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

FPL

April 6, 2012

Proprietary Information – Withhold From Public Disclosure Under 10 CFR 2.390.
The balance of this letter may be considered non-proprietary upon removal of Attachment 2.

L-2012-150
10 CFR 50.90
10 CFR 2.390

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

Re: St. Lucie Plant Unit 2
Docket No. 50-389
Renewed Facility Operating License No. NPF-16

Response to Request for Additional Information Identified During Audit of the Non-Loss of Coolant Accident Safety Analyses Calculations for the Extended Power Uprate License Amendment Request

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-021), "License Amendment Request for Extended Power Uprate," February 25, 2011, Accession No. ML110730116.

(2) NRC Reactor Systems Branch Audit Conducted at Westinghouse Electric Company Facilities in Rockville, MD, February 14 and 15, 2012.

By letter L-2011-021 dated February 25, 2011 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. NPF-16 and revise the St. Lucie Unit 2 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).

A001
A002
NRR

During the course of the NRC staff audit conducted at the Westinghouse Electric Company (Westinghouse) facilities in Rockville, MD on February 14 and 15, 2012 [Reference 2], the NRC staff requested additional information to support the review of the non-loss of coolant accident (non-LOCA) safety analyses calculations used in the St. Lucie Unit 2 EPU license amendment request (LAR).

Additional information related to following non-LOCA events was requested. The events included: steam generator tube rupture, station blackout, loss of condenser vacuum, feedwater line break and loss of normal feedwater, asymmetric steam generator transient, reactor coolant pump rotor seizure/shaft break and control element assembly withdrawal from subcritical, and inadvertent opening of a power operated relief valve.

Attachment 1 contains the non-proprietary responses for each of the events listed. Attachment 2 contains proprietary responses for the reactor coolant pump rotor seizure/shaft breaks and control element assembly withdrawal from subcritical events, as these responses contain information that is proprietary to Westinghouse Electric Company (Westinghouse).

Attachment 3 contains the Proprietary Information Affidavit. The purpose of this attachment is to withhold the proprietary information contained in the response to the reactor coolant pump rotor seizure/shaft breaks and control element assembly withdrawal from subcritical events (Attachment 2) from public disclosure. The Affidavit, signed by Westinghouse Electric Company (Westinghouse) as the owner of the information, sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of § 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

The attachment to this letter provides the requested information and the FPL responses for the events.

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-2011-021 [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 06-April-2012

Very truly yours,

Richard L. Anderson

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachments (3)

cc: Mr. William Passetti, Florida Department of Health

**Response to Request for Additional Information
Identified During Audit of the EPU LAR
Non-Loss of Coolant Accident 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 review of the Extended Power Uprate (EPU) License Amendment Request (LAR) for St. Lucie Unit 2 submitted to the NRC by FPL via letter L-2011-021 dated February 25, 2011, Accession Number ML110730116.

The NRC Reactor Systems Branch conducted an audit of the St. Lucie Unit 2 EPU non-loss of coolant accident (non-LOCA) safety analyses calculations at the Westinghouse Electric Company (Westinghouse) facility in Rockville, MD on February 14 and 15, 2012. Additional information related to following non-LOCA events was requested. The events included:

- Steam generator tube rupture (SGTR),
- Station blackout (SBO),
- Loss of condenser vacuum LOCV),
- Feedwater line break (FWLB) and loss of normal feedwater (LONF),
- Asymmetric steam generator transient (ASGT),
- Reactor coolant pump (RCP) rotor seizure/shaft break and control element assembly (CEA) withdrawal from subcritical, and
- Inadvertent opening of a power operated relief valve (IOPORV).

The non-proprietary responses for these events are provided below. The responses to the RCP rotor seizure/shaft break and CEA withdrawal from subcritical events contain information proprietary to Westinghouse Electric Company (Westinghouse). The proprietary responses are provided in Attachment 2.

Steam Generator Tube Rupture (SGTR)

RAI SRXB-01 and SRXB-08, responses provided in FPL letter L-2011-441 (Reference SGTR-1), followup request regarding SGTR margin to overfill (MTO) and mass releases analysis Figure 1 shows the steam generator (SG) liquid volume as a function of time for the time up to 2700 seconds (45 minutes). As shown in Table 1, after 2700 seconds, operator actions begin.

a. **Discuss the break flow rate from the reactor coolant system (RCS) primary side to the affected SG at 2700 seconds following the SGTR event initiation. If the break flow is not terminated, provide a discussion of the plant procedures, operator training program and training records to show that the after 2700 seconds, the operator actions will ensure that the SG MTO exists when the break flow is terminated. Discuss the systems for the operator actions to mitigate for consequences of the SGTR during the period from 2700 seconds to the break flow termination. If non-safety grade systems are credited by the operator in SG overfill prevention after 2700 seconds, justify the use of the non-safety systems.**

b. **Item 5 of the SRXB-08 response indicates that the atmospheric dump valve (ADV) on the affected SG is used for plant cooldown. An assumption of the single failure causing the ADV on the unaffected SG fail to open will disable the ADV for plant cooldown. Discuss the effects of the single failure of the ADV on the MTO and mass releases analysis for the SGTR event after 2700 seconds until the break flow is terminated by operator actions.**

c. **Page 15 of the SRXB-08 response indicates that "once the ruptured SG is isolated, emergency operating procedure (EOP) 2-EOP-4 directs operators to maintain level in the isolated SG less than 90% NR." To maintain SG level, the response indicates that the EOP provides four methods including steaming the isolated SG to atmosphere. During the period from 2700 seconds to break flow termination, if the flow paths for mass releases from the affected SG (such as steaming the isolated (affected) SG to atmosphere) are required to reopened in order to control the water level within the procedure-specified range for SG overfill prevention, discuss the effects of the mass releases from the affected SG on the results of the dose analysis and demonstrate that the case for calculating mass releases discussed in the response to SRXB-08 remains bounding, resulting in limiting mass and dose releases. (It should be noted that the mass release analysis discussed in the response to SRXB-08 covers the first 45 minutes (2700 seconds) following the SGTR event initiation and assumes that the affected SG is isolated by the operator to close the main steam isolation valve. No information is provided to address if mass releases from the affected SG for overfill prevention will occur, and the associated effects of potential mass releases from affected SG on the dose releases for the period after 2700 seconds are not considered.)**

<u>Response</u>

In response to the follow-on question regarding SRXB-01 and SRXB-08 for the steam generator (SG) margin to overfill (MTO) and mass release analyses, the following additional information is provided.

At the end point of the 45 minute EPU SG tube rupture (SGTR) mass release event, the SG MTO is approximatly 6600 ft^3. The primary-to-secondary ruptured tube leakage rate at the end of the transient is approximately 35 lbm/sec or ~0.78 ft^3/sec. If the operator takes no actions to initiate backflow from the secondary to primary side, as described in Emergency Operating Procedure (EOP) 2-EOP-04, Steam Generator Tube Rupture (SGTR), it would require over 2 hours to eliminate the available MTO.

2-EOP-04 provides the operator actions that must be accomplished in the event of a SGTR. One of the goals of the procedure is to maintain control over the isolated (or affected) SG. Specific operator actions are provided in 2-EOP-04 to maintain the isolated SG level less than 90% narrow range indication. This can be accomplished by any of the following methods (listed in the order presented in the EOP):

- Lowering reactor coolant system (RCS) pressure to below the isolated SG pressure, thus enabling back flow. 2-EOP-04 identifies this as the preferred method to control isolated SG level. The back flow method can be accomplished using safety-related equipment (use of charging pumps and auxiliary spray valves to depressurize the RCS).
- Blowing down the isolated SG to the monitor storage tanks.
- Steaming the isolated SG to the condenser.

- Steaming the isolated SG to the atmosphere via the atmospheric dump valves (ADVs). 2-EOP-04 notes that this is the least preferred method to control isolated SG level. A caution note is also provided in the EOP stating, "Steaming the isolated SG to atmosphere should only be performed as a last resort."

2-EOP-04 details operator actions which will maintain SG level within the control band. Additionally, 2-EOP-04 notes that steaming from the safety grade ADVs on the isolated SG is the least preferred option. Therefore, modeling the opening of the isolated SG's ADV would be contrary to the instructions provided to the operator. The additional MTO timeframe of approximately 2 hours (following the initial 45 minutes) is sufficient for the operators to initiate mitigative actions prior to the loss of SG MTO.

In conclusion, the steam releases provided in the EPU SGTR analysis and described in EPU LAR Attachment 5, Section 2.8.5.6.2 and Table 2.8.5.6.2-2 continue to be bounding.

References

SGTR-1 R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-441), Response to NRC Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request, January 18, 2012, Accession No. ML12023A031.

Station Blackout (SBO)

RAI SRXB-40, response provided in FPL letter L-2011-532 (Reference SB0-1), followup request for reactor coolant pump (RCP) seal leakage rate documentation.

Last paragraph of the RAI response indicates that "an additional analysis performed by Combustion Engineering (CE), simulated an 8 hour SBO event to test the upgraded Byron Jackson N-9000 seals, as described in WCAP-16175-P-A. Test data from this analysis illustrates that maximum seal leakage observed during this test was approximately 14 gph (0.233 gpm)."

Specify the page number in WCAP-16175-P-A showing that the maximum seal leakage is 14 gph (0.233 gpm) for the seal leakage test of Byron Jackson N-9000 seals simulating 8 hour SBO event and address the applicability of the test data to the RCP seals during an SBO event at St. Lucie Unit 2 in support of the EPU application.

Response

The requested documentation from WCAP-16175-P-A, "Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants," supporting the seal leakage values was provided to the NRC during the audit meeting as it was currently part of the St. Lucie Unit 2 docket. WCAP-16175-P-A page 7-4 was identified as the reference document supporting the 0.25 gpm reactor coolant pump (RCP) seal leak rate assumption in the station blackout (SBO) analysis and page B-29 of WCAP-16175-P-A was identified as the reference for the Byron Jackson N-9000 seal test simulating the SBO conditions at St. Lucie Unit 2 and the corresponding seal leakage rate.

Reference

SBO-1 R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-532), Response to NRC Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request, January 14, 2012, Accession No. ML12019A074.

Loss of Condenser Vacuum (LOCV)

Note that there are three sets of supplemental information for the LOCV event provided below.

a. **RAI SRXB-48, response provided in FPL letter L-2011-532 (Reference LOCV-1), followup request to the initial water level in the pressurizer assumed in the heatup transient analyses**

The response indicates that "an initial pressurizer level of 66% span is assumed for the LOCV. This consists of the nominal pressurizer level of 63% span plus 3% uncertainty. Initiating from 66% span as opposed to 71% span delays the reactor trip and provides a longer increase in pressure before reactor, ultimately leading to a higher observed pressurizer pressure."

The quoted statement implies that use of a lower value of initial pressurizer water level will result in a higher peak pressurizer pressure for the LOCV event. Address the effect of including a negative 3% uncertainty in the nominal initial pressurizer water level of 63% (i.e., 60% span for the initial pressurizer level) on the peak pressuriser pressure during heatup transients (including the LOCV event) as discussed in the SRXB-48 response, and show that the applicable RCS pressure boundary limits are not exceeded.

Response

a. The loss of condenser vacuum analysis (LOCV) was performed consistent with the current approved methodology and analysis of record (AOR) for St. Lucie Unit 2. The EPU LOCV overpressure analysis described in LAR Attachment 5, Section 2.8.5.2.1.2.1 is initialized at 66% pressurizer level (nominal value of 63% plus 3% uncertainty). Initializing at this pressurizer level results in a peak pressure of 2669.14 psia, which is below the safety limit of 2750 psia for reactor coolant system (RCS) pressure.

Initializing at nominal pressurizer level minus uncertainty (60% initial level) for the overpressure case slows the pressure buildup in the pressurizer and results in a slight trip delay on high pressurizer pressure as compared to the case initializing at the higher pressurizer level of 66%. Initializing the pressurizer level at 60% decreases the peak pressure by ~0.5 psia. If the pressurizer level is initialized at the upper limit plus uncertainty (68% plus 3%), the peak pressure increases by ~0.5 psia from the 66% level case. Therefore, the higher initial pressurizer level of 66% (63% plus 3%) is more conservative for the LOCV overpressure case than initializing the pressurizer at a level of 60% (63% minus 3%).

The impact from initializing at 71% as opposed to 60% would increase the primary peak pressure by approximately 1 psia. This limited impact demonstrates that for heatup events, the initial pressurizer level is not a dominant input for overpressure.

In conclusion, the current approved methodology of selecting the nominal initial pressurizer level plus uncertainty is justified.

References

LOCV-1 R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-532), Response to NRC Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request, January 14, 2012, Accession No. ML12019A074.

b. A question was asked with regard to identifying the magnitude of the second pressure peak associated with the EPU primary and secondary overpressure events for St. Lucie Unit 2. The EPU LOCV event duration captured the first pressure peak; however, the event timing did not indicate the impact of auxiliary feedwater (AFW) flow addition and the second pressure peak that could be associated with AFW initiation. Please evaluate the peak pressure event to determine the impact of the second pressure peak.

Response

b. The EPU LOCV event is the bounding primary and secondary peak pressure event and is described in LAR Attachment 5, Section 2.8.5.2.1.2. LAR Attachment 5, Tables 2.8.5.2.1-2 and 2.8.5.2.1-3 provide the results of the LOCV analysis and it is shown that the peak pressures reported are below the acceptable design limits. However, the LOCV event is primarily analyzed for peak pressure and as such, is a short duration event.

To determine the magnitude of any second pressure peak, the LOCV event was reanalyzed by extending the end time of the event past the point of auxiliary feedwater (AFW) initiation where the second pressure peak could occur. The Section 2.8.5.2.1.2 event was reanalyzed utilizing the current licensed LOCV methodology. There is no second peak in pressure as the main steam safety valves (MSSVs) are adequately sized to provide sufficient cooling to remove the decay heat of the event.

Table LOCV-1 provides a summary of the initial conditions modeled in the LOCV event. Table LOCV-2 provides the sequence of events for the extended LOCV event and indicates that the peak primary pressure is the same as that listed in Table 2.8.5.2.1-2. Figures LOCV-1 through LOCV-5 provide additional details for the extended LOCV event.

The MSSVs demonstrated that they are adequately sized to provide sufficient cooling to offset the decay heat generated; therefore, the secondary overpressure case listed in Table 2.8.5.2.1-3 remains bounding. Therefore, the primary and secondary side peak pressures discussed in Section 2.8.5.2.1.2 remain bounding and the MSSV relief capacity is sufficient to preclude any second pressure peaks during the event.

The loss of normal feedwater (LONF) event, however, does produce a second primary system pressure peak. To determine the magnitude of the second peak, LONF was run for a time period beyond the time of second peak (500 seconds). Table LONF-1 provides the initial conditions modeled in the LONF event. The sequence of events for LONF is presented in Table LONF-2 and reports a peak RCS pressure of 2627.91 psia, with a corresponding pressurizer pressure of 2575 psia (safety valves setpoint pressure). This pressure is bounded by the LOCV results in Table LOCV-2. Therefore, the limiting LOCV overpressure case bounds the pressure peaks seen in Figure LONF-2 for the LONF event. Figures LONF-1 through LONF-5 provide additional details of the LONF case.

Table LOCV-1
LOCV Second Peak Pressure for Overpressure

Parameter		Value
Core power		100% + Uncertainty (3030 MWt)
RCS loop flow rate		Total Design Flow (TDF) (187,500 gpm)
Vessel T_{avg} temperature		Low-T_{avg} – Uncertainty (560°F)
Pressurizer	Initial pressure	Low Nominal – Uncertainty (2180 psia)
	Initial water level	Nominal + Uncertainty (66% NRS)
	Charging/letdown	Unavailable
	Heater	Unavailable
	Power operated relief valve (PORV)	Unavailable
	Spray	Unavailable
	Pressurizer safety valve (PSV)	Design + Uncertainty (2575 psia)
Steam generator	Initial water level	Nominal (65% span)
	Tube conditions Tube plugging (%)	Fouled 10%
	MSSV setpoint	Design + Uncertainty Bank 1 @ 1030 psia Bank 2 @ 1060.8 psia
Auxiliary feedwater (AFW)	Pumps	2 motor driven AFW pumps (MDAFP)
	Flowrate	275 gpm per MDAFP
	Delay	330 seconds
	Initiation trip setpoint	Low Nominal – Uncertainty (14.5% NRS)
Reactor trip setpoint	High pressurizer pressure trip (HPPT)	Nominal + Uncertainty (2415 psi)
Decay Heat		ANS-1979 + 2σ

Table LOCV-2
LOCV-Second Pressure Peak Sequence of Events

Event	Time (seconds)
Turbine trip	10.110
Main feedwater terminates (both loops)	10.110
High pressurizer pressure trip (HPPT) setpoint reached	16.300
Reactor trip on high pressurizer pressure	17.455
Rod motion begins	18.195
Pressurizer safety valve (PSV) Opens	18.195
Time of peak RCS pressure	18.700
First main steam safety valve (MSSV) opens	20.281
AFW signal on steam generator 2 on low level	586.981
AFW signal on steam generator 1 on low level	591.772
AFW initiated (330 second delay)	916.981
Results	
Peak RCS pressure [@ 18.7 seconds]	2669.14 psia
RCS pressure maximum limit	2750 psia

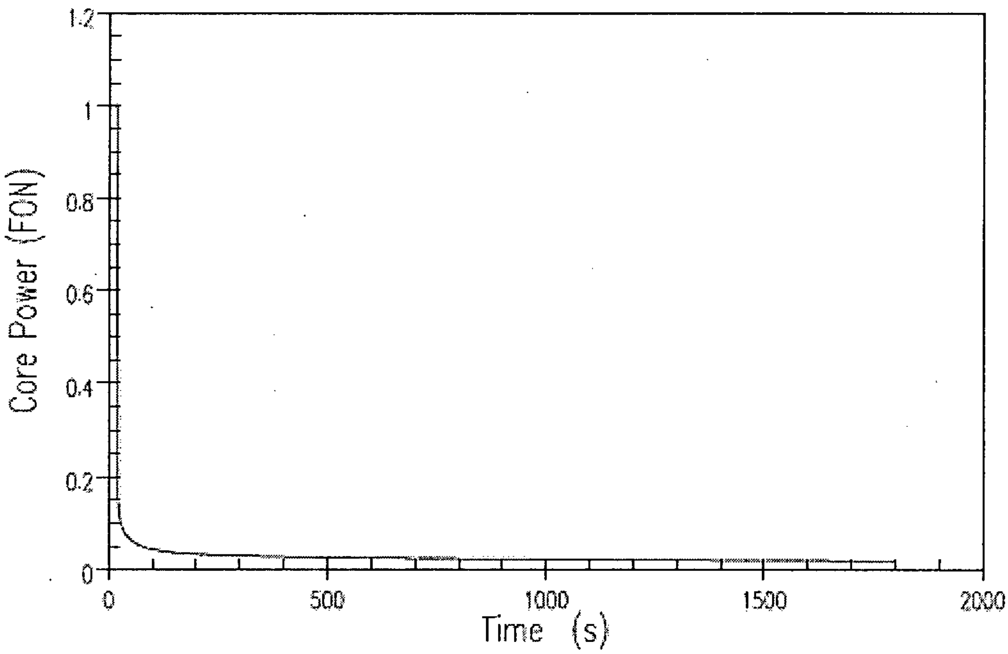

Figure LOCV-1
Core Power vs. Time

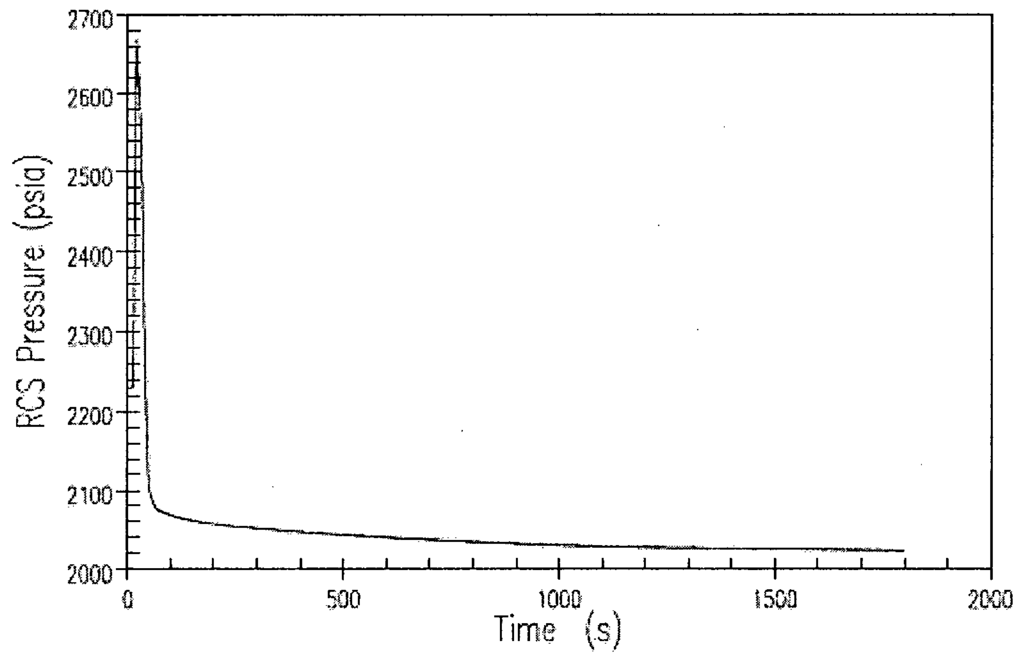

Figure LOCV-2
RCS Pressure vs. Time

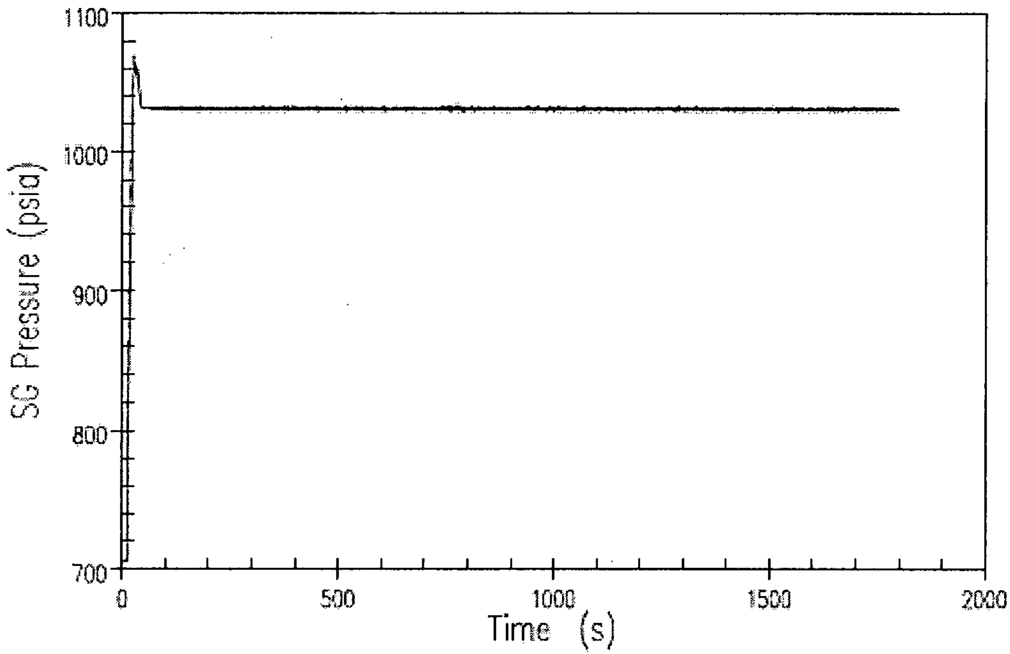

Figure LOCV-3
SG Pressure vs. Time

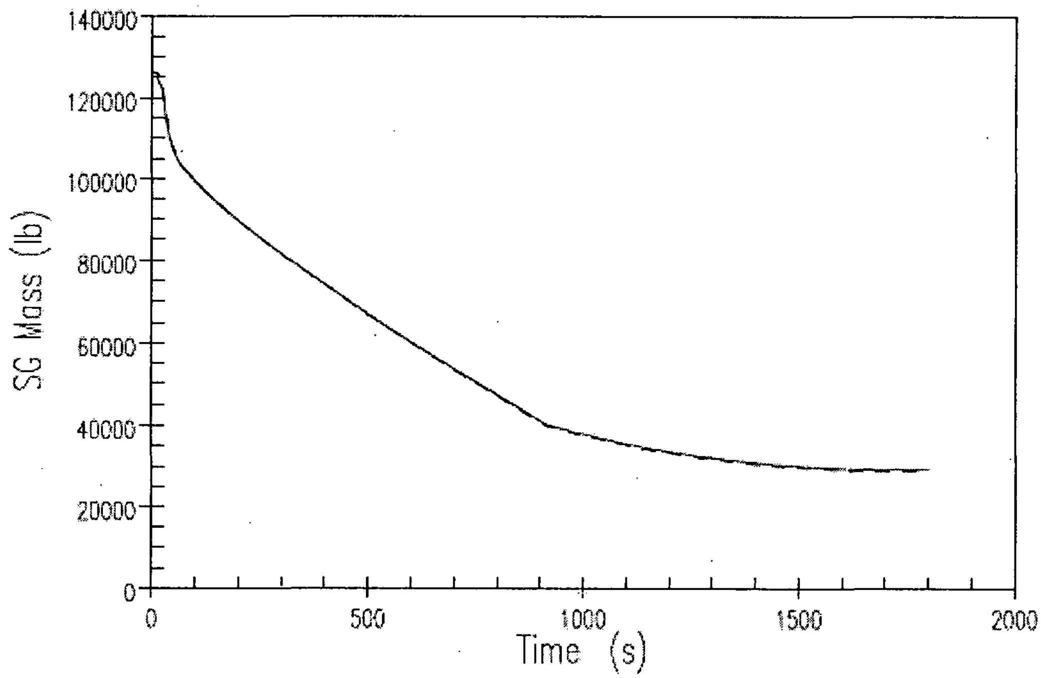

Figure LOCV-4
SG Mass vs. Time

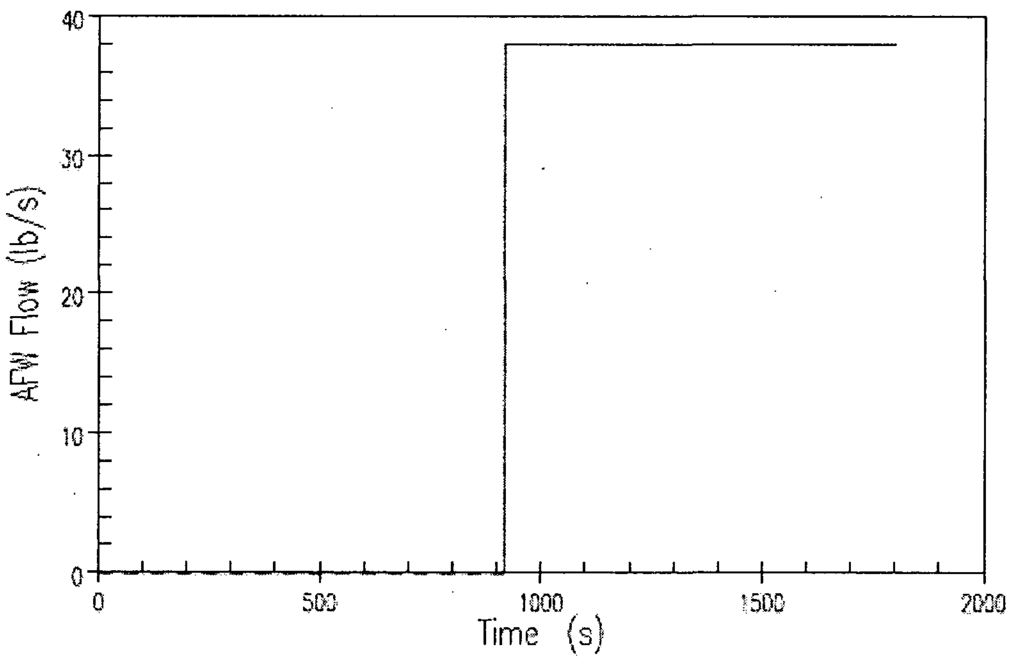

Figure LOCV-5
AFW Flow vs. Time

Table LONF-1: Initial Conditions
LONF Second Peak Primary Pressure Case

<table>
<tr><th colspan="2">Parameter</th><th>Value</th></tr>
<tr><td colspan="2">Core power</td><td>100% + Uncertainty
(3030 MWt)</td></tr>
<tr><td colspan="2">RCS loop flow rate</td><td>Total Design Flow (TDF)
(187,500 gpm)</td></tr>
<tr><td colspan="2">RCS temperature</td><td>High Nominal – Uncertainty
(581.5°F)</td></tr>
<tr><td rowspan="6">Pressurizer</td><td>Initial pressure</td><td>Nominal – Uncertainty
(2180 psia)</td></tr>
<tr><td>Initial water level</td><td>Nominal + Uncertainty
(66% NRS)</td></tr>
<tr><td>Charging/letdown</td><td>Unavailable</td></tr>
<tr><td>Heater</td><td>Available</td></tr>
<tr><td>Power operated relief valve (PORV)</td><td>Unavailable</td></tr>
<tr><td>Spray</td><td>Unavailable</td></tr>
<tr><td rowspan="4">Steam generator</td><td>Initial water level</td><td>Nominal
(65% span)</td></tr>
<tr><td>Tube conditions
Tube plugging (%)</td><td>Fouled
10%</td></tr>
<tr><td>Atmospheric dump valve (ADV)</td><td>Unavailable</td></tr>
<tr><td>MSSV setpoint</td><td>Design + Uncertainty
Bank 1 @ 1030 psia
Bank 2 @ 1060.8 psia</td></tr>
<tr><td rowspan="4">Auxiliary feedwater (AFW)</td><td>Pumps</td><td>2 motor driven AFW pumps (MDAFP)</td></tr>
<tr><td>Flowrate</td><td>275 gpm per MDAFP</td></tr>
<tr><td>Delay</td><td>330 seconds</td></tr>
<tr><td>Initiation trip setpoint</td><td>Low Nominal – Uncertainty
(14.5% NRS)</td></tr>
<tr><td>Reactor trip setpoint</td><td>High pressurizer pressure trip (HPPT)</td><td>Nominal + Uncertainty
(2415 psi)</td></tr>
<tr><td colspan="2">Decay Heat</td><td>ANS-1979 + 2σ</td></tr>
</table>

Table LONF-2
Sequence of Events

Event	Time (seconds)
Main feedwater terminates (both loops)	20.00
High pressurizer pressure trip (HPPT) setpoint reached	51.11
Reactor trip on high pressurizer pressure	52.26
Rod motion begins	53.00
AFW signal on steam generators 1 and 2 on low level	75.98
Pressurizer safety valve (PSV) opens	378.65
AFW initiated (330 second delay)	405.97
Time of peak RCS pressure	444.40
Results	
Peak RCS pressure	2627.91 psia
RCS pressure maximum limit	2750 psia

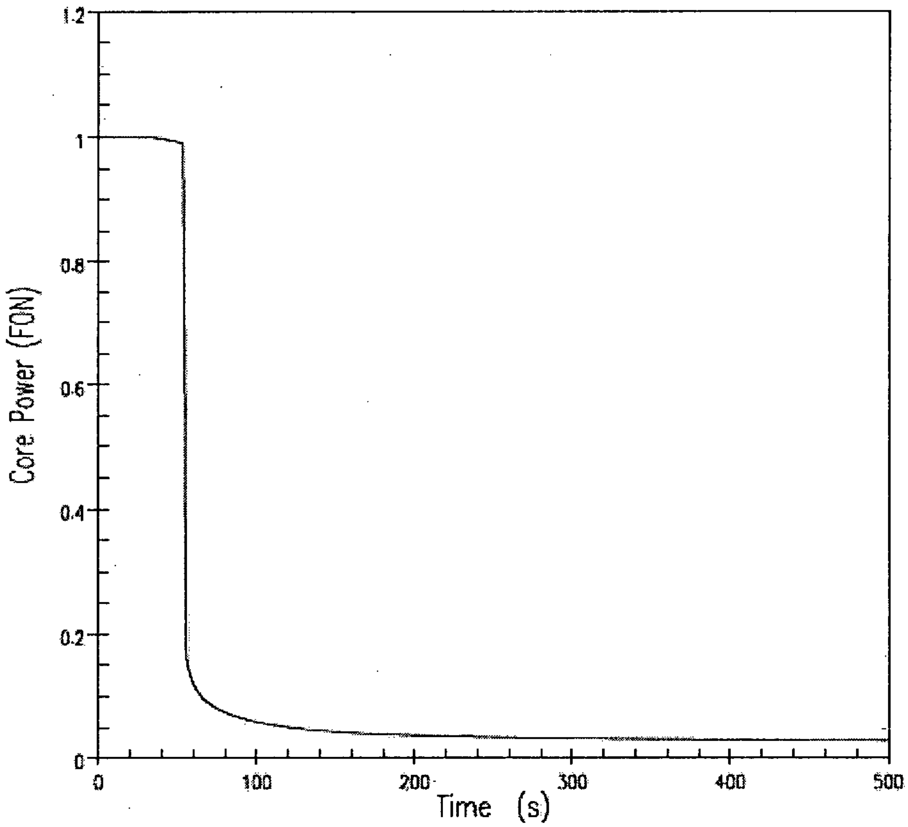

Figure LONF-1
Core Power vs. Time

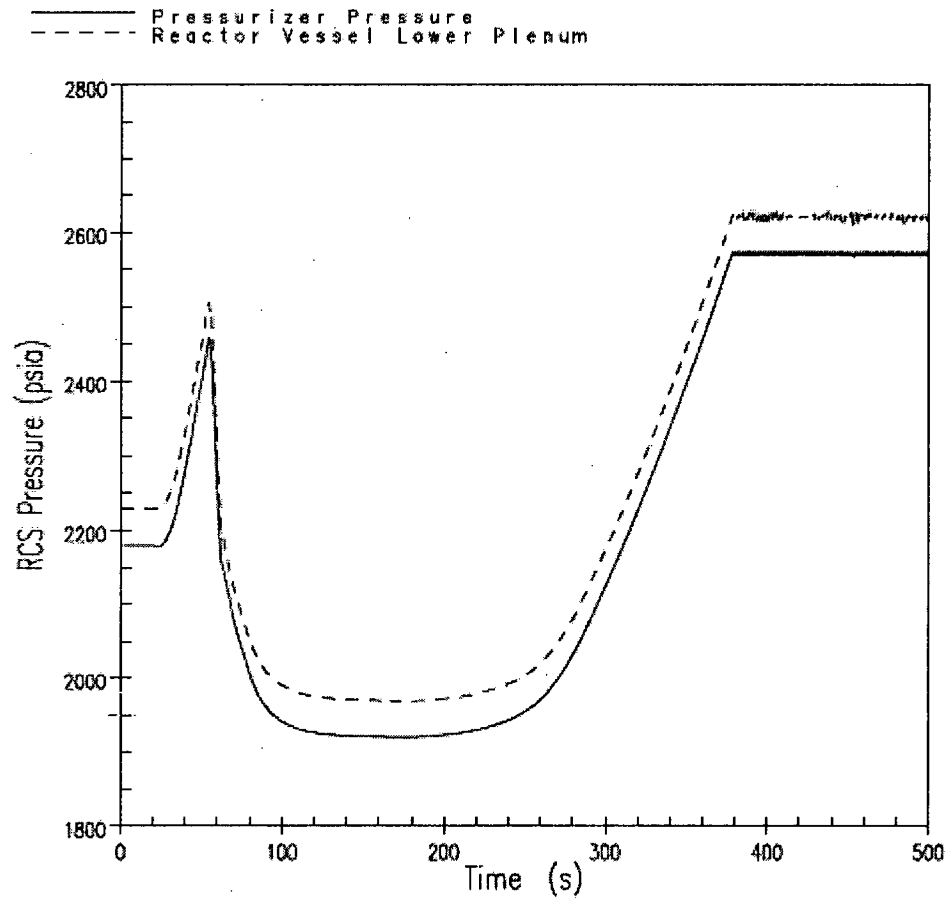

Figure LONF-2
RCS Pressure vs. Time

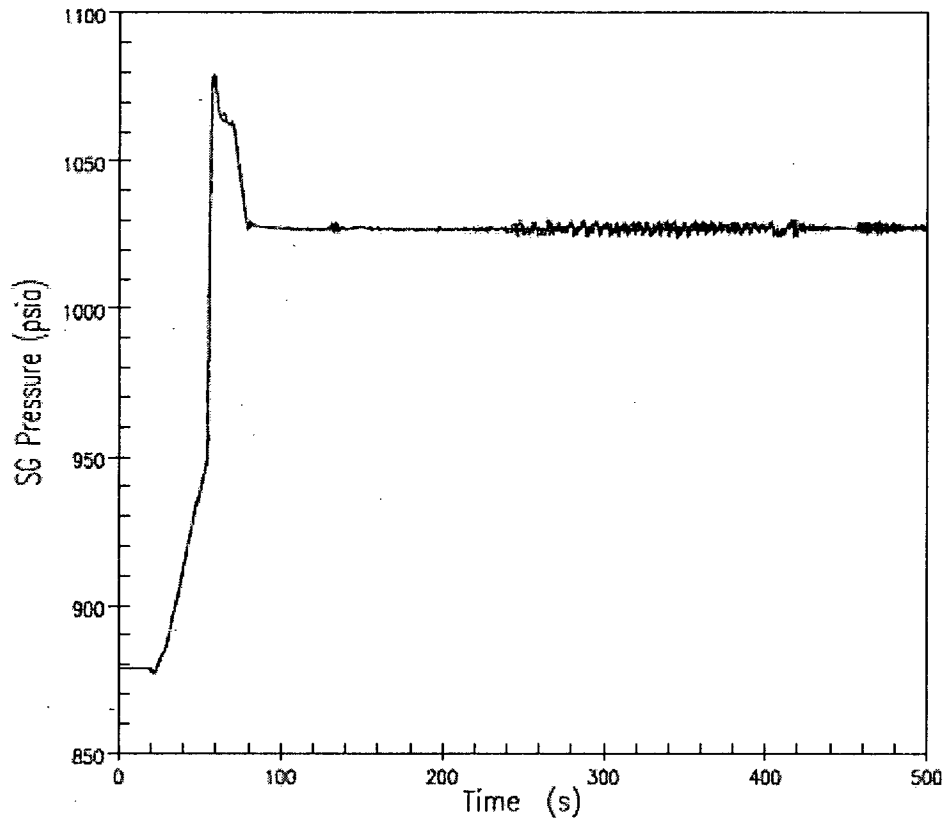

Figure LONF-3
SG Pressure vs. Time

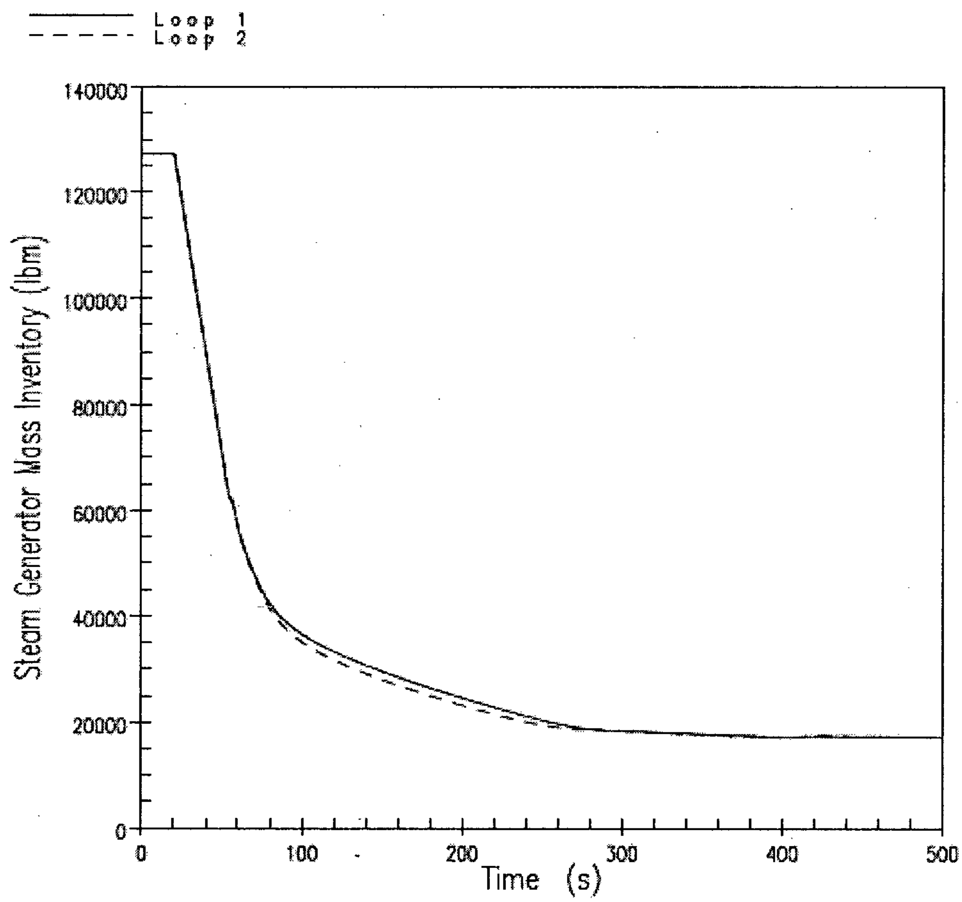

Figure LONF-4
SG Mass vs. Time

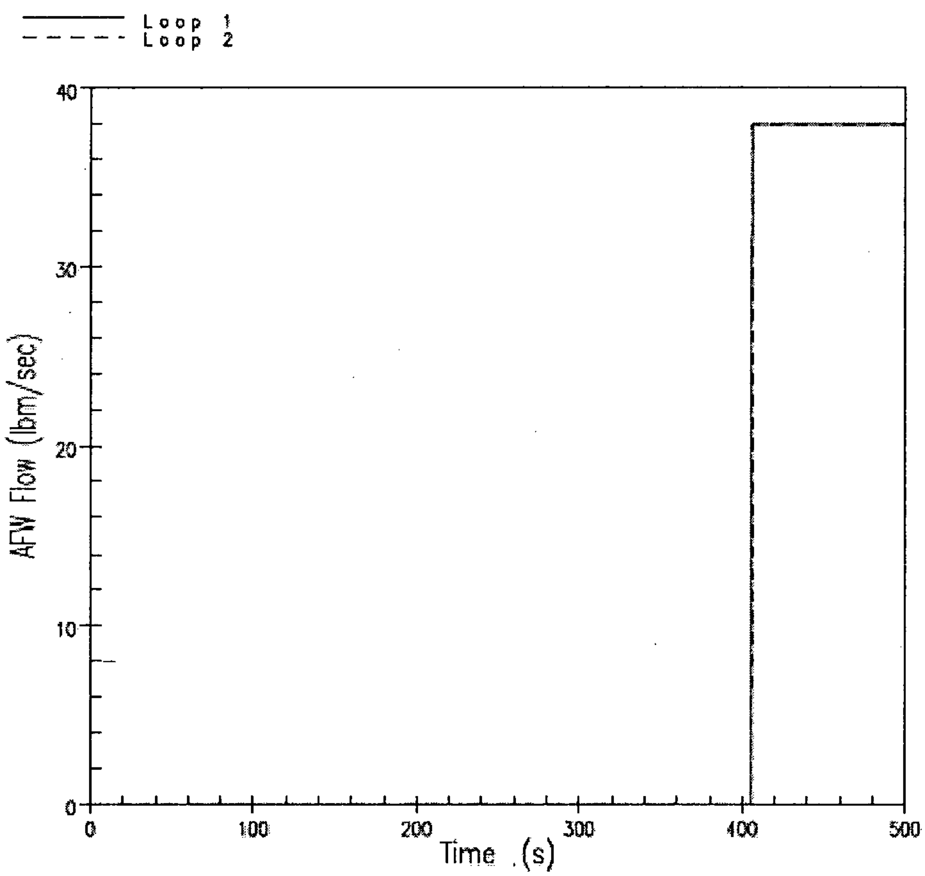

Figure LONF-5
AFW Flow vs. Time

c. A question was asked with regard to the impact of crediting the second safety grade reactor trip function on the primary overpressure analysis. Specifically, what would the impact on peak primary pressure be if the first safety grade reactor trip was bypassed and only the second safety grade reactor trip was credited as noted in SRP Section 5.2.2?

Response

c. Standard Review Plan (SRP) (NUREG-0800) Chapter 5.2.2 details in the Acceptance Criteria Section 3 (for PWRs) Item B that states:

> "The design of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB [reactor coolant pressure boundary] design pressure during the most severe AOO [anticipated operational occurrence] with reactor scram, as specified by ASME Code Article NB-7000. Also, sufficient available margin should account for uncertainties in the design and operation of the plant assuming:
>
> i. The reactor is operating at a power level that will produce the most severe overpressurization transient.
>
> ii. All system and core parameters have values within normal operating range, including uncertainties and technical specification limits that produce the highest anticipated pressure.
>
> iii. The second safety-grade signal from the reactor protection system initiates the reactor scram."

The EPU LOCV event is the bounding primary and secondary peak overpressure event and is detailed in LAR Attachment 5, Section 2.8.5.2.1. LAR Attachment 5, Tables 2.8.5.2.1-2 and 2.8.5.2.1-3 provide the results of the EPU LOCV analysis and it is shown that the peak pressures reported are below the acceptable design limits. The EPU LOCV event in Section 2.8.5.2.1 utilizes the second available reactor trip, as the reactor trip on turbine trip is not a safety grade function and is not credited by delaying the reactor trip until the safety grade high pressurizer pressure trip setpoint is obtained. In response to the NRC question, the LOCV peak pressure event was analyzed with no credit taken for the first safety grade reactor trip.

The event was analyzed utilizing the current licensed LOCV methodology and Table LOCV-3 provides a summary of the initial conditions modeled. The reactor trip was delayed from the first safety grade reactor trip on high pressurizer pressure until the second safety grade reactor trip signal on steam generator low level. The reactor trip setpoint credited for steam generator (SG) low level is 30% NR, which has been reduced from the nominal value to account for the SG level uncertainty. Table LOCV-4 lists peak primary overpressure for the analyzed event and demonstrates that the peak pressure of ~2712 psia remains below the acceptance criteria of 2750 psia.

Figures LOCV-6 through LOCV-11 provide additional details for the analyzed LOCV event assuming only the second safety grade reactor trip is credited.

The analyzed LOCV event based on the second reactor safety grade trip demonstrates that the peak primary overpressure criterion is met and therefore, the design and sizing of the pressurizer safety valves meets the overpressure design criterion cited in the SRP Chapter 5.2.2.

TABLE LOCV-3
SECOND SAFETY TRIP FOR LOCV OVERPRESSURE
INITIAL CONDITIONS

Parameter		Value
Core power		100% + Uncertainty 3030 MWt
RCS loop flow rate		Total Design Flow (TDF) 187,500 gpm
Vessel T_{avg} temperature		Low T_{avg} – Uncertainty 560°F
Pressurizer	Initial pressure	Low Nominal – Uncertainty 2180 psia
	Initial water level	Nominal + Uncertainty 66% NRS
	Charging	Unavailable
	Letdown	
	Heater	
	Power operated relief valve (PORV)	
	Spray	
	Pressurizer safety valve (PSV)	Design + Uncertainty 2575 psia
Steam generator	Initial water level	Nominal 65% NRS
	Tube conditions Tube plugging	Fouled 10%
	Main steam safety valve (MSSV) setpoint	Design + Uncertainty Bank 1 – 1030 psia Bank 2 – 1060.8 psia
Reactor trip setpoint	High pressurizer pressure trip (not credited) SG low level trip	------- Nominal – Uncertainty 30% NRS
Decay heat		ANS-1979 + 2σ

<u>Control Grade Systems Credited for the Event</u>

No control grade systems are modeled as they would benefit the transient response.

<u>Operator Actions Credited for the Event</u>

No operator actions are credited for this transient.

TABLE LOCV-4
SECOND SAFETY GRADE REACTOR TRIP
SEQUENCE OF EVENTS AND TRANSIENT RESULTS

Without pressurizer pressure control (for primary RCS overpressure)	
Event	**Time (seconds)**
Turbine trip	10.110
Main feedwater terminates (both loops)	10.110
High pressurizer pressure reactor trip (not credited)	17.455
SG low level setpoint reached	17.900
Pressurizer safety valve (PSV) opens	18.198
Reactor trip on SG low level	19.246
Control rod motion begins	19.986
First main steam safety valve (MSSV) opens	20.280
Peak RCS pressure	21.500
Peak secondary pressure	24.600
Results	
Peak RCS pressure	2711.66 psia
RCS pressure maximum limit	2750 psia
Peak secondary pressure	1073.86 psia
Secondary pressure maximum limit	1100 psia

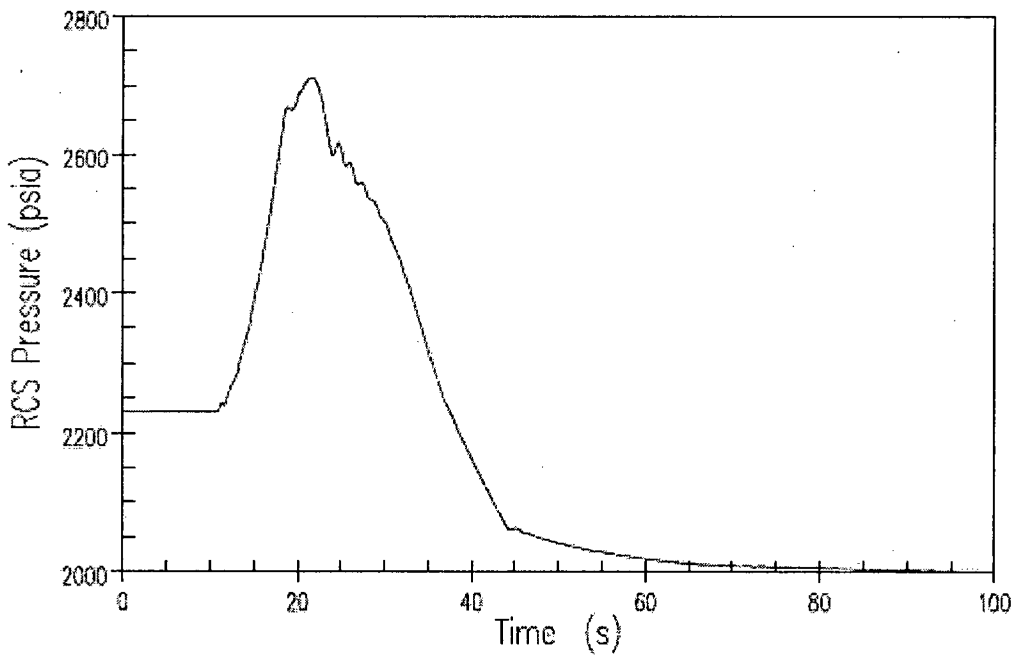

Figure LOCV-6
RCS Pressure vs. Time

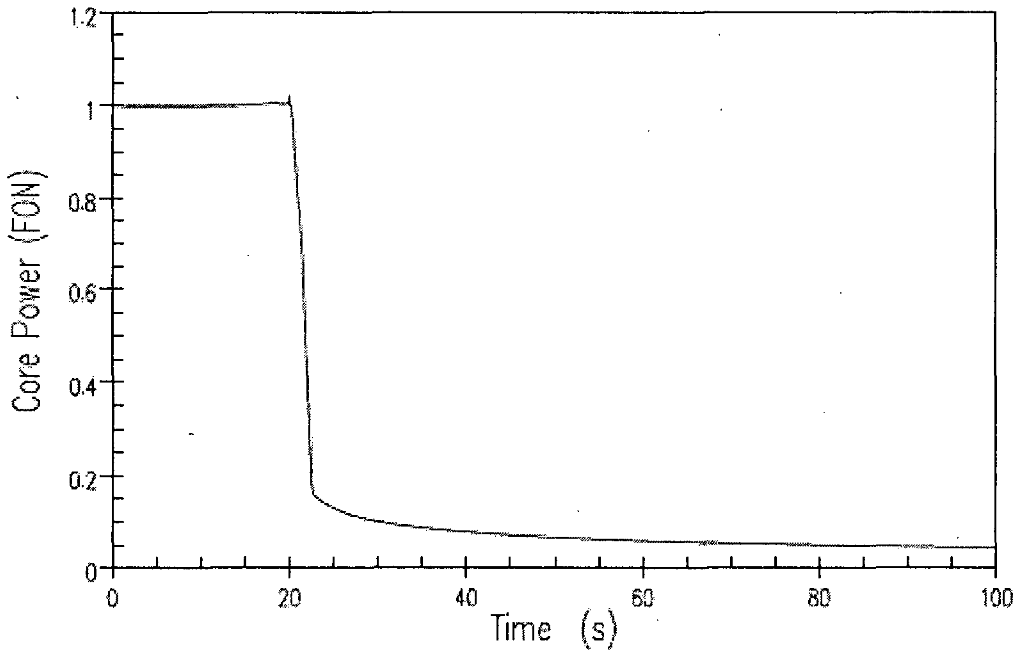

Figure LOCV-7
Core Power vs. Time

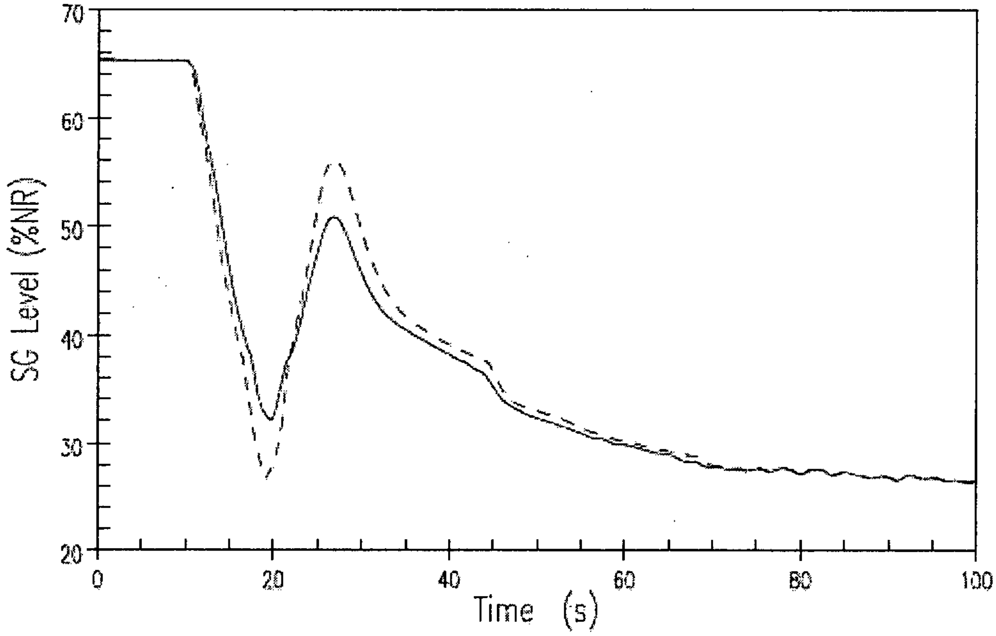

Figure LOCV-8
SG Level vs. Time

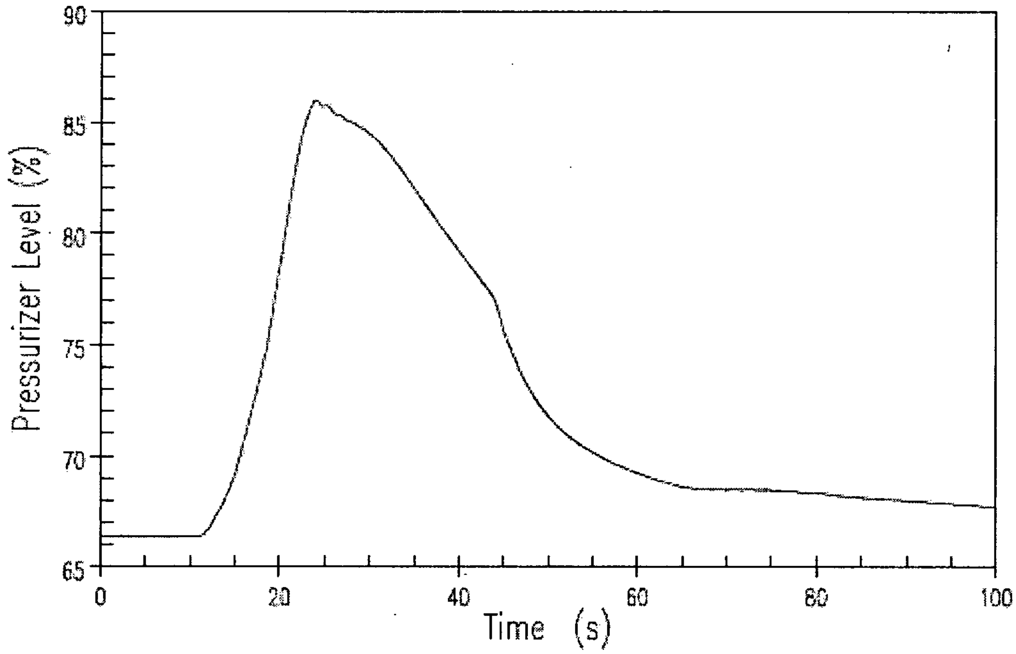

Figure LOCV-9
Pressurizer Level vs. Time

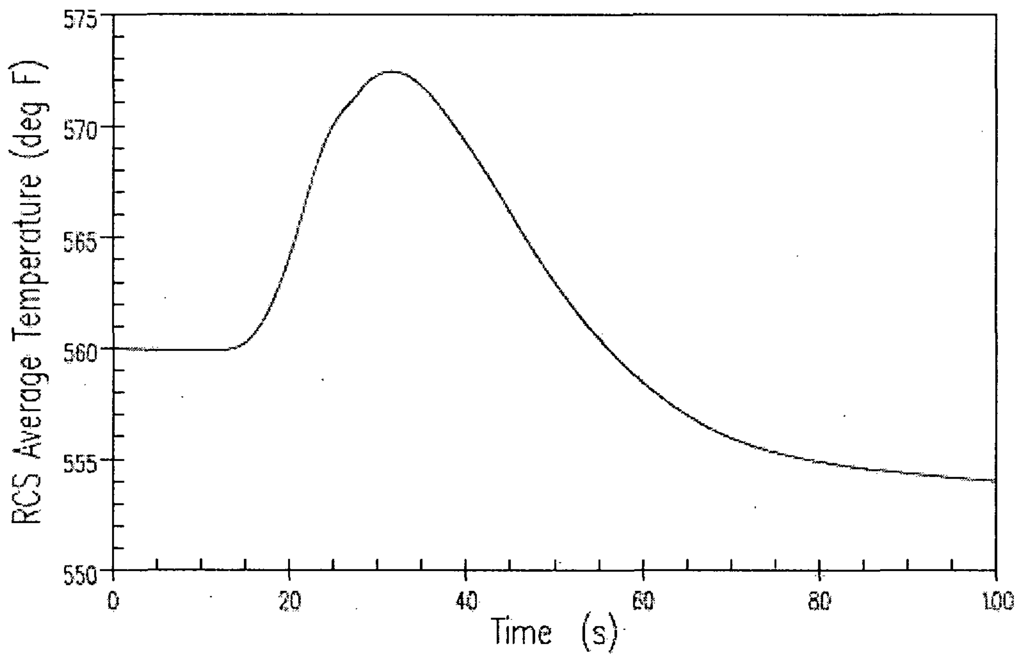

Figure LOCV-10
T_{avg} vs. Time

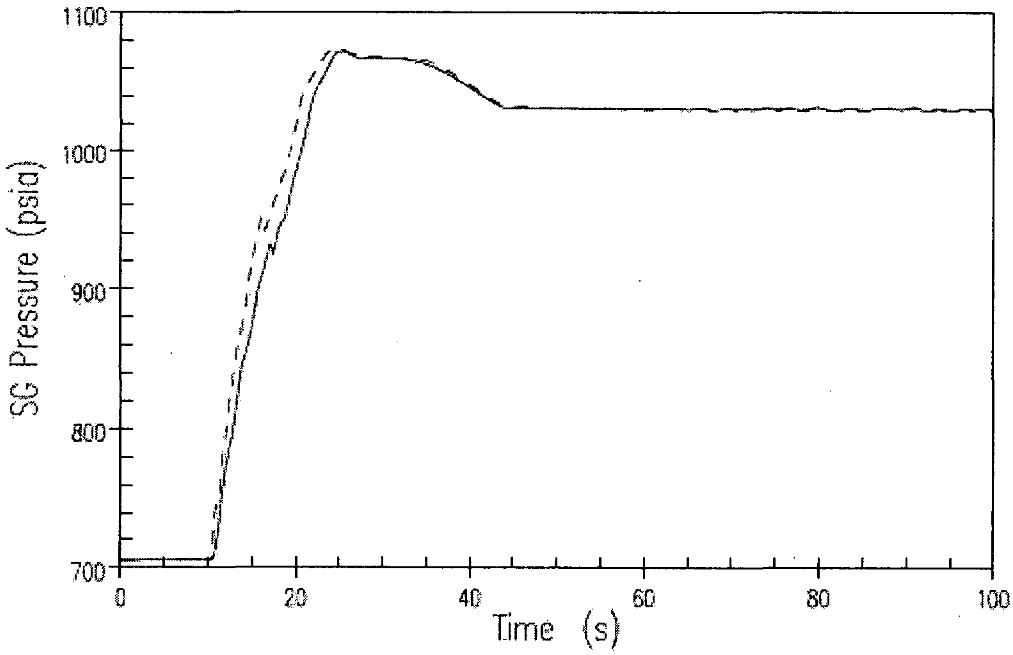

Figure LOCV-11
SG Pressure vs. Time

Feedwater Line Break (FWLB) and Loss of Normal Feedwater (LONF)

RAI SRXB-61, response provided in FPL letter L-2011-532 (Reference FWLB-1), followup request regarding the long term cooling (LTC) analyses for the FWLB and LONF events

The RAI response discusses the LTC analyses for the LONF and FWLB events with and without a loss of offsite power (LOOP). The NRC staff finds that the discussed LTC analyses do not contain the same level of the conservatisms for the LTC analyses required to demonstrate compliance with the applicable acceptance criteria for the LONF and FWLB analyses, which are part of the non-loss of coolant accident (non-LOCA) transient analyses included in Updated Final Safety Analysis Report (UFSAR) Chapter 15. The guidance of performing analyses of the UFSAR Chapter 15 events is provided in the Standard Review Plan (SRP, NUREG-0800). SRP 15.0 specifies that: (1) the NRC-approved computer codes should be used for the analysis; (2) only safety-related systems or components are allowed for use in mitigating Chapter 15 events; (3) the effects of single active failures and operator errors need to be included in the analysis; and (4) the values used in the analysis for the key plant parameters should include permitted fluctuations and uncertainties, (5) the appropriate conditions, within the operating band, should be considered as initial conditions in the safety analysis, and (6) the limiting setpoint and delay time for each protection or safety system function should be used.

Provide the results of reanalysis of the LTC analysis for the LONF and FWLB with and without LOOP events and demonstrate compliance with the applicable acceptance criteria. The requested information should include the following:

1) **Discuss the methods or computer codes used for the LTC analyses. If the methods or computer codes were previously reviewed and approved by the NRC, list the NRC safety evaluations approving the methods or computer codes, and address the compliance with the conditions or restrictions. If the methods or codes were not previously reviewed and approved by NRC, address acceptability of the methods or codes.**

2) **List the nominal values with measurement uncertainties and values for the key plant parameters, initial conditions or setpoint of the protective system used in the analysis. Discuss the bases (including the degree of conservatisms) used to select the numerical values of the input parameters, initial conditions, and setpoints.**

3) **List the single failures considered in the LTC analyses and identify the worst single failure used in the analyses that result in the minimum margin to the applicable acceptable criteria.**

4) **Discuss the systems that are credited for consequences mitigation. If non-safety grade systems are used, provide justification of the use.**

Response

Supplemental Information Regarding Long Term Cooling (LTC) Feedwater Line Break (FWLB) Analysis

The LTC FWLB for auxiliary feedwater (AFW) adequacy was reanalyzed to determine subcooling margin while considering additional conservatisms in key plant parameters. Uncertainties applied to these key parameters to maximize the subcooling degradation throughout the event are tabulated in Table FWLB-1. The RETRAN code as described in LAR

Attachment 5, Section 2.8.5.0.9 was used to analyze the LTC FWLB for AFW adequacy. As noted in Section 2.8.5.0.9, the NRC previously approved the RETRAN computer code's application at St. Lucie Unit 2 as part of the 30% steam generator (SG) tube plugging and WCAP-9272 methodology change program. The three conditions of acceptance for RETRAN identified in the RETRAN safety evaluation report (SER) are described in LAR Attachment 5, Appendix A, Safety Evaluation Report Compliance. The analysis of the LTC FWLB event is in compliance with the conditions of acceptance, as denoted in Appendix A.

The FWLB event assumes the loss of the turbine driven AFW pump as the single failure, which is the highest capacity AFW pump. Control systems are assumed to function only if their normal operation yields more severe accident analysis results. Therefore, the pressurizer power operated relief valve (PORV) is modeled to minimize the reactor coolant system (RCS) pressure, which is conservative for subcooling margin. No other non-safety grade systems are used in this analysis.

The systems available in the LTC FWLB for AFW adequacy analysis are the safety grade reactor protection system (RPS) reactor trip on SG low level, the pressurizer safety valves (PSVs), the secondary side main steam safety valves (MSSVs), and the AFW system, as noted above. The SG low level setpoint on the affected SG has been reduced from the nominal setpoint (20.5% NR) to account for harsh environment uncertainites. The RPS reactor trip on high containment pressure is expected to be the first reactor trip signal; however, it is not modeled and therefore, the later reactor trip on SG low level is credited. Table FWLB-1 lists the setpoint values utilized in this analysis. Operator actions are credited as described below and include a manual trip of the reactor coolant pumps (RCPs) and manually opening one of the two safety grade atmospheric dump values (ADVs) on the intact SG.

The operator takes action at fifteen minutes into the transient to manually trip the RCPs and again at twenty five minutes to actuate an ADV to the intact SG. The ADV will maintain the intact SG pressure at 850 psia.

Table FWLB-1 provides a summary of the initial conditions utilized in the LTC FWLB event analysis. The results of the LTC FWLB analysis demonstrate that subcooling margin is maintained throughout the event, including the time in which the RCS temperatures begin to decrease. Figures FWLB-1 through FWLB-9 and Table FWLB-2 display the transient responses and sequence of events for the LTC FWLB analysis for AFW adequacy with offsite power available.

Figure FWLB-5 shows that the pressurizer reaches the maximum volume at approximately 1100 seconds. With the PORV open at that time, there is a potential that water from the RCS may escape into containment and thereby creating a radiological release situation. However, this situation is bounded by the radiological dose calculation for FWLB presented in LAR Attachment 5, Section 2.9.2.9. The analysis presented in Section 2.9.2.9 assumes a conservative release of both SG inventories with maximum RCS leakage directly to atmosphere through an outside of containment break. Additionally, the RCS leakage is assumed to occur for a 12.4 hour period before steam release is terminated to maximize the reactivity released through the outside of containment FWLB. The LTC FWLB analysis described in this response is bounded by Section 2.9.2.9 as only the inventory from the affected SG is released to the inside of the containment via the break.

The liquid discharge of LTC FWLB for AFW adequacy would be limited to the release of the RCS liquid through the PORV during the time in which the PORV is open and the pressurizer is filled. Figure FWLB-9 shows that at approximately 1600 seconds, the pressurizer pressure decreases to a point where the PORV closes. As such, the pressurizer has the potential of releasing liquid to containment for 500 seconds. Based on the integrated PORV mass release depicted in Figure FWLB-9 for the period noted, an approximate total of 22,000 lbm of liquid is released via the PORV. Any radiological release to atmosphere from this liquid would be limited by the containment leakage rate. This radiological release would be bounded by the analysis presented in Section 2.9.2.9, as it assumed a conservative 12.4 hour maximum RCS leakage to the secondary system and the release of both SG inventories to atmosphere through an outside of containment FWLB.

Note that only the event with off-site power available is provided. The difference associated with the off-site power available and the loss of off-site power (LOOP) events is the status of the RCPs. The LOOP event models the RCPs as coasting down with the LOOP at the time of reactor trip, resulting in a decrease of 20 MWt of pump heat. The decrease of 20 MWt of pump heat throughout the duration of the event associated with the LOOP events decreases the demand on the AFW system. Therefore, the long-term cooling feedline break event with off-site power available bounds the event that models a LOOP.

References

FWLB-1 R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-532), Response to NRC Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request, January 14, 2012, Accession No. ML12019A074.

TABLE FWLB-1
LTC FWLB
Initial Conditions Summary

Parameter		Value
Core power		100% + Uncertainty 3030 MWt
RCS loop flow rate		Total Design Flow (TDF) 187,500 gpm
Vessel T_{avg} temperature		High T_{avg} – Uncertainty 581.5°F
Pressurizer	Initial pressure	Nominal – Uncertainty 2180 psia
	Initial water level	Nominal + Uncertainty 66% NRS
	Charging/letdown	Unavailable
	Heater	Unavailable
	Power operated relief valve (PORV)	Available
	Spray	Unavailable
Steam generator	Initial intact water level	Nominal – Uncertainty 60% Span
	Initial faulted water level	Nominal – Uncertainty 70% Span
	Tube conditions Tube plugging	Fouled 10%
	Atmospheric dump valve (ADV)	Available
	Steam bypass control system (SBCS)	Unavailable
Auxiliary feedwater (AFW)	Pumps	1 motor driven AFW pumps
	Flowrate	275 gpm @ 1000 psia 356 gpm @ 900 psia 410 gpm @ 800 psia
	Delay	330 seconds
	Trip setpoint	Nominal + Harsh Environment 4% NRS
Loss of offsite power		Not Assumed
Reactor trip setpoint	High pressurizer pressure trip setpoint	2370 psia
	SG low level (affected)	Nominal AFAS – Harsh Environment 4% NRS
	Low steam pressure	546 psia
Decay heat		ANS-1979 + 2σ

Control Systems Credited for the Event

Control systems are assumed to function only if their normal operation yields more severe accident analysis results. Therefore, the PORV is modeled to minimize the RCS pressure, which is conservative for subcooling margin. No other non-safety grade systems are used in this analysis.

Operator Actions Credited for the Event

The operator takes action at fifteen minutes into the transient to manually trip the RCPs and again at twenty five minutes to actuate an ADV to the intact SG. The ADV will maintain the intact SG pressure at 850 psia.

Single Failure Applied to the Event

The event assumes the loss of the turbine driven AFW pump as the single failure, which is the highest capacity AFW pump.

TABLE FWLB-2
LTC FWLB
SEQUENCE OF EVENTS

Time (sec)	Event	Setpoint/Value
20.00	FWLB occurs in the main feedwater line between Loop 1 SG and last check valve	1.23 ft^2
38.90	Loop 1 affected SG low level trip setpoint reached	4% NRS
40.08	Reactor trip breaker opens	1.18 second delay
40.82	Control element assembly (CEA) release	0.74 second delay
113.67	Safety injection actuation system (SIAS) generated on low pressurizer pressure setpoint	1638 psia
131.48	Loop 2 unaffected SG main steam isolation actuation setpoint	487 psia
135.00	Minimum pressurizer volume	0 ft^3
138.23	Main steam isolation valve (MSIV) completely closed	------
139.35	Loop 2 unaffected SG level reaches AFW actuation system (AFAS) setpoint	4% NRS
167.50	Loop 1 affected SG dryout	< 500 lbm
469.35	AFW reaches loop 2 unaffected SG	330 second delay
920.00	Operator action – All RCPs manually tripped	15 minutes
1520.00	Operator action - Open ADV on unaffected SG to reduce unaffected SG pressure to 850 psia	25 minutes
1650.00	Hot leg temperature begins to decrease	------

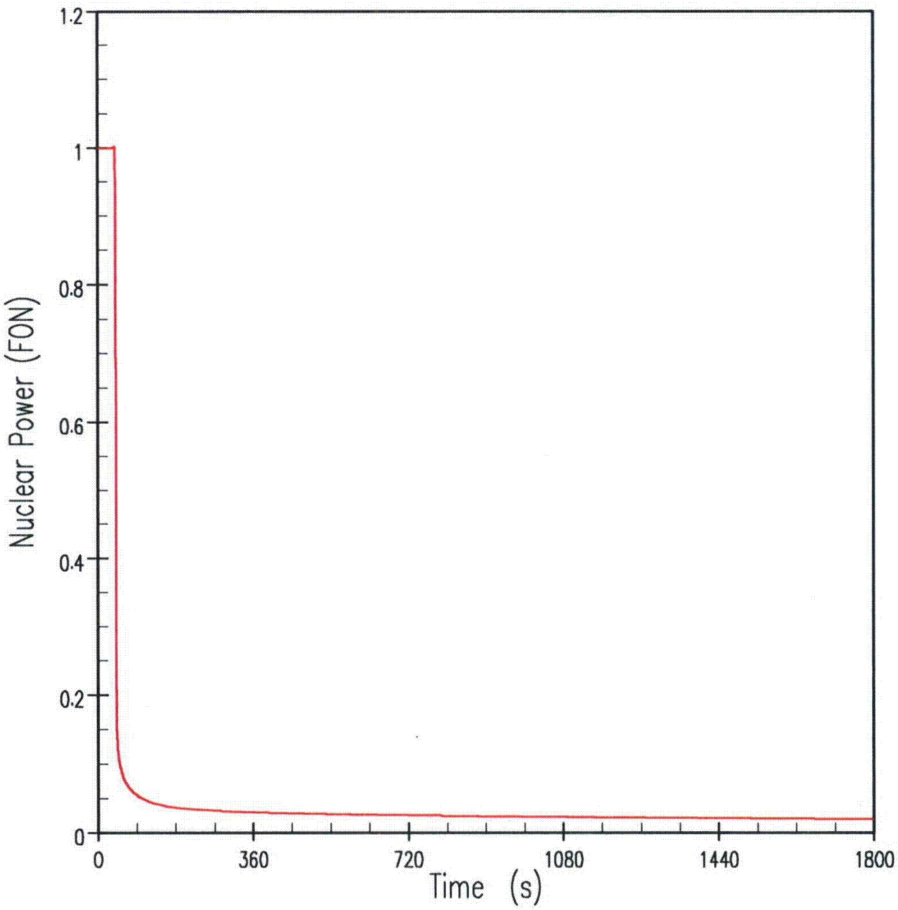

Figure FWLB-1
Nuclear Power vs. Time

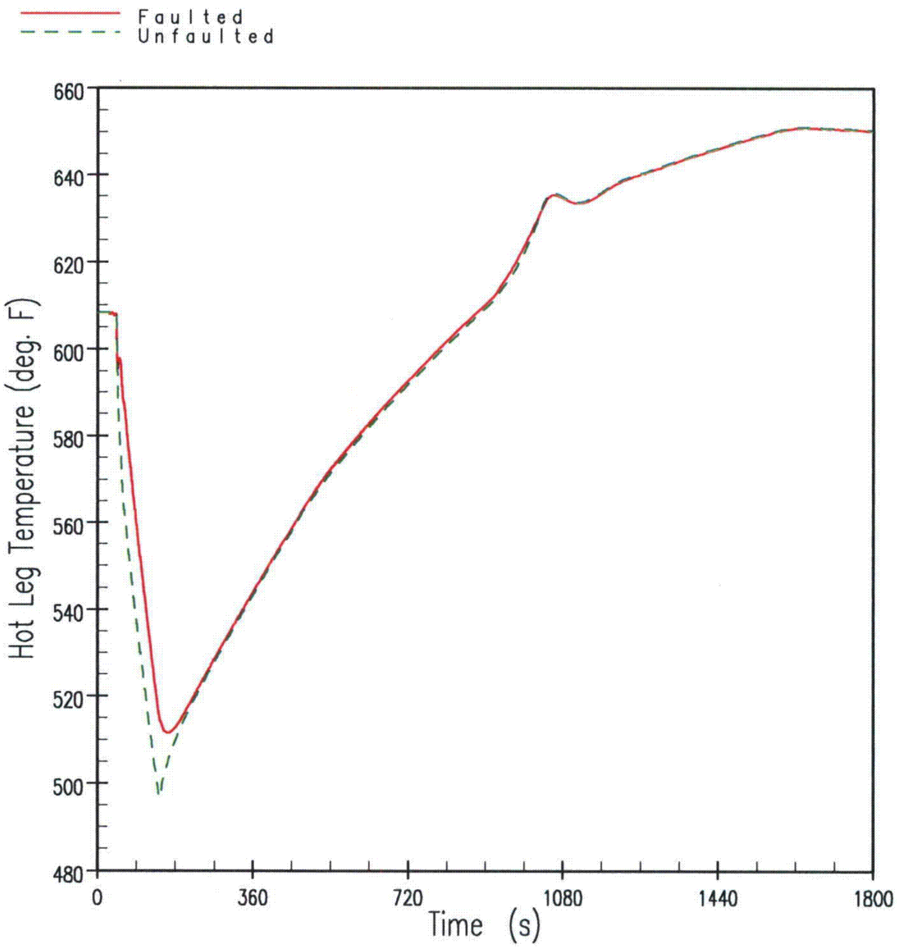

Figure FWLB-2
Hot Leg Temperature vs. Time

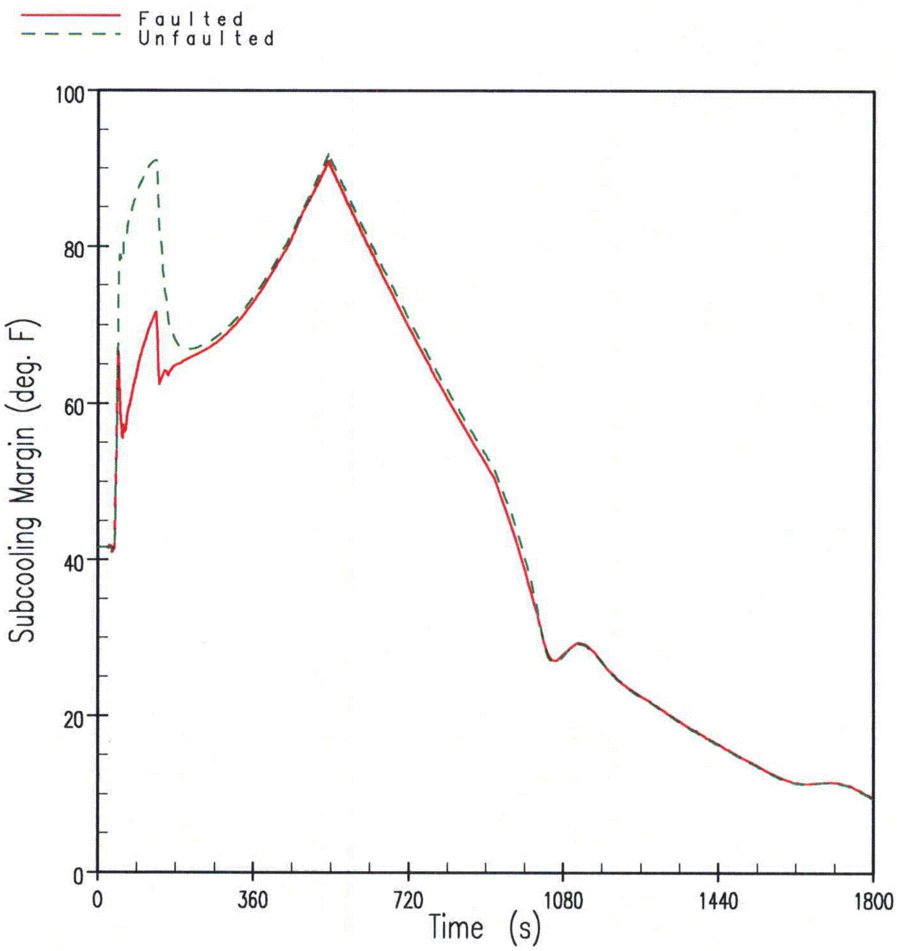

Figure FWLB-3
Subcooling Margin vs. Time

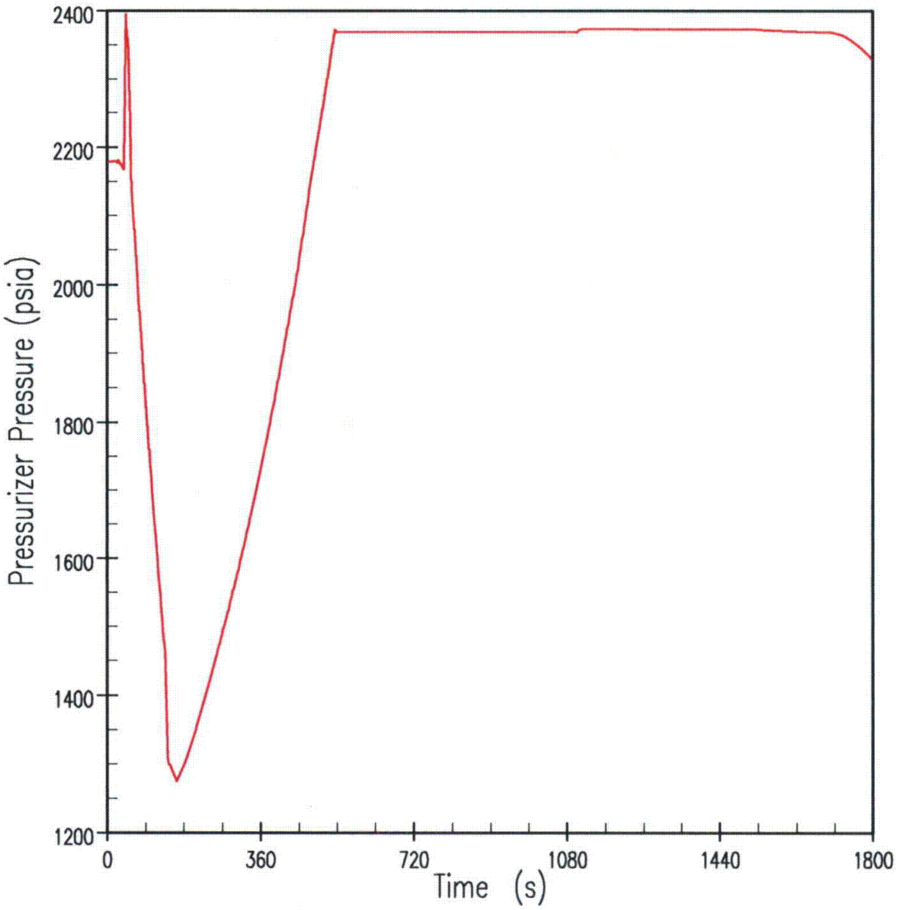

Figure FWLB-4
Pressurizer Pressure vs. Time

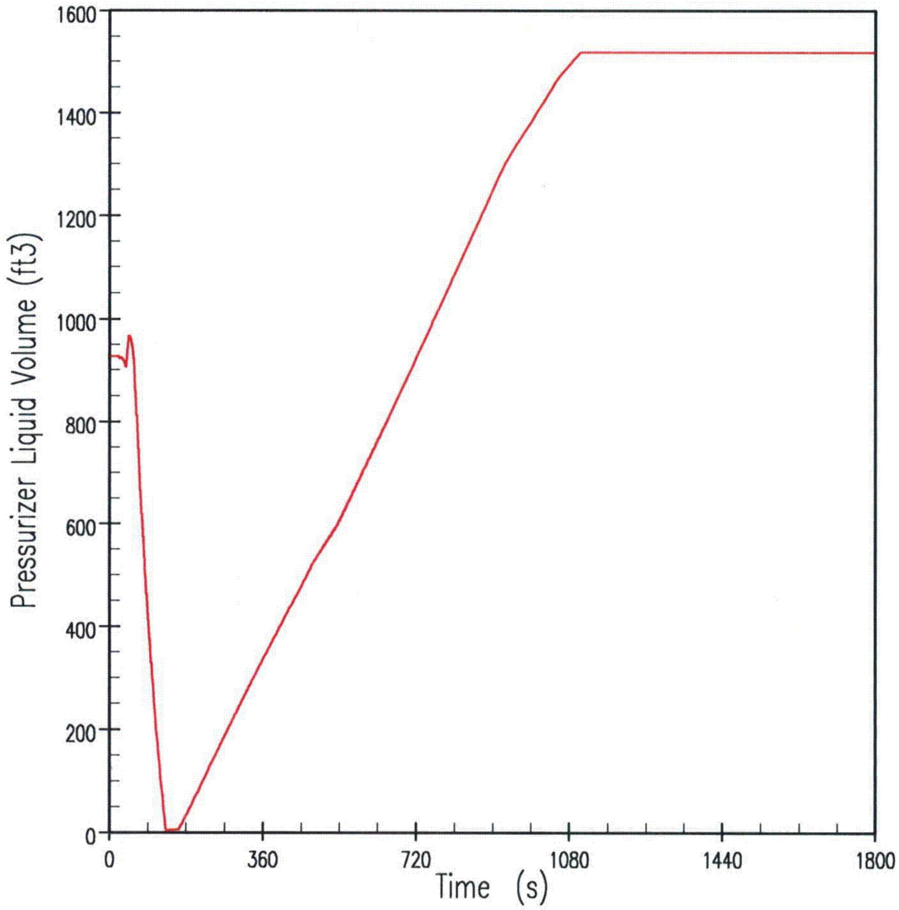

Figure FWLB-5
Pressurizer Liquid Volume vs. Time

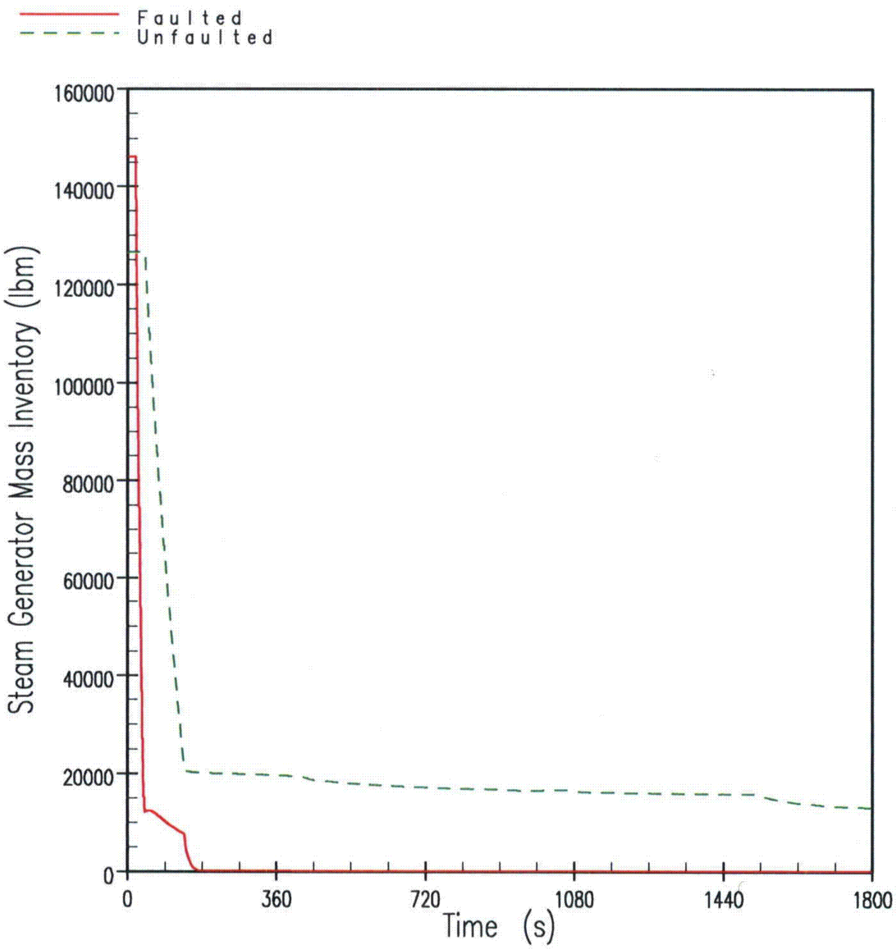

Figure FWLB-6
Steam Generator Inventory vs. Time

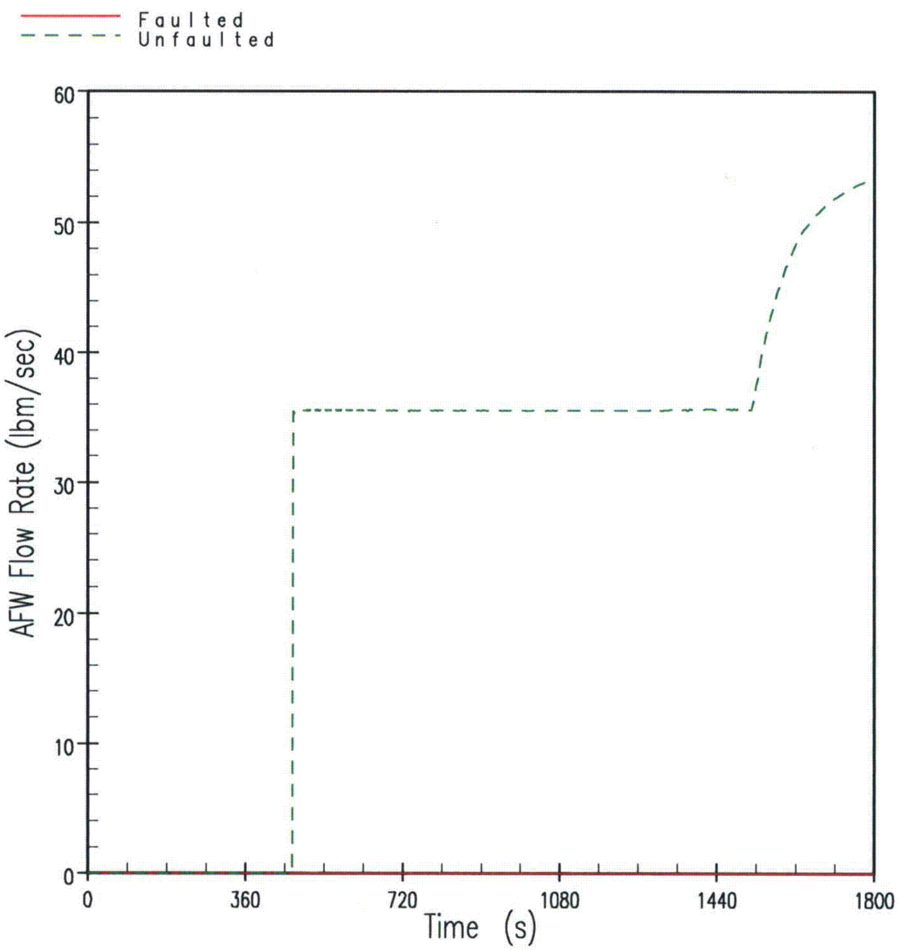

Figure FWLB-7
AFW Flow Rate vs. Time

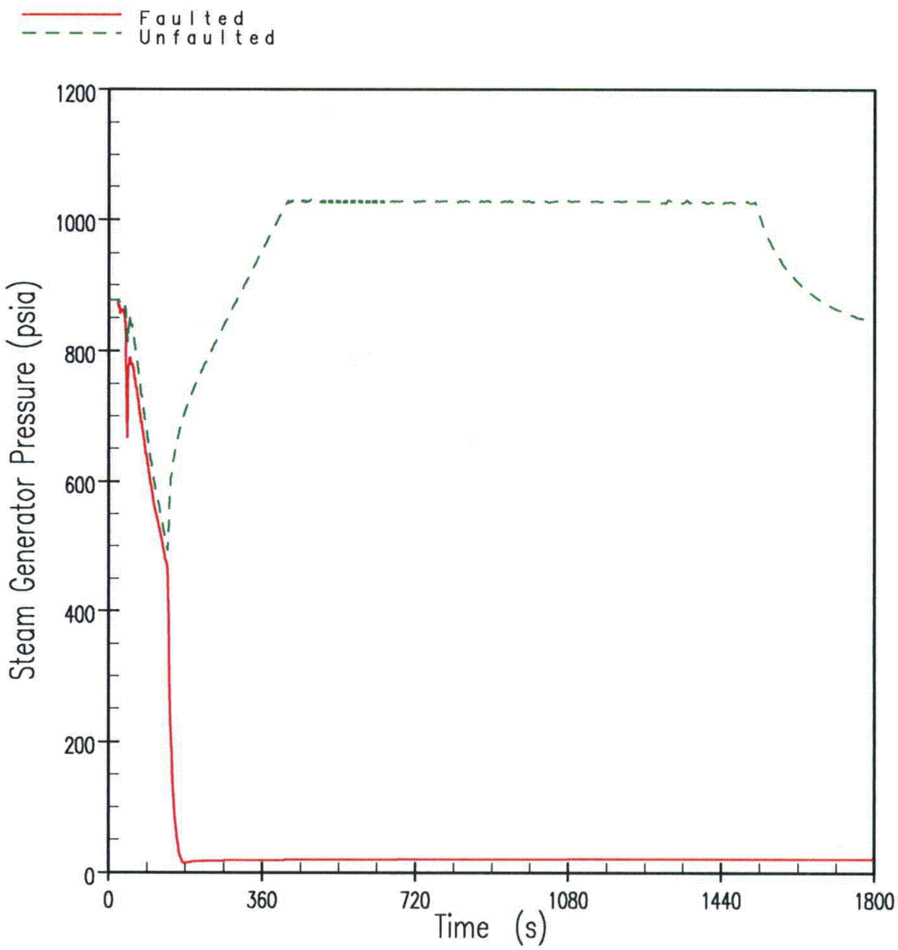

Figure FWLB-8
Steam Generator Pressure vs. Time

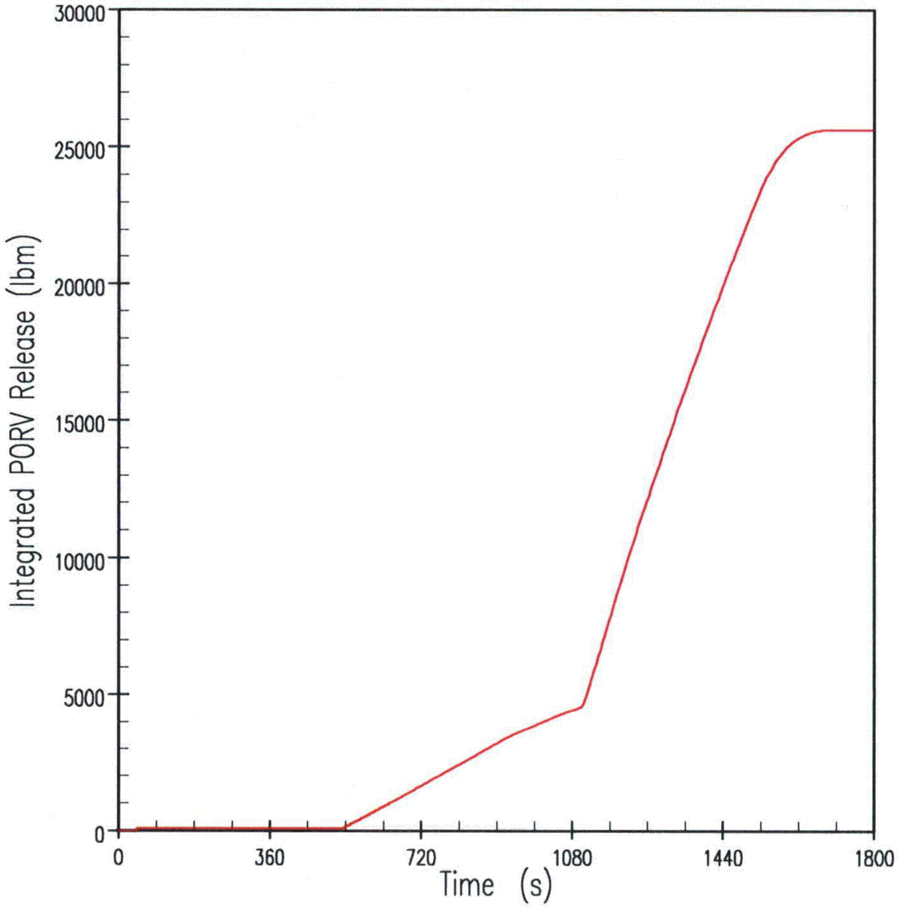

Figure FWLB-9
Integrated PORV Release vs. Time

Supplemental Information Regarding Long Term Cooling (LTC) Loss of Normal Feedwater (LONF) analysis

The LTC LONF analysis for auxiliary feedwater (AFW) adequacy was reanalyzed to determine subcooling margin while considering additional conservatisms in key plant parameters. Uncertainties were applied to these key parameters to maximize the subcooling degradation throughout the event are tabulated in Table LONF-1. The RETRAN code as described in LAR Attachment 5, Section 2.8.5.0.9 was used to analyze the LTC LONF for AFW adequacy. As noted in Section 2.8.5.0.9, the NRC previously approved the RETRAN computer code's application for St. Lucie Unit 2 as part of the 30% steam generator (SG) tube plugging and WCAP-9272 methodology change program. The three conditions of acceptance for RETRAN identified in the RETRAN safety evaluation report (SER) are described in LAR Attachment 5, Appendix A. The analysis of the LTC LONF event is in compliance with the conditions of acceptance as denoted in Appendix A.

As described in LAR Attachment 5, Section 2.5.4.5.2.4, the LTC LONF event assumes the loss of the turbine driven AFW pump as the single failure, which is the highest capacity AFW pump. Control systems are assumed to function only if their normal operation yields more severe accident analysis results. Therefore, the pressurizer power operated relief valve (PORV) is modeled to minimize the reactor coolant system (RCS) pressure, which is conservative for subcooling margin. No other non-safety grade systems are credited in this analysis.

The systems credited in the LTC LONF for AFW adequacy analysis are the safety grade reactor protection system (RPS) reactor trip on high pressurizer pressure, pressurzier safety valves (PSVs), the secondary side main steam safety valves (MSSVs), and the AFW system as noted above. No other systems are credited for the LTC LONF analysis.

Operator action was not credited for the LTC LONF event.

Table LONF-3 provides a summary of the initial conditions utilized in the LTC LONF event analysis. The results demonstrate that subcooling margin is maintained throughout the event, including the time in which the RCS temperatures begin to decrease. Figures LONF-6 through LONF-14 and Table LONF-4 display the transient responses and sequence of events for the LTC LONF analysis for AFW adequacy with offsite power available.

Consistent with the LTC feedwater line break (FWLB) event, only the LTC LONF event with off-site power available is provided. The difference associated with the off-site power available and the loss of off-site power (LOOP) event is the status of the reactor coolant pumps (RCPs). The LOOP event models the RCPs as coasting down with the LOOP at the time of reactor trip, resulting in a decrease of 20 MWt of pump heat. The decrease of 20 MWt of pump heat throughout the duration of the event associated with the LOOP events decreases the demand on the AFW system. Therefore the LTC LONF event with off-site power available bounds the event that models a LOOP.

TABLE LONF-3
LTC LONF
Initial Conditions

Parameter		Value With AC Power
Core power		100% + Uncertainty 3030 MWt
RCS loop flow rate		Total Design Flow (TDF) 187,500 gpm
Vessel T_{avg} temperature		High Nominal – Uncertainty 581.5°F
Pressurizer	Initial pressure	Nominal – Uncertainty 2180 psia
	Initial water level	Nominal + Uncertainty 66% NRS
	Charging/letdown	Unavailable
	Heater	Unavailable
	Power operated relief valve (PORV)	Available (1)
	Spray	Unavailable
Steam generator	Initial water level	Nominal 65% NRS
	Tube conditions Tube plugging	Fouled 10%
	Atmospheric dump valve (ADV)	Not available
	Main steam safety valves (MSSVs)	Design + Uncertainty Bank 1 – 1030 psia Bank 2 – 1060.8 psia
Auxiliary feedwater (AFW)	Pumps	2 motor driven AFW pumps (MDAFP)
	Flowrate	275 gpm per MDAFP
	Delay	330 seconds
	Initiation trip setpoint	Low Nominal – Uncertainty 14.25% NRS
Reactor trip setpoint	High pressurizer pressure	Nominal – Uncertainty 2370 psia
Decay heat		ANS-1979 + 2σ

<u>Control Systems Credited for the Event</u>

Control systems are assumed to function only if their normal operation yields more severe accident analysis results. Therefore, the PORV is modeled to minimize the RCS pressure, which is conservative for subcooling margin. No other non-safety grade systems are used in this analysis.

Operator Actions Credited for the Event

Operator actions are not credited for this event.

Single Failure Applied to the Event

The event assumes the loss of the turbine driven AFW pump as the single failure, which is the highest capacity AFW pump.

TABLE LONF-4
LTC LONF
SEQUENCE OF EVENTS

Time (sec)	Event	Setpoint/Value
20.0	Loss of feedwater to both SGs	
47.3	High pressurizer pressure setpoint reached	2370 psia
48.0	Maximum RCS pressure	2428 psia
48.1	Pressurizer PORV actuates	
48.5	Reactor trip breaker opens	1.2 second delay
49.3	Reactor trip (CEA) release	0.74 second delay
52.3	MSSV bank 1 opens (both SGs)	1030 psia
53.7	MSSV bank 2 opens (both SGs)	1060.8 psia
79.8	Loop 2 SG low level AFW actuation setpoint	14.25% NRS
409.4	AFW flow reaches SGs	330 second delay
1463.9	Maximum pressurizer liquid volume	1263 ft^3
1467.2	Minimum SG inventory	16,800 lbm/SG
~1500	Hot leg temperature begins to decrease	------

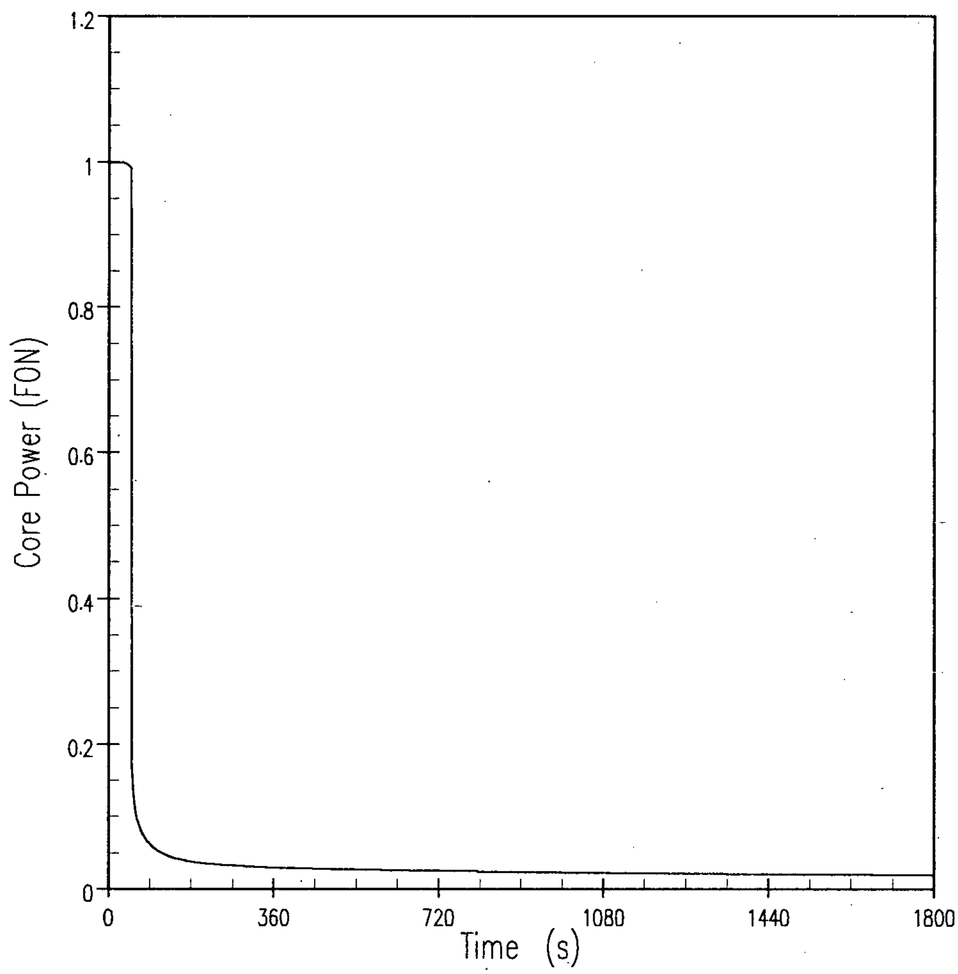

Figure LONF-6
Core Power vs. Time

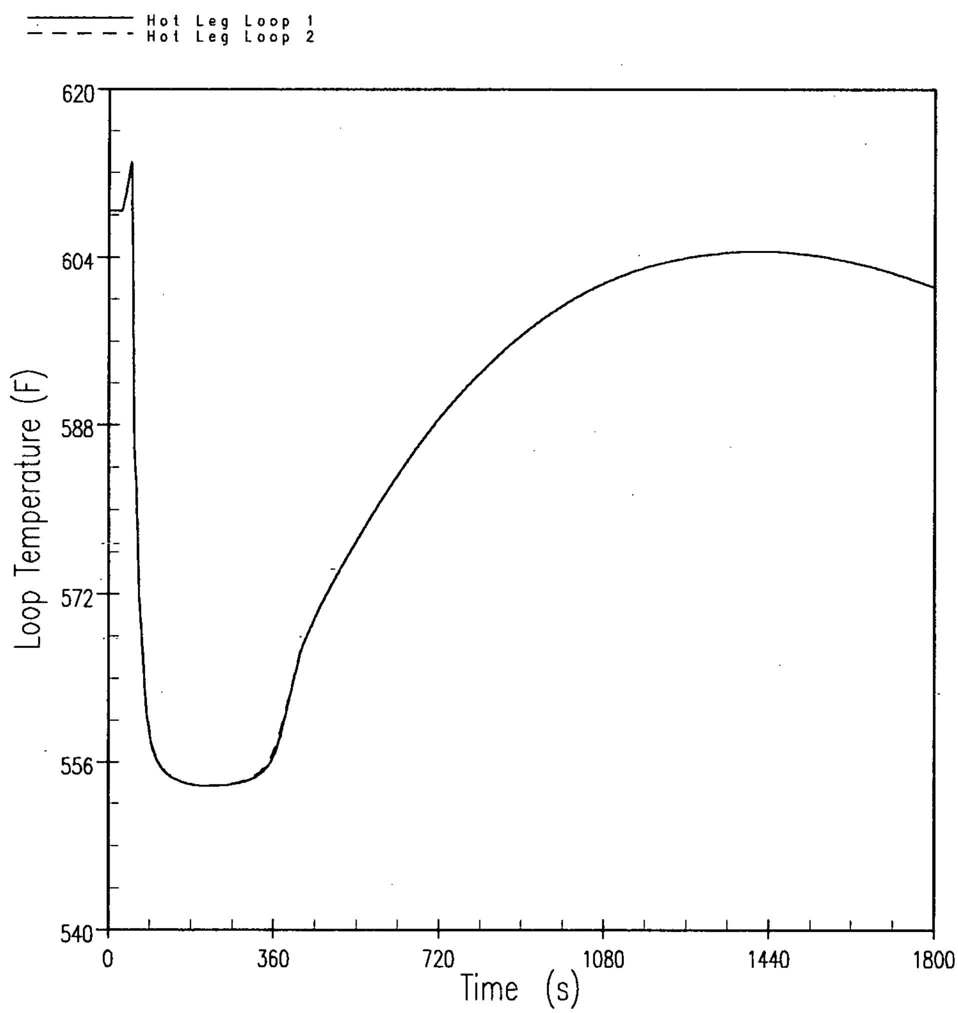

Figure LONF -7
RCS Hot Leg Temperature vs. Time

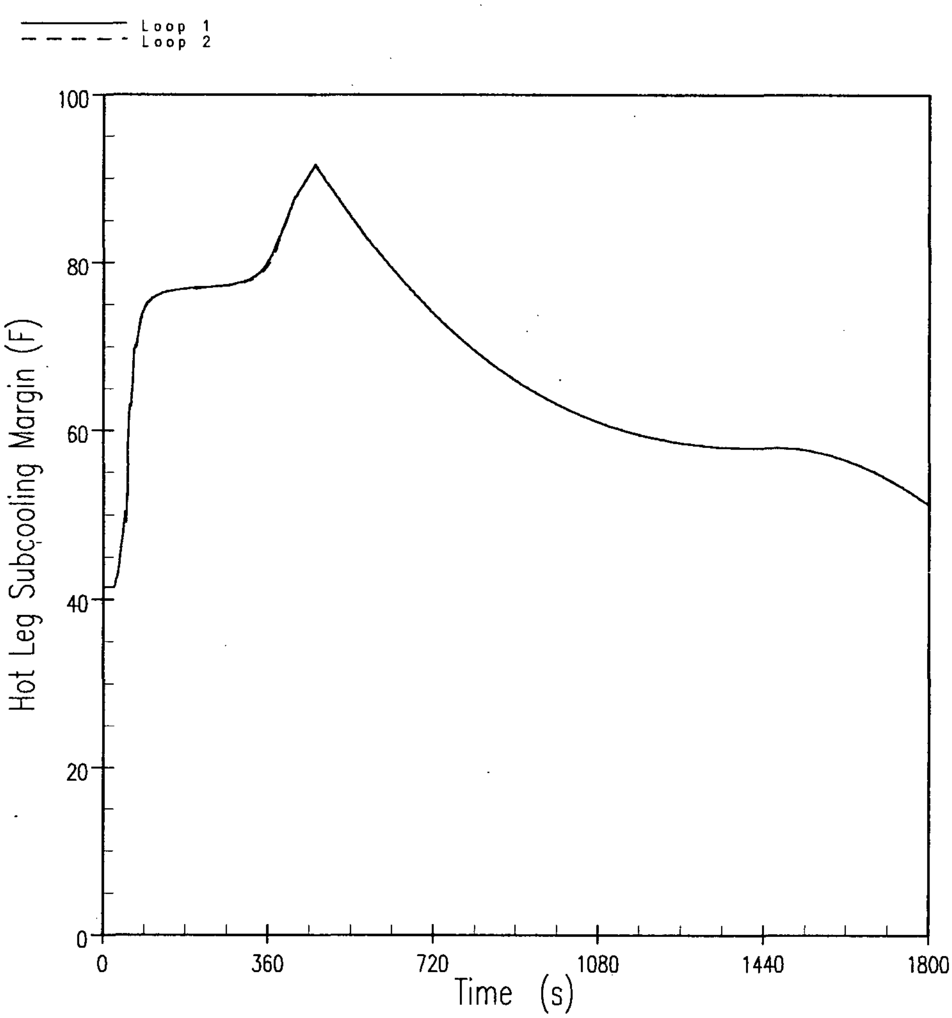

Figure LONF -8
Hot Leg Subcooling Margin vs. Time

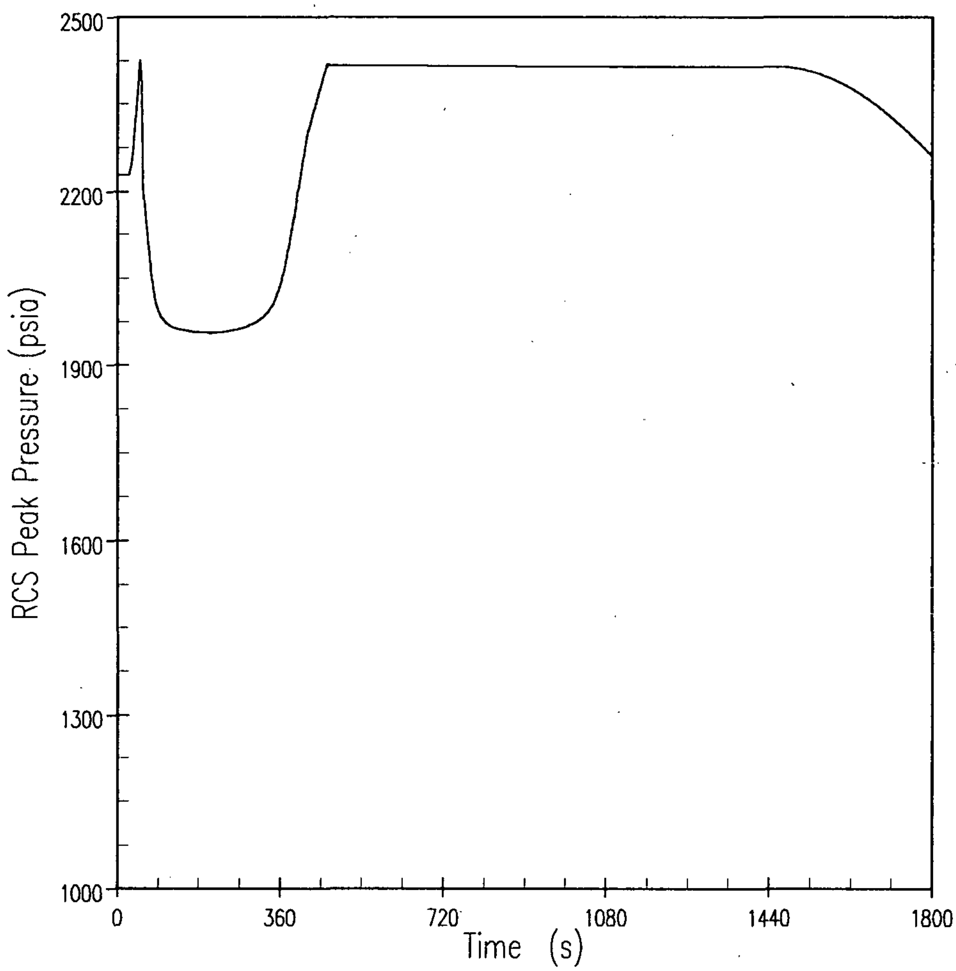

Figure LONF -9
RCS Peak Pressure vs. Time

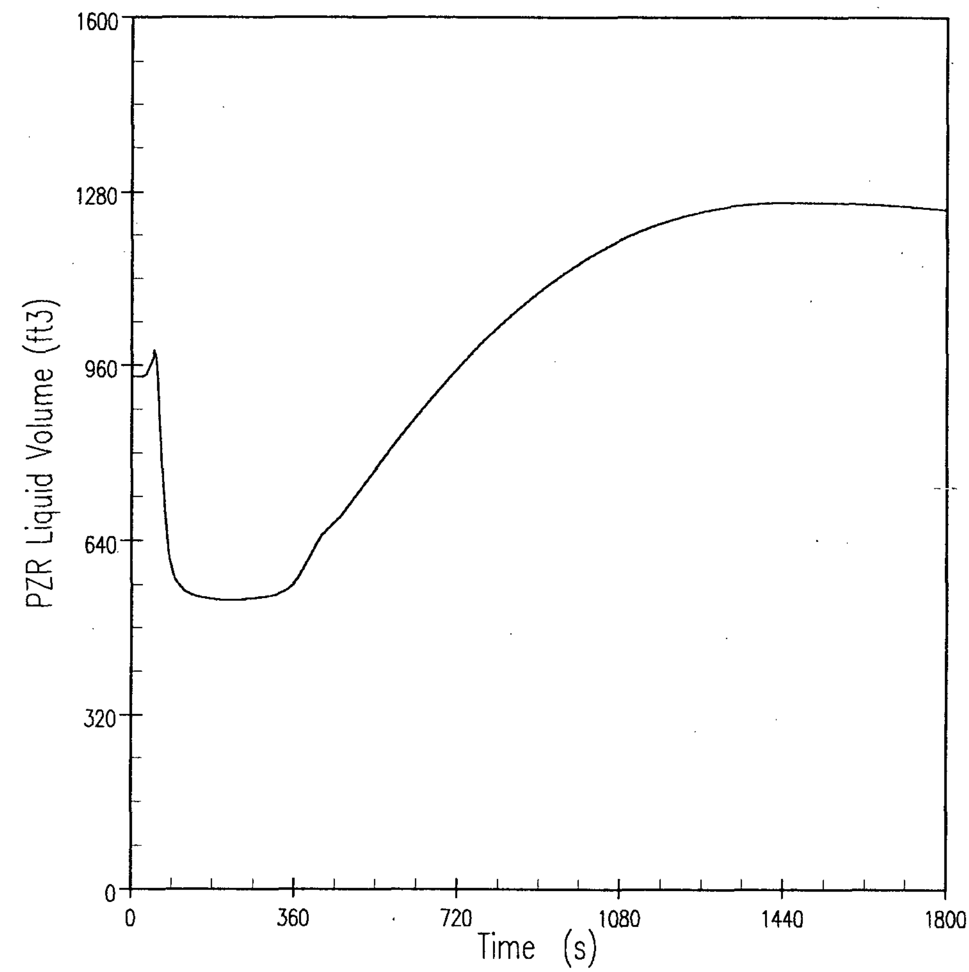

Figure LONF -10
Pressurizer Liquid Volume vs. Time

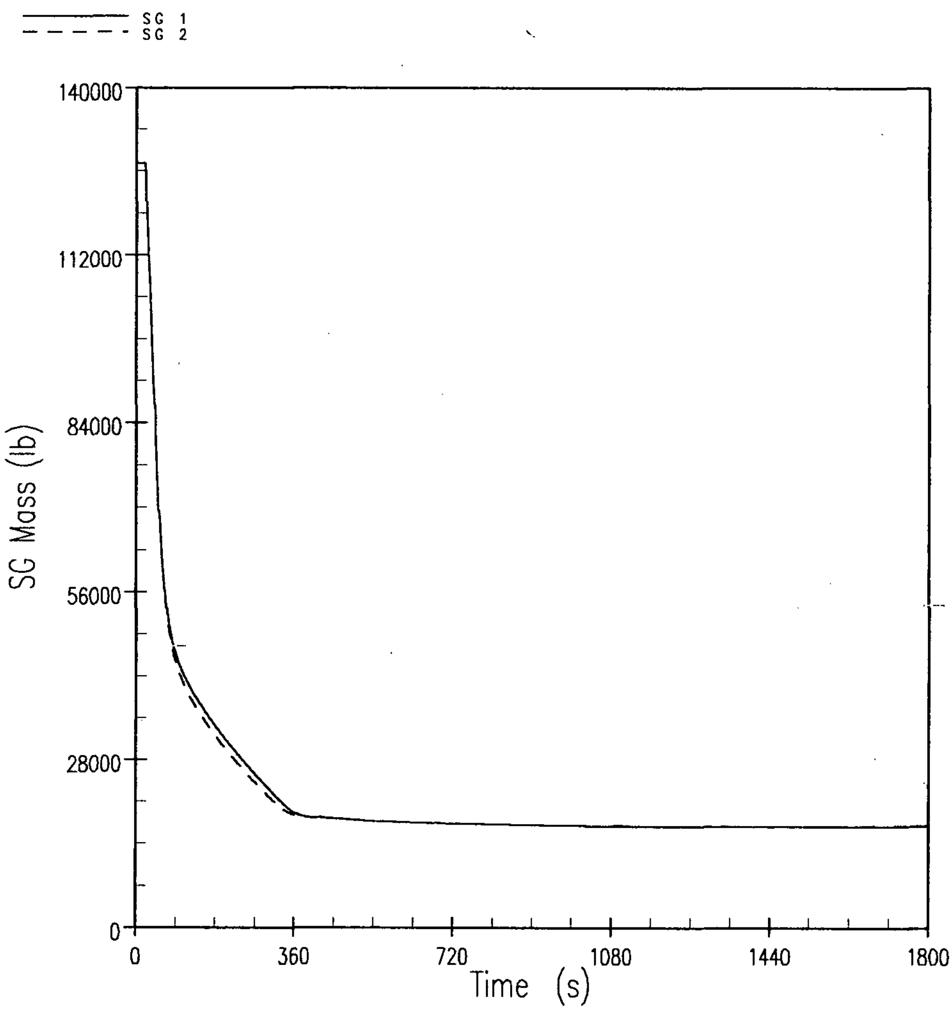

Figure LONF -11
Steam Generator Mass vs. Time

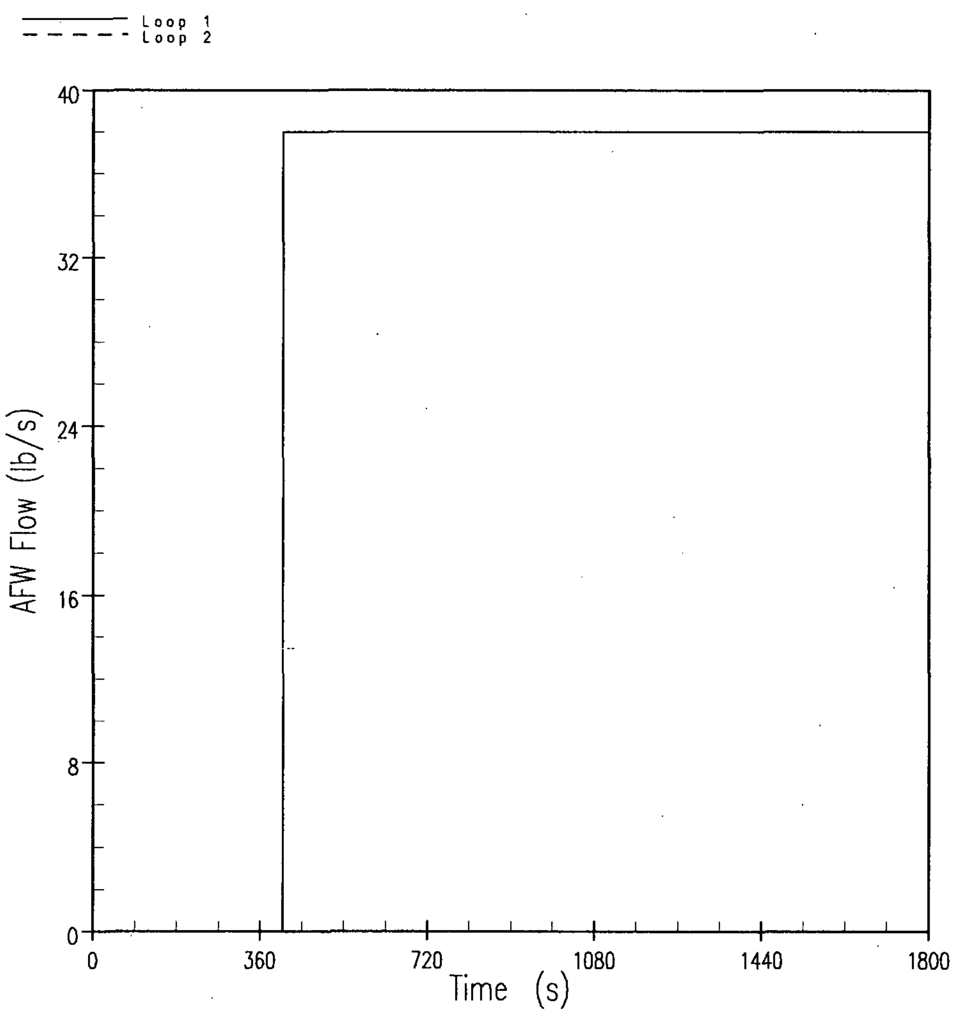

Figure LONF -12
AFW Flow vs. Time

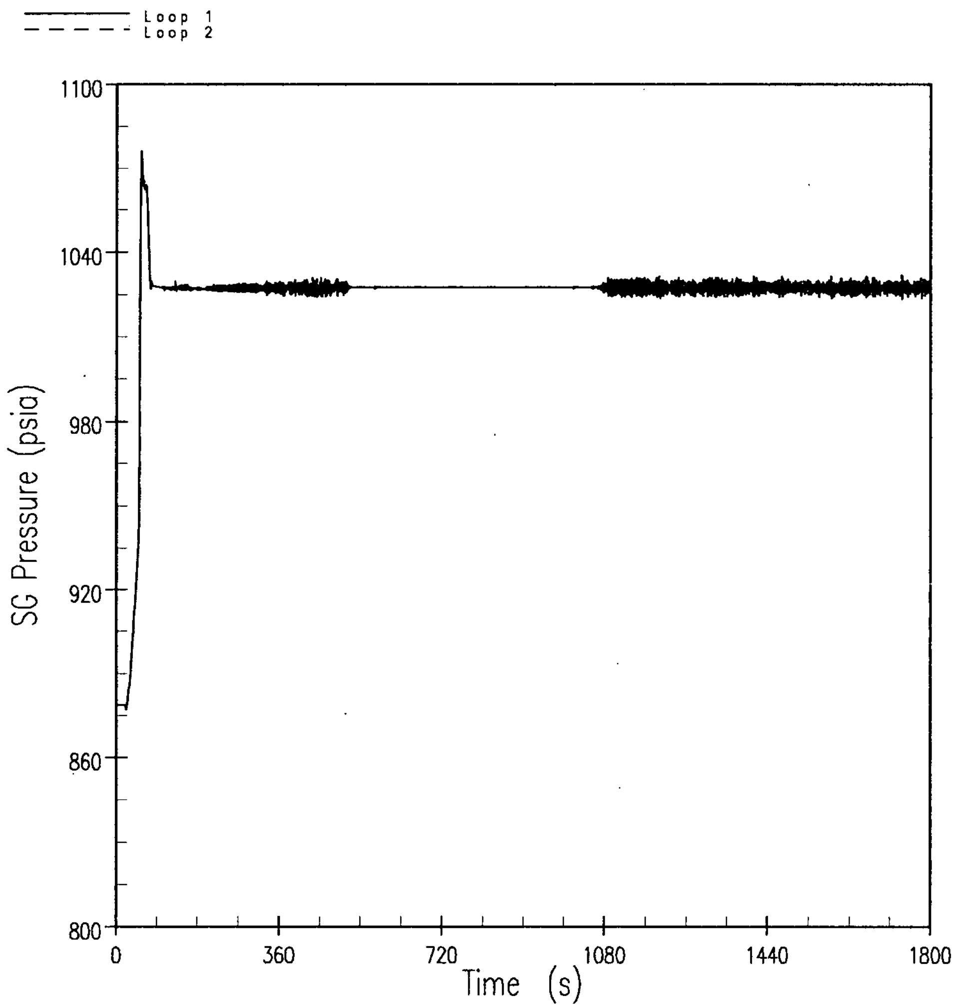

Figure LONF -13
Steam Generator Pressure vs. Time

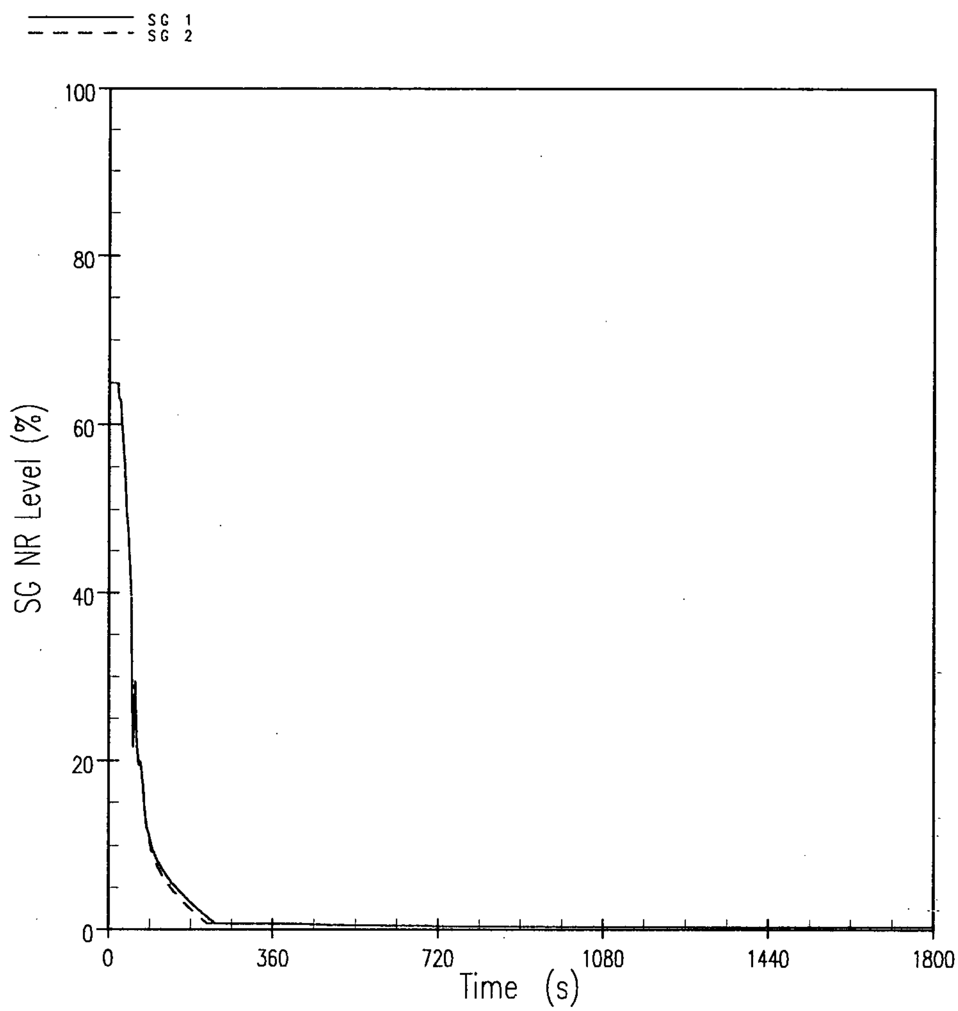

Figure LONF -14
Steam Generator Narrow Range Level vs. Time

Asymmetric Steam Generator Transient (ASGT)

A question was asked with regard to the inputs to the RETRAN departure from nucleate boiling ratio (DNBR) model and whether the DNBR model inputs reflect the asymmetry of the event. This question was raised as the current methodology utilizes the VIPRE thermal-hydraulics code to evaluate DNBR and the EPU method is based on the RETRAN DNBR model.

Response

The RETRAN model, which was approved for St. Lucie Unit 2 as part of the 30% steam generator tube plugging and WCAP-9272 methodology change program, was reviewed to determine the inputs and overall methodology comprising the departure from nucleate boiling ratio (DNBR) model. The RETRAN DNBR is calculated as part of every iteration utilizing the following inputs:

1) Current pressurizer pressure minus the reference pressure;
2) Lower unheated core node temperature for each of the four core quadrants;
3) Upper unheated core node temperature for each of the four core quadrants; and
4) Core heat flux as a fraction of nominal power.

The average of each core sector temperature is then calculated utilizing the lower and upper core quadrant temperatures. Each core quadrant's average temperature is then used to calculate (auctioneer high) the maximum of the loop oriented core sector temperatures.

The DNBR algorithm is based on partial derivatives of pressure and of power with respect to temperature. The DNBR algorithm derivatives are determined as part of the core limits evaluation.

Based on these inputs, the RETRAN DNBR algorithm utilizes the maximum average core quadrant temperature, the change in heat flux (power), and the change in pressurizer pressure during the transient.

With respect to the asymmetric steam generator transient (ASGT), the RETRAN DNBR algorithm will incorporate the maximum effects of the temperature increase based on utilizing the maximum average core quadrant temperature, the impact of the pressure increase by comparing it to the initial pressure during the transient, and the impact of the power increases.

The RETRAN algorithm provides a conservative DNBR as it selects the limiting (highest) average core quadrant temperature and accounts for other transient affects. The algorithm therefore provides a conservative DNBR value post ASGT event initiation.

Reactor Coolant Pump (RCP) Rotor Seizure/Shaft Break and Control Element Assembly (CEA) Withdrawal from Subcritical

A question was asked with regard to the change of the departure from nucleate boiling ratio (DNBR) safety analysis limit (SAL) for the RCP rotor seizure/shaft break event and for the CEA withdrawal from subcritical analyses. The request was to justify that the change in the DNBR SAL was made by only reducing the available plant specific margin and that all necessary components were still accounted for in the SAL DNBR limit.

Response

Reactor Coolant Pump (RCP) Rotor Seizure and Shaft Break Supplemental Information

EPU LAR Attachment 5, Section 2.8.5.3.2 discusses the RCP rotor seizure and shaft break event. LAR Attachment 5, Table 2.8.5.3.2-2 presents the results of this analysis and provides the current Updated Final Safety Analysis Report (UFSAR) departure from nucleate boiling ratio (DNBR) limit of [ ](a,c). For the RCP rotor seizure event, Table 2.8.5.3.2-2 states that the DNBR safety analysis limit (SAL) for the EPU analysis is [ ](a,c).

Based on the results of the RCP rotor seizure event provided in Section 2.8.5.3.2 and listed on Table 2.8.5.3.2-2, the current SAL DNBR of [ ](a,c) was reduced to [ ](a,c). The reduction was performed through the removal of a portion of the discretionary plant specific margin that was initially added to the 95/95 Revised Thermal Design Procedure (RTDP) design DNBR limit of 1.32. Reducing the level of discretionary plant specific margin results in no rods-in-DNB.

The thermal-hydraulic design section of the LAR Attachment 5, Section 2.8.3.2.2.2, discusses the basis for the RTDP design DNBR limit and lists the various uncertainties included therein. Section 2.8.3.2.2.2 also discusses that the rod bow penalty and discretionary plant specific margin is applied to the RTDP design DNBR limit to produce the safety analysis DNBR limit for the RCP rotor seizure event.

LAR Attachment 5, Table 2.8.3-5 provides the various DNBR limits applicable to the EPU. The RTDP DNBR limit is provided on Table 2.8.3-5 as 1.32, this RTDP DNBR limit is then increased by [ ](a,c) to obtain the SAL DNBR limit of [ ](a,c). As noted in Table 2.8.3-5 and in Section 2.8.3.2.3.7, the rod bow DNBR penalty of [ ](a,c) is also included in the SAL DNBR limit of [ ](a,c) and therefore the available discretionary plant specific margin is reduced to [ ](a,c) to account for the rod bow DNBR penalty.

A reduction of the SAL DNBR limit from [ ](a,c) to [ ](a,c) for the EPU RCP rotor seizure event maintains [ ](a,c) discretionary margin, in addition to the margin required for offsetting the rod bow DNBR penalty of [ ](a,c).

Therefore, the EPU RCP rotor seizure SAL DNBR limit value of [ ](a,c) listed in Table 2.8.5.3.2-2 remains conservative with respect to the 95/95 DNB acceptance criterion as provided in Table 2.8.3-5.

Control Rod Withdrawal from a Subcritical Condition Supplemental Information

EPU LAR Attachment 5, Section 2.8.5.4.1 discusses the control rod withdrawal from a subcritical condition. LAR Attachment 5, Table 2.8.5.4.1-3 presents the results of this analysis and provides the Standard Design Thermal Procedure (STDP) safety analysis limit (SAL) departure from nucleate boiling ratio (DNBR) limit of [ ](a,c) for the EPU analysis. Table 2.8.5.4.1-3 also provides the STDP SAL DNBR limit for the current analysis of [ ](a,c). Both DNBR limits in Table 2.8.5.4.1-3 retain discretionary plant specific margin above the 95/95 DNB acceptance criterion.

Based on the results of the control rod withdrawal from a subcritical condition analysis provided in Section 2.8.5.4.1 and listed on Table 2.8.5.4.1-3, the current SAL DNBR of [ ](a,c) was reduced to [ ](a,c). The reduction was performed through the removal of a portion of the discretionary plant specific margin that was initially added to the STDP DNBR correlation limit of 1.13, as listed on LAR Attachment 5, Table 2.8.3-5. By reducing the amount of discretionary plant specific margin in the STDP SAL DNBR limit, all acceptance criteria for the event were satisfied.

The thermal-hydraulic design section of the LAR Attachment 5, Section 2.8.3.2.2.2.1, discusses the basis for the STDP correlation DNBR limit. The STDP methodology is based on the DNBR correlation limit of 1.13 as listed on Table 2.8.3-5. The engineering factors and other uncertainties are applied directly into the VIPRE-W model or applied as multipliers on the calculated DNBRs for the event. The rod bow DNBR penalty is the only necessary penalty for which the retained margin between the DNBR correlation limit and the SAL DNBR must account.

Based on the STDP DNBR correlation limit of 1.13 and the SAL DNBR limit of [ ](a,c) applicable to the current rod withdrawal from a subcritical condition event, the plant specific margin retained in the current SAL DNBR limit of [ ](a,c) is [ ](a,c). A reduction of the current SAL DNBR limit from [ ](a,c) to [ ](a,c) for the EPU rod withdrawal from a subcritical condition event still retains the plant specific margin of [ ](a,c).

The rod bow DNBR penalty applicable to the STDP DNBR correlation is [ ](a,c) as stated in LAR Attachment 5, Section 2.8.3.2.3.7. Therefore the available discretionary plant specific margin is reduced to [ ](a,c) after accounting for the rod bow DNBR penalty as applied to the EPU rod withdrawal from a subcritical condition event.

In conclusion, the EPU rod withdrawal from a subcritical condition STDP SAL DNBR limit of [ ](a,c) listed on Table 2.8.5.4.1-3 remains conservative with respect to the 95/95 DNB acceptance criterion as described in LAR Attachment 5, Table 2.8.3-1.

Inadvertent Opening of a Power Operated Relief Valve (IOPORV)

A question was asked with regard to the time required to fill the pressurizer as a result of the IOPORV event. Specifically a question was asked to provide the timing of the pressurizer fill for the IOPORV event and to describe the control room operator's actions used to mitigate the event.

Response

The IOPORV event is discussed in LAR Attachment 5, Section 2.8.5.6.1. As described in Section 2.8.5.6.1.2.5, the purpose of the IOPORV analysis is to demonstrate that the reactor protection system (RPS) functions and mitigates the consequences of a reactor coolant system (RCS) depressurization event at the EPU conditions utilizing the currently approved methodology. The event is analyzed to ensure that the departure from nucleate boiling ratio (DNBR) criterion is met.

A question was asked as part of the NRC audit of the EPU application regarding the time to fill the pressurizer to a water solid condition during an IOPORV event. As the event is analyzed for DNBR criterion, the timeframe of the event is very short and is terminated prior to the overfill condition of the pressurizer. The event was reanalyzed by extending the end time of the transient beyond that required to fill the pressurizer to a water solid condition. A set of sensitivity runs were completed to determine the impact of various input conditions on the time to fill. Table IOPORV-1 provides the final set of analysis input assumptions for the most limiting (shortest) time to fill the pressurizer.

Table IOPORV-2 provides the sequence of events for the IOPORV fill event. The limiting case demonstrates that the pressurizer will fill to a water solid condition in just under three minutes from the start of the event. Figures IOPORV-1 through IOPORV-4 provide additional details of the IOPORV event modeling the overfill condition.

An IOPORV will result in one or more of the following control room annunciators:

- H-9 – PZR CHANNEL X PRESS HIGH/LOW,
- H-10 – PZR CHANNEL Y PRESS HIGH/LOW,
- H-16 – QUENCH TANK PRESS HIGH,
- H-17 – PZR CHANNEL X LEVEL HIGH/LOW,
- H-18 – PZR CHANNEL Y LEVEL HIGH/LOW,
- H-20 – PORV V1475 RELIEF LINE TEMP HIGH,
- H-24 – QUENCH TANK TEMP HIGH,
- H-29 – PZR PROPORTIONAL HTR LOW LEVEL TRIP/INTERLOCK,
- H-30 – PZR BACKUP HTR LOW LEVEL TRIP/SS ISOL/INTLK,
- H-32 – QUENCH TANK LEVEL HIGH/LOW,
- H-36 – PORV V1474 RELIEF LINE TEMP HIGH, and
- LC-1 – PZR PORV/SAFETY OPEN.

The annunciator response procedures for these alarms provide direction to the operator to go to abnormal operating procedure 2-AOP-01.10, Pressurizer Pressure and Level. The first immediate operator action for 2-AOP-01.10 is to verify operating pressure is stable. The first contingency action requires determining if a PORV is open or leaking and provides direction to place the PORV in OVERRIDE and close the PORV block valve.

For the limiting case that was analyzed, the safety injection actuation system (SIAS) actuates at 40.9 seconds. As discussed above, isolation of a PORV is addressed in 2-AOP-01.10 as an immediate action. Operators respond to all alarms, expected and unexpected, and perform immediate operator actions from memory. Simulator experience has demonstrated that the operator would respond in approximately 10 seconds. Assuming the operator was not in the vicinity of the PORV switch on the control board or needs to use the procedure, the PORV will be closed or isolated prior to water passing through the PORV or the pressurizer becoming water solid. If the event is terminated prior to SIAS, additional charging pumps and the high pressure safety injection pumps are not actuated and pressurizer overfill is not a concern.

TABLE IOPORV-1
IOPORV OVEFILL EVENT
INITIAL CONDITIONS

Parameter		Value
Core power		100% + Uncertainty 3030 MWt
RCS loop flow rate		Minimum Measured Flow (MMF) 195,000 gpm
Vessel T_{avg} temperature		Low T_{avg} 563°F
Pressurizer	Initial pressure	Nominal – Low 2225 psia
	Initial water level	Nominal 63% NRS
	Charging	Available (2 pumps)
	Letdown	Unavailable
	Heater	
	Power operated relief valve (PORV)	Available (1)
	Spray	Available
Steam generator	Initial water level	Nominal 65% Span
	Tube conditions Tube plugging	Fouled 10%
	Atmospheric dump valve (ADV)	Unavailable
	Steam bypass control system (SBCS)	
High pressure safety injection (HPSI)	HPSI pumps	2
	Flowrate	Maximum
	Setpoint	1683 psia
Reactor trip setpoint	Thermal margin/low pressure (TM/LP)	1855 psia
Decay heat		ANS-1979 + 2σ

Control Grade Systems Credited for the Event

Control systems are assumed to function only if their normal operation yields more severe accident analysis results. Pressurizer spray and charging flow are credited as these are conservative for the overfill case. No other non-safety grade systems are used in this analysis.

Operator Actions Credited for the Event

No operator actions were assumed for this event.

TABLE IOPORV-2
IOPORV OVEFILL EVENT
SEQUENCE OF EVENTS

Event	Time (seconds)
Transient initiation (PORV spuriously opens	10.110
Thermal margin/low pressure (TM/LP) trip setpoint reached	42.374
Reactor trip on TM/LP	43.514
Safety injection signal	50.986
Safety injection initiated	71.000
Pressurizer fills	184.00
First main steam safety valve (MSSV) opens	222.470

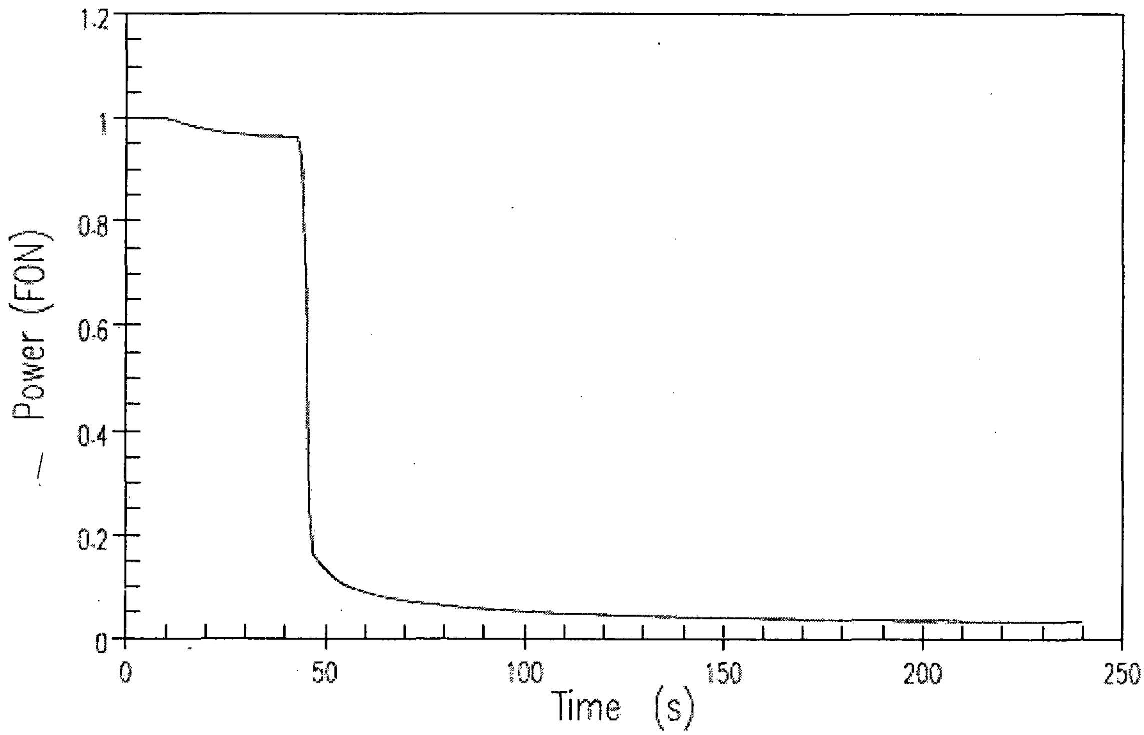

Figure IOPORV-1
Nuclear Steam Supply System (NSSS) Power vs. Time

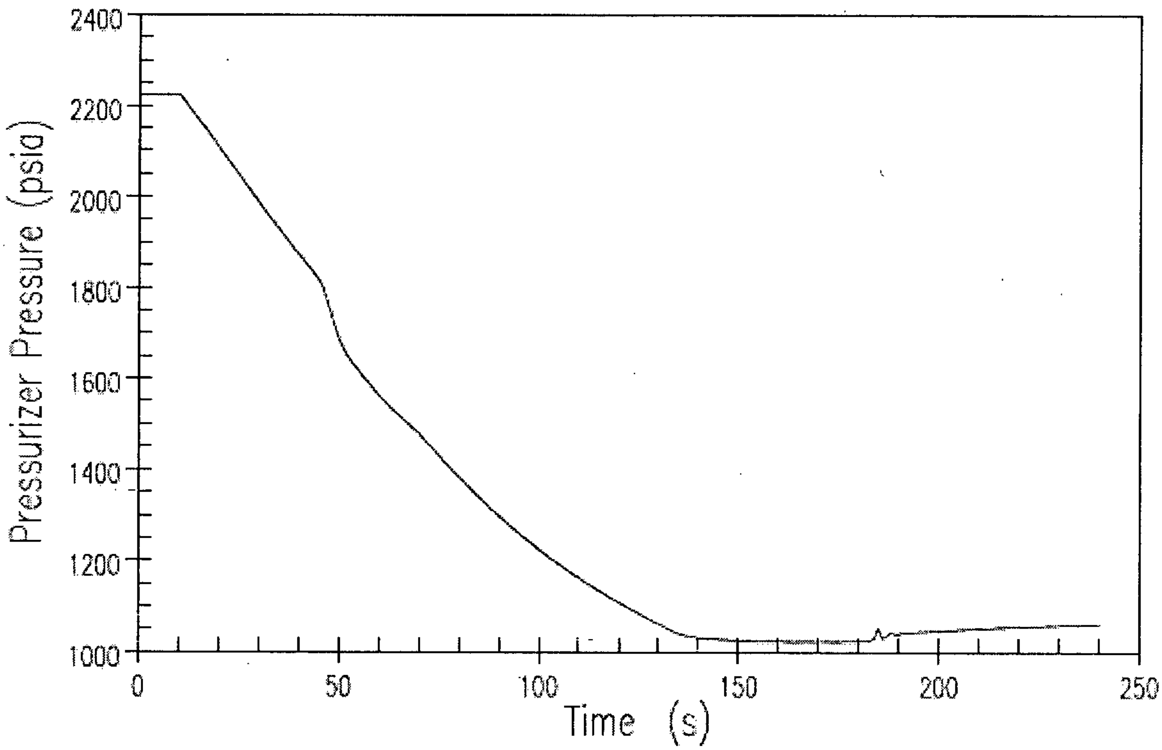

Figure IOPORV-2
Pressurizer Pressure vs. Time

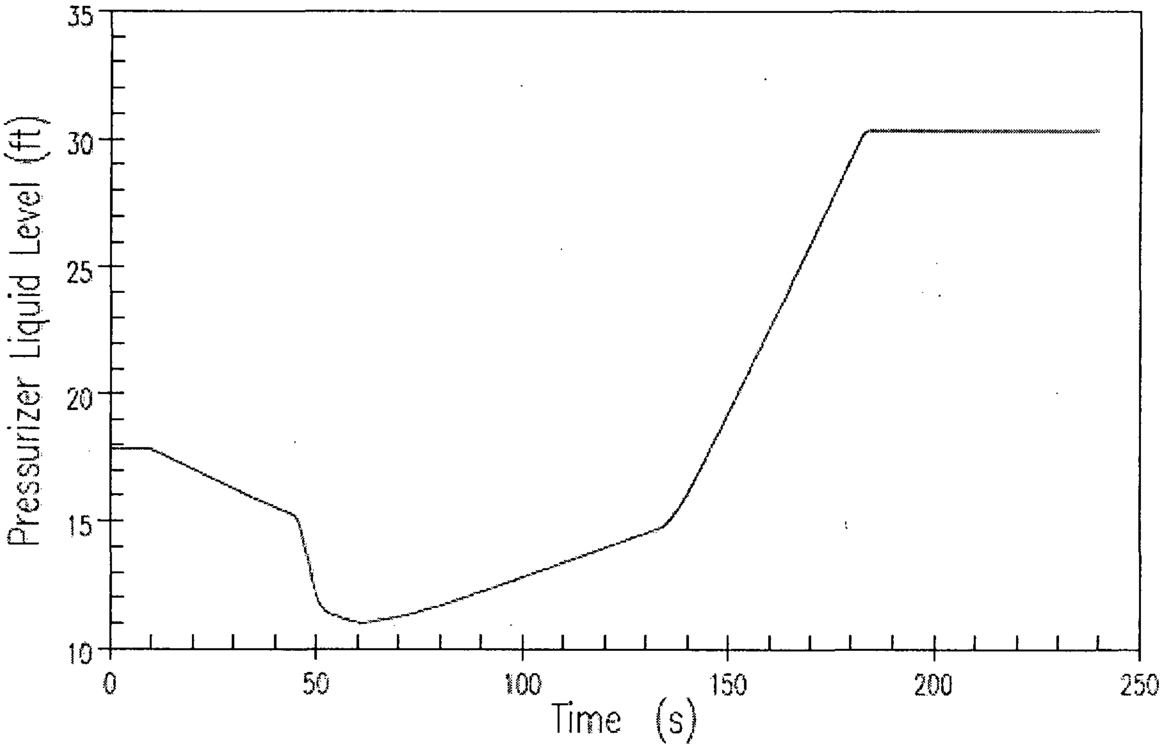

Figure IOPORV-3
Pressurizer Liquid Level vs. Time

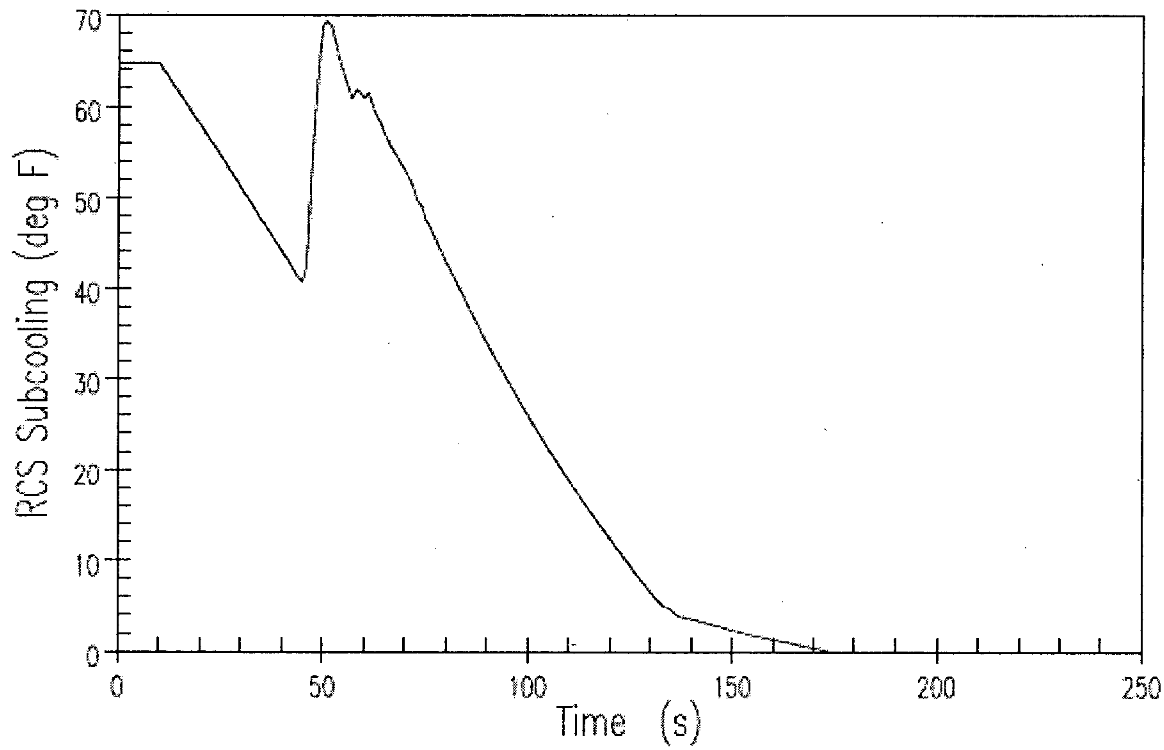

Figure IOPORV-4
RCS Subcooling vs. Time

ATTACHMENT 3

EXTENDED POWER UPRATE – RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION IDENTIFIED DURING AUDIT OF THE NON-LOSS OF COOLANT ACCIDENT SAFETY ANALYSES CALCULATIONS

Affidavit to Withhold from Public Disclosure
Proprietary Information
Under 10 CFR 2.390

(Cover page plus 7 pages)

Westinghouse

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Proj letter: FPL-12-100

CAW-12-3447

March 23, 2012

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Responses to NRC's Information Request Regarding the St. Lucie Unit 2 Extended Power Uprate Non-LOCA Transient Analyses Audit (Proprietary)

The proprietary information in the subject audit question responses for which withholding is being requested is further identified in Affidavit CAW-12-3447 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. Specifically, the proprietary information is contained in the response to the question, "Reactor Coolant Pump (RCP) Rotor Seizure/Shaft Break and Control Element (CEA) Withdrawal from Subcritical." The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Florida Power and Light

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference CAW-12-3447, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

for J. A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

STATE OF CONNECTICUT:

ss WINDSOR LOCKS

COUNTY OF HARTFORD:

Before me, the undersigned authority, personally appeared C. M. Molnar, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

C. M. Molnar, Senior Engineer
Regulatory Compliance

Sworn to and subscribed before me
this 23rd day of MARCH 2012

Notary Public

Subscribed and Sworn to before me, a Notary Public, in and for County of Hartford and State of Connecticut, this 23rd day of MARCH 2012.

JOAN GRAY
Notary Public
My Commission Expires January 31, 2017

(1) I am Senior Engineer, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in the response to the NRC's question, "Reactor Coolant Pump (RCP) Rotor Seizure/Shaft Break and Control Element (CEA) Withdrawal from Subcritical" (Proprietary), asked during the NRC's non-LOCA transient analysis audit of the St. Lucie Unit 2 Extended Power Uprate license amendment request, for submittal to the Commission, being transmitted by Florida Power and Light letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the St. Lucie Unit 2 Extended Power Uprate license amendment application and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Support Florida Power and Light in obtaining approval of the St. Lucie Unit 2 Extended Power Uprate license amendment request.

Further this information has substantial commercial value as follows:

(a) The information reveals available margins under Extended Power Uprate conditions and, therefore, would enhance a competitor's ability to provide future analytical services to Florida Power and Light.

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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