UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 19, 2011

Mr. Mano Nazar
Executive Vice President, Nuclear and
Chief Nuclear Officer
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT ST. LUCIE PLANT, UNIT 1 - REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATE (TAC NO. ME5091)

Dear Mr. Nazar:

By letter dated November 22, 2010, Florida Power & Light Company submitted a license amendment request for St. Lucie Plant, Unit No. 1. The proposed amendment would increase the licensed core power level from 2700 megawatt thermal (MWt) to 3020 MWt.

The Nuclear Regulatory Commission staff is reviewing your submittal and has determined that additional information is needed to complete its review. The specific questions are found in the enclosed request for additional information (RAI). It is requested that your RAI response be provided within 30 days of the date of this letter.

Sincerely,

Tracy J. Orf

Tracy J. Orf, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-335

Enclosure:
Request for Additional Information

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REQUEST FOR ADDITIONAL INFORMATION (RAI)

REGARDING LICENSE AMENDMENT REQUEST FOR

EXTENDED POWER UPRATE

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

CVIB-1:

The limiting material with regard to adjusted reference temperature (ART) and the pressurized thermal shock reference temperature (RT_{PTS}) for the St. Lucie, Unit 1 reactor vessel (RV) is Lower Shell Axial Weld 3-203 A/C, Heat Number 305424. This material heat is not contained in the St. Lucie, Unit 1 surveillance program but is contained in the Beaver Valley, Unit 1 surveillance program. Reference 1, Table 2.1.2-4 provides a copper content of 0.27 weight percent, a nickel content of 0.63 weight percent, and a chemistry factor (CF) of 188.8 degrees Farenheit (F) determined using Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, Position 1.1. The reference given for the CF is a letter dated August 28, 1997 (Reference 2), forwarding updated information in response to Generic Letter 92-01 Revision 1, "Reactor Vessel Structural Integrity." Reference 2 references Combustion Engineering (CE) report CE NPSD-1039, Revision 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds" (Reference 3).

However, Beaver Valley, Unit 1 reported three additional chemistry measurements for Heat Number 305424 in Reference 4 that are not included in Reference 3.

The staff, therefore, requests the following information:

1. Provide a discussion of how the additional chemistry measurements reported in Reference 4 were accounted for in the determination of the best estimate CF for weld 3-203 A/C (Heat Number 305424), reported in the St. Lucie, Unit 1 extended power uprate (EPU) Licensing Report, or provide a justification for not using the additional chemistry data.

2. Revise the CF for weld 3-203 A/C (heat number 305424) if necessary.

CVIB-2:

Reference 5 provides the basis for the revised pressure-temperature (P-T) limits for St. Lucie, Unit 1, incorporating revised neutron fluence values that account for the EPU. Figures 3.4-2a and 3.4-2b of the technical specifications have been revised to incorporate the new P-T limits.

The P-T limits must meet the minimum temperature requirements of Title 10, *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, "Fracture Toughness Requirements." For normal operation, including heatups and cooldowns, and anticipated operational occurrences, 10 CFR Part 50, Appendix G, requires that the RV pressure may not exceed 20 percent of the preservice hydrostatic test (PSHT) pressure until the RV temperature exceeds by 120 degrees F the highest reference temperature of the material in the closure flange region of the RV that is highly stressed by bolt preload.

For St. Lucie, Unit 1, 20 percent of the PSHT pressure is 636.25 pounds per square inch gauge (psig). With the indicated pressure correction factor applied, this becomes 557.3 psig. Section 2.7 of Reference 5 states the maximum RT_{NDT} of the closure flange region is 50 degrees F, which means that 557.3 pounds per square inch differential should not be exceeded until a temperature of 170 degrees F is reached. However, revised Technical Specification Figures 3.4-2a and Figure 3.4-2b as well as Figures 2-3 and 2-4 of Reference 5 show the heatup curves exceeding the 20 percent PSHT pressure at 165 degrees F, which Reference 5 indicates is the lowest service temperature as defined by the American Society of Mechanical Engineers Code, Section III, Paragraph NB-3211 (158 degrees F), plus 7 degrees F to account for instrument uncertainty.

Given the information above, the staff, therefore, requests the licensee to explain how the P-T limits for St. Lucie, Unit 1 meet the requirements of 10 CFR Part 50, Appendix G.

<u>CVIB-3</u>:

Clarify whether the minimum temperature for core critical operation and the hydrostatic test temperature is 268.2 degrees F, as shown on Figures 2-1 and 2-2 of Reference 5, or 270.7 degrees F, as stated at the top of page 2-17 of Reference 5.

<u>CVIB-4</u>:

Section 61 of 10 CFR Part 50, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," defines RT_{PTS} as the reference temperature, RT_{NDT}, evaluated for the end of life (EOL) fluence, where EOL fluence means the best-estimate neutron fluence projected for a specific RV beltline material at the clad-base metal interface on the inside surface of the RV at the location where the material receives the highest neutron fluence on the expiration date of the operating license. In Reference 1, Section 2.1.3, the RT_{PTS} evaluation was based on the neutron fluence for 52 effective full power years (EFPY). Additionally, Reference 1, Table 2.1.1-2, "Comparison of Peak 0° and 15° Azimuth Vessel ID Fluence Values at 52 EFPY," implies that 52 EFPY is the expected maximum EFPY value for St. Lucie, Unit 1 corresponding to 60 calendar years, when the renewed operating license expires. However, for the ART evaluation supporting the revised P-T limits, ART values were projected for both the 52 EFPY and 54 EFPY neutron fluences. Reference 1, Section 2.1.2.2 states that new 60-year P-T limits have been generated based on the neutron fluence projected to 54 EFPY to provide margin for fuel management.

Based on the above information, given that the 54 EFPY neutron fluence was used as a basis for the ART used to develop the P-T curves to provide margin for fuel management, the staff requests the licensee discuss whether the 54 EFPY neutron fluence should also be applied to the pressurized thermal shock evaluation.

CVIB-5:

In Reference 1, the effects of EPU are evaluated for the following aging mechanisms of the reactor vessel internals (RVI): fuel cladding corrosion, irradiation assisted stress corrosion cracking (IASCC), stress corrosion cracking (SCC), irradiation embrittlement, thermal embrittlement, void swelling, and irradiation-enhanced stress relaxation.

The susceptibility of the St. Lucie, Unit 1 RVI components to these mechanisms (with the exception of fuel cladding corrosion) was assessed for license renewal as documented in the St. Lucie, Units 1 and 2 License Renewal Application (LRA, Reference 6) and the associated Safety Evaluation Report (Reference 7). The LRA identified the following aging effects and the mechanisms that cause the aging effect: 1) cracking due to SCC and IASCC, 2) reduction in fracture toughness due to irradiation embrittlement and thermal embrittlement, 3) loss of material due to wear, 4) loss of mechanical closure integrity due to cracking (SCC and IASCC) and stress relaxation, 5) loss of preload due to stress relaxation, and 6) dimensional change due to void swelling.

Neutron fluence and temperature are important parameters with respect to assessing the susceptibility of RVI components to many of these aging mechanisms. In particular, threshold neutron fluence levels are identified for certain aging mechanisms in industry guidance documents and topical reports such as WCAP-14577, Revision 1, "License Renewal Evaluation: Aging Management for Reactor Internals (Reference 8), and similar threshold neutron fluence values are also identified in Reference 1 for several of the aging mechanisms evaluated, including IASCC, irradiation embrittlement, void swelling, and stress relaxation. It is not clear to the staff whether any additional components were identified as susceptible to these mechanisms as a result of EPU, compared to those identified in the LRA. For example, Reference 1, Section 2.1.4.2.3.D lists components that are susceptible to irradiation embrittlement. This list does not exactly match the components listed as susceptible to irradiation embrittlement in Section 3.1.4.2.2 of the LRA. It is not clear how the screening for susceptibility to these mechanisms was accomplished.

Based on the above, the staff requests the following information:

a. Describe the method of determining if additional RVI components become susceptible to the aging effects of 1) cracking due to SCC or IASCC, 2) reduction of fracture toughness due to irradiation embrittlement; 3) loss of material due to wear; 4) loss of mechanical closure integrity due to IASCC, irradiation embrittlement, irradiation creep, or stress relaxation; 5) loss of preload due to stress relaxation; and 6) dimensional change due to void swelling. The discussion should address whether a detailed neutron fluence and temperature map was used, and whether stresses in individual components were reevaluated.

b. Confirm whether the design projections of gamma heating bound the projected amount of gamma heating of the RVI components under EPU conditions. Discuss the acceptability of the effects of gamma heating on the RVI components under EPU conditions.

c. Clarify whether any additional RVI components were determined to be susceptible to the aging effects listed in part "a" of this question as a result of EPU, compared to those listed as susceptible to these mechanisms in the LRA (Reference 6).

CVIB-6:

In Section 3.1.4.2.1 of Reference 7, the staff concurred with the licensee's conclusion from the LRA (Reference 6) that loss of material due to wear was an aging effect requiring management for certain St. Lucie, Units 1 and 2 RVI components. Reference 7 states that loss of material from wear occurs due to relative motion between the interfaces and mating surfaces of components caused by flow-induced vibration during plant operation, differential thermal expansion and contraction movements during plant heat up and cool down, and changes in power operating cycles.

Additionally in Section 3.1.4.2.1 of Reference 7, the staff concurred with the licensee's conclusion from the LRA that for the St. Lucie, Unit 1 RVI, loss of mechanical closure integrity of fuel alignment plate guide lug bolts, fuel alignment plate guide lug insert bolts, and control element assembly shroud bolts can occur due to cracking and stress relaxation, and that loss of mechanical closure integrity associated with the core shroud tie rods and snubber bolts can occur due to cracking, reduction in fracture toughness due to irradiation embrittlement, and stress relaxation. However, loss of material due to wear and loss of mechanical closure integrity are not included among the relevant degradation (aging) mechanisms evaluated in Section 2.1.4 of Reference 1. The staff therefore requests the licensee provide an evaluation of the following aging mechanisms considering EPU:

- loss of mechanical closure integrity
- loss of material

The evaluation should address whether additional RVI components (compared to those listed as susceptible to these aging effects in the LRA) become susceptible to these aging effects as a result of EPU.

References

1. Licensing Report, Attachment 5 to Letter from Richard L. Anderson to NRC dated November 22, 2010, Re: St. Lucie Plant Unit 1, Docket No. 50-335, Renewed License No. DPR-67, License Amendment Request for an Extended Power Uprate, Florida Power & Light (FPL) Letter No. L-2010-259 (ADAMS Accession No. ML103560429).

2. FPL Letter, L-97-223, St. Lucie Units 1 and 2, Docket Nos. 50-335 and 50-389, NRC Reactor Vessel Integrity Generic Letter 92-01, Revision 1 Updated Information, August 28, 1997 (ADAMS Legacy Accession No. 9709040378).

3. CE NPSD-1039, Revision 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," prepared for the CE Owners Group, June 1997.

4. WCAP-15770, Revision 2, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation, April 2001, Westinghouse Electric Co. (ADAMS Accession No. ML011870482).

5. WCAP-17197-NP Revision 0, "St. Lucie Unit 1 RCS [Reactor Coolant Systems] Pressure and Temperature Limits and Low-Temperature Overpressure Protection Report For 54 Effective Full Power Years," Appendix G to the EPU Licensing Report for Saint Lucie, Unit 1 (ADAMS Accession No. ML103560511).

6. Application for Renewed Operating Licenses, St. Lucie Units 1 & 2 (ADAMS Accession No. ML013400292).

7. NUREG-1779, "Safety Evaluation Report Related to the License Renewal of St. Lucie Nuclear Plant, Units 1 and 2, Docket Nos. 50-335 and 50-389, Florida Power & Light Company," September, 2003, (ADAMS Accession No. ML032940205).

8. WCAP-14577, Rev. 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals," March 2001 (ADAMS Accession No. ML011080790).

9. "Application for Exemption from Section IV.A.2 of Appendix G to 10 CFR 50 Requirements when Computing Pressure-Temperature Limits for St. Lucie 1," Attachment 1 to Letter from Richard L. Anderson to NRC dated March 3, 2011, Re: St. Lucie Plant Unit 1, Docket No. 50-335, Renewed Facility Operating License No. DPR-67, Response to NRC Request for Additional Information (RAI) Regarding Extended Power Uprate License Amendment Request (ADAMS Accession No. ML110660300).

10. Westinghouse Report, CE-NPSD-683-A Task-1174, Revision 06, "Development of a RCS Pressure and Temperature Limits Report for the Removal of P-T Limits and LTOP [Low-Temperature Overpressure Protection] Requirements from the Technical Specifications," April 2001 (ADAMS Accession No. ML011350387).

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