Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

March 6, 2012

L-2012-085
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Re: St. Lucie Plant Unit 1
Docket No. 50-335
Renewed Facility Operating License No. DPR-67

Response To Request For Additional Information Identified During Audit Of The Safety Analyses Calculations for the Extended Power Uprate License Amendment Request

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2010-259), "License Amendment Request (LAR) for Extended Power Uprate," November 22, 2010, Accession No. ML103560419.

(2) NRC Reactor Systems Branch Audit Conducted at AREVA NP Inc. Facilities in Lynchburg, VA, January 30 and 31, 2012.

By letter L-2010-259 dated November 22, 2010 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. DPR-67 and revise the St. Lucie Unit 1 Technical Specifications (TS). The proposed amendment will increase the unit's licensed core thermal power level from 2700 megawatts thermal (MWt) to 3020 MWt and revise the Renewed Facility Operating License and TS to support operation at this increased core thermal power level. This represents an approximate increase of 11.85% and is therefore considered an Extended Power Uprate (EPU).

During the course of the NRC audit conducted at the AREVA NP Inc. facilities in Lynchburg, VA on January 30 and 31, 2012 [Reference 2], the NRC staff requested additional information to support the review of the safety analyses used in the St. Lucie Unit 1 EPU LAR. Additional information related to five events was requested. The events included: feedwater line break (FWLB), inadvertent opening of a power operated relief valve (IOPORV), chemical and volume control system (CVCS) malfunction, loss of electrical load (LOEL)/loss of condenser vacuum (LOCV), and realistic large break loss of coolant accident (RLBLOCA).

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The attachment to this letter provides the requested information and the FPL response for the FWLB event. The response to the requested information for the other four events is being provided in separate correspondence.

This submittal contains no new commitments and no revisions to existing commitments.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-259 [Reference 1].

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the designated State of Florida official.

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate LAR Project Manager, at 772-467-7138.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed on 6 - March - 2012

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachment

cc: Mr. William Passetti, Florida Department of Health

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION IDENTIFIED DURING AUDIT OF THE SAFETY ANALYSES CALCULATIONS

The following information is provided by Florida Power & Light (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support the Extended Power Uprate (EPU) License Amendment Request (LAR) for St. Lucie Unit 1 submitted to the NRC by FPL via letter L-2010-259 dated November 22, 2010, Accession Number ML103560419.

The NRC Reactor Systems Branch conducted an audit of the St. Lucie Unit 1 EPU safety analyses calculations at the AREVA NP Inc. (AREVA) facility in Lynchburg, VA on January 30 and 31, 2012. The NRC identified five events that require additional information. The events are:

- Feedwater line break (FWLB),
- Inadvertent opening of a power operated relief valve (IOPORV),
- Chemical and volume control system (CVCS) malfunction,
- Loss of electrical load (LOEL) / loss of condenser vacuum (LOCV), and
- Realistic large break loss of coolant accident (RLBLOCA).

The response to RAIs for IOPORV, CVCS malfunction, LOEL/LOCV and RLBBLOCA events is being provided in separate correspondence. The response to FWLB event is provided below.

Feedwater Line Break (FWLB)

Submit a feedwater line break (FWLB) heatup analysis. Provide one case with offsite power available and one case without offsite power available.

Response

1.0 Introduction

The current licensing basis for St. Lucie Unit 1 does not include an analysis for the FWLB event as a heatup event. The evaluation and results presented below:

- Demonstrate the adequacy of the auxiliary feedwater (AFW) system to remove post-trip decay heat and maintain reactor coolant system (RCS) subcooling margin until RCS heatup is terminated with the AFW heat removal capacity exceeding the decay heat; and
- Verify the steam generator (SG) low level reactor trip setpoint to be sufficiently conservative to perform the reactor protection function, after accounting for the uncertainties associated with the harsh environment that could be created by the break of a feedwater line inside containment.

1.1 Summary of Findings

Key analysis inputs for this representative case are consistent with those that are typically limiting for Combustion Engineering (CE) plants with feedring-type SGs. These inputs include modeling the largest double-ended break possible for the St. Lucie Unit 1 SG model and minimum reactivity feedback parameters.

The evaluation demonstrates that a FWLB event under EPU conditions, with input assumptions typical of limiting analyzed cases, affords adequate margin to hot leg saturation. This information provides reasonable assurance that the consequences of a FWLB event do not present a safety concern for operation of St. Lucie Unit 1 at EPU conditions.

2.0 Event Description

The FWLB incident is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to maintain shell-side fluid inventory in the SGs. If the break is postulated in a feedline between the check valve and the SG, fluid from the SG will be discharged through the break. (In contrast, if a break occurred upstream of the feedline check valve, the transient would progress as a loss of normal feedwater event.) Furthermore, because the AFW piping connects to the main feedline, a break between the check valve and the SG could preclude the subsequent addition of AFW to the affected SG. Depending upon the size of the break and the plant operating conditions at the time of the rupture, the break first causes a cooldown (by excessive energy discharge through the break), followed by a heatup of the RCS. Because the consequences of an RCS cooldown resulting from a FWLB are bounded by the cooldown consequences of a steam system piping failure, the FWLB event was analyzed with respect to RCS heatup effects.

As the subcooled feedwater flow to the SGs is reduced by a FWLB, the long-term capacity of the secondary system to remove heat from the RCS is diminished. The feedwater flow reduction can cause RCS temperatures to increase prior to reactor trip. -Additionally, fluid inventory of the faulted SG may be discharged through the break, which will reduce the heat sink volume available for decay heat removal following a reactor trip. The FWLB event is analyzed to demonstrate the ability of the AFW system to adequately remove long-term decay heat and prevent excessive heatup of the RCS.

In the analysis performed, the break was assumed to be located in a feedline between the check valve and the SG. A break in this location results in the discharge of fluid from the associated SG. The size of the break and the functionality of the main feedwater (MFW) control system are two important factors during a FWLB transient. Some breaks may be small enough that a properly functioning MFW control system will be able to completely make up for the resultant inventory loss. In contrast, larger feedline breaks can cause a sizeable blowdown (inventory loss) that prevents the MFW control system from being able to supply enough feedwater to maintain shel-side fluid inventory in the SGs. This then leads to a SG low level reactor trip and AFW actuation. Another important factor during a FWLB transient is the shell-side fluid inventory in the intact SG at the time of reactor trip. It is conservative to employ analysis assumptions that minimize this fluid inventory because it minimizes the heat removal capability of the SG, which in turn maximizes the RCS heatup. For this purpose, the analysis performed assumes that MFW is completely terminated (to both SGs) at the time the break occurs. Furthermore, the initial water level in the faulted SG is assumed at its highest level consistent with full-power conditions (to delay reactor trip on SG low level), while the initial water level in the intact SG is at its lowest (to minimize inventory available for long-term heat removal).

Early in the event, there is a rapid decrease in reactor coolant temperature as over-cooling temporarily occurs in the faulted SG. When the SG water level in the affected SG reaches the analysis-assumed conservative SG low level reactor protection system (RPS) setpoint, a reactor trip occurs. A turbine trip shortly after reactor trip causes a sudden reduction in steam flow and a further reduction in the heat removal capacity of the SG. The steam bypass control system (SBCS) was modeled in the offsite power available case to deplete the SGs of inventory slightly faster. With the reduced steam flow, the steam pressure in the intact SG rapidly increases to the setpoint of the first (lowest setpoint) main steam safety valves (MSSVs), and remains there until the RCS heatup ceases, i.e., until the heat removal capability of the intact SG being fed AFW is sufficient to remove the decay heat generated in the core (also known as the time of event turnaround). During the heatup period after reactor trip, the pressurizer pressure increases to, and is maintained near, the pressurizer power operated relief valve (PORV) setpoint. At event turnaround, the RCS temperature and pressure, and the pressurizer water level begin to decrease again, and the heatup transient is over. Subsequently, the plant operators can follow the applicable emergency operating procedures (EOPs) to bring the plant to a stabilized condition.

The intent of the analysis was to maximize the potential for reaching saturated conditions in the RCS hot legs. Some of the key characteristics of the analysis are described below.

3.0 Analytical Methodology

The FWLB transient is analyzed by employing the S-RELAP5 computer code. The code simulates the neutron kinetics, RCS, pressurizer, pressurizer PORVs and safety valves, pressurizer spray, SG, and MSSVs. The code computes pertinent plant variables including temperatures, pressures, and power level.

Normal reactor control systems are not required to function. The RPS functions to trip the reactor on the appropriate signal, while the engineered safeguards features actuation system (ESFAS), primarily the auxiliary feedwater actuation signal (AFAS), is actuated to provide long-term decay heat removal capability via AFW. No single active failure will prevent the RPS from functioning properly. A limiting single active failure in the AFW system was included to minimize the available AFW flow to the intact SG.

Inputs were modeled to maximize the potential for reaching saturated conditions in the hot and/or cold legs, as provided in Table 3-1 below.

The analysis of this event was performed with the reactor initially operating at the EPU rated thermal power plus uncertainty. This is bounding because the stored energy of the primary system is maximized at full power conditions, and therefore the disparity between the heat generated in the RCS and the heat removal capacity of the SGs is maximized.

The cases analyzed investigated the largest possible break size, i.e., 1.12 ft^2, which is consistent with a break occurring at the SG nozzle. For smaller breaks, the MFW system would tend to result in MFW penetrating the intact SG, thus mitigating the loss of heat sink. The largest break size possible will result in the most MFW being diverted away from the intact SG, through the piping network to the break.

Cases were analyzed for both (a) offsite power available and (b) loss of offsite power (LOOP) following reactor trip. Reactor trip was assumed to occur on a faulted SG low level signal received when level reached 5% of the narrow range span (NRS). Additional available RPS trips include the high pressurizer pressure (HPP) and SG low pressure RPS trips. All RPS trips setpoints included allowance for harsh containment conditions and maximum signal processing time delays. Also, a maximum holding coil release time was assumed to conservatively delay the initiation of rod motion upon scram.

It was assumed that the flow from the motor-driven AFW pump supplying the faulted SG was lost through the break. The other motor-driven AFW pump (supplying the intact SG) is capable of supplying a minimum flow of 296 gpm following a 330-second delay. (This delay was assumed for both the offsite power available and LOOP cases.) No credit was taken for operator action to re-direct the other motor-driven AFW pump supplying the faulted SG to the intact SG. Since the AFW control logic would isolate the faulted SG, the turbine-driven AFW pump was assumed to be the single active failure because that pump would otherwise feed the intact SG with a higher capacity than the single motor-driven pump. AFW flow was assumed to be at the maximum temperature of 100°F plus 4°F uncertainty. SG blowdown was assumed to be on for 30 minutes at 65 gpm.

Table 3-1
Analysis Parameters

Parameter	Bias	Biased Value	Comment
Initial reactor power	Rated thermal power plus uncertainty	3029.06 MWt (3020 MWt + 0.3%)	Maximum core power increases the mismatch between the RCS and intact SG. Decay heat is also maximized.
Initial core inlet temperature	Technical Specification (TS) maximum plus uncertainty	554°F (551 + 3°F)	Initial value does not significantly affect results.
Initial pressurizer pressure	Low	2185 psia (2225 – 40 unc.)	Initial value does not significantly affect results.
Initial pressurizer level	High	68.6% (65.6 + 3 unc.)	Initial value does not significantly affect results.
Initial reactor coolant flow rate	TS minimum	375,000 gpm	Decreases heat transfer between the RCS and the intact SG.
Initial SG liquid level	Faulted: high Intact: low	Faulted: 70% (65 + 5 unc.) Intact: 60% (65 – 5 unc.)	High faulted SG level delays reactor trip; low intact SG level limits heat removal capability.
SG tube plugging	High	10%	Decreases heat transfer area between the RCS and the intact SG.
Decay heat	Per methodology	ANS 73 + actinides	Conservatively high decay heat.
Moderator temperature coefficient (MTC)	TS most-positive HFP limit	+2 pcm/°F	Increases core power prior to trip.
SG low level RPS trip	Low	Nominal = 20.5% Analysis value = 5% (exceeds 14% harsh condition uncertainty)	Minimum setpoint delays reactor scram. Value used of 20.5% has margin to the TS setpoint value of 35%.
SG low level RPS trip circuit delay	High	0.9 s	Maximum instrumentation actuation delays reactor trip.
High pressurizer pressure RPS trip	Low	2320 psia (2400 -80 psi harsh condition uncertainty)	Biased low because it uses same signal as PORV. (Lower RCS pressure provides challenge to subcooling.)
High pressurizer pressure RPS trip delay	High	0.9 s	Maximum instrumentation actuation delays reactor trip.

Table 3-1 (continued)
Analysis Parameters

Parameter	Bias	Biased Value	Comment
SG Low pressure RPS trip	Low	400 psia (600-200 psi for harsh conditions)	Minimum setpoint delays reactor trip.
SG low pressure RPS trip circuit delay	High	0.9 s	Maximum instrumentation actuation delays reactor trip.
Scram delay	High	0.5 s	Maximum holding coil release time delays reactor power reduction.
Main Steam Isolation System (MSIS)	SG low pressure	400 psia with 6.9 s valve stroke time	600-200=400 psia (See SG low pressure above.)
SG low level AFAS trip	Low	5.0% (19.0% - 14% nominal less harsh conditions uncertainty)	Biased low setpoint delays AFW to the intact SG.
SG low level AFAS trip delay time	High	330 s (irrespective of availability of offsite power)	Diesel generator starting and sequence loading delays included. Maximum time delays availability of AFW to the intact SG.
AFW flow rate	Minimum	296 gpm (one motor driven pump)	No credit was taken for re-direction of flow from motor driven AFW pump associated with the faulted SG.
AFW temperature	TS maximum plus uncertainty	104°F (100°F + 4°F unc.)	Slightly reduces the heat removal capability of the AFW.
MFW delivery and termination	Conservative	Instantaneously isolated at event initiation (both SGs).	Conservative assumption that maximizes the challenge to the AFW in the intact SG.
Single-failure	--	Loss of turbine-driven AFW pump	AFW isolation logic would divert all turbine driven pump flow to the intact SG. Turbine driven flow rates are much greater than motor driven, >600 gpm vs. 296 gpm. Thus, a failure of the turbine driven pump is worse than a failure of the motor driven pump to the intact SG.

Table 3-1 (continued)
Analysis Parameters

Parameter	Bias	Biased Value	Comment
High pressure safety Injection (HPSI) actuation pressure	Low	Modeled to actuate at 1520 psia (1600-80 psi for harsh conditions)	The setpoint was biased low to prevent HPSI injection (which would benefit subcooling). HPSI did not actuate due to the high RCS pressures.
Charging	Not modeled	Not modeled	Charging would provide a benefit to the subcooling margin by cooling the RCS.
MSSV setpoints	Nominal plus tolerance	Bank 1: 1,000 psia + 3% Bank 2: 1,040 psia +3%	Bank 2 tolerance is +2%, however, for this analysis the opening setpoint was determined using a conservatively large +3% tolerance.
Reactor coolant pump (RCP) heat generation	High	21.7 MW_t total for 4 RCPs	Maximum RCP heat increases the heat load on the intact SG.
Pressurizer heaters	N/A	Not modeled	Heat addition to the pressurizer would benefit subcooling by raising the saturation temperature.
PORV setpoint	Nominal less uncertainty	Open: 2320 psia Reset: 2296 psia	RCS pressure controlled by the PORVs during time period that challenges subcooling. The setpoints were biased low to minimize the pressure and, consequently, the saturation temperature.
PORV flow rate	Design value + 10%	153,000 lbm/hr per valve + 10% Analysis value: 168,300 lbm/hr per valve at 2400 psia	PORV capacity maintains pressure at the opening setpoint during the heatup period.

Table 3-2
Reactor Trip Status

Parameter	Assumed Status
Thermal margin/low pressure (TM/LP)	Disabled
High pressurizer pressure (HPP)	Available
Reactor trip on turbine trip	Disabled
Steam generator low level	Available
Low primary flow	Disabled
Steam generator low pressure	Available
Steam generator differential pressure	Disabled

Table 3-3
Equipment Status

Equipment / System	Assumed Status
Rod position controller	Manual
Pressurizer heaters	Disabled
Pressurizer spray	Available
Pressurizer PORVs	Available
SG blowdown flow	Available
Steam dump and bypass valves	Modeled (no LOOP case)
Steam atmospheric dump valves	Not modeled (Manual)
Reactor coolant pumps	Operating per Mode 1
Main feedwater	Isolated at event initiation
Auxiliary feedwater	Available, consistent with single-failure assumptions
Charging pumps	Not modeled
Letdown flow	Not modeled
Rod block system	Disabled
Turbine control valve	Automatic
Operational mode	Mode 1, Full Power

The PORVs were activated at nominal pressure less uncertainty for the analysis herein. This was done because the RCS pressure is controlled by the PORV for a significant period in this transient. The pressure (and the figure of merit subcooling) would be higher during that period if the PORVs were disabled.

For the no-loss of offsite power (LOOP) case, it was assumed that the operator trips all four RCPs at 15 minutes after reactor trip in accordance with the EOPs, with expected SIAS on high containment pressure. For the LOOP case, the RCPs are assumed to trip at the time of turbine trip on reactor trip.

4.0 Summary of Results

In this analysis, two scenarios were considered:

- Offsite power available (no LOOP), and
- Loss of offsite power (LOOP).

This evaluation demonstrates that a FWLB event, with input assumptions typical of the limiting analyzed case, affords adequate margin to hot leg saturation under EPU conditions. This information provides reasonable assurance that the consequences of a FWLB event do not present a safety concern for operation at EPU conditions.

The sequence of events and selected graphical results are provided below in Table 4-1, Table 4-2, and Figures 4-1 through 4-12. Table 4-3 provides a summary of the analysis.

Table 4-1
Sequence of Events for FWLB with Offsite Power Available

Event	Time (s)	Comment
Double ended guillotine break (DEGB) of MFW nozzle occurred, resulting in assumed complete loss of MFW to both SGs	0	Area = 1.12 ft^2
SG low level setpoint reached	11.0	5%NR in faulted SG
SG low level trip occurred	11.9	
Rod motion began	12.4	
High SG differential pressure (DP) trip setpoint reached (not credited)	20.7	335 psid
SG low pressure trip setpoint reached[(1)]	24.2	400 psia
AFAS setpoint reached in intact SG	38.7	5% NR in intact SG
High pressurizer pressure trip and PORV setpoint reached[(1)]	364	2320 psia
AFW began to intact SG	369	Signal plus 330 s delay
RCPs tripped by operator	912	15 minutes after reactor trip
AFW heat removal matched core decay heat and peak RCS temperature occurred	2090	9°F subcooling at that time
Calculation terminated	3000	

(1) Reactor trips on SG low level RPS trip, which occurs earlier.

Table 4-2
Sequence of Events for FWLB with Loss of Offsite Power

Event	Time (s)	Comment
DEGB of MFW nozzle occurred, resulting in assumed complete loss of MFW to both SGs	0	Area = 1.12 ft^2
SG low level setpoint reached	11.0	5%NR in faulted SG
SG low level trip occurred; RCPs began coastdown based on assumed loss of offsite power	11.9	
Rod motion began	12.4	
High SG DP trip setpoint reached (not credited)	17.0	335 psid
SG low pressure trip setpoint reached(1)	24.5	400 psia
AFAS setpoint reached in intact SG	33.7	5% NR
High pressurizer pressure trip and PORV setpoint reached(2)	116	2320 psia
AFW began to intact SG	364	Signal plus 330 s delay
AFW heat removal matched core decay heat and peak RCS temperature occurred	2276	28°F subcooling at that time
Calculation terminated	3000	

(1) Reactor trips on SG low level RPS trip, which occurs earlier.

Table 4-3
Summary of Results

AFW Heat Removal Matched Decay Heat (1)		
Case	**Time (s) (2)**	**Subcooling (°F)**
No LOOP	2090	9
LOOP	2276	28

(1) The time that AFW heat removal capability matched decay heat production was defined as the time at which RCS temperature peaked.

(2) In each case, the calculation was run to 3000 seconds to provide an estimate of available time for additional operator action.

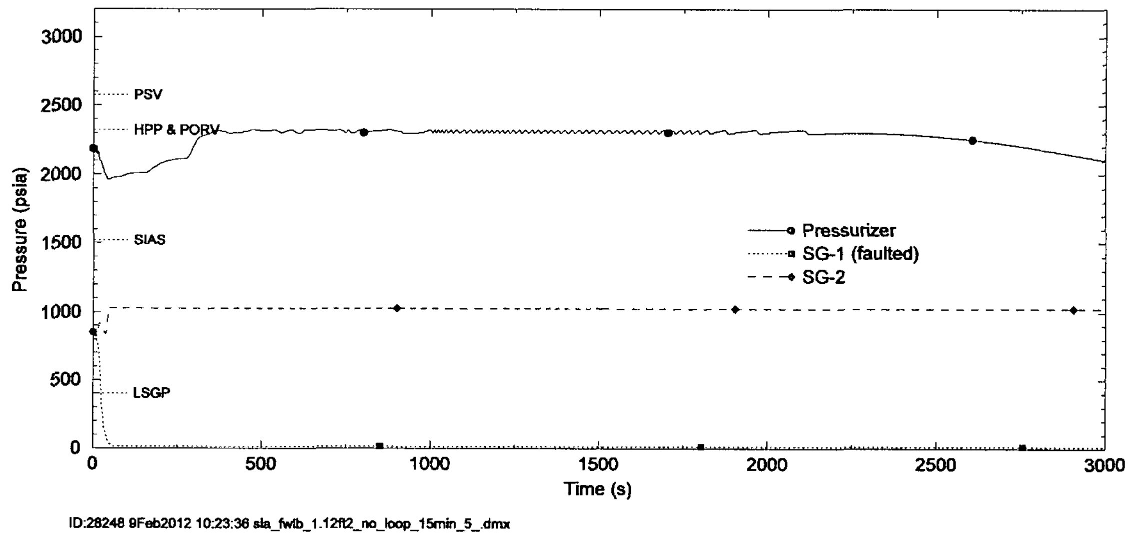

Figure 4-1
No LOOP Case: Pressurizer and SG Pressures

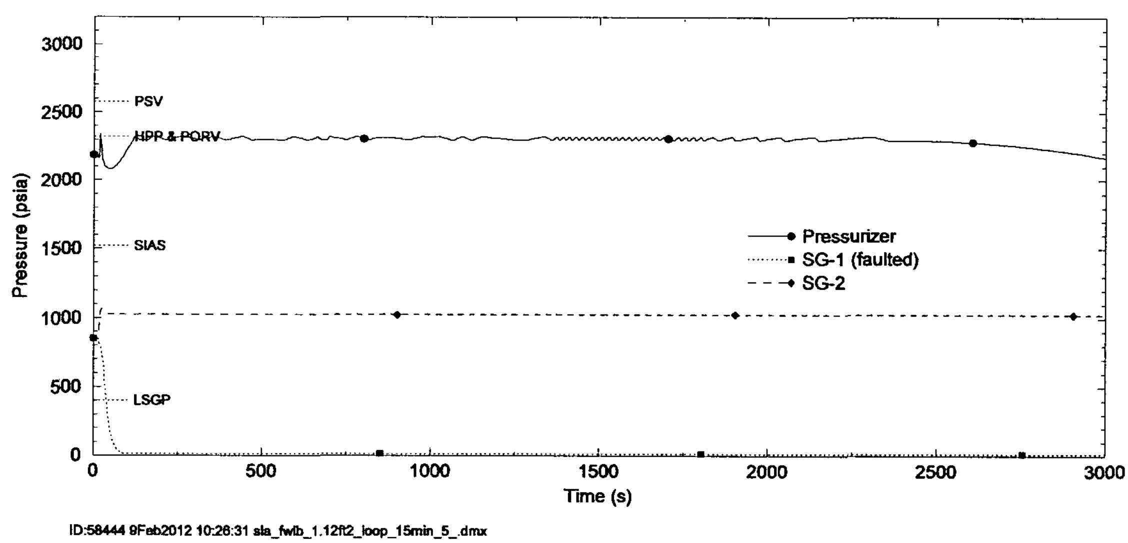

Figure 4-2
LOOP Case: Pressurizer and SG Pressures

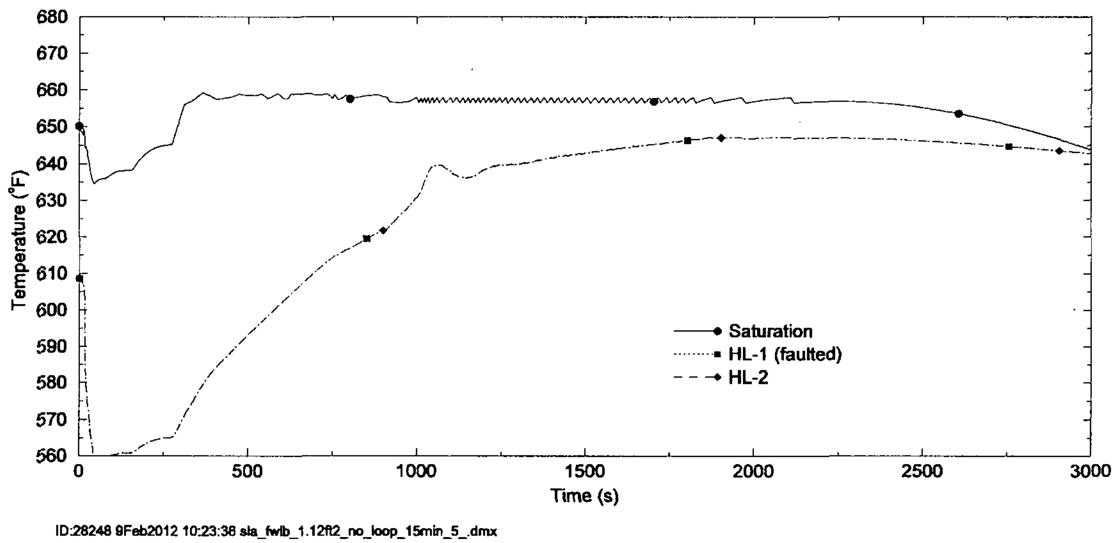

Figure 4-3
No LOOP Case: Hot Leg Temperatures

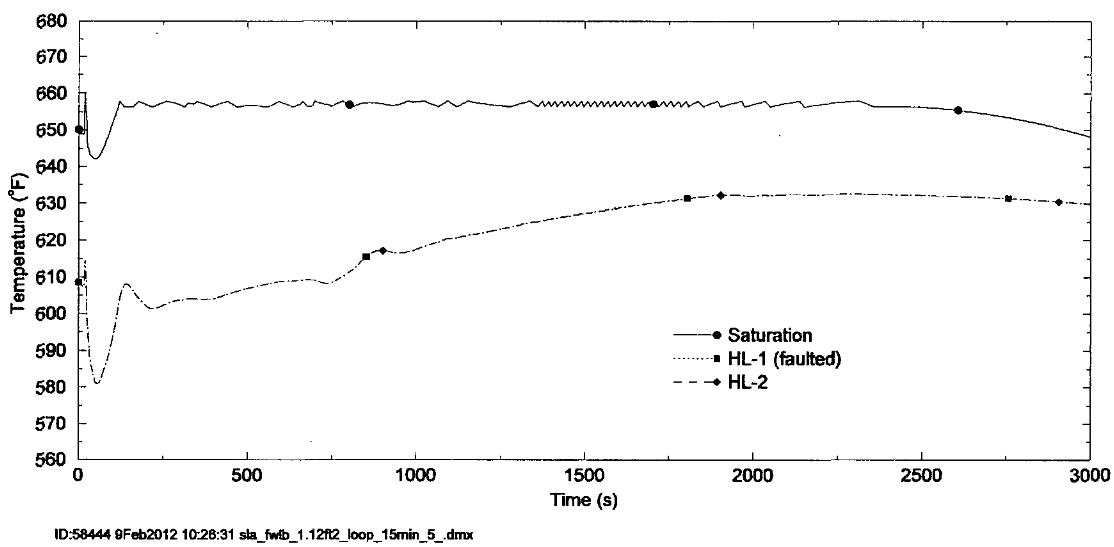

Figure 4-4
LOOP Case: Hot Leg Temperatures

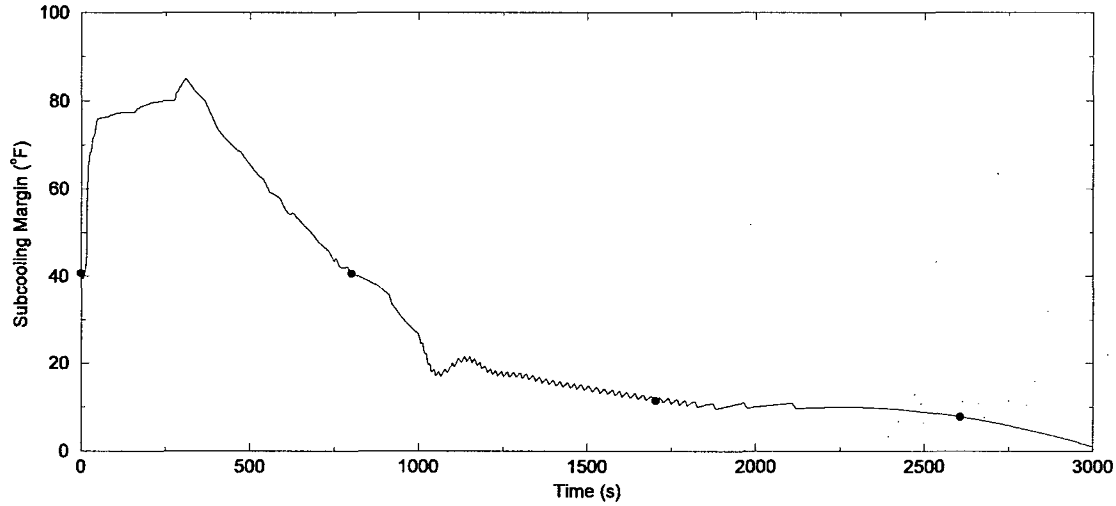

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Figure 4-5
No LOOP Case: Hot leg Subcooling

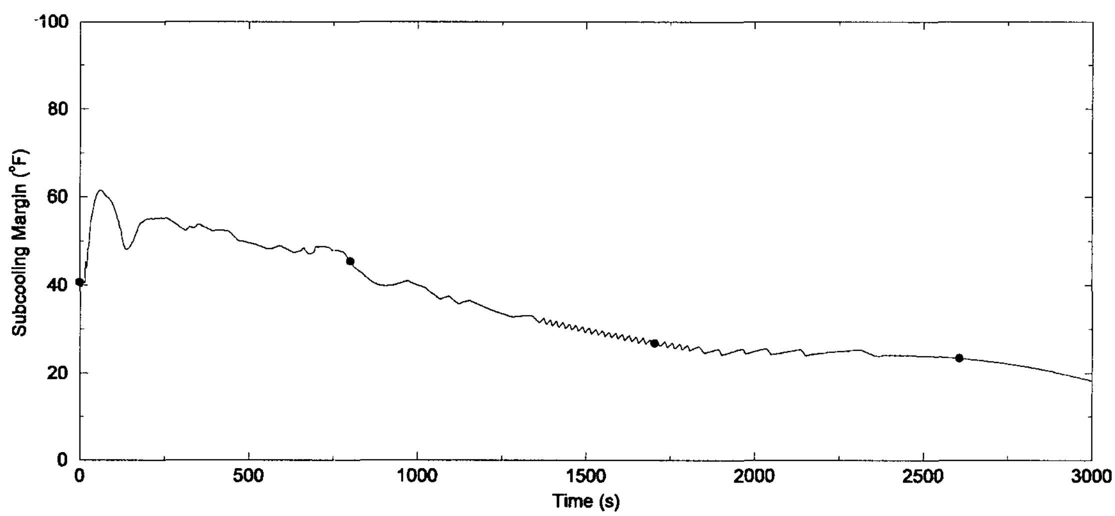

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Figure 4-6
LOOP Case: Hot Leg Subcooling

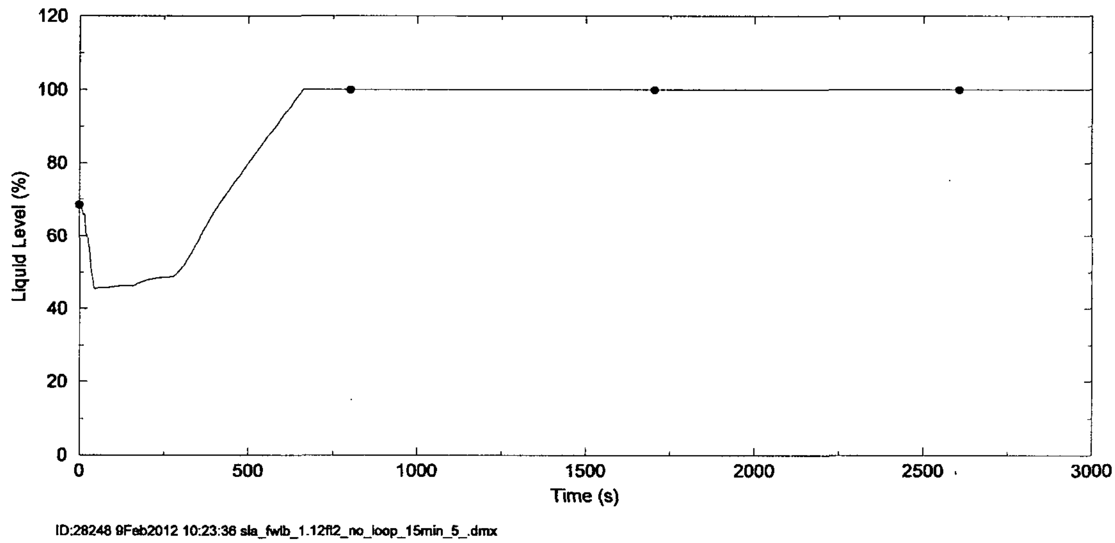

Figure 4-7
No LOOP Case: Pressurizer Liquid Level

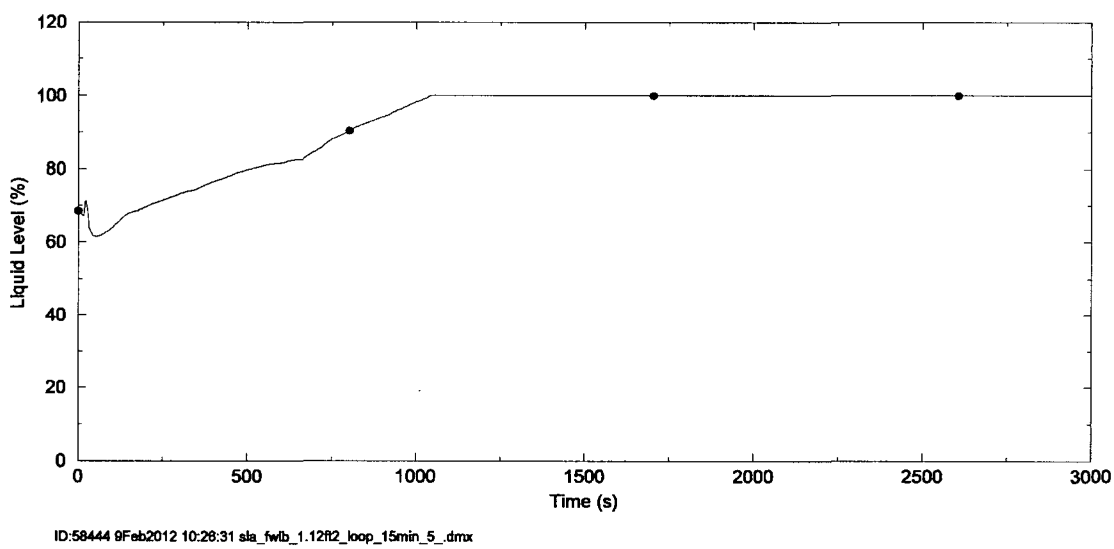

Figure 4-8
LOOP Case: Pressurizer Liquid Level

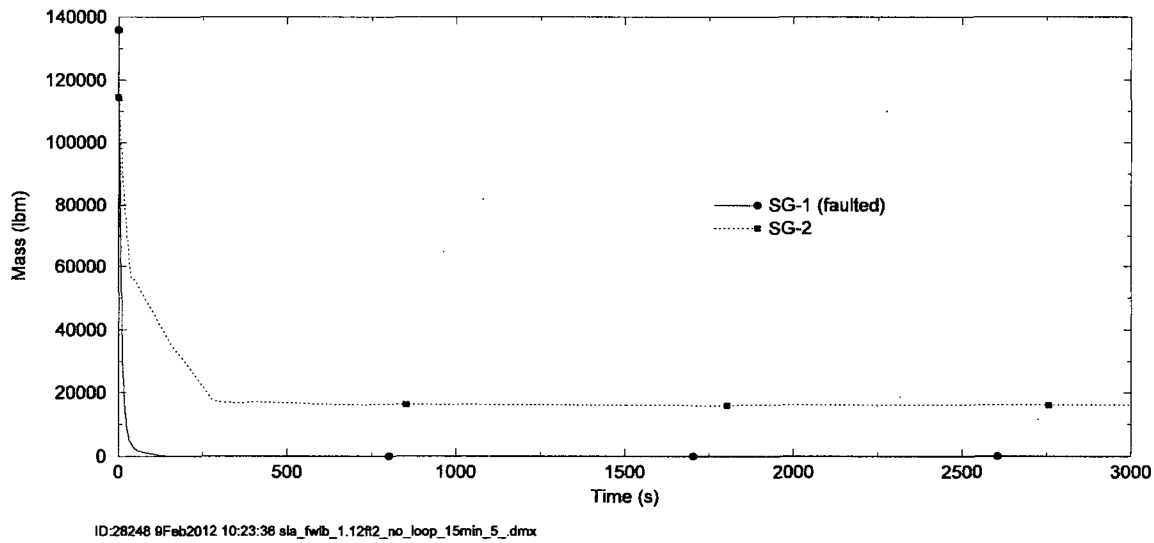

Figure 4-9
No LOOP Case: SG Inventories

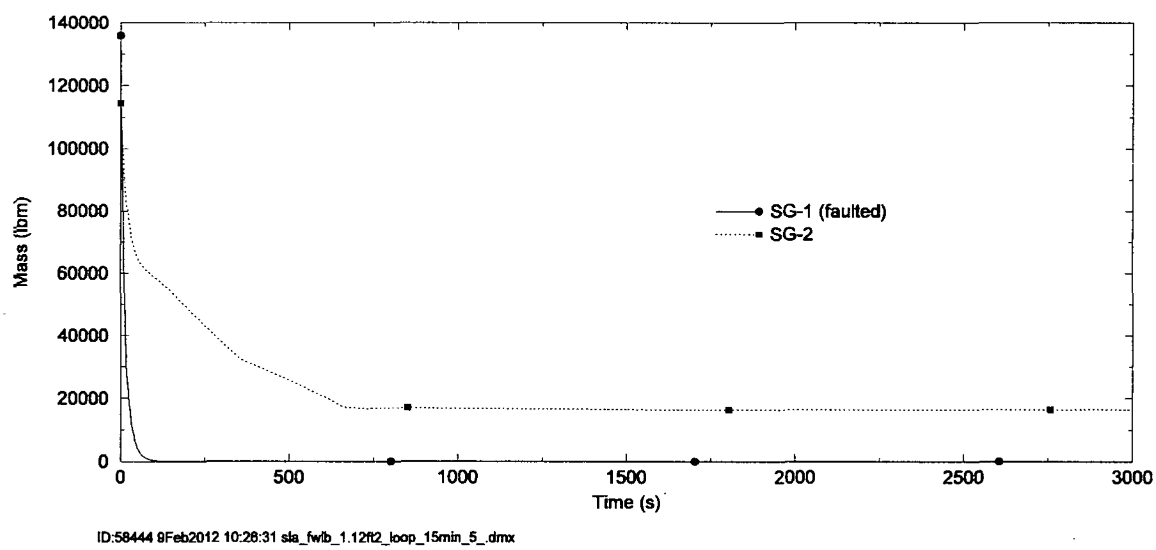

Figure 4-10
LOOP Case: SG Inventories

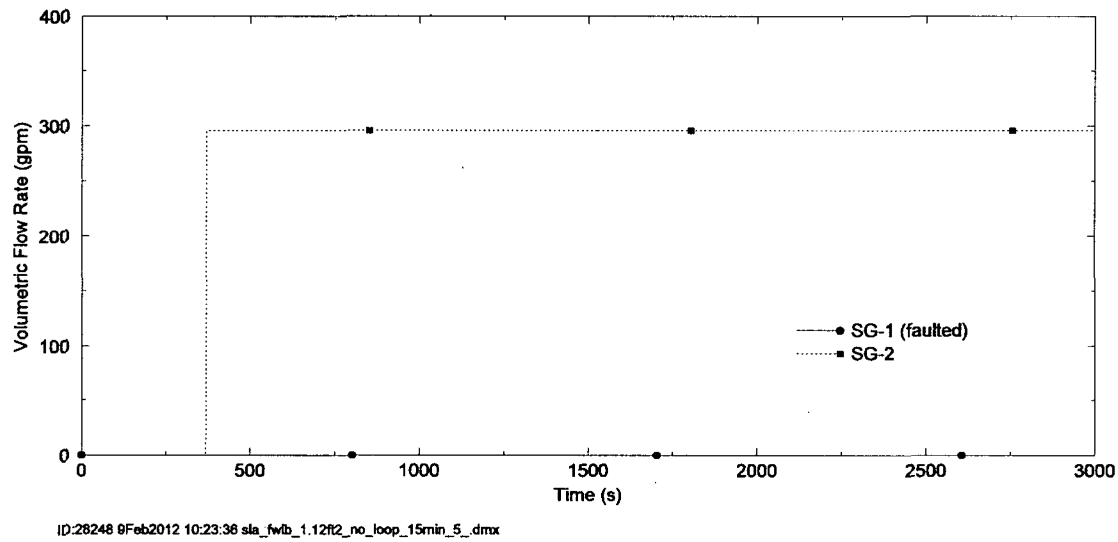

Figure 4-11
No LOOP Case: AFW Flow Rates

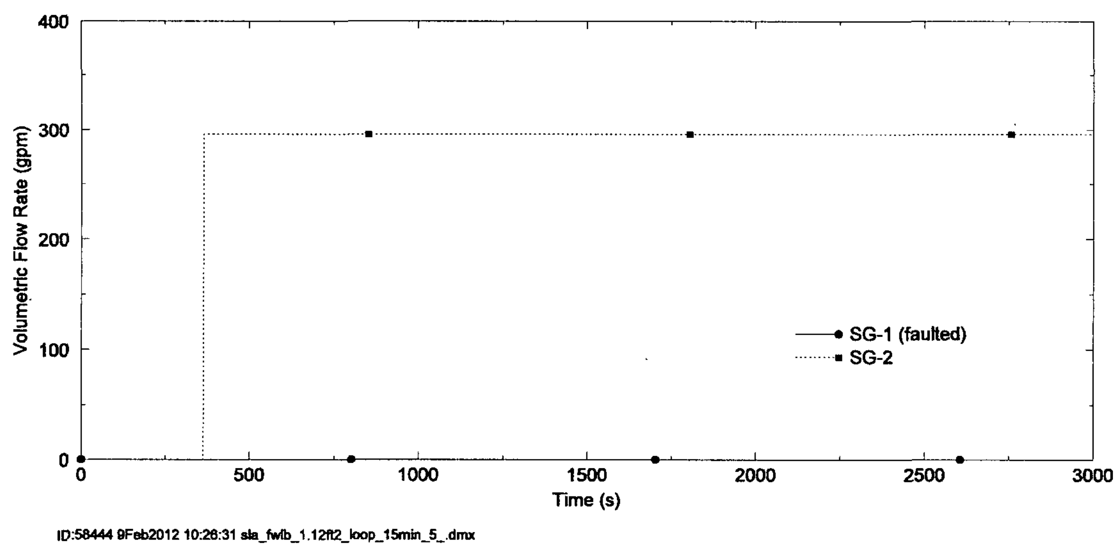

Figure 4-12
LOOP Case: AFW Flow Rates