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

FPL.

April 10, 2012

L-2012-157
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 Request for Additional Information Identified During Audit of the Reactor Systems Branch (SRXB) Fluid System Analyses 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.

(3) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2012-116), "Response to Request for Additional Information Identified During Audit of the Reactor Systems Branch (SRXB) Fluid System Analyses for the Extended Power Uprate License Amendment Request," March 25, 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).

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 selected fluid system analyses performed for the St. Lucie Unit 2 EPU license amendment request (LAR).

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The fluid system analyses reviewed during the audit included; 1) low temperature overpressure protection (LTOP), 2) boric acid delivery, 3) natural circulation cooldown, and 4) shutdown cooling system performance. The FPL response to address fluid system audit items 1, 2, and 4 was previously transmitted to the NRC via letter L-2012-116 [Reference 3]. This letter provides the FPL response to address the fluid system audit item 3 associated with the natural circulation cooldown analysis. This letter also provides FPL's supplemental response to RAI SRXB-77.

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 10-April-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 Request for Additional Information
Identified During Audit of the EPU LAR
Reactor Systems Branch Fluid System Analyses**

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 (SRXB) conducted an audit of selected St. Lucie Unit 2 EPU fluid system analyses at the Westinghouse Electric Company (Westinghouse) facility in Rockville, MD on February 14 and 15, 2012. The fluid system analyses reviewed during the audit included; 1) low temperature overpressure protection (LTOP), 2) boric acid delivery, 3) natural circulation cooldown, and 4) shutdown cooling system performance. This transmittal provides the additional information requested to address fluid system audit item 3. This transmittal also provides a supplemental response to RAI SRXB-77.

<u>SRXB-77 (RAI 2.8.7.2-1)</u>

Table 2.8.7.2-2 includes the results of the natural circulation cooldown (NCC) analysis using the CENTS based on cooldown rates of 30°F/hr and 50°F/hr.

Provide the following information in support of the results in Table 2.8.7.2-2:

1. **a discussion addressing acceptability of use of CENTS for the NCC analysis, and justifying adequacy of any changes to the NRC-approved version of CENTS**

2. **a discussion to show acceptability of the assumptions used and worst single failure considered in the NCC analysis**

3. **a discussion of the results of the NCC analysis to show that the predicted thermal-hydraulic response is within the range approved by the NRC for use of the CENTS code, and there is no unexplainable thermal-hydraulic phenomena for parameters**

4. **justification for use of the decay heat rates based on ANI/ANS-5.1-1979**

5. **a derivation of the required CST water volume for the NCC analysis to show that the required CST water volume is within the TS limits**

6. **a discussion of compliance with the branch positions F and G in BTP RSB 5-4 (SRP, Revision 3).**

<u>Supplemental Response</u>

1. As stated in the previous response to RAI SRXB-77, the CENTS code is not used in the Current Licensing Basis (CLB) Natural Circulation Cooldown (NCC) analysis, but is an approved code that is acceptable for referencing in licensing applications for Combustion Engineering pressurized water reactors (PWRs). The version of the CENTS code used for this analysis is consistent with the NRC approved version of the code described in topical

report WCAP-15996-P-A Revision 1, which has been accepted for use in the analysis for natural circulation cooldown by the US NRC (ML032790634).

While performing case runs to confirm the consistency of the NCC analyses with St. Lucie Unit 2 (SL2) procedures, FPL determined that a feature of the CENTS computer code had been unintentionally activated. The application of the modeling feature results in a non-conservative conclusion for the NCC analyses. Accordingly, FPL is providing updated information based on revised calculations performed with the modeling feature disabled, using the CENTS code in a configuration consistent with the version previously approved by the NRC (ML032790634). The updated information includes the quantity of condensate storage tank (CST) inventory required and the expected duration of the NCC event. The results of the updated analyses do not change the conclusions of EPU LAR Attachment 5, Section 2.8.7.2 with respect to the capability to perform a NCC to Shutdown Cooling (SDC) entry conditions without voiding following implementation of the EPU.

The primary change to the subject NCC analyses was to ensure that all modeling features used in the EPU NCC capability analyses are consistent with the NRC approved version of CENTS. The re-analysis performed used operator limited cooldown rates at various points during the event in order to ensure that there is no voiding in the reactor vessel upper head (RVUH) while simultaneously maintaining the plant within appropriate operational limits. A reduction in cooldown rate results in an extension in the duration of the event and an increase in the condensate requirements. The impact to both the time to SDC entry and the condensate requirements are summarized in Table 1 below for each of the reported cooldown rates.

The EPU re-analysis values reported in Table 1 supersede those reported in EPU LAR Attachment 5, Table 2.8.7.2-1.

2. The plant conditions and assumptions used in the limiting NCC analysis are listed below.

- Plant power is initially at 100.5% of rated power to account for indicated power uncertainty.
- 1979 ANS 5.1 Standard Decay Heat Curve including long term actinides is used.
- Letdown is disabled.
- Main feedwater is disabled.
- The Safety Injection System (SIS) is not used.
- Heat losses from the reactor vessel upper head to containment are set to zero, pressurizer heat losses to containment are calculated by CENTS and the balance of the heat losses from the RCS are also set to zero.
- Main steam isolation valves are closed.
- The main steam safety valves (MSSVs) provide the initial heat removal path.
- Manual control of atmospheric dump valves (ADVs) is available 15 minutes after plant trip and is used to maintain steam generator (SG) pressure within procedural limits.
- The AFW flow is controlled to maintain steam generator level within procedural limits.
- Feedwater enthalpy initially accounts for maximum operating feedwater conditions until the feedwater piping is purged, subsequently feedwater enthalpy accounts for the maximum CST temperature, including uncertainty, and conservative AFW pump heat.
- Charging is available 15 minutes following the plant trip, and is controlled to maintain pressurizer level within procedural limits.

- The minimum Technical Specification required quantity of pressurizer heaters, consistent with the loss of a single electrical train, are available 45 minutes following the plant trip, and are used to maintain pressurizer pressure within procedural limits.

As stated in the previous response to RAI SRXB-77, the most limiting single failure for the NCC analysis remains a loss of one direct current (DC) emergency power train. A loss of one DC emergency train causes the loss of one train of electrical power. The single failure disables one train of components associated with the ADVs, Chemical and Volume Control System, AFW System, and SDC system. Only two of the four DC powered ADVs (one per steam generator) are used in the NCC analysis.

3. The temperature and pressure conditions considered in the revised NCC analysis are within the bounds of the NRC approved CENTS code. The RCS is kept above the saturation pressure corresponding to the RVUH temperature; therefore no two-phase conditions are present during the NCC analysis and no unexpected thermal-hydraulic phenomena are predicted.

4. The decay heat table in the St. Lucie Unit 2 CENTS code is based on the 1979 ANS 5.1 Standard Decay Heat Curve, including 2σ uncertainty for the limiting 50°F/hr cooldown case, and accounts for the affects of neutron capture and long term actinides. The decay heat curve bounds fuel designs with up to 5 weight percent fuel enrichment, fuel burnups to 73,000 MWd/MTU, and operating cycles up to 24 months in duration. Therefore the basis for the decay heat curve used in the NCC analysis bounds the fuel design and operating cycle lengths anticipated as part of the St. Lucie Unit 2 EPU design.

5. The updated values for both the time to SDC entry and the total condensate requirements associated with the changes to the NCC analysis are summarized in Table 1 below. The condensate usage for each phase of the NCC cooldown event is provided in Table 2.

   The increase in the NCC condensate requirements for EPU is acceptable, as it remains bounded by a CST usable volume value of 293,567 gallons. This usable volume is equivalent to the minimum Technical Specification volume of 307,000 gallons for the CST minus the unusable volume of 13,433 gallons consistent with EPU LAR Attachment 5, Table 2.5.4.5-1.

6. The revised NCC analysis assumes that the operators do not depressurize the RVUH below a 20 degree F subcooling margin throughout the cooldown (to preclude drawing a void in the upper head). The revised analysis demonstrates that the plant can be cooled to shutdown cooling entry conditions using only safety grade equipment. Additional details regarding this question are provided in the audit follow-up response below.

<u>SRXB-77 Audit Follow-Up</u>

The response to SRXB-77 provides the information related to the natural circulation cooldown (NCC) analysis. It is not clear whether the NCC analysis is performed in accordance with the plant cooldown procedures or not. As stated in Items B.6 and B.7 of BTP 5-4 of SRP Chapter 5, Revision 3 (which is the same as items F and G of BTP RSB 5-1 of SRP 5.4.7, Revision 2) indicate that the information of plant procedures should be considered in the NCC analysis and the auxiliary feedwater (AFW) supply for cooldown should be based on the longest cooldown time needed to maximize the required AFW

supply for cooldown conditions with or without a loss of offsite power considering effects of a single failure.

If the NCC analysis does not base on the information in the cooldown procedures, provide a discussion of the plant NCC procedures and demonstrate that the NCC analysis results in a longer cooldown time than that for the NCC analysis performed in accordance with the cooldown procedures. Alternatively, the reanalysis of the NCC should be performed in compliance with the plant cooldown procedures with credit of only safety grade systems for cooldown and demonstrate that the required AFW supply in the TS can provide sufficient water for cooldown.

SRXB-77 Audit Follow-Up Response

While the primary change to the St. Lucie Unit 2 EPU NCC analyses described above was to ensure that all modeling features are consistent with the NRC approved version of CENTS, changes were also made to the sequence of operator actions applied during the simulated event. These changes were required as a direct consequence of disabling the affected modeling feature, which influenced the dynamics of the simulated event. The operator actions performed in the updated analyses are consistent with the limitations in the current plant specific NCC procedures. Specifically, these actions include hold periods and reductions in cooldown and depressurization rates, up to and including a soak period. These actions are performed to maintain plant operational limits associated with RCS sub-cooling margins, pressurizer level, steam generator level and margin to upper head voiding. These actions are also performed using only safety grade equipment which is operable from the control room, assuming the loss of off-site power with a limiting single failure of a loss of one emergency power train. The inclusion of such operational and equipment limitations results in a longer duration cooldown than is expected during an actual plant event, ensuring bounding conclusions with respect to both the duration of the event response and the required condensate to support that response.

Operational limitations on the plant cooldown rate, up to and including the use of a soak period, are required based on the revised EPU NCC analysis to ensure voids are not formed in the RVUH. The soak time included in the current St. Lucie Unit 2 NCC procedures will be updated as part of the EPU implementation process based on the updated EPU NCC analysis.

Table 1
Summary of Results for the Updated Natural Circulation Cooldown Analysis

Cooldown Rate Limit (°F/hr)		Calculated Loop ΔT (°F)	Time to Shutdown Cooling Entry (hr)	Condensate Requirement (gallons)
Pre-EPU	50°F/hr	-	-	276,000
EPU	30°F/hr	30	22.8	265.000
EPU	50°F/hr*	36	23.6	281,000

**The 50°F/hr case includes additional conservatism in the decay heat input.*

Table 2
Updated Auxiliary Feedwater Usage per Natural Circulation Cooldown Analysis Segment

Cooldown Rate Limit (°F/hr)	Hot Standby Hold (gallons)	Cooldown: Hot Standby to SDC Entry Temperature (gallons)	Soak to SDC Entry Pressure (gallons)	Total (gallons)
30°F/hr	74,000	117,000	74,000	265,000
50°F/hr*	76,300	164,200	40,500	281,000

**The 50°/hr case includes additional conservatism in the decay heat input.*