FPL

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

January 18, 2012

L-2011-441
10 CFR 50.90

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 NRC Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information Regarding 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) Email from T. Orf (NRC) to C. Wasik (FPL), St. Lucie 2 EPU - draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB), September 6, 2011.

(3) Email from L. Abbott (FPL) to T. Orf (NRC), RE: St. Lucie 2 EPU - draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) – Question numbering, September 28, 2011.

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
NRR

In an email dated September 6, 2011 from NRC (T. Orf) to FPL (C. Wasik), Subject: St. Lucie 2 EPU - draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) [Reference 2], the NRC staff requested additional information regarding FPL's license amendment request (LAR) to implement the EPU. FPL email dated September 28, 2011 from FPL (L. Abbott) to NRC (T. Orf), Subject: RE: St. Lucie 2 EPU - draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) – Question numbering [Reference 3], provided specific numbers (SXRB-01 through SRXB-102) for the questions included in the September 6, 2011 email. The Attachment to this letter provides the FPL responses to RAI questions SRXB-01 through SRXB-08 related to steam generator margin to overfill and steam generator tube rupture.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2011-021 [Reference 1].

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

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 18-January-2012

Very truly yours,

Richard L. Anderson

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachment

cc: Mr. William Passetti, Florida Department of Health

Response to NRC Reactor Systems Branch Request for Additional Information

The following information is provided by Florida Power & Light 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 Nuclear Plant Unit 2 that was submitted to the NRC by FPL via letter (L-2011-021) dated February 25, 2011 (Accession Number ML110730116).

In an email dated September 6, 2011 from NRC (T. Orf) to FPL (C. Wasik), Subject: St. Lucie 2 EPU - draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB), the NRC staff requested additional information regarding FPL's license amendment request (LAR) to implement the EPU. FPL email dated September 28, 2011 from FPL (L. Abbott) to NRC (T. Orf), Subject: RE: St. Lucie 2 EPU - draft RAIs Reactor Systems Branch and Nuclear Performance Branch (SRXB and SNPB) – Question numbering, provided specific numbers (SXRB-01 through SRXB-102) for the questions included in the September 6, 2011 email. This attachment documents the FPL responses to RAI questions SRXB-01 through SRXB-08 related to steam generator margin to overfill (SG MTO) and steam generator tube rupture (SGTR).

RAI 2.8.5.6.2-1

Section 2.8.5.6.2 discusses the analysis of the steam generator tube rupture (STGR) event. However, it does not include an analysis to validate an assumption that the SG margin-to-overfill (MTO) exits. In addition, although the analysis of mass releases is discussed, additional information is needed as follows.

SRXB-01

Perform an analysis to show that the SG MTO exists. Confirm that the computer code and methods used in the analysis are approved by the NRC, and address compliance with restrictions and conditions specified in the NRC safety evaluation report approving the computer code and methods. The requested information should include:

Response

The results of the analysis provided below demonstrate that SG MTO exists for St. Lucie Unit 2.

The sequence of events is presented in Table 1. Figure 1 shows the SG liquid volume as a function of time. The maximum SG liquid inventory is calculated to be 6631 ft^3, which is below the SG overfill limit of ~7984 ft^3, providing a margin of greater than 1,200 ft^3 to SG overfill.

Figure 1 - Total SG Liquid Volume vs. Time

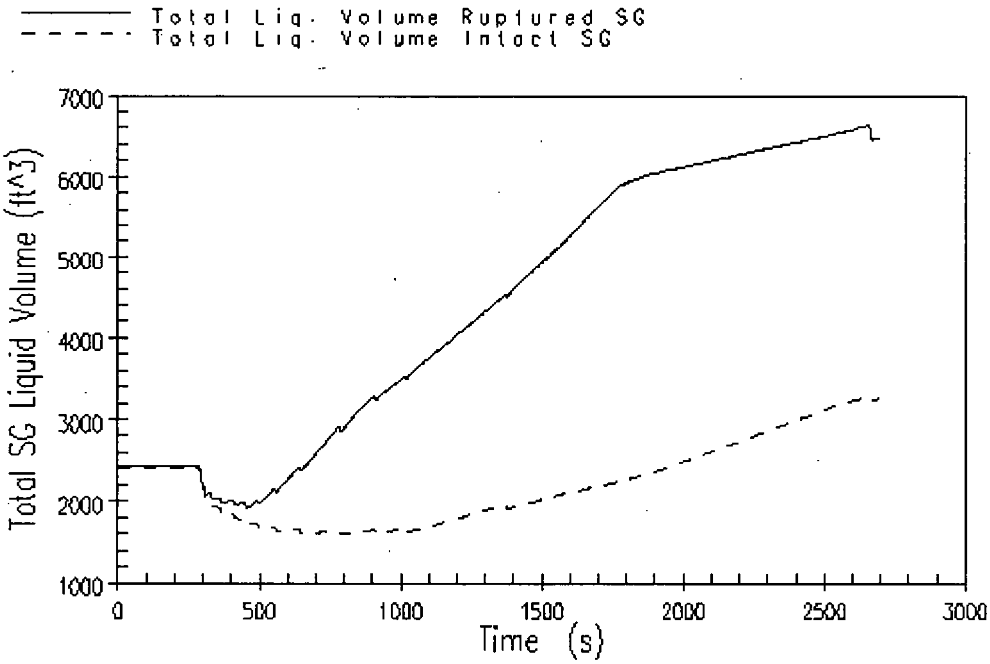

Table 1 - Sequence of Events

Time (sec)	Event	Analysis Setpoint
0.1	Tube Rupture	---
282.2	TMLP Low Pressure Setpoint Reached	2122 psia
283.0	TMLP Signal for Reactor Trip	2122 psia
283.4	Reactor Trip Signal	---
284.1	CEA Release (0.74 sec delayed)	---
285.1	Main Steam Safety Valves Open	970.0 psia
462.0	AFW Flow Initiated (Intact SG 120 sec Delayed)	SG Narrow Range (NR) Level 25.5%
464.3	AFW Flow Initiated (Ruptured SG 120 sec Delayed)	SG NR Level 25.5%
514.1	Pressurizer Empties	---
571.8	SIAS on Low Pressurizer Pressure	1578 psia*
871.3	AFAS Reset Setpoint Reached & TDAFW Continues	39.1% NR
871.6	SI Flow Begins (RCS pressure below shutoff)	1454.8 psia (shutoff)
1771.3	TDAFW Flow Terminated (900 sec delayed)	---
2660.9	Maximum SG Level Loop 1	---
2700.0	Operator Action Begins	---

* The analysis is insensitive to this value since the pressure drops to below the SI shut-off head much later, and an earlier SIAS would not initiate HPSI earlier.

The SG MTO analysis utilizes the safety analysis code RETRAN. The Nuclear Regulatory Commission's Safety Evaluation Report for the RETRAN Topical Report WCAP-14882 states that RETRAN is qualified to analyze the SGTR MTO event.

The Westinghouse standard methodology for a Steam Generator Tube Rupture (SGTR) Margin to Overfill (MTO) analysis is documented in WCAP-10698-P-A. This document was evaluated and deemed to be conservative with respect to SG MTO via NRC letter (Mr. Charles E. Rossi) to Westinghouse Owner's Group (Mr. Alan E. Ladieu) dated March 30, 1987 (contained in WCAP-10698-P-A). The analysis for St. Lucie Unit 2 follows this methodology, with the following conservatisms:

1. The turbine-driven auxiliary feedwater (TDAFW) flow to the ruptured steam generator is terminated 15 minutes after the steam generator level reaches the Auxiliary Feedwater Actuation Signal (AFAS) reset setpoint. The St. Lucie Unit 2 AFW design includes two safety related, redundant series isolation valves in the AFW pump discharge lines for each of the three AFW pumps. The safety related AFAS design employs four channels and utilizes a 2-out-of-3 channel logic initiation signal. When the SG level rises to the AFAS reset level, the redundant isolation valves close, diverting AFW flow to the condensate storage tank (CST). The two isolation valves for each AFW pump are powered from separate and diverse safety related power supplies, providing protection for a single failure of a power supply. By conservatively delaying the isolation of AFW to the ruptured steam generator for 15 minutes beyond the time of AFAS reset, MTO is minimized.

2. AFW flow to the ruptured steam generator was assumed to be equal to the total flow supplied by all (1 motor-driven and 1 turbine-driven) AFW pumps. AFW flow to the intact steam generator was assumed to be from one motor-driven AFW pump. Maximizing AFW flow to the ruptured steam generator will maximize SG inventory. Minimizing AFW flow to the intact steam generator minimizes cooldown, which keeps break flow rate high, thus minimizing MTO.

The SG MTO analysis does not credit any operator actions typically performed during a SGTR event. If credited, several of the typical operator actions would provide greater margin to overfill. These include:

1. <u>Initiation of a cooldown using SG Atmospheric Dump Valves (ADVs).</u> Following the isolation of feedwater flow, the operator would typically initiate a plant cooldown by dumping steam through the ADVs. A cooldown of the Reactor Coolant System (RCS) presents a benefit to SG MTO as the cooldown of the RCS combined with a depressurization will bring the primary and secondary sides to a temperature and pressure equilibrium, essentially isolating any SG tube break flow. In the analysis, no cooldown of the RCS via operator action is credited in the first 45 minutes, and as such, break flow remains high throughout the event. This break flow, combined with continuing AFW flow, conservatively limits SG MTO.

2. Initiation of depressurization with Pressurizer Power Operated Relief Valve (PORV). During the cooldown of the RCS, the operator would typically depressurize the RCS to within 50 psi of the ruptured SG pressure. This minimizes the break flow. In the analysis, no operator action to depressurize the RCS is credited in the first 45 minutes, and as such, break flow remains conservatively high throughout the event. This continuing break flow, combined with maximum AFW flow, conservatively limits SG MTO.

Through the discussion above, the methodology used for the St. Lucie Unit 2 SG MTO analysis is significantly more conservative than the typical Westinghouse methodology used for the SG MTO analysis provided in WCAP-10698-P-A.

SRXB-02

1. List the nominal values with measurement uncertainties and the corresponding values for key plant parameters used in the SG MTO analysis. This discussion should include rationale to show that the value of each of the key parameters used in the SG MTO analysis is conservative, resulting in a minimum SG MTO. In addition, provide a basis for the target cooldown temperature used in the analysis.

Response

Table 2 presents the SG MTO analysis parameter values. With respect to the basis for the target cooldown temperature, as discussed above, no credit is taken for RCS cooldown in the St. Lucie Unit 2 SG MTO analysis. This is conservative as it maintains higher break flow for the full duration of the event. Table 4 displays information on the conservatism for each analysis parameter.

Table 2 – Key Parameters used in the SG MTO Analysis

Parameter	Analysis Value Used
Initial Power Level	3050 MWt (100% RTP + 0.3% uncertainty + 20 MWt pump heat)
Initial RCS Pressure	2395 psia (Maximum of 2350 psia + 45 psia uncertainty)
Initial SG Pressure	Corresponds to initial SG level, maximum SG tubes plugged, and thermal design flow.
Initial SG Water Inventory	Set to a value corresponding to the initial nominal SG level of 60% NR (65% - 5% uncertainty).
Total Available SG Volume (per SG)	7984.80 ft^3
Initial Pressurizer Water Volume	71% Span (high level alarm setpoint including uncertainty)
SI Actuation Pressure Setpoint	1578 psia (HPSI pump head is 1452 psia)
SI Flow vs. RCS Pressure	See Table 3 (maximum SI flow)
SI Pump Delay Time	0.0 Seconds
AFW Actuation Setpoint	25.5% NR (20.5% NR nominal + 5% uncertainty)
AFW Delay Time	120 seconds
AFW Temperature	40°F
Auxiliary Feedwater Actuation Signal (AFAS) Reset Setpoint	35% NR (30% + 5% uncertainty) Note: The liquid volume value corresponding to 35% NR is calculated with assumption that all the water in the SG is collapsed liquid and contains no bubbles. This is a conservative assumption to minimize MTO.
Initial AFW Flow Rate (per SG)	Intact SG – 325 gpm Ruptured SG – 860 gpm At 35% NR, the ruptured SG TDAFW flow is conservatively continued for 15 minutes beyond the AFAS reset level, sending maximum AFW flow to the ruptured SG.
MSSV Opening Setpoints	970.0 psia (4 valves) and 1008.8 psia (4 valves). Maximum 10% blowdown also modeled.
MSSV Flow Rates (per valve)	744,210 lbm/hr at 1000 psia and 774,000 lbm/hr at 1040 psia
Time of Loss of Offsite Power	Concurrent with Reactor Trip
Atmospheric Dump Valve (ADV)	No ADV operation is credited in the analysis.
Decay Heat Model	ANSI 1979 +2σ Model
Initial Value of Decay Heat	7.99%

Table 3 – Maximum Safety Injection Flow vs. RCS Pressure

RCS Pressure (psia)	SI Flow (lbm/s)
0	394.50
146.8	263.55
157.1	248.05
166.5	232.54
175.2	217.06
183.2	201.58
190.5	186.10
197.4	170.65
203.8	155.19
209.8	139.74
215.3	124.29
220.3	108.85
224.9	93.42
228.9	77.99
232.2	62.58
233.7	54.87
234.9	47.18
383.3	44.19
592.3	40.16
773.6	36.15
929.8	32.13
1063.1	28.12
1175.4	24.10
1268.1	20.09
1342.1	16.07
1397.7	12.06
1434.7	8.03
1451.8	4.02
1454.8	0.0

Table 4 – Conservatisms for Key SG MTO Analysis Parameters

Parameter	Conservatism
Initial Power Level	Highest power plus uncertainties is conservative to maximize initial energy and therefore maximize break flow.
Initial RCS Pressure	Highest possible initial pressure is conservative to maximize break flow.
Initial SG Pressure	The initial SG pressure is set to the conservative value corresponding to the highest SG tube plugging level and largest tube fouling.
Initial SG Water Inventory	Lowest initial level is conservative, resulting in the earliest AFW actuation.
Initial Pressurizer Water Volume	A higher initial pressurizer level will result in maintaining RCS pressure and higher break flow, thus limiting MTO.
SI Actuation Pressure Setpoint	Earliest possible SI actuation is conservative as it maintains RCS pressure and inventory which maximizes break flow.
SI Flow vs. RCS Pressure	Set to the highest possible SI flow to maximize break flow, thus limiting MTO.
SI Pump Delay Time	No delay is conservative as it provides SI flow instantaneously, maintaining RCS pressure which maximizes break flow.
AFW Actuation Setpoint	AFW is modeled such that it initiates earlier in the event. Earlier AFW initiation time is conservative as it decreases the SG pressure which increases break flow, resulting in increased inventory in the ruptured SG, thus limiting MTO.
AFW Delay Time	A minimum delay time provides AFW sooner, which increases inventory, thus limiting MTO.
AFW Temperature	Minimum AFW temperature is conservative as is drives down secondary pressure thus creating a higher RCS to main steam system pressure differential and higher break flow. Also, the higher the density, more mass remains in the SG.
AFW Flow Rate (per SG)	Ruptured SG: Maximum AFW flow maximizes inventory, thus limiting MTO. Intact SG: Minimum AFW flow minimizes cooldown ability which maintains higher break flow.
MSSV Opening Setpoints	The MSSVs are set to minimum opening setpoints and maximum blowdown (10%). This is conservative as it keeps secondary pressure lower, thus creating a higher RCS to main steam system pressure differential and higher break flow.
MSSV Flow Rates	The MSSV flow rates are set to default values.
Time of Loss of Offsite Power (LOOP)	A LOOP concurrent with reactor trip is conservative as it will maintain a higher RCS pressure, increasing break flow, thus limiting MTO.
ADV Setpoint	Not modeled.
Decay Heat Model	The $+2\sigma$ model is conservative as higher decay heat levels will help maintain high RCS pressure.
Initial Value of Decay Heat	See Table 2

SRXB-03

2. **Identify operator actions and associated action times credited in the analysis. Where an operator action is credited, confirm that such action is consistent with station procedure and action times are conservative, resulting in a minimum SG MTO. Discuss the acceptance criteria for the results of simulator exercises in support of operator action times credited in the analysis.**

Response

There are no operator actions credited in the St. Lucie Unit 2 SG MTO analysis.

SRXB-04

3. **Under the assumed loss-of-offsite-power (LOOP) conditions, address the functionality of each ADV. Discuss what, if any, mitigating function of the ADV provides, and its capability to perform that function under assumed LOOP conditions. If the valve's actuation must be manual, provide information to show that the operator is capable of actuating the valve within the analytical assumed time.**

Response

St. Lucie Unit 2 has two safety related ADVs per steam line, which shall be in manual operation above 15% power. Each ADV is dual powered by independent sources of safety grade onsite AC and DC electric power. The ADVs are capable of automatic modulating service using AC power and are capable of open/close service from the control room using DC power. ADV operation is powered by emergency diesel generators in the event of a LOOP. Each ADV has a dedicated safety related motor-operated block valve, which may be used to isolate the ADV if it is stuck open. In the SG MTO analysis, no credit is taken for the ADVs for the first 45 minutes. Crediting the ADVs in the SG MTO analysis for RCS cooldown and depressurization would provide additional margin to overfill by reducing the break flow and providing additional steaming during the cooldown, which would reduce the overall SG inventory.

SRXB-05

4. One of key parameters that will significantly affect the results of the SG MTO analysis during an SGTR event is the initial SG water level, which is a function of the initial power level. The MTO analysis should consider the effects of the initial SG water levels corresponding to power levels that capture 95 percent of the operating time during a fuel cycle. Also, for the range of the power levels that envelop 95 percent of the operating time, provide trending data for the corresponding SG water level to show that conservative initial SG water level (with inclusion of measurement uncertainty, plant conditions perturbation, and SG mass addition due to the turbine runback) have been used in the MTO analysis.

Response

The nominal SG level of 65% Narrow Range (NR) minus the level uncertainty of 5% is used. This reduces the delta between SG level and the AFAS setpoint, thus maximizing the AFW flow provided to the ruptured steam generator. The SG MTO analysis was analyzed at full power. Analyzing at full power maximizes break flow into the ruptured SG, and initializing at 60% NR SG level begins delivery of AFW into the ruptured SG earlier. These parameters, along with other inputs used in the SG MTO analysis, result in a limiting MTO of greater than 1,200 ft^3.

Since the initiation of AFW flow and the subsequent AFAS reset is based on fixed SG levels, as specified in Table 2 above, the initial SG mass plays no significant role in determining the minimum margin to overfill. Using a higher initial mass would delay the AFW flow initiation; however, AFW flow initiation would be at the same SG level setpoint, irrespective of the initial mass. Therefore, the power level range associated with 95% of the operating time is not relevant and was not used in the SG MTO analysis.

<u>SRXB-06</u>

5. List systems, components and instruments that are credited in for consequence mitigation of the SGTR MTO analysis in accordance with the St Lucie 2 SGTR emergency operating procedures (EOP). Discuss whether each system and component specified is safety grade.

For SG ADVs and control valves, specify the valves motive power and discuss if the motive power and valve controls are safety grade. For non-safety grade systems and components, discuss whether safety grade backups are available which can be expected to function or provide the desired information within a time frame compatible with prevention of SGTR overfill or justify that non-safety grade systems and components can be used for the SGTR overfill analysis. Provide a list of all radiation monitors that could be used for identification of the SGTR event and the affected SG, and specify the quality and reliability of the instrumentation, as appropriate. If the EOP specifies SG sampling as a means of affected SG identification, provide the expected time period for obtaining the sample results and discuss the effects on the duration of the SGTR event.

<u>Response</u>

Systems, components and instruments credited in the SG MTO analysis are discussed in the above RAI responses. Specifically, the response to RAI SRXB-01 discusses the safety grade design features credited for isolation of AFW flow to the ruptured SG upon an increasing SG level reaching the AFAS reset setpoint. Although not credited in the SG MTO analysis, the response to RAI SRXB-04 discusses the safety grade design of the ADVs and associated power sources. No additional systems, components or instruments, including radiation monitors and SG sampling, were credited for consequence mitigation in the SG MTO analysis.

<u>SRXB-07</u>

6. List the single failure events considered in the SGTR overfill analysis and identify the worst single failure used in the analysis that resulted in a minimum MTO.

7. Update the licensing basis radiological release analysis to reflect radiological consequences of the above identified limiting release.

<u>Response</u>

6. As a result of the safety grade design features credited for event mitigation as discussed in the responses to RAIs SRXB-01, 04 and 06, there is no single failure that further minimizes SG MTO. Conservative analysis assumptions were used in an effort to minimize MTO. Analysis results demonstrate that there is sufficient margin to overfill.

7. The primary objective of the SG MTO analysis is a SG overfill condition, which was evaluated based on conservative accumulation of SG inventory using conservative AFW flow. Bottling-up the SG to accumulate as much inventory as possible results in steam release rates less than those recorded in the SGTR dose analysis while also diluting the concentration of activity in the SG. Consequently, the dose impact from the SG MTO

analysis event would be less severe. The most adverse dose conditions are evaluated by the SGTR dose case, which remains bounding with respect to dose considerations.

SGTR radiological releases were evaluated for EPU conditions. The results of the evaluation, presented in letter R. L. Anderson (FPL) to NRC Document Control Desk, L-2011-467, "Response to NRC Accident Dose Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request," November 14, 2011, (ML11320A286) demonstrate that at EPU conditions, the radiological consequences remain acceptable.

SRXB-08

This RAI is also applicable to the analysis of the mass releases during an SGTR event. Provide the applicable requested information discussed above to show that (1) acceptable computer code and methods are used, (2) numerical values with consideration of the uncertainties and fluctuations around the nominal values are conservative, resulting in a maximum mass release, (3) the worst single failure is considered, (4) the operator actions credited in the analysis are consistent with SL2 EOP, (5) the operator action times are adequately validated by simulator exercises, (6) safety grade equipment is used for consequence mitigation, and (7) non-safety systems can be used in the SGTR analysis.

Response

The analysis of the SGTR mass release is discussed following the response to Items 1-7 below.

(1) Acceptable computer code and methods are used.

The SGTR analysis utilizes the safety analysis code RETRAN. The NRC's Safety Evaluation Report for the RETRAN Topical Report WCAP-14882 states that RETRAN is qualified to analyze the SGTR event. The methods discussed in Updated Final Safety Analysis Report (UFSAR) Section 15.6.3 were maintained for the SGTR analysis.

(2) Numerical values with consideration of the uncertainties and fluctuations around the nominal values are conservative, resulting in a maximum mass release.

Steam Generator Tube Rupture (SGTR)
Initial Conditions and Assumptions

Parameter	Condition/Assumption
Power	Maximum Nuclear Steam Supply System (NSSS) Power Level Including Pump Heat + Uncertainties (retains a higher reactor coolant system (RCS) temperature and pressure to maximize the break flow and flashing fraction)
RCS Average Temperature	Maximum Nominal + Uncertainty
Pressurizer Pressure	Highest Pressurizer Pressure + Uncertainties
RCS Flow	Thermal Design Flow (TDF)
Pressurizer Level	Maximum Level + Uncertainty (maintains higher RCS pressure)
Steam Generator (SG)	Nominal Level - Uncertainty (maintains higher steam release rate)
Main Steam Safety Valves	Minimum Tolerance and Account for 10% Blowdown (maximizes steam release rate)
SG Tube Plugging	Maximum Level (reduced initial SG pressure to maximize break flow)
Physics Parameters	The SGTR event results in a small variation in the primary temperature during the period of primary coolant release and a steady decrease in primary pressure. This results in a small decrease in primary fluid density that does not have an impact on the initial physics parameters. Therefore kinetics parameters associated with minimum feedback were selected.
SGTR Location and Break Size	A double-ended rupture of a single SG tube at the cold end of the shortest tube such that the mass flow rate from the break is maximized. However, the enthalpy of the break flow is based on the hot leg, and therefore, the flashing of the break flow is conservatively modeled.
Loss-of-Offsite Power	At Time of Reactor Trip
Operator Actions	During the first 45 minutes of SGTR event, no operator actions are assumed.
High Pressure Safety Injection (HPSI)	Flow from two (2) HPSI pumps is used to maximize RCS pressure and hence the break flow.
Main Feedwater Flow	A loss of offsite power is assumed to occur at reactor trip, which results in the main feedwater pumps tripping and coasting down. The feedwater flow rate is assumed to rapidly decrease to zero minimizing the SG inventory. The lower inventory is conservative, as it increases the flashing fraction and the amount of the steam release to the atmosphere
Auxiliary feedwater (AFW)	Automatic initiation of one of the AFW pumps on low SG level is required for the 45-minute case to prevent SG dry out on the intact loop.

Subsequent to the isolation of the ruptured SG at 45 minutes, the steam releases resulting from the cooldown of the plant to a RCS temperature of 212°F were calculated, from a heat balance, for various cooldown rates (20°F/hr to 100°F/hr). These steam releases were subsequently used in the dose analyses.

(3) The worst single failure is considered.

The single failure assumption for the SGTR event with the atmospheric dump valves (ADVs) in manual mode, as described in EPU LAR Attachment 5, Section 2.8.5.6.2, is the failure of one train of the reactor protection system (RPS). The protection function is carried out by the other train of the RPS, which remains functional during the event. For impact of operator action using ADVs, see response below for Item 4.

(4) Operator actions credited in the analysis are consistent with SL2 EOP.

The Emergency Operating Procedure (EOP) for the SGTR event specifies the use of ADVs on both steam generators (SGs) to cool down the RCS. The RCS depressurization is coincident (or slightly delayed) with RCS cooldown which allows for the termination of break flow and the isolation of the ruptured SG. These EOP actions result in cooling the RCS to 510°F while depressurizing the RCS to within 50 psi of the SG pressure (while maintaining the subcooling requirement). These actions result in a significant reduction in break flow and consequently the dose consequences.

The analysis performed for the EPU did not model the ADVs during the first 45 minutes of the SGTR event. St. Lucie Unit 2 has two safety related ADVs per steam line, which are in manual operation above 15% power. Each ADV is dual powered by independent sources of safety grade onsite AC and DC electric power. The ADVs are capable of automatic operation from the control room. ADV operation is powered by emergency diesel generators (EDGs) in the event of a LOOP. Each ADV has a dedicated safety related motor-operated block valve, which may be used to isolate the ADV if it is stuck open. A stuck open ADV can thus be isolated in a short time.

To assess the impact of a stuck open ADV, supplemental analyses were performed with an ADV on the ruptured and the intact steam generators kept open for 5 minutes. Since LOOP cases result in increased steam releases, these supplemental analyses considered ADV operation for the loss of offsite power (LOOP) cases with variation in the ADVs opening time within the first 45 minutes of the transient (early and late opening). Both cases analyze the ADVs in the full open position for the same time duration. The single failure for these SGTR events is a failed open ADV on the ruptured and intact SGs with isolation assumed in 5 minutes.

Opening the ADVs causes a slight depressurization of the SGs during the time ADVs are open. The analyses show that the impact on the total steam release from the ruptured SG and the break flow is not significant. Additionally, the break flow and the steam release from the ruptured SG (up to 45 minutes) from these analyses remain comparable but below those used in the EPU LAR dose analyses. Therefore, the case with respect to break flow and steam releases presented in letter R. L. Anderson (FPL) to NRC Document Control Desk, L-2011-467, "Response to NRC Accident Dose Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request," November 14, 2011, (ML11320A286) remains acceptable for dose consequences.

The SGTR analysis, for steam releases in the first 45 minutes, shows that without AFW, the level in the ruptured SG does not increase. Thus the steam release rate used in the analysis is seen to be greater than the break flow rate and no additional steaming beyond that used in the dose analysis will be needed to control the SG level.

SGTR Dose Analysis Input Data Comparison

Item	Description	Cases with ADVs Opening Time after Reactor Trip		Dose Analysis Input Values
		5 minutes (Case 1)	30 minutes (Case 2)	
1	Reactor Trip Time (sec)	284		
2	RCS Liquid Volume – Maximum (ft^3)	10,441		
3	Ruptured SG Tube Flow Prior to Trip (lb)	17,190		18,900
4	Ruptured SG Tube Flow from Trip to 45 Minutes (lb)	90,661	92,086	91,700
5	Ruptured SG Tube Flow (total) Prior to 45 Minutes (lb)	107,851	109,276	110,600
6	Steam Release from Ruptured SG via Turbine to Condenser, Prior to Trip (Total) (lb)	527,327	527,327	578,100
7	Steam Release from Intact SG via Turbine to Condenser, Prior to Trip (Total) (lb)	524,684	524,684	575,200
8	Ruptured SG Steam Release from Trip Time to 45 Minutes (lb)	116,496	126,158	147,300
9	Intact SG Steam Release from Trip Time to 45 Minutes (lb)	116,633	126,336	136,800

(5) Operator action times are adequately validated by simulator exercises.

The SGTR analysis does not credit operator actions to cooldown and depressurize the reactor coolant system (RCS) for the first 45 minutes. The auxiliary feedwater actuation system (AFAS) logic will prevent auxiliary feedwater (AFW) flow to the ruptured steam generator (SG) when the nominal SG level is above the AFAS reset setpoint. With respect to the mass release analysis, an operator action time of up to 45 minutes was analyzed. Operators were assumed to isolate the ruptured SG by closing the main steam isolation valve (MSIV) and begin cooldown using the atmospheric dump valve (ADV) on the unaffected SG. This represents an increase from the operator action time of 30 minutes currently presented in Updated Final Safety Analysis Report (UFSAR) Section 15.6.3.2. With respect to the above, the conservatisms assumed in the SGTR analysis ensures that operator delay times are adequate and acceptable.

The emergency operating procedures (EOPs) have been developed to provide direction to the operating crews during post-trip situations. They provide the means to successfully deal with operating conditions while protecting the health and safety of the public. 2-EOP-4, “Steam

Generator Tube Rupture," provides the actions to mitigate the effects of a SGTR, initiate plant cooldown, isolate and control the ruptured SG, and place the plant in shutdown cooling.

2-EOP-4 directs operators to isolate the most affected SG and refers to 2-EOP-99, Appendix R, "Steam Generator Isolation." Appendix R directs operators to isolate AFW flow by ensuring AFW pump discharge isolation valves are closed and that the steam supply to the turbine driven (TD) AFW pump is isolated. These valves can be operated from the control room. In addition to isolating the most affected SG, the EOP Safety Function Status Check Sheet requires operators to monitor SG levels and to maintain unisolated SG levels between 60 and 70% narrow range (NR) span. The Safety Function Status Check Sheet is required to be completed every 15 minutes and is commenced in Step 1 of 2-EOP-4, which is identified as a continuous step.

Once the ruptured SG is isolated, 2-EOP-4 directs operators to maintain level in the isolated SG at less than 90% NR. The EOP provides the following methods to maintain SG level:

- Lowering reactor coolant system (RCS) pressure to below isolated SG pressure (identified as the most preferred method),
- Blowing down the isolated SG to a monitor storage tank,
- Steaming the isolated SG to the condenser, and
- Steaming the isolated SG to atmosphere (identified as the least preferred method).

Isolation of a ruptured SG is included in Licensed Operator Continuing Training and is identified on St. Lucie's INPO-accredited Licensed Operator Continuing Training program Task List at a frequency of every two years. The SGTR event scenario is included as a simulator training exercise and isolation of a ruptured SG is accomplished by implementing 2-EOP-99, Appendix R. Every licensed operator participates in this exercise during the course of the two-year Licensed Operator Continuing Training program. Additionally, there is a critical task contained in the St. Lucie Training Department's Simulator Evaluation Guides and Training Department Guidelines require satisfactory completion of critical tasks in order to receive a satisfactory grade during a simulator evaluation.

(6) Safety grade equipment is used for consequence mitigation.

The following safety grade equipment is assumed to function during a SGTR event:

- Thermal margin low pressurizer pressure reactor trip,
- High pressure safety injection initiation,
- Opening of the SG main steam safety valves,
- AFW initiation, and
- Atmospheric dump valves (ADVs) for reactor coolant system cooldown.

(7) Non-safety systems can be used in the SGTR analysis.

No non-safety systems were credited as part of this analysis.

Analysis of Mass Releases During a Steam Generator Tube Rupture (SGTR) Event

Below are tables showing the steam releases for the cooldown from 45 minutes to 8 hours (300°F) over various cooldown rates. A cooldown of rate of 20°F/hr is used from 300°F to 212°F. A larger cooldown rate is more adverse early on in the cooldown (i.e., 45 minutes to 2 hours). However, as the time elapses, a lower cooldown rate extends the cooldown and results in a higher total steam release.

Steam Releases for St. Lucie Unit 2 EPU SGTR Dose Analysis

Steam releases from 45 minutes to 2 hours
for varying cooldown rates:

Initial Cooldown Rate	Steam Release over Period
20°F/hr	327,500 lb
25°F/hr	338,600 lb
30°F/hr	349,500 lb
38°F/hr	366,800 lb
75°F/hr	443,200 lb
100°F/hr	492,200 lb

Steam releases from 2 hours to 8 hours to
shutdown cooling (SDC) entry (300°F)
for varying cooldown rates:

Based on Initial Cooldown Rate	Steam Release over Period
20°F/hr	1,134,000 lb
25°F/hr	1,119,300 lb
30°F/hr	1,104,400 lb
38°F/hr	1,081,200 lb
75°F/hr	977,700 lb
100°F/hr	911,300 lb

The SGTR dose consequences evaluation was performed with conservative steam releases to ensure worst dose consequences. All steam mass releases generated by the thermal-hydraulic event analyses, as presented above, were augmented by approximately 10% for conservatism prior to the use in the dose analysis. The steam mass release totals from the ruptured and the intact SGs were used to provide the pre-trip steam release rates for the dose analysis.

From reactor trip to isolation of the ruptured SG (conservatively assumed to be 45 minutes), the total steam mass releases were used with the duration of this time interval to determine steam release rates.

From 45 minutes to 2 hours, the thermal-hydraulic event results from the 100°F/hr cooldown rate were conservatively used to generate the mass release rate. This maximum cooldown rate maximizes the steam released during the early part of the event, which is conservative for the limiting 2-hour exclusion area boundary (EAB) dose. This cooldown rate was used to determine the RCS temperature at 2 hours.

From 2 hours to 8 hours, the RCS is cooled to RHR entry conditions of 300°F. The cooldown rate was determined from the time and temperature differences to be 18.3°F, and thus the mass release from the cooldown rate of 20°F/hr was used for this interval.

The steam releases from 8 hours to RCS conditions of 212°F were determined using 20°F/hr cooldown rate, as the slowest cooldown rate was determined to provide conservative dose consequences.

In summary, conservatively long times were used between dose analysis event break points (45 minutes for operator action to isolate, 8 hours to reach SDC conditions), and appropriate mass release rates for these intervals were determined. For the last interval, the slowest cooldown rate was chosen for dose analysis, based on a sensitivity study which determined that this choice would maximize the predicted dose. In all cases, the thermal-hydraulic event predictions were augmented by approximately 10% for conservatism before use in the dose evaluation.

Consideration of an Integrated Break in the Dose Calculations and Its Effect on the Determination of the Worst Dose Release Case

The integrated break flow is instigated by maximizing primary-to-secondary break flow. Sensitivity cases were performed to determine the maximum break flow. Parameters included maintaining a higher RCS pressure as well as a lower SG level.

Maximizing the integrated break flow also maximizes the transport of RCS mass (with its higher radionuclide inventory, relative to the secondary side coolant) into both the affected and intact SGs. In addition to the thermal-hydraulic event break flow mass transport, the dose analysis modeled the additional flow from normal steam generator tube leakage. The normal leakage transport was modeled based on the RCS and secondary conditions from the thermal-hydraulic event, and was continued from event initiation to the time when the RCS conditions reached 212°F.

Using the RCS and secondary thermal hydraulic conditions, an enthalpy based flashing fraction of the primary to secondary break flows was developed for the applicable time periods. The total leakage flow was then separated into the unflashed flow and the flashed flow. Flashed flow is released to the environment immediately, without scrubbing. Unflashed flow is transferred to the SG liquid volume, and is then released to the environment at the steaming rate (with credit for SG or condenser scrubbing, if applicable) of the respective SGs.

A combination of this break flow with the steam releases specified above produced the worst dose consequences.