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

FPL

February 29, 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 4.

L-2012-072
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 1
Docket No. 50-335
Renewed Facility Operating License No. DPR-67

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

References:

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

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

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

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

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Attachments 1 through 4 to this letter provide the requested information and the FPL responses for all of the events except FWLB. The FWLB response is being provided in a separate submittal.

Attachments 1 and 2 contain the responses to staff questions related to the IOPORV. Attachment 1 also contains the responses to CVCS malfunction and LOEL/LOCV events. Attachment 3 contains the non-proprietary and Attachment 4 contains the proprietary response to questions related to the RLBLOCA event.

Attachment 5 contains the Proprietary Information Affidavit. The purpose of this attachment is to withhold the proprietary information contained in the response to the RLBLOCA event (Attachment 4) from public disclosure. The Affidavit, signed by AREVA 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 AREVA be withheld from public disclosure in accordance with 10 CFR 2.390.

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

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

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

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

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

Executed on 29-February-2012

Very truly yours,

Richard L. Anderson

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachments (5)

cc: Mr. William Passetti, Florida Department of Health

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

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

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

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

The responses to RAIs for IOPORV, CVCS malfunction, and LOEL/LOCV events are provided below. The response RLBLOCA event contains information proprietary to AREVA. The non-proprietary response for the RLBLOCA event is provided in Attachment 3 and the proprietary response is provided in Attachment 4. Note that the RLBLOCA response is separated into four questions identified as NRC Audit Q1 through NRC Audit Q4. The response to FWLB event is being submitted in separate correspondence.

Inadvertent Opening of a Power Operated Relief Valve (IOPORV)

Submit pressurizer fill case, commencing with description of the analysis.

a. **Identify more than one single failure required to open both power operated relief valves (PORVs).**

b. **Explain why hot full power (HFP) is more limiting than hot zero power (HZP) and that using HZP moderator temperature coefficient (MTC) bounds the HZP case.**

c. **Confirm that operators can secure charging pumps without having to reset safety injection actuation system (SIAS) if it occurs.**

d. **Provide discussion of procedures and immediateness of responses.**

e. **Justify that the operators will take action before 107 seconds from an operator responses standpoint, simulator experience, operating procedures, etc. Justify that operators will secure charging pump before pressurizer is filled.**

f. **If operators do not close PORVs in 107 seconds, justify using chemical and volume control (CVCS) malfunction to bound or address the remainder of the event.**

Response

Attachment 2, ANP-3067, Revision 1, St. Lucie Unit 1 EPU – Information to Support NRC Review of RCS Depressurization with Pressurizer Overfill, February 2012 contains the IOPORV pressurizer fill case, including the description of the analysis.

a. As described in response to SRXB-38 submitted in FPL letter L-2011-448 [Reference IOPORV-1], the St. Lucie Unit 1 PORVs are direct current (dc) power operated solenoid valves. The two PORVs are mechanically independent of each other. Accordingly, there is no single mechanical failure that could cause both valves to open.

   The PORV actuation logic can be summarized as follows:

   1. The PORV actuation logic is driven from four reactor protective system (RPS) measurement channel high pressurizer pressure bistables.
   2. On high pressure, a bistable output contact energizes a relay in the associated RPS auxiliary logic assembly.
   3. The relay contacts from each of the four redundant RPS auxiliary logic relays are combined to form a two out of four PORV actuation logic.
   4. The RPS auxiliary logic output will energize a single relay (63X1P-1102) on two out of four high pressurizer pressure signals.
   5. Contacts from the 63X1P-1102 relay then energize the contactors for each PORV.

   Spurious energization of the 63X1P-1102 relay due to a short circuit would cause both PORVs to open.

   There is no control switch in the control room to operate the PORVs. Therefore, there is no operator error that could cause both PORVs to open.

b. Refer to Attachment 2 (ANP-3067) Section 2.3 Input Parameters and Assumptions for the discussion of why hot full power (HFP) is more limiting than hot zero power (HZP) and that using HZP moderator temperature coefficient (MTC) bounds the HZP case.

c. The control circuitry for each charging pump will allow the charging pump to be stopped prior to resetting a safety injection actuation signal (SIAS). With the charging pump control switch in the stop (maintained contact) position, the pump will not run.

d. Inadvertent opening of a PORV will result in one or more of the following control room annunciators:

   - H-11 – PZR SAFETY/PORV OPEN,
   - H-13 – PZR CHANNEL X PRESS HIGH/LOW,
   - H-14 – PZR CHANNEL Y PRESS HIGH/LOW,
   - H-19 – PZR CHANNEL X LEVEL HIGH/LOW,
   - H-20 – PZR CHANNEL Y LEVEL HIGH/LOW,
   - H-24 – QUENCH TANK TEMP/PRESS HIGH,
   - H-25 – PZR CHANNEL X LEVEL LOW/LOW,
   - H-26 – PZR CHANNEL Y LEVEL LOW/LOW,
   - H-28 – PZR RELIEF LINE TEMP HIGH,
   - H-30 – QUENCH TANK LEVEL HIGH/LOW,
   - H-31 – PZR PROPORTIONAL HTR LOW LEVEL TRIP/SS ISOL, and
   - H-32 – PZR BACKUP HTR LOW LEVEL TRIP/SS ISOL.

The annunciator response procedures for these alarms provide direction to the operator to go to abnormal operating procedure 1-AOP-01.10 Pressurizer Pressure and Level. The first immediate operator action for 1-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.

e. For the limiting case that was analyzed, the SIAS actuates at 107 sec. As discussed in d. above, isolation of a PORV is addressed in 1-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 actuating a SIAS and thus prior to water passing through the PORV or the pressurizer becoming water solid. Since 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.

f. Not applicable, operators will close or isolate the PORV in less than 107 seconds.

References

IOPORV-1 R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-448), Response to NRC Reactor Systems Branch and Nuclear Performance Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request, October 31, 2011 (ML11269A221).

Chemical and Volume Control System (CVCS) Malfunction

Identify what is a credible single failure as opposed to what is analyzed. Identify the indications of the event that would occur prior to the pressurizer high level alarm. Identify that this event addresses inadvertent safety injection because shutoff head of safety injection pumps.

Response

The limiting credible single failure for the CVCS Malfunction event is the analyzed failure described below. As discussed in FPL Letter L-2011-389 [Reference CVCS-1], the CVCS Malfunction event is initiated by the failure of the pressurizer level transmitter which results in an erroneous low-low level signal for pressurizer level. The generated signal is transmitted to the controller, which responds by starting the two standby charging pumps (the third charging pump is assumed to already be operating) and closing the letdown flow control valve to the minimum flow position. The analysis conservatively assumes the letdown to be isolated. With the mismatch between letdown and charging flow, the pressurizer level and pressure increase. The pressurizer sprays limit the pressure increase. The operators are alerted by a pressurizer high level alarm or a high pressurizer pressure reactor trip or one of the indications described below.

A CVCS malfunction will result in running indication for all charging pumps and one or more of the following control room annunciators prior to the pressurizer high level alarm include:

- H-13 – PZR CHANNEL X PRESS HIGH/LOW,
- H-14 – PZR CHANNEL Y PRESS HIGH/LOW,
- H-19 – PZR CHANNEL X LEVEL HIGH/LOW,
- H-20 – PZR CHANNEL Y LEVEL HIGH/LOW,

- H-25 – PZR CHANNEL X LEVEL LOW/LOW,
- H-26 – PZR CHANNEL Y LEVEL LOW/LOW,
- H-31 – PZR PROPORTIONAL HTR LOW LEVEL TRIP/SS ISOL,
- H-32 – PZR BACKUP HTR LOW LEVEL TRIP/SS ISOL.
- M-3 - VCT LEVEL/HIGH LOW,
- M-5 - LETDOWN PRESSURE HIGH/LOW, and
- M-11 - VCT LEVEL LOW-LOW.

For St. Lucie Unit 1, the consequences of this event are also applicable to an inadvertent operation of the emergency core cooling system (ECCS) where inadvertent operation of the charging pumps results in an increase in pressurizer level. As described in FPL letter L-2011-471 [Reference CVCS-2], a sensitivity study was performed to confirm that the limiting case for the CVCS malfunction event and inadvertent ECCS has been identified. An inadvertent safety injection actuation signal (SIAS) causes the high pressure safety injection (HPSI) pumps, low pressure safety injection (LPSI) pumps, and charging pumps to start, however, since the reactor coolant system (RCS) pressure during the event remains above the shutoff head for the HPSI and LPSI pumps, an inadvertent SIAS causes only the charging pumps to inject. Thus, the injection into the RCS for both the CVCS malfunction and inadvertent ECCS actuation events is the same and is from the charging pumps only.

<u>References</u>

CVCS-1 R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-389), Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request, September 22, 2011 (ML11269A221).

CVCS-2 R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-471), Information Regarding CVCS Malfunction Provided in Support of the Extended Power Uprate License Amendment Request, October 31, 2011 (ML11306A014).

<u>Loss of Electrical Load (LOEL) / Loss of Condenser Vacuum (LOCV)</u>

Clarify the discussion of the limiting overpressure event relative to the Standard Review Plan (SRP) Section 5.2.2 discussion of crediting the second safety grade trip for overpressure protection.

<u>Response</u>

FPL Letter L-2011-206 [Reference LOEL-1] (Section 2.3.6) provided a discussion of the licensing basis for St. Lucie Unit 1 relative to overpressure protection documented in EPU LAR Attachment 5, LR Section 2.8.4.2, Overpressure Protection During Power Operation. The following discussion clarifies the application of the limiting overpressure event to overpressure protection.

St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.2.1, Loss of External Electrical Load, Turbine Trip, and Loss of Condenser Vacuum, and Reference LOEL-1 (Section 2.3.1) discussed the results of the analyses which produce the limiting peak reactor coolant system (RCS) and peak main steam system (MSS) pressures. The limiting overpressure event, Loss of External Electrical Load (LOEL), bounds the Turbine Trip and Loss of Condenser Vacuum (LOCV) events.

Overpressure event analyses, addressed in the LR Section 2.8.5.2.1 and Reference LOEL-1 (Section 2.3.1), were performed by not crediting the safety-grade steam generator low level (SGLL) reactor protection system (RPS) trip and relying on the RPS trip signal from the safety-grade high pressurizer pressure (HPP) trip. To bound the LOEL, Turbine Trip, and LOCV events, the limiting overpressure analysis assumed that the non-safety grade steam bypass control system (SBCS) was unavailable, the main feedwater (MFW) pumps were tripped on turbine trip at event initiation and the turbine stop valve closure time was conservatively short. In addition, no credit is taken for non-safety grade equipment to mitigate the overpressure challenge, specifically, the pressurizer sprays, and the pressurizer power-operated relief valves (PORVs). The analyses assume that the reactor is operating at the EPU power level, and that key system and core parameters are biased within their allowed operating range, including measurement uncertainty. These initial conditions and assumptions produce the highest anticipated pressure.

Tables LOEL-1 and LOEL-2 present the sequence of events for the limiting RCS and MSS overpressure cases, respectively. The results for the RCS overpressure analysis show that the safety-grade SGLL RPS trip, based on the current Technical Specification (TS) setpoint of 20.5%, would occur at 4.4 seconds at an analysis setpoint of 15.5% after accounting for uncertainties. For the MSS pressurization case, the safety-grade SGLL RPS trip would occur slightly after the HPP RPS trip and some credit for the revised TS trip value for EPU is needed. The results show that with a trip setpoint value of 21.0%, the SGLL RPS trip would occur at 5.7 seconds at an analysis setpoint of 16.0% after accounting for the uncertainties (prior to the HPP trip). Since the SGLL RPS trip setpoint is being revised to 35.0% by the EPU LAR, assuming a trip setpoint of 21.0% with uncertainties is conservative and ensures the SGLL trip occurs prior to the HPP trip. The RPS trip on HPP, which is credited in these analyses, occurs at 7.0 seconds and 6.8 seconds for the RCS and MSS overpressure cases, respectively. The credited HPP RPS trip is thus the second safety grade RPS trip based on the above discussion of SGLL RPS trip.

The characteristics of tripping the MFW pumps would not significantly affect the relative timing of the RPS trips since the steam generator (SG) downcomer levels decrease rapidly as a consequence of the termination of steam flow at event initiation, i.e., turbine trip with no SBCS. The analyses demonstrate that the plant will have sufficient pressure relief capacity to ensure that RCS and MSS pressure limits will not be exceeded at the EPU conditions. The peak RCS pressure was found to be below 110% of design pressure or 2750 psia at the limiting RCS location. Peak main steam system pressure was found to be below 110% of design pressure or 1100 psia in the SG dome location.

Therefore, the analysis of the limiting overpressure event, under EPU conditions, demonstrates that the pressurizer safety valves, main steam safety valves (MSSVs), and the RPS provide the requisite overpressure protection during power operation.

<u>References</u>

LOEL-1 R. L. Anderson (FPL) letter to U.S. Nuclear Regulatory Commission (L-2011-206), Information Regarding Areva LOCA and Non-LOCA Methodologies Provided in Support of the St. Lucie Unit 1 License Amendment Request for Extended Power Uprate, May 27, 2011 (ML11153A048).

Table LOEL-1
Sequence of Events for Limiting RCS Overpressure Case

Event	Time (sec)
Event initiation (turbine trip)	0.0
SG low level trip setpoint reached **(20.5% nominal – 5% uncertainty, not credited)**	4.4
High pressurizer pressure trip setpoint reached	7.0
Reactor trip occurred on high pressurizer pressure (including trip response delay)	7.9
Control element assembly (CEA) insertion begins	8.4
Peak reactor power occurred	8.4
Pressurizer safety valves opened	8.9
Peak primary pressure occurred **(2744 psia)**	9.9
Peak core-average RCS temperature occurred	10.3
SG Bank 1 MSSVs opened (both SGs)	10.9
Peak pressurizer level occurred	13.0
Peak MSS pressure (SG dome) occurred	17.2

Table LOEL-2
Sequence of Events for Limiting MSS Overpressure Case

Event	Time (Sec)
Event initiation (turbine trip)	0.0
Pressurizer spray begins	2.5
SG Bank 1 MSSVs opened (both SGs)	4.3
SG low level trip setpoint reached **(partial credit for the high trip setpoint value of 21% -5% uncertainty, not credited)**	5.7
SG Bank 2 MSSVs opened (both SGs)	6.3
High pressurizer pressure trip setpoint reached	6.8
Reactor trip occurred on high pressurizer pressure (including trip response delay)	7.7
Peak reactor power occurred	8.2
CEA insertion begins	8.2
Pressurizer safety valves opened	9.3
Peak MSS pressure (SG dome) occurred **(1092 psia)**	14.0