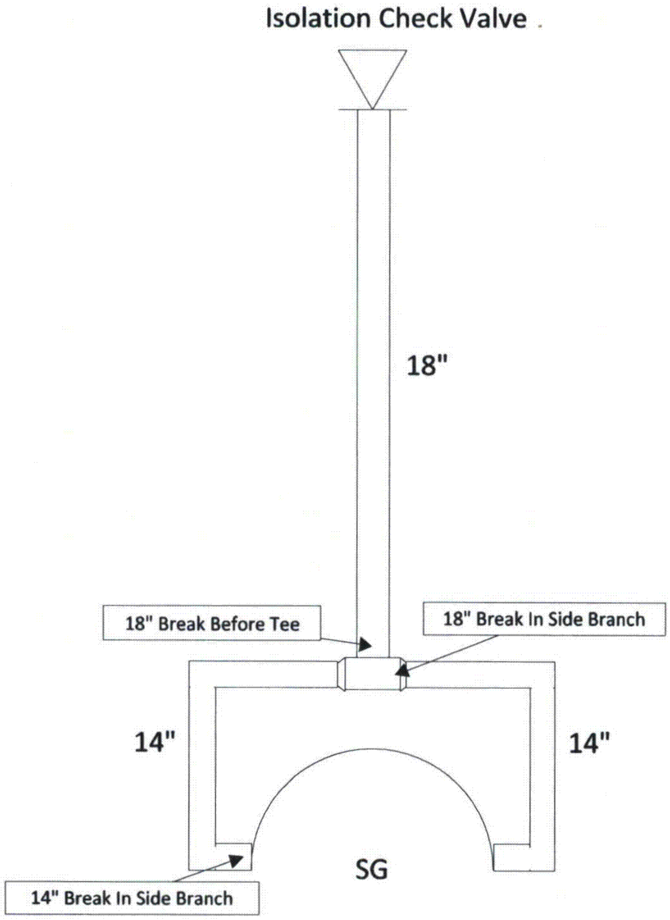


Table 3-1: Input to Feedwater Line Break Analysis with First Trip Failed

Parameter	Value
RCS Conditions	
Core Power, MWt	3014 * 1.004 = 3026.1
Decay Heat	1.0*ANS71 plus B&W Actinides
Total Net Reactor Coolant Pump (RCP) Heat, MWt	16.4
Average RCS Temperature, °F	585
Initial Hot Leg Pressure, psia	2170
Total RCS Flow Rate, gpm	398,850
Pressurizer	
Initial Indicated Pressurizer Level, in	290
Pressurizer Spray	Not Modeled
Pressurizer Heaters	Not Modeled
Pilot Operated Relief Valve (PORV)	Not Modeled
Pressurizer Safety Valve (PSV) Setpoints, psig	2500 * (1 + 0.03) (open) 2500 * (1 – 0.04) (close)
Total PSV Rated Capacity, lbm/hr	2 * 317,973 @ 2750 psig
Secondary Side	
Initial Main Feedwater (MFW) Temperature, °F	460 °F
Tube Plugging, %	5
Initial SG Level, %OR	80
EFW Temperature, °F	120
EFW Minimum Required Flow, gpm	660
EFW Delay Time, sec	40
Turbine Trip Delay Time, s	0.0
TSV Stroke Time, s	0.2
Number of Main Steam Safety Valves (MSSVs) per SG	8
MSSV Capacity per SG	7 @ 845,759 lbm/hr 1 @ 583,574 lbm/hr
MSSV Nominal Setpoints	2 @ 1050 psig 2 @ 1070 psig 2 @ 1090 psig 2 @ 1100 psig including small MSSV
MSSV Setpoint Tolerance	+3%
MSSV Accumulation	+3%

Parameter	Value
MSSV Blowdown	-5%
Core Kinetics Parameters	
Doppler Temperature Coefficient (Δk/k/°F)	-1.30 E-5
Moderator Temperature Coefficient (Δk/k/°F)	0.0 E-4
Prompt Neutron Generation Time, (μs)	24.8
Effective Delayed Neutron Fraction	0.0070
DSS Insertable Rod Worth, %Δk/k	2.000
RPS High RCS Pressure Trip	Assumed Failed
DSS High RCS Pressure Trip Setpoint, psia	2465
DSS Trip Delay Time, s	1.23
Critical Flow Models	
Subcooled Liquid	Homogeneous Equilibrium Model
Two-Phase and Superheated Fluid	Moody
Miscellaneous	
Offsite Power	Available
Single Failures	One Train of EFIC
Operator Actions	None

4.0 RESULTS / CONCLUSIONS

The sequence of events for the FWLB accident with the first safety grade trip failed is listed in Table 4-1 and the calculated results are tabulated in Table 4-2. Plots that demonstrate the transient response following a FWLB are provided in Figures 4-1 through 4-15.

Following initiation of the FWLB, the blowdown of the affected SG results in a reduction in the secondary heat removal. The mismatch between energy addition to the reactor coolant and the secondary heat removal causes the reactor coolant to heat up and pressurize. The pressure increases to the RPS high RCS pressure trip setpoint, but the trip is assumed to fail. The pressure continues to increase to the DSS high RCS pressure trip setpoint. After the appropriate delay time, the DSS trip inserts the regulating control rod banks.

After reactor trip, the RCS pressure continues to increase until the PSVs lift. Shortly after the PSVs lift, the pressure begins to decrease. The peak RCS pressure occurs in the bottom of the reactor vessel and does not exceed 120% of the design pressure of 2500 psig (3000 psig).

EFIC actuates on low SG level in the affected SG. After considering the possible 40 second delay, EFW is provided to the unaffected SG at the flow rate of one EFW pump (660 gpm). During this time, the PSVs maintain the RCS pressure based on the PSV open and close setpoints. The FWLB event is sufficiently severe that the pressurizer fills. As a result, the PSVs begin to pass single-phase liquid. The PSVs of the type installed at CR-3 achieve satisfactory performance for fluid temperatures greater than ~550°F. An additional check is performed to show that the PSV fluid inlet temperature remains greater than 600 °F to ensure that the PSVs operate as intended. Figure 4-4 demonstrates that at all times throughout the FWLB transient, the liquid temperature at the top of the pressurizer remains above 600 °F. The PSVs close for the final time at ~170 seconds. At ~8 minutes, the secondary heat removal from EFW causes the pressurizer level to drop.

A sensitivity study was performed modeling the FWLB transient with the first safety grade trip failed and a loss of offsite power (LOOP). The LOOP is conservatively considered to occur coincident with the turbine trip that follows reactor trip. The case with a LOOP included the insertion of the safety control rod banks on LOOP. The sensitivity study determined that the peak RCS pressure from a LOOP case is less limiting than the peak RCS pressure without a LOOP. Therefore, the FWLB evaluation using the DSS trip function and no LOOP is the bounding case.

Table 4-1: Sequence of Events for FWLB with First Trip Failed

Parameter	Time, sec
Transient Initiated	0.0
MFW to unaffected SG Interrupted	0.01
Peak Thermal Power Occurs	7.59
DSS High RCS Pressure Trip Setpoint Reached	10.26
EFIC Actuated on Low SG-B Level	10.42
Regulating Control Rod Groups Begin to Insert	11.49
Turbine Trip, TSVs Begin to Close	11.49
Initial PSV Lift	~12.5
Peak RCS Pressure occurs	15.02
Pressurizer becomes liquid solid	~50
Affected SG depressurization complete	~50
EFW to Unaffected SG Begins	50.43
Final PSV closure	~170
Peak Tave occurs	~240
Transient Analysis Ends	600

Table 4-2: Results for FWLB with First Trip Failed

Parameter	Value
Peak RCS pressure (psia)	2915.39
Peak thermal power (%RTP)	100.51
Peak Tave (°F)	622.22

Figure 4-1: FWLB with Fail First Trip – RCS Pressure

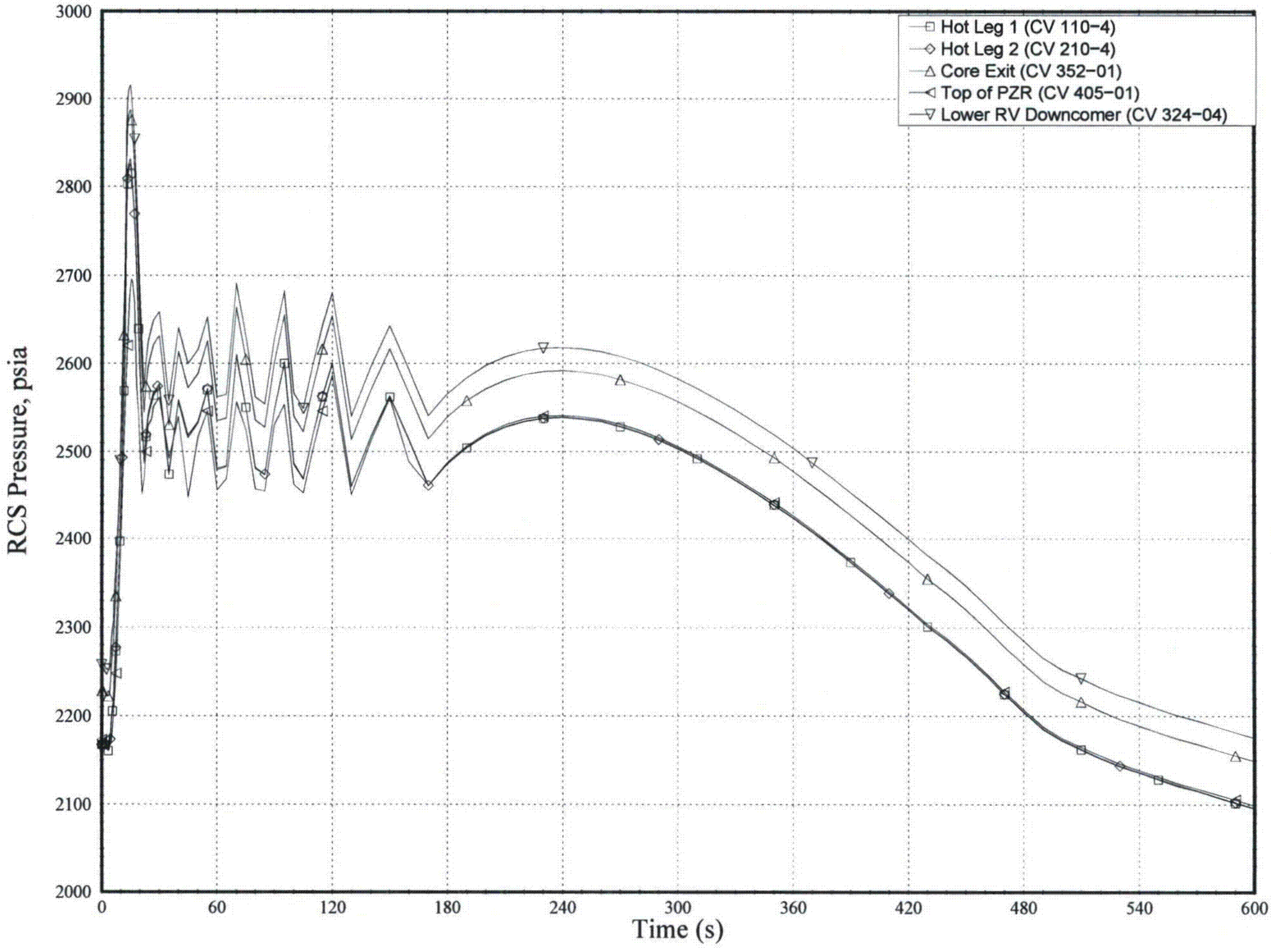


Figure 4-2: FWLB with Fail First Trip – Reactor Power

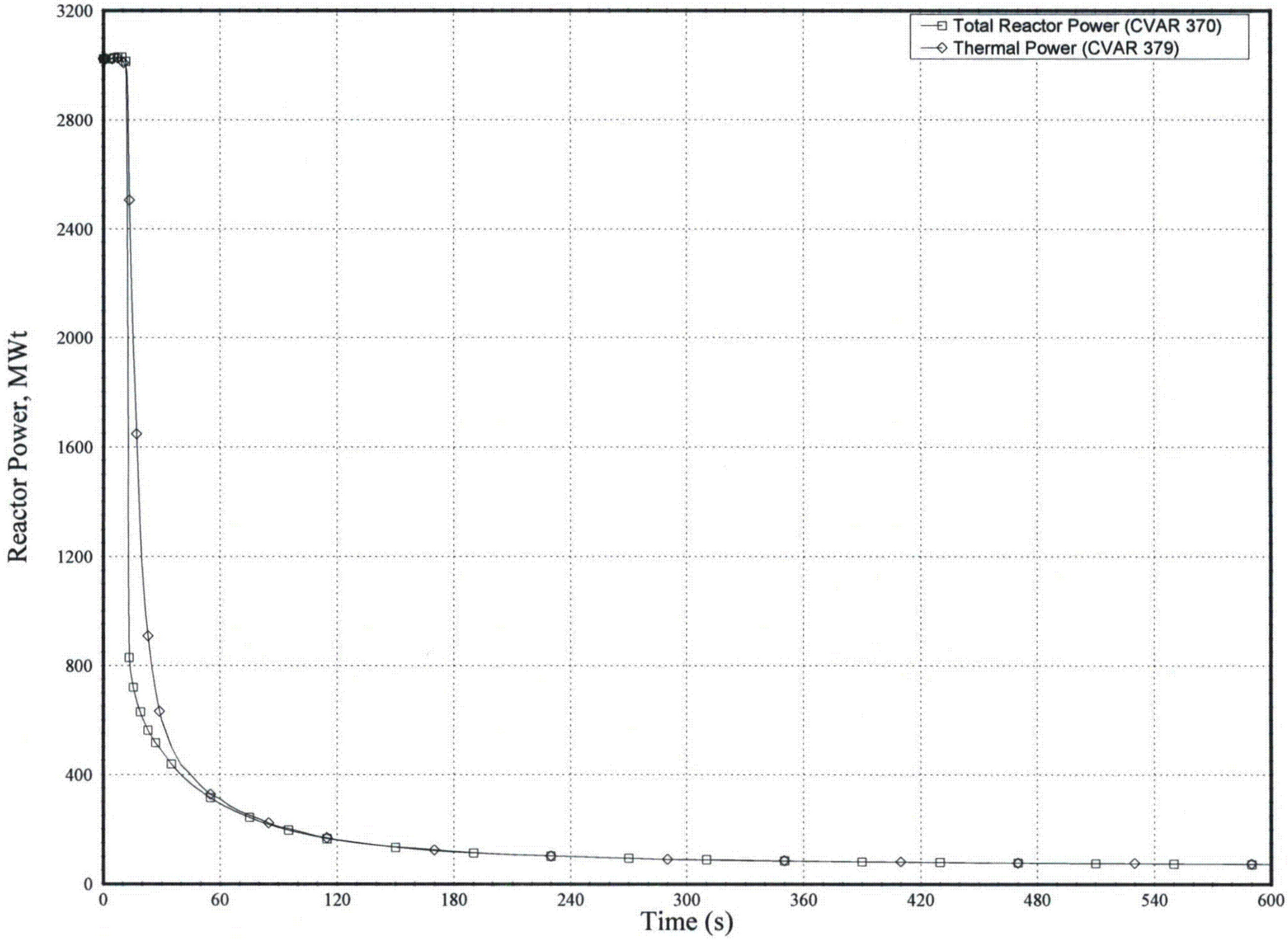


Figure 4-3: FWLB with Fail First Trip – Reactivity

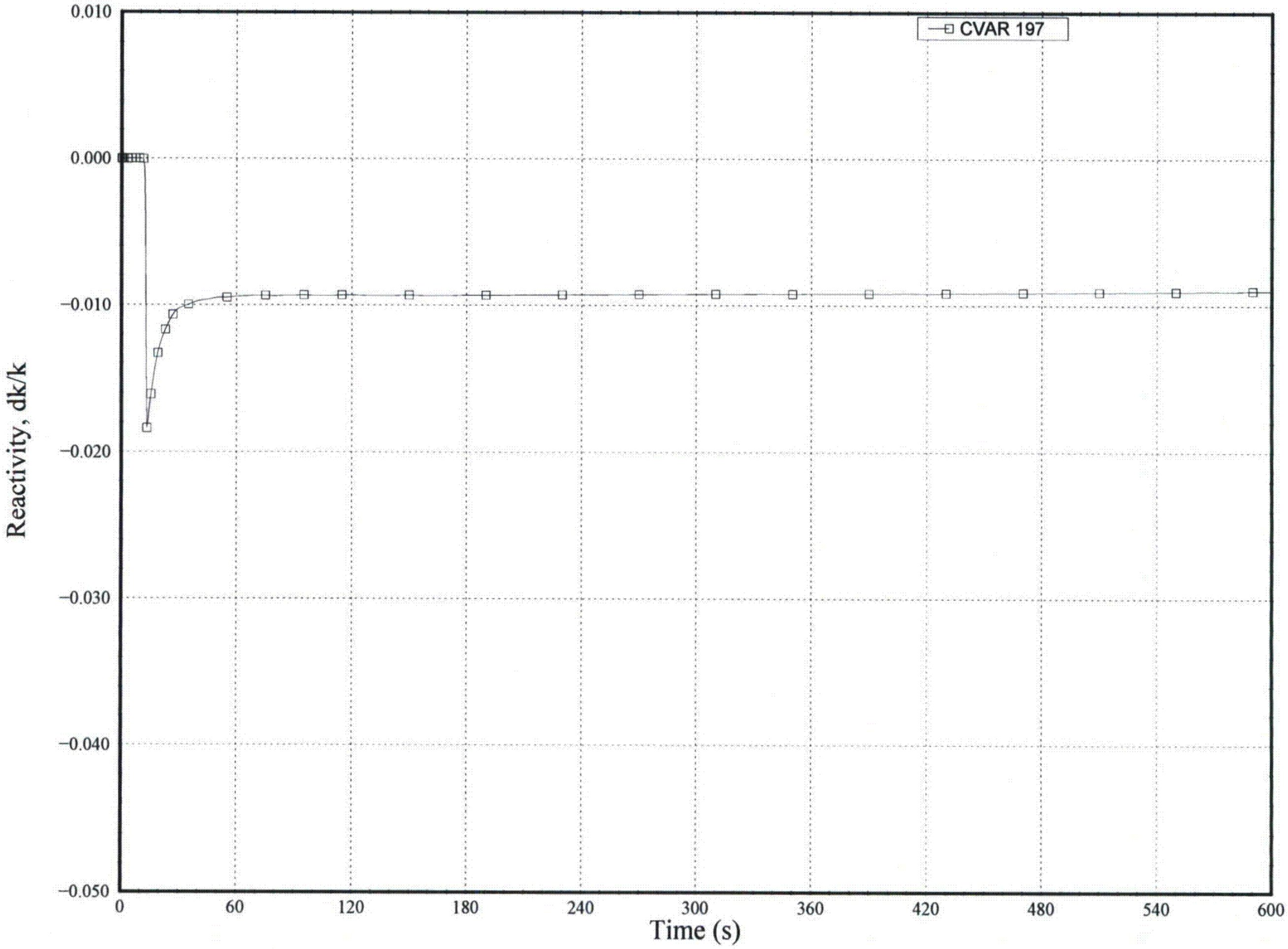


Figure 4-4: FWLB with Fail First Trip – Primary System Temperatures

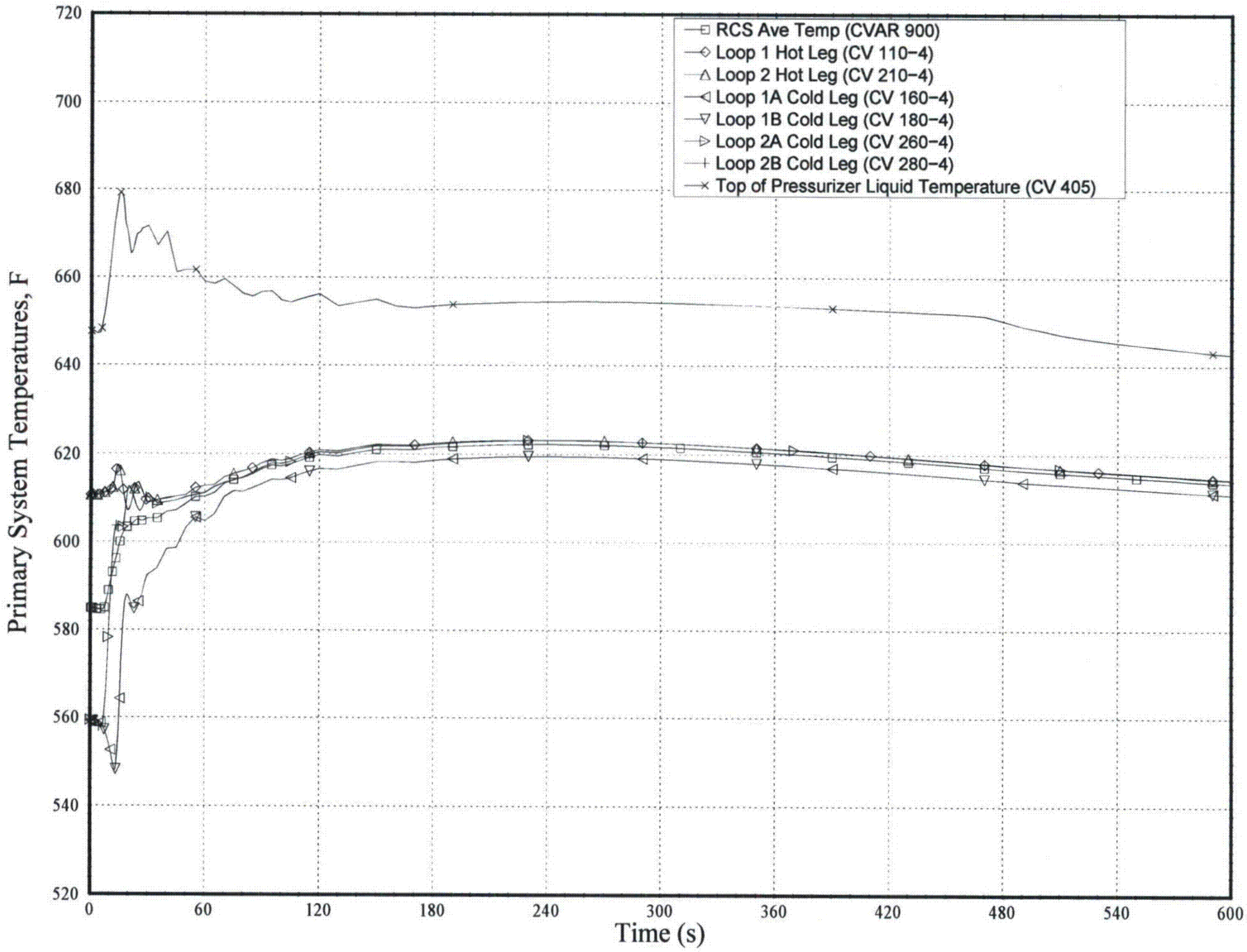


Figure 4-5: FWLB with Fail First Trip – Indicated Pressurizer Level

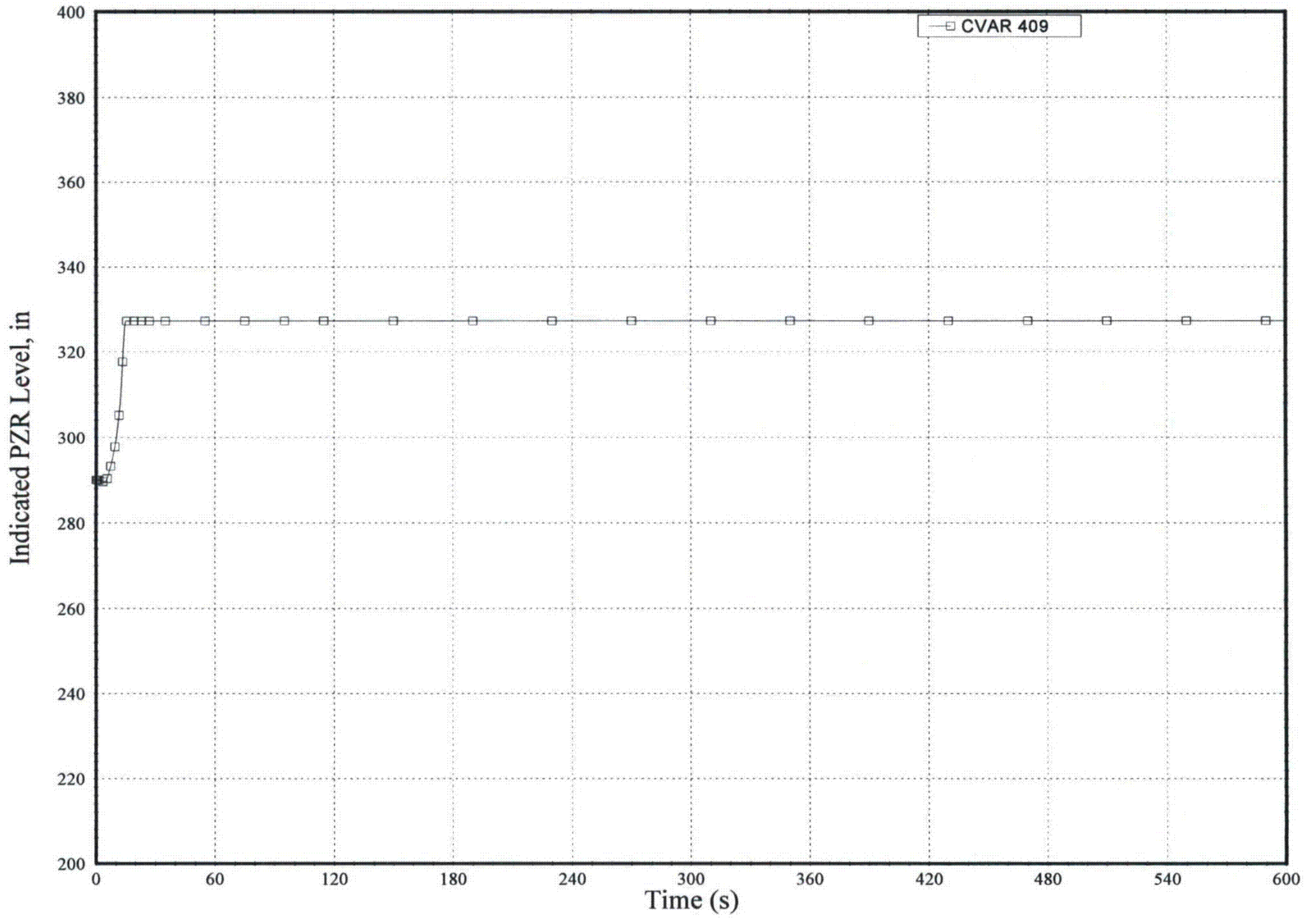


Figure 4-6: FWLB with Fail First Trip – Pressurizer Collapsed Liquid Level

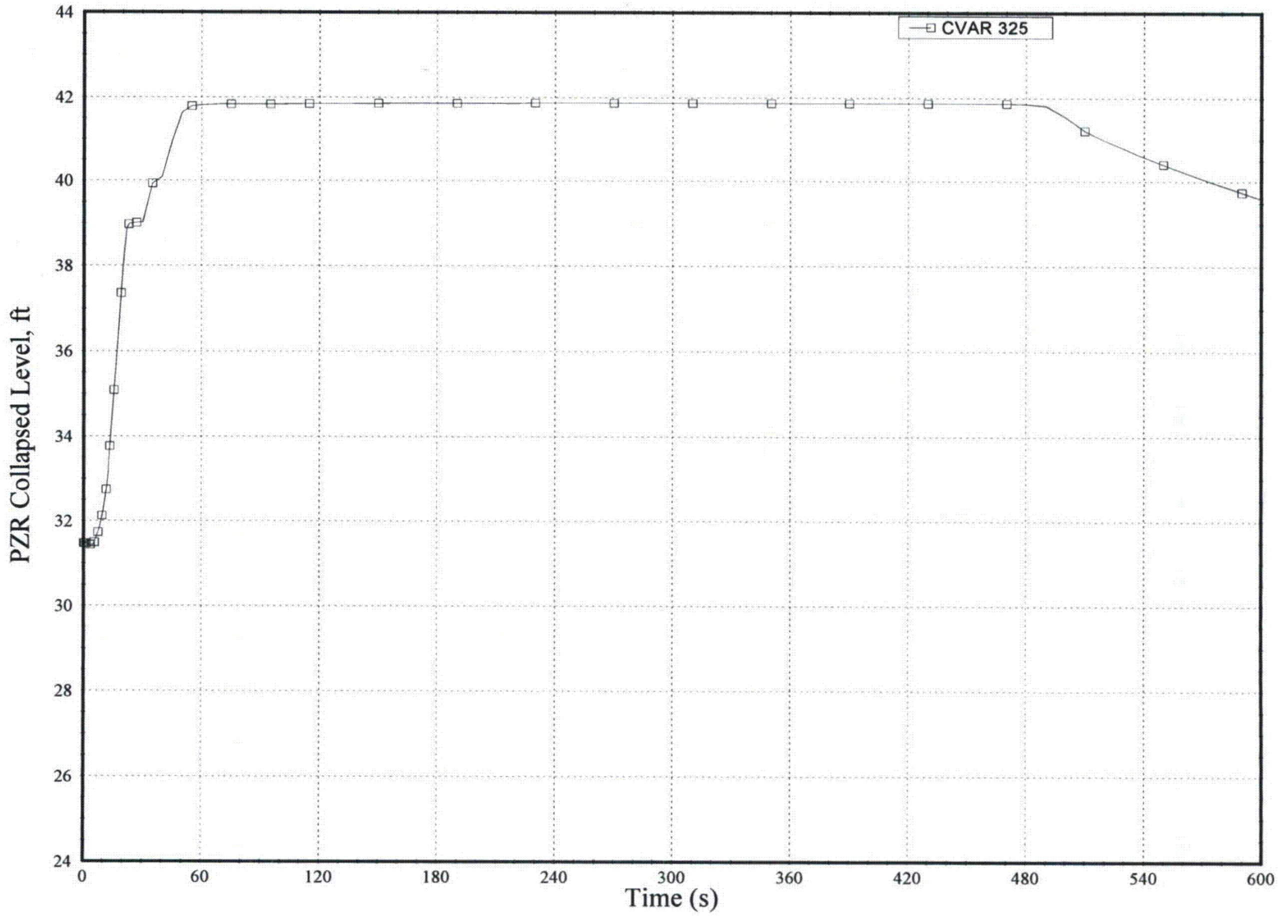


Figure 4-7: FWLB with Fail First Trip - Pressurizer Surge Line Flow

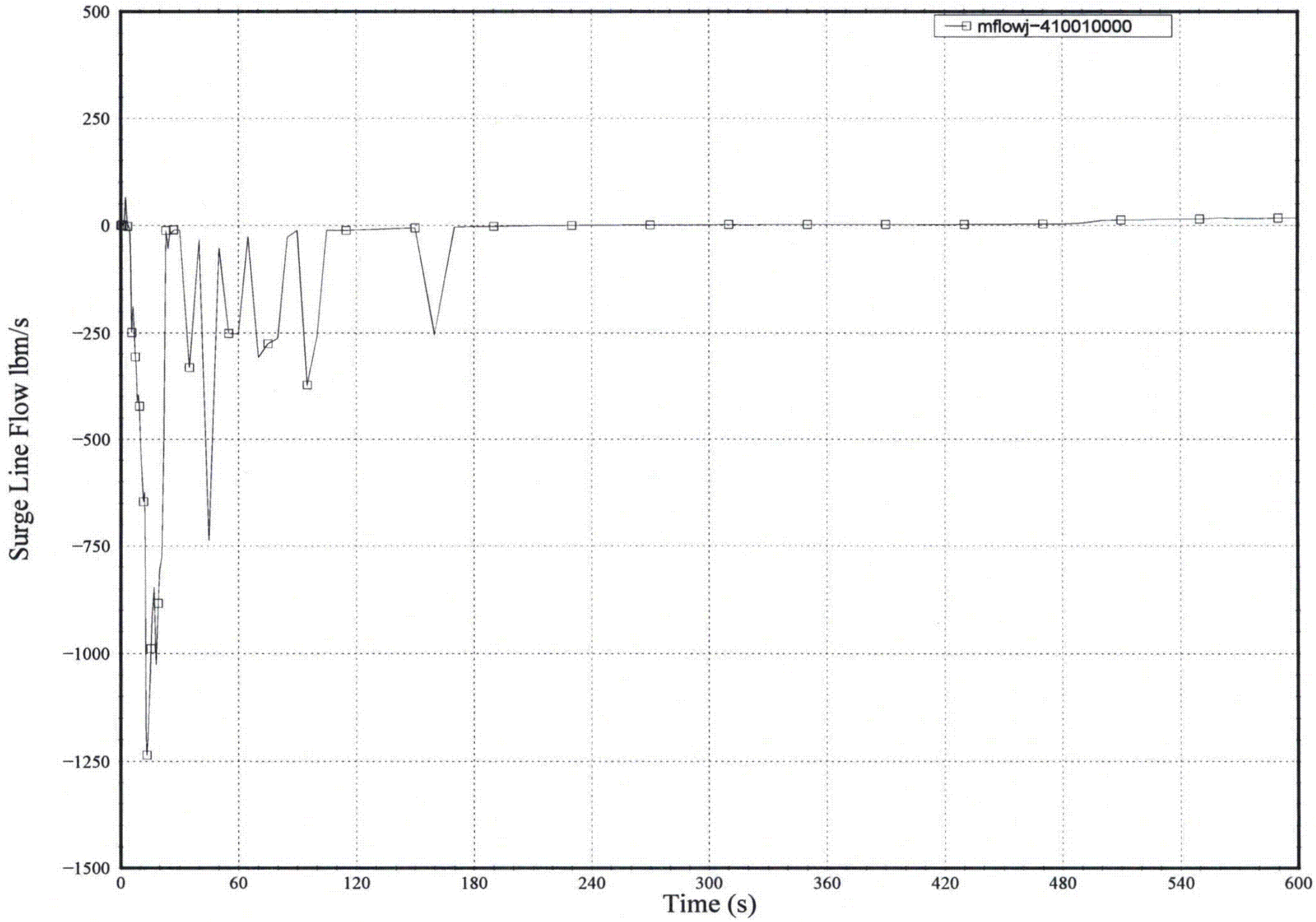


Figure 4-8: FWLB with Fail First Trip – RCS Volumetric Flow Rate

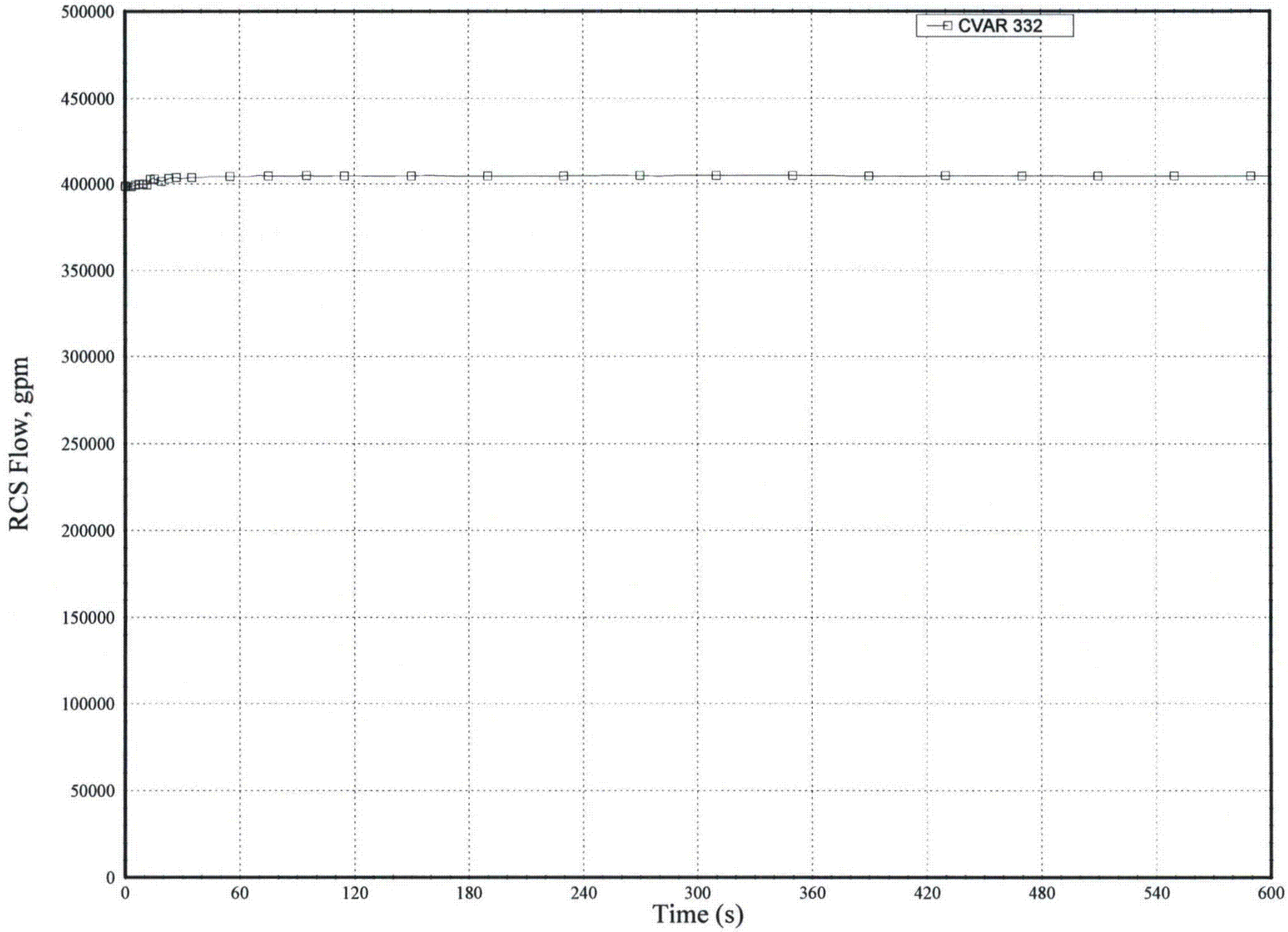


Figure 4-9: FWLB with Fail First Trip - Pressurizer Safety Valve Flow

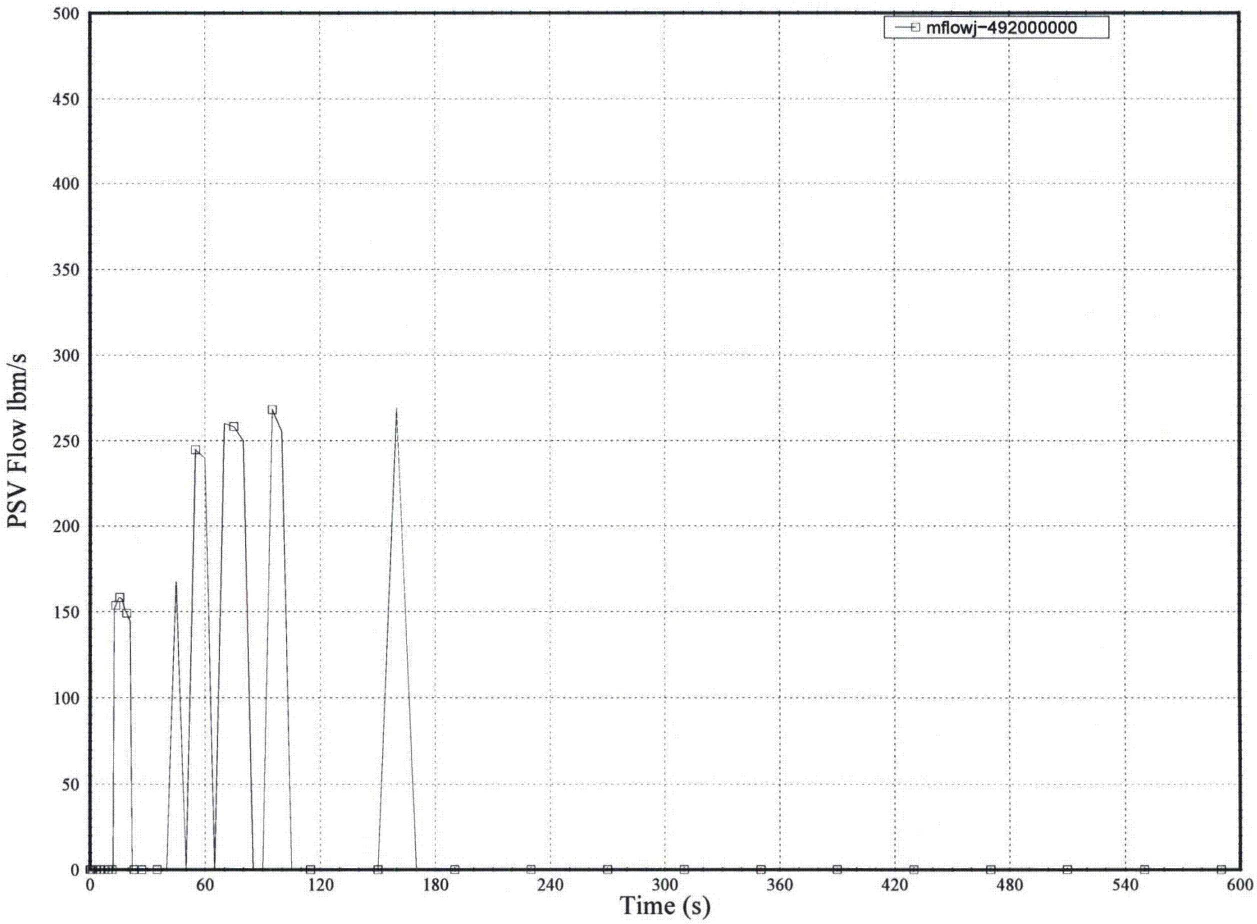


Figure 4-10: FWLB with Fail First Trip – SG Secondary Side Liquid Level

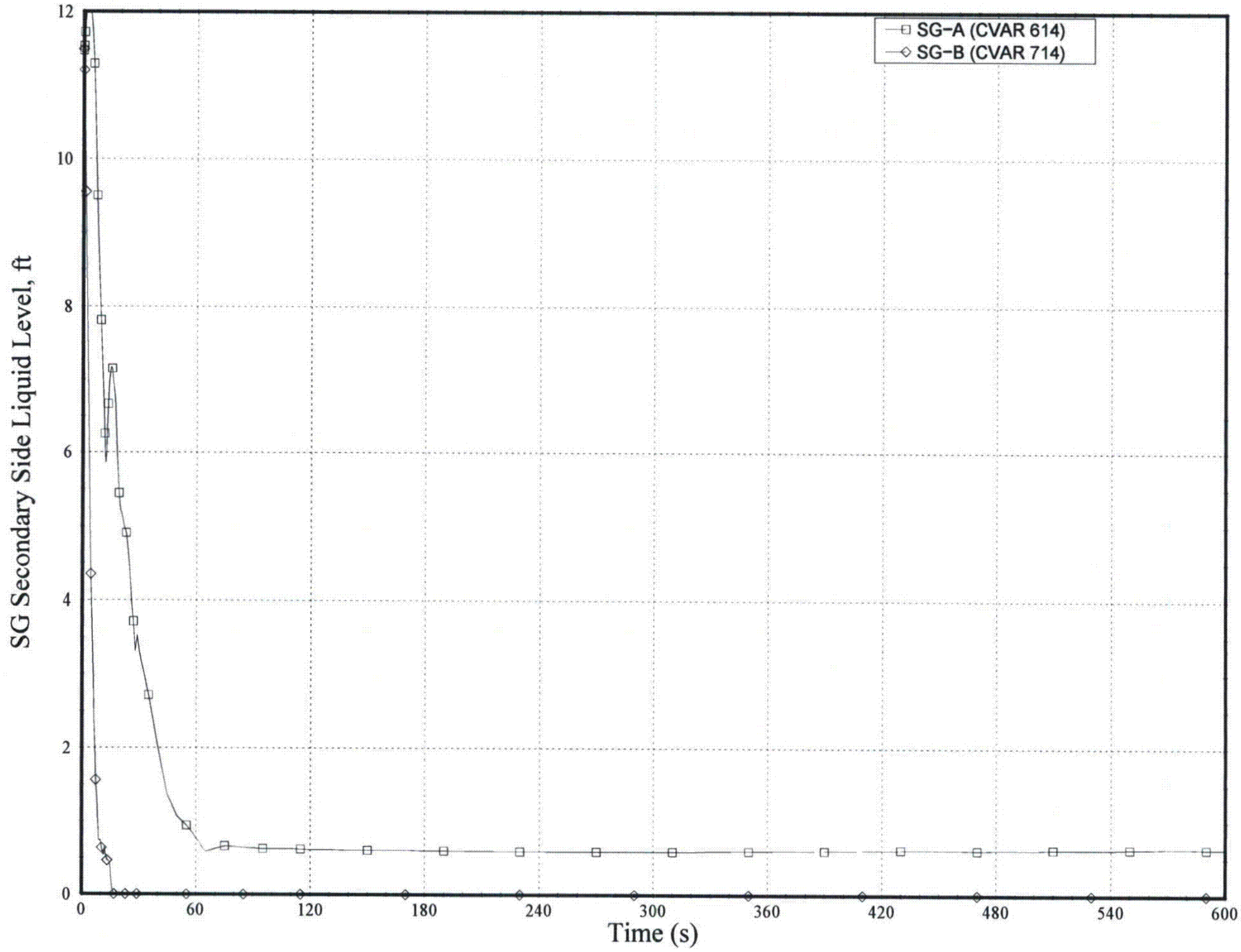


Figure 4-11: FWLB with Fail First Trip – SG Secondary Side Inventory

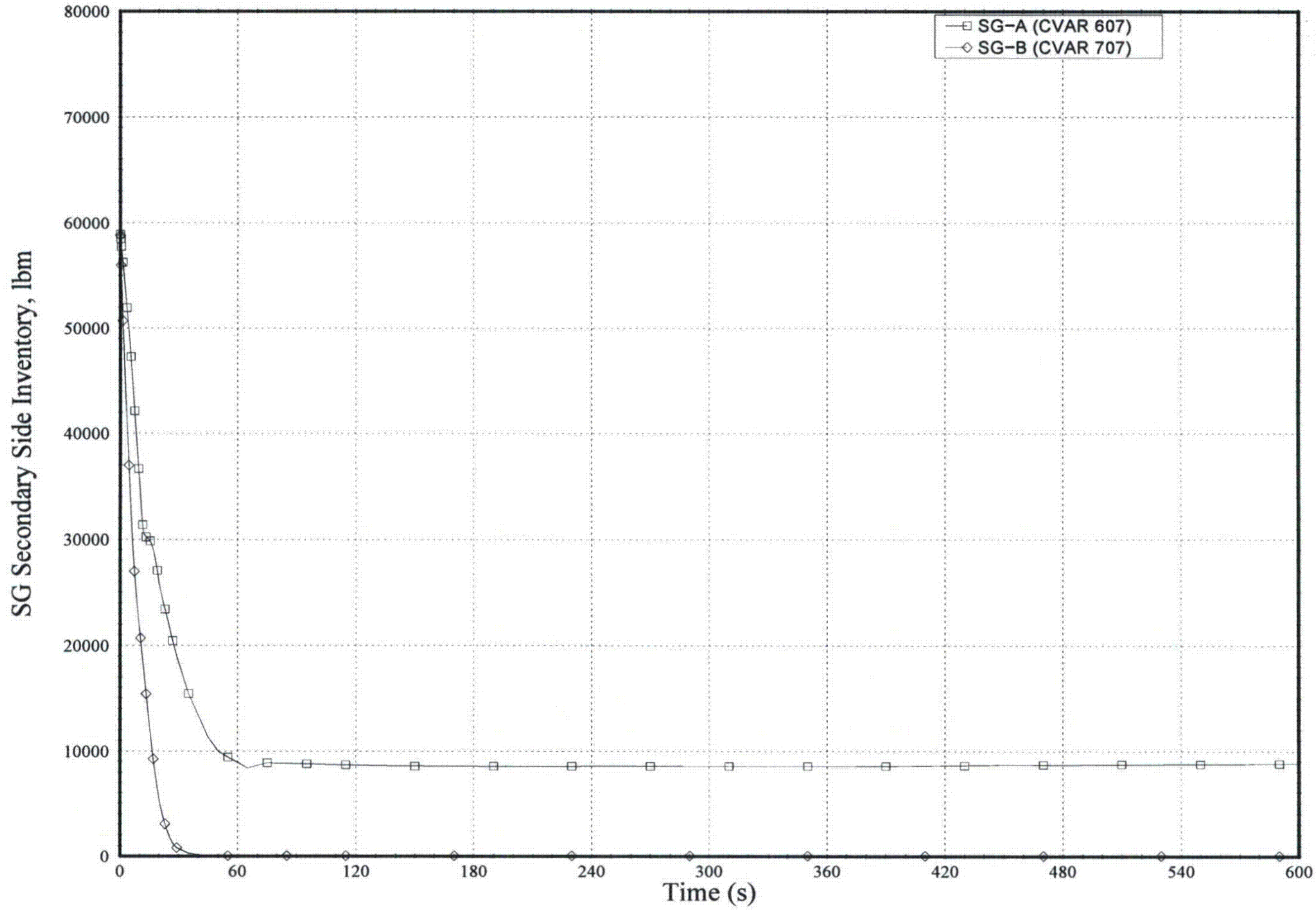


Figure 4-12: FWLB with Fail First Trip – SG % Operating Range

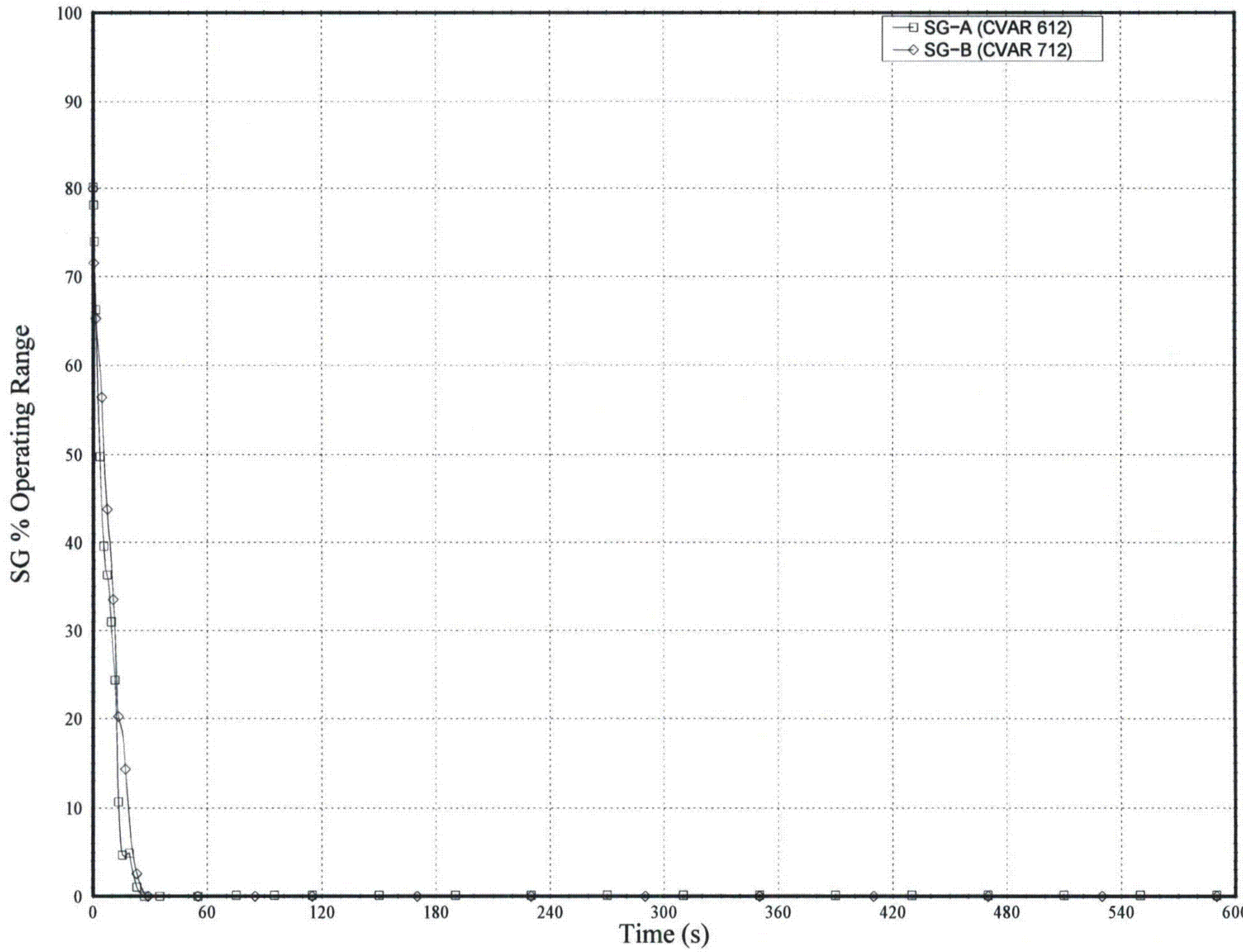


Figure 4-13: FWLB with Fail First Trip – SG Pressure

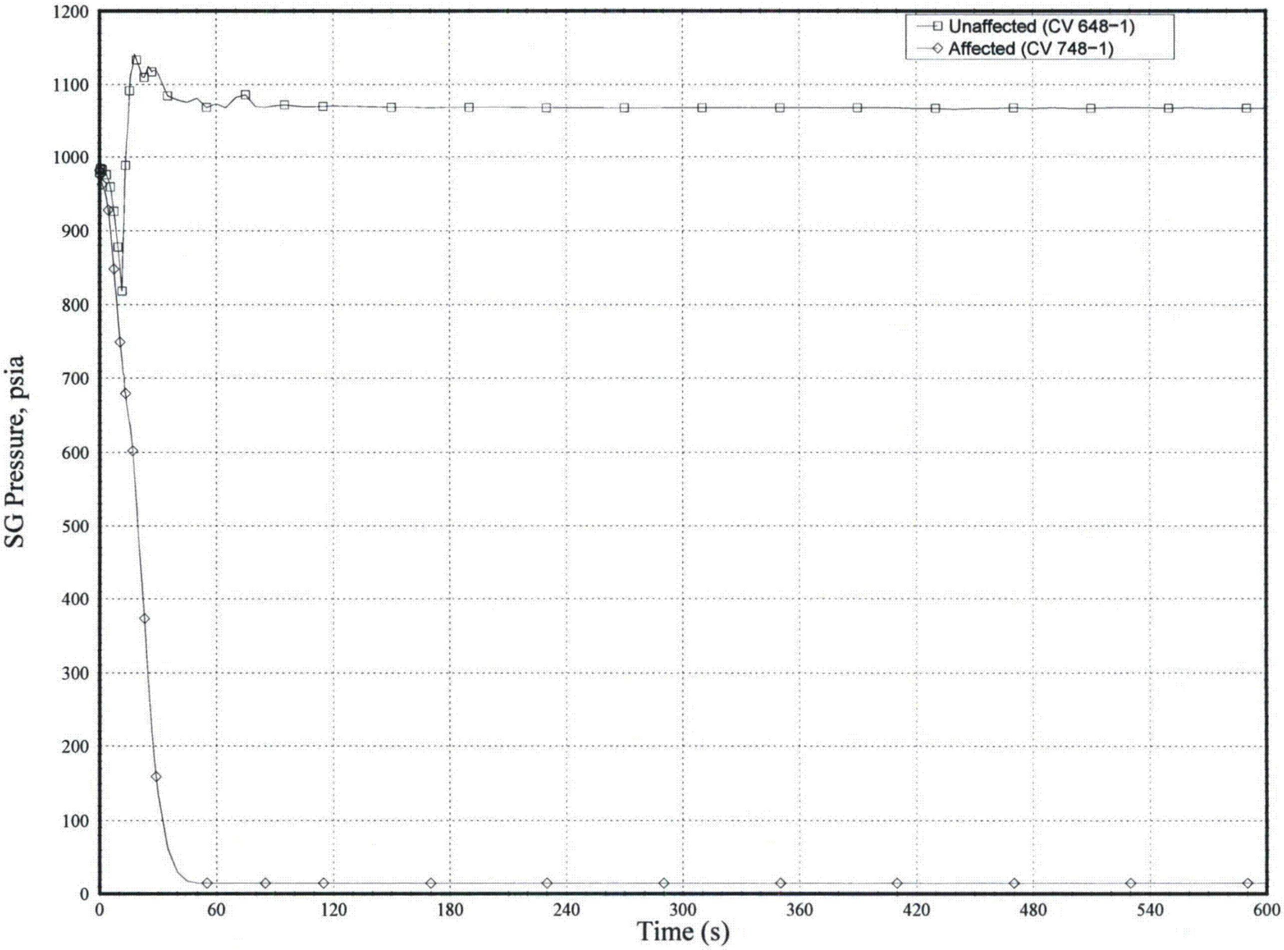


Figure 4-14: FWLB with Fail First Trip – EFW Flow

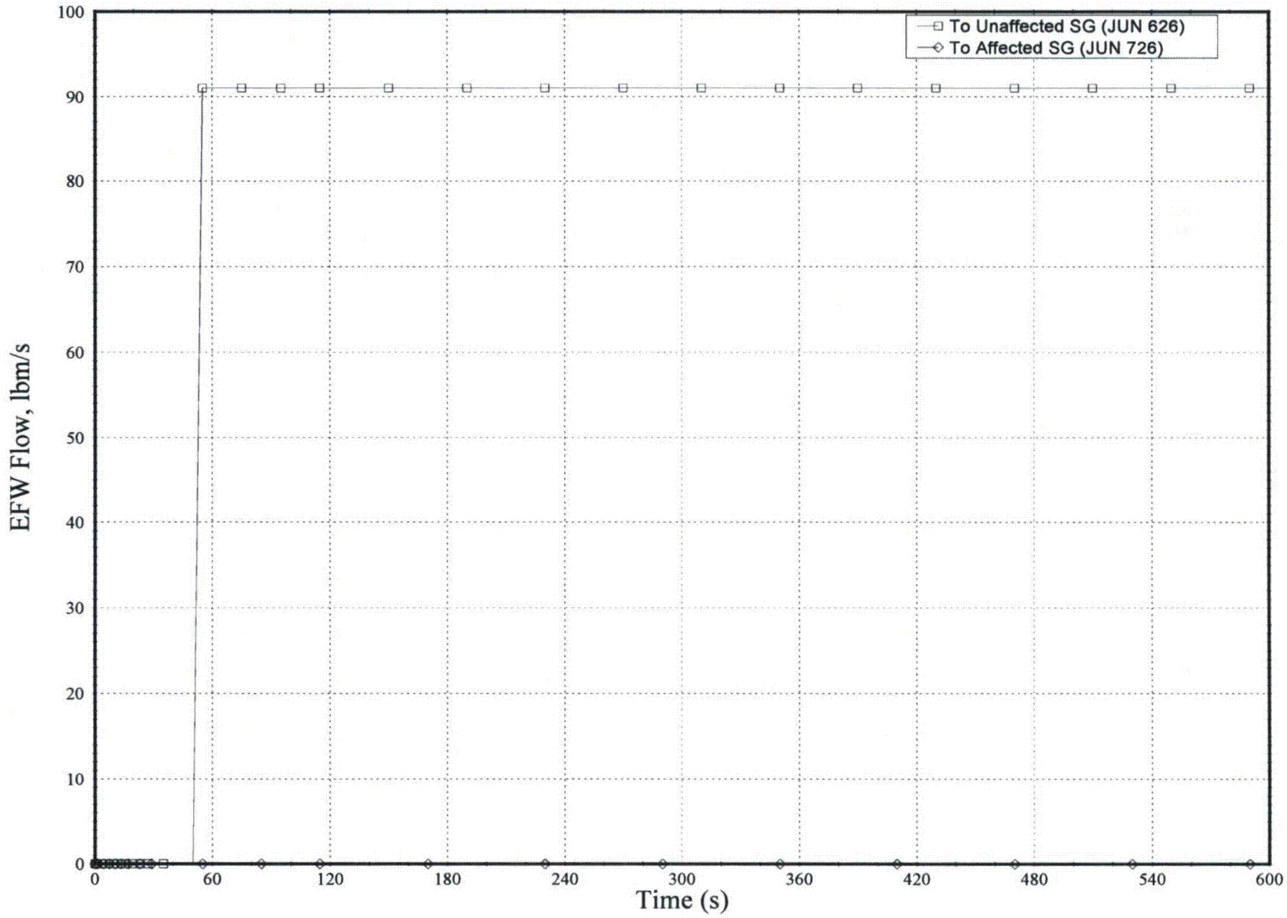
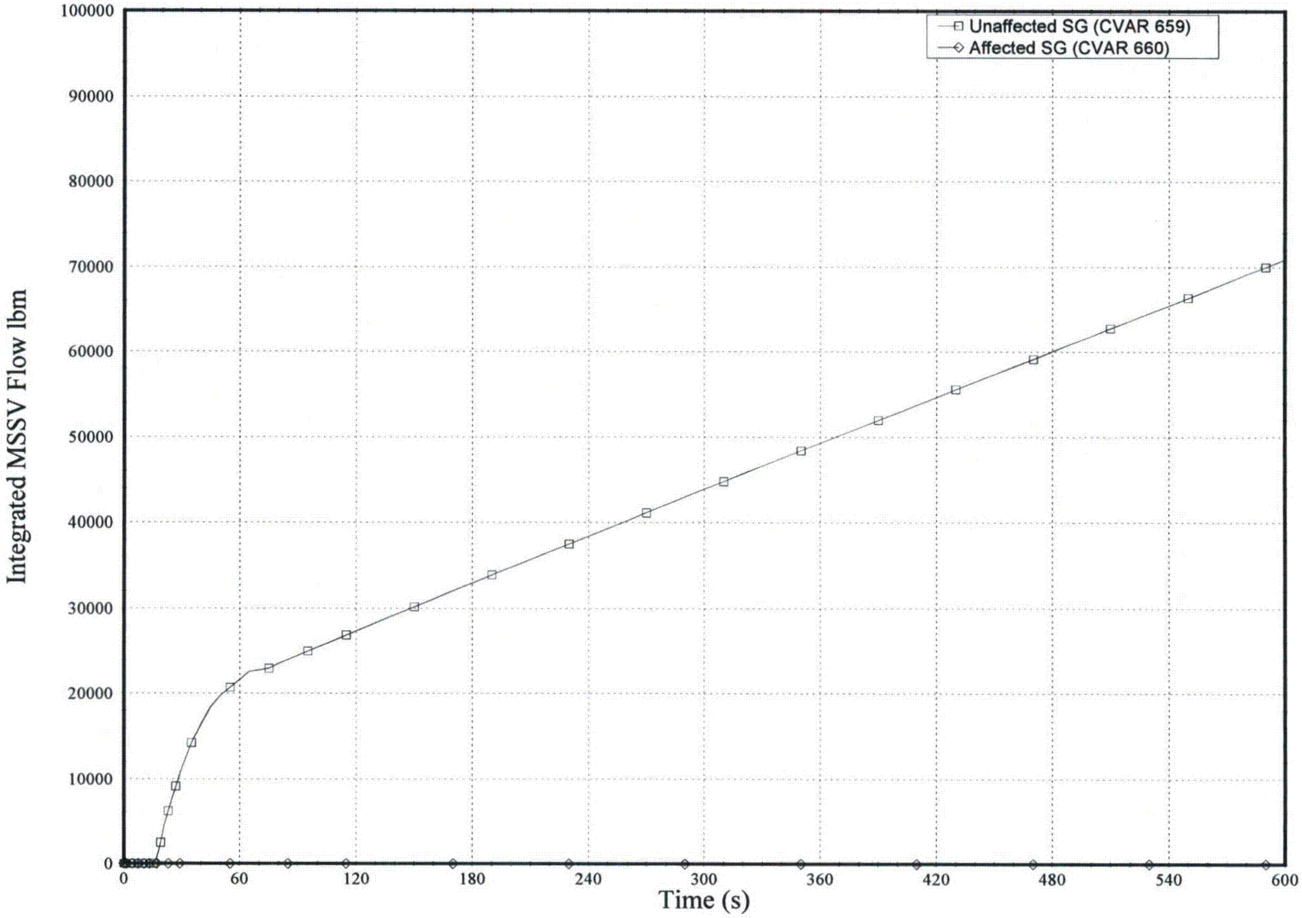


Figure 4-15: FWLB with Fail First Trip – Integrated MSSV Flow



5.0 REFERENCES

1. BAW-10164PA-06, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

2. BAW-10193PA-00, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurizer Water Reactors."

3. ANP-3114NP-000, "CR-3 EPU – Feedwater Line Break Analysis Sensitivity Studies."