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

FPL

November 29, 2011

L-2011-473
10 CFR 50.90

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

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

Response to NRC Reactor Systems 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-2010-259), "License Amendment Request (LAR) for Extended Power Uprate," November 22, 2010, Accession No. ML103560419.

(2) T. Orf (NRC) email to C. Wasik (FPL), "St. Lucie 1 EPU draft RAI -- Reactor Systems (SRXB)," September 23, 2011.

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

In an email from the NRC Project Manager dated September 23, 2011 [Reference 2], additional information related to reactor systems was requested by the NRC staff in the Reactor Systems Branch (SRXB) to support their review of the St. Lucie Unit 1 EPU License Amendment Request (LAR). Reference 2 contains six RAIs numbered SRXB-1 through SRXB-6. Since these RAI numbers have been assigned previously, FPL renumbered SRXB-1 through SRXB-6 as SRXB-52 through SRXB-57 respectively. FPL's response to these RAIs is presented in the attachment to this letter.

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

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

A001
NRR

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

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate License Amendment Request (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 29-November-2011

Very truly yours,

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

In an email dated September 23, 2011 from T. Orf (NRC) to C. Wasik (FPL), "St. Lucie 1 EPU draft RAI – Reactor Systems (SRXB)," the NRC staff requested additional information regarding FPL's request to implement the EPU. The RAI consisted of six questions from the NRC Reactor Systems Branch. As presented in the NRC email, these six RAIs are numbered SRXB-1 through SRXB-6. Since these RAI numbers have been assigned previously, FPL has renumbered SRXB-1 through SRXB-6 as SRXB-52 through SRXB-57 respectively. The responses are provided below.

SRXB-52

Page 8 indicates for the MTO analysis that the operator action delay times are 28.7 minutes and 45 minutes to terminate the AFW flow and break flow to the ruptured SG, respectively. Also, page 44 indicates for the mass release analysis that the operator action delay times are 45 minutes to isolate the ruptured SG, and 45 minutes to locally open the ADVs or to restore instrument air to ADVs.

Discuss administrative control procedures, training program, and acceptance criteria for the results of the plant simulator exercises to ensure that the above operator action delay times assumed in the SGTR analysis are adequate and acceptable. Also, confirm that the equipment and systems for operator actions are safety grade systems.

Response

The analysis discussed in Reference 1 does not credit operator actions to cooldown and depressurize the reactor coolant system (RCS) to equilibrate RCS and ruptured steam generator (SG) pressures and terminate break flow. The only operator action directly accounted for in the analysis is termination of turbine-driven (TD) auxiliary feedwater (AFW) flow to the ruptured SG following auxiliary feedwater actuation system (AFAS) reset when the ruptured SG narrow range (NR) level reaches 35%, plus a 15 minute delay time for operator action. The AFAS reset logic will prevent AFW flow to the ruptured SG when the nominal SG level is above 29% narrow range (NR). The value used in the analysis is 35% NR. By accounting for the full calculated break flow and AFW flow for 15 minutes after the AFAS reset setpoint (35% NR) is reached during the transient, the MTO analysis discussed herein is more conservative than an analysis that would model all operator actions in conjunction with St. Lucie Unit 1 Emergency Operating Procedures (EOPs). Note that the 15 minute delay time after reaching the 35% NR corresponds to 28.7 minutes in the analysis performed for MTO. With respect to the mass release analysis; an operator action time of up to 45 minutes was analyzed. Operators were assumed to isolate the affected 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 UFSAR Section 15.4.4.5.4. 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. EOP-4, "Steam Generator Tube Rupture," provides the actions to mitigate the effects of a SGTR, initiate plant cooldown, isolate and control the affected SG, and place the plant in shutdown cooling.

EOP-4 directs operators to isolate the most affected SG and refers to 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 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% NR. The Safety Function Status Check Sheet is required to be completed every 15 minutes and is commenced in Step 1 of EOP-4, which is identified as a continuous step. This 15-minute cycle is the basis for the operator action time assumed in the Reference 1 analysis and is discussed further in the response to SRXB-57.

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

- Lowering 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 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.

Most of the systems associated with the operator actions identified in this RAI are safety grade. The AFW system and the main steam system steam supply to the TD AFW pump are safety grade systems. The instrument air system serves no safety function since it is not required to achieve safe shutdown or to mitigate the consequences of a design basis event. In the MTO analysis it is assumed that ADVs can be operated for RCS cooldown within 45 minutes, either from the control room if instrument air is available, or locally if instrument air is not available. Although ADVs are not part of a safety grade system, EOPs have enough guidance to require that operator action be taken to manually open the ADV locally as required, while parallel activities are being performed to restore instrument air in the field. The use of ADVs is acceptable in this mode of operation.

References (SRXB-52)

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-311), "Information Regarding Steam Generator Tube Rupture Steam Release and Margin to Overfill Provided in Support of the Extended Power Uprate License Amendment Request," August 18, 2011, Accession No. ML11231A946.

SRXB-53

Page 26 indicates that the initial SG water level corresponding to operation at 85 percent rated thermal power is assumed for the MTO analysis.

Provide the basis for the assumption.

Response

The steam generator (SG) margin to overfill (MTO) analysis documented in ANP-3019(P) and transmitted in Reference 1 is insensitive to the initial SG mass, as the limiting conditions are dominated by the high Auxiliary Feedwater (AFW) flow subsequent to the assumed worst single failure of flow control valve for the steam driven AFW pump flow to affected SG. This condition occurs, in the analysis, at a fixed SG level (AFAS reset setpoint) and is independent of the initial SG mass. This high AFW flow for 15 minutes from the time of reaching the AFAS reset setpoint, along with the break flow, determined the minimum MTO.

The initial SG mass is higher at lower powers. Using the power history of St. Lucie Unit 1 for each of the last two cycles, the power level corresponding to one-sided 95/95 percent probability was estimated to be greater than 85%. Although not important for the analysis as stated above, this power level was used to set the initial SG mass in the MTO analysis.

References (SRXB-53)

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-311), "Information Regarding Steam Generator Tube Rupture Steam Release and Margin to Overfill Provided in Support of the Extended Power Uprate License Amendment Request," August 18, 2011, Accession No. ML11231A946.

SRXB-54

Page 31 indicates that after the isolation of ruptured SG, the steam releases resulting from the cooldown of the plant to a RCS temperature of 212 degree F are calculated for various cooldown rates (20, 25, 30 , 38 and 100 degree/hr).

List the calculated steam releases for various cooldown rates and address their effects on the dose release calculations.

Response

The steam releases for various cooldown rates are provided below:

Steam Releases from 45 Minutes to 2 Hours

Cooldown Rate (°F/hr)	Steam Release over Period (lbm)
100	197,511

Steam Releases from 45 Minutes to 212 °F

Cooldown Rate (°F/hr)	Steam Release over Period (lbm)
20	2.89E+06
25	2.53E+06
30	2.28E+06
38	2.00 E+06
100	1.25 E+06

The steam generator tube rupture (SGTR) event steam releases were evaluated via sensitivity study for the above cooldown rates of 100°F/hr, 38°F/hr, 30°F/hr, 25°F/hr, and 20°F/hr. The most conservative cooldown rate for the limiting hot leg break was used to determine the radiological consequences reported in EPU LAR Attachment 5, Section 2.9.2, as supplemented by FPL letter L-2011-360 dated September 2, 2011 (Accession No. ML11251A159).

From event initiation to isolation of the affected steam generator (SG) (45 minutes), the cooldown rate and the total steam released were calculated by the thermal hydraulic analysis.

The steam releases from 45 minutes to 2 hours (end of first X/Q dose input interval), at a specified 100°F/hour cooldown rate, and from 45 minutes to 212°F at the various cooldown rates (100°F/hr, 38°F/hr, 30°F/hr, 25°F/hr, and 20°F/hr) were used from the above table. Higher cooldown rates reduce the total steam releases due to the shorter time to get to 212°F, whereas slower cooldown rates increase the total steam releases.

Based on the RADTRAD-NAI sensitivity studies, a combination of these releases was used to get conservative dose results. The key X/Q dose input intervals are 0-2 hours, 2-8 hours and 8-

24 hours. The average cooldown rate of 25°F/hour from 45 minutes to 8 hours is used for the 45 minutes to 2 hour interval, with a 1000lbm/min adder to account for the expected higher early release rate. Steam releases to get to 300°F in 8 hours are used for 2 to 8 hour period, and a slow 20°F/hr cooldown rate is used from 300°F at 8 hours, to the end point of steam releases (Reactor Coolant System at 212°F).

These release rates yielded conservative dose consequence results for the SGTR event.

SRXB-55

Page 35 of Table 7 lists the integrated break flow during a period from the reactor trip through 45 minutes when the ruptured SG is isolated.

Discuss how the integrated break is considered in the dose calculations and its effect on the determination of the worst dose release case.

Response

The dose results using the Reference Case analysis (Table 7) steam releases are presented in EPU LAR Attachment 5, Section 2.9.2, as supplemented by FPL letter L-2011-360 dated September 2, 2011 (Accession No. ML11251A159). In the dose analysis, the total normal maximum allowed tube leakage is added to the integrated break flow to obtain the total primary to secondary mass flow into the ruptured steam generator (SG). Using the 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.

Based on the results in Table 7, Cases A, B, & C have the integrated break flow and the ruptured SG steam releases, from trip time to 45 minutes, less than those of the Reference Case and thus remain bounded by the Reference Case dose. Cases D & E, however, have larger integrated break flow but lower steam releases from the ruptured SG. The combined effect of the break flow and steam releases for these cases on the dose was evaluated with RADTRAD-NAI models, and compared to the dose from the Reference Case. The Reference Case dose was found to remain bounding.

SRXB-56

Page 23 (table 5) indicates that for high SG water level, the operator is require by the EOP to reduce the water level within 60 to 70% NR by steaming with ADVs.

Discuss the effect of the EOP step controlling SG water level by steaming with ADVs on the dose releases calculations and worst case determination in Table 7.

Response

The steam generator tube rupture (SGTR) steam release analysis assumed the steam releases occurred through the Main Steam Safety Valves (MSSVs) which were modeled to open shortly after turbine trip on reactor trip. To maximize the steam releases from the ruptured steam generator, and consequently the offsite dose, the steam release analysis conservatively assumed that the ruptured steam generator did not receive auxiliary feedwater (AFW) flow. The steam releases in this analysis exceeded the break flow resulting in the liquid mass in the ruptured steam generator (SG) decreasing through the event.

If AFW flow was modeled to the ruptured SG and the SG level was being maintained between 60% and 70% narrow range (NR), this would be achieved by controlling the AFW flow and/or opening the atmospheric dump valves (ADVs). Since the steam releases in the worst case analysis exceeded the break flow and there was no AFW flow to the ruptured SG, no additional steaming through the ADVs was needed to control the SG level. If the analysis included AFW flow, the SG level could be maintained by controlling AFW flow without additional steam release through the ADVs.

Thus, the results of the steam release analysis bound those assuming Emergency Operating Procedure (EOP) action to control SG water level between 60% and 70% NR by steaming with the ADVs and with AFW flow.

SRXB-57

Pages 7 and 9 indicate that the turbine driven (TD) AFW flow to the ruptured SG is terminated 15 minutes (for operator action delay time) after the SG narrow range (NR) level reach 35 %.

Justify acceptance of the assumed operator action delay time of 15 minutes. Describe how the TS AFW is terminated. Confirm that the SG NR system and the equipment for the TD AFW termination are safety grade systems.

Response

Justification for acceptance of the operator action delay time of 15 minutes is based upon direction provided in Emergency Operating Procedures (EOP) which requires an operator to verify safety function status by checking that acceptance criteria are satisfied every 15 minutes.

As discussed in the response to SRXB-52 above, AFAS automatically secures feed to the steam generators (SGs) at 29% narrow range (NR) rising. The MTO analysis assumes that this automatic isolation fails to occur and that an additional 15 minutes passes before AFW flow to the ruptured SG is assumed to be terminated. The basis for the 15-minute assumption is the

EOP requirement to complete the EOP Safety Function Status Check Sheet (SFSC) every 15 minutes. Step 1 of EOP-4, Steam Generator Tube Rupture, directs operators to verify SFSC acceptance criteria are satisfied every 15 minutes (a continuous EOP step). The SFSC includes monitoring of SG levels.

Appendix R of EOP-99 directs operators to isolate AFW flow by ensuring AFW pump discharge isolation valves are closed and that the steam supply to the TD AFW pump is isolated. Either one of these actions is sufficient to terminate TD AFW flow to the ruptured generator. Note that for Combustion Engineering designed plants, SG overfill has not historically been a concern due to the large size of the SGs and completion of these EOP actions, supporting the MTO analysis assumption of 15 minutes, is considered acceptable for the following reasons:

- SFSC acceptance criteria, including SG level, is required to be verified every 15 minutes;
- EOP-4 includes steps for ensuring SG levels are maintained within specified limits;
- EOP-99 Appendix R provides instructions for isolating a ruptured SG;
- Isolation of a ruptured SG is considered a critical task in operator training and is required in order to satisfactorily complete the training scenario;
- Isolation is accomplished via closing of valves that can be operated from the control room.

Additionally, the MTO analysis results demonstrate there is significant margin to an overfill condition, with over 1,800 ft^3 of margin available at the time of TD AFW flow termination. This margin would accommodate over 15 additional minutes of TD AFW flow.

The SG NR system and the equipment for the TD AFW termination are safety grade systems.