FPL

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

May 17, 2011

L-2011-178
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 Vessels & Internals Integrity 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 for Extended Power Uprate, November 22, 2010, Accession No. ML103560419.
(2) T. Orf (NRC) to M. Nazar (FPL), "St. Lucie Plant, Unit 1 EPU – Request for Additional Information Regarding License Amendment Request for Extended Power Uprate," April 19, 2011, Accession No. ML111010098.

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).

By letter from the NRC Project Manager dated April 19, 2011 [Reference 2], additional information related to the proposed EPU was requested by the NRC staff in the Vessels & Internals Integrity Branch (CVIB) to support their review of the EPU LAR. The request for additional information (RAI) identified six questions. The response to these RAIs is provided in Attachment 1 to this letter.

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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].

This submittal requires a revision to plant license renewal documentation associated with the aging management of the fuel alignment plate as discussed in the response to RAI CVIB-5. The revisions will be completed prior to the period of extended operation (March 1, 2016).

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate LAR Project Manager, at 772-429-7138.

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

Executed on May 17, 2011.

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachment

cc: Mr. William Passetti, Florida Department of Health

Response to 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 Extended Power Uprate (EPU) License Amendment Request (LAR) for St. Lucie Nuclear Plant Unit 1 that was submitted to the NRC by FPL via letter (L-2010-259) dated November 22, 2010 (Accession Number ML103560419).

In a letter dated April 19, 2011 from NRC (Tracy Orf) to FPL (Mano Nazar), Accession Number ML 111010098, Subject: St. Lucie Plant, Unit 1 – Request for Additional Information Regarding License Amendment Request for Extended Power Uprate, the NRC staff requested additional information regarding FPL's request to implement the EPU. The RAI consisted of six (6) questions from the NRC's Vessels & Internals Integrity Branch (CVIB). These six RAI questions and the FPL responses are documented below. References identified in the NRC request are provided at the end of this attachment.

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 Fahrenheit (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.**

Response

The value of 0.27 wt% copper (Cu) for weld 3-203 A/C, heat number 305424, has its roots in the best estimate value generated by CE NPSD-1039, Rev. 2, and reaffirmed in CE NPSD-1119, Rev. 1. This best estimate value has been accepted by the NRC in previous Generic Letter 92-01 submittals, and has been incorporated into the NRC Reactor Vessel Integrity Database, Version 2.0.1.

The best estimate Cu used by CE NPSD-1039, Rev. 2 for this weld was a coil-weighted average. Specifically, three "groups" of measurements were available, each corresponding to a weld. For two of the three groups ("a" and "b"), a single measurement was obtained for each group. For the third group ("c"), 32 measurements were obtained. To resolve this difference in measurement quantity, the number of coils required for the fabrication of each weld was determined, and the measurements were weighted accordingly. Groups "a" and "b" each represented a single arc weld fabricated with one coil, with one measurement available for each. Group "c", with 32 measurements, was a tandem arc weld for which 8 coils were used. Application of the coil-weighted methodology accounted for relative volume of the weld deposits, as well as sample-to-sample variations, and was thus determined to be the best estimate for Cu.

The Beaver Valley Unit 1, Capsule Y Cu measurements range from 0.22 – 0.23 wt% Cu. When these results are incorporated into the coil-weighted average of CE NPSD-1039, Rev. 2, the best estimate Cu content is slightly reduced. However, the difference is negligible when the result is rounded to two significant digits. Therefore, the best-estimate Cu content of 0.27 wt% reported by St. Lucie Unit 1 is conservative, and no change is necessary to the reported chemistry factor for weld 3-203 A/C, heat number 305424.

<u>CVIB-2</u>

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.

Response

After reviewing the documentation associated with the pressure temperature limits for St. Lucie Unit 1, it was determined that the value of the maximum RTNDT for the closure flange region is 30°F. The value identified in WCAP-17197-NP, Revision 0, Section 2.7 should therefore be 30°F and the margin to the minimum bolt-up temperature of 80°F should be 50°F. Therefore, the sum of the RTNDT for the closure flange, the 120°F limit and the uncertainty is equal to 157°F. This is less than the limitation due to the lowest service temperature requirement of 165°F which is therefore the overall bounding temperature limit under which the 557.3 psi differential cannot be exceeded. Therefore, the figures discussed are correct as presented in WCAP-17197-NP, Revision 0 and in the proposed Technical Specifications.

The correct value for this limit has been used to determine the requirements associated with the proposed licensing change with the exception of the noted error. This error has been determined to be typographical in nature and has only been identified in the text of WCAP-17197-NP Revision 0 and has no further impact on the proposed changes.

CVIB-3

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.

Response

The minimum temperature for core critical operation and the hydrostatic test temperature is 268.2°F, as shown in Figure 2-1, Figure 2-2, and Table 2-8 of Reference 5. The first and third paragraphs of Reference 5, page 2-17, contain incorrect temperatures. The 270.7°F value should be 268.2°F; the 277.7°F value should be 275.2°F.

CVIB-4

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 (PTS) evaluation.

Response

52 effective full power years (EFPY) corresponds to 60 calendar years of operation for St. Lucie Unit 1. Time-limited aging analyses performed to 52 EFPY at extended power uprate conditions meet the requirements of 10 CFR 50.61 and 10 CFR 50, Appendix G. A fuel management margin (a margin to account for future changes in operating factors that could affect EFPY calculations) of 2 EFPY (for a total of 54 EFPY) was added to the 60-year P-T limits.

An update to the P-T limit curves would entail a revision to the technical specifications and require training operators to new procedures. In contrast, an update to the pressurized thermal shock (PTS) or upper shelf energy (USE) projections would require a submittal to the U.S. NRC in the event of a significant deviation from current projections. If future operating conditions (due to any future plant modifications after EPU) differ from the uprated projections such that the reported 52 EFPY PTS and USE projections are no longer adequate, then St. Lucie Unit 1 will review these evaluations and update them, as required by existing programs.

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).**

<u>Response</u>

The material degradation mechanisms that are potentially affected by changes in plant operating conditions and environments due to the EPU were evaluated with regard to the various components in the RVI. A detailed fluence analysis of the reactor pressure vessel (from the interior of the core shroud plates through the vessel wall around the mid-plane) was used to determine fluence through the various RVI components. The fluence calculation adhered to the requirements of Regulatory Guide 1.190 with regard to method and uncertainty. For the materials evaluation, the fluence values in the detailed map were used to evaluate potential fluence conditions at other locations within the RVI. For temperature, the gamma heating rates (based on fluence) were evaluated to find the areas of highest temperature within the internals. These temperatures were inputs to the environmental conditions considered in the materials evaluation.

a. To determine if additional RVI components would become susceptible to degradation for the EPU, the EPU conditions were considered relative to the prior plant conditions. Screening criteria for degradation mechanisms were applied to the components in the RVI. Components that newly exceeded threshold criteria were considered to be potentially susceptible to degradation. Certain components that exceeded threshold criteria under EPU conditions had also previously exceeded threshold criteria under prior conditions. The screening criteria of MRP-175 (*Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values.* 1012081; ADAMS Accession Number ML061880278) were consulted in evaluating the susceptibility of the RVI components to various age-related degradation mechanisms:

   - SCC – Comparison of the chemistry and temperatures were made to determine susceptibility. Credit for positive changes was also given, such as for new implementation of zinc injection.
   - IASCC – Comparison of plant conditions were made, especially with regard to the fluence threshold values.
   - Thermal embrittlement – Austenitic stainless steel materials with high delta ferrite were considered as potentially susceptible. NUREG-1801 (GALL) provides screening criteria for cast austenitic stainless steel.

- Wear – Wear occurs as a result of reduced preload and changing flow patterns. The preload changes were identified in the discussion. Prior aging management reviews reviewed this mechanism for applicability.
- Irradiation embrittlement – Screening criteria for irradiation embrittlement included fluence.
- Stress Relaxation – Conservative calculations were performed for the various potentially susceptible threaded structural fasteners.
- Loss of mechanical closure integrity – The threaded structural fasteners that were potentially affected by uprate conditions (higher fluence) had a calculation performed for remaining preload.
- Void swelling – This degradation was examined for the areas of highest temperature and fluence in the core shroud against screening criteria.

A fluence and temperature map was used, but beyond the core shroud area, the fluence was extrapolated from the core shroud model (using the decreasing fluence values at increasing distances from the mid-plane core shroud plate interior surface). Fluence values at the various components were used to determine susceptibility. Temperatures at various points within the internals (reactor vessel inlet, reactor vessel outlet, peak core outlet, and peak gamma heating) were used.

SCC is a synergistic degradation mechanism that requires high tensile stress (including residual and operating), an aggressive environment, and a susceptible material (or material condition). As identified in LRA Table 3.1-1, all RVI components have already been identified as requiring aging management to control SCC. The St. Lucie Unit 1 chemistry controls program maintains rigorous control of reactor coolant chemistry; this program manages contaminant concentrations to a low level that will not cause SCC of stainless steel in primary water. The increases in temperature or stress for EPU conditions will not increase the susceptibility to SCC for the extended license period.

For IASCC to occur, both sufficient fluence and stress are required. Temperature is not among the current industry standard threshold values for evaluating IASCC in MRP-175. Fluence calculations for EPU-specific conditions were included in the evaluation of the St. Lucie Unit 1 RVI materials.

For thermal embrittlement, cast austenitic stainless steels are potentially susceptible. The guidance of NUREG-1801 was followed to evaluate the potential susceptibility of cast stainless steel components.

Loss of material due to wear is a flow dependent phenomenon. (Wear is also dependent on thermal expansion and movement due to differential thermal expansion during heatup and cooldown, which are not affected by the change to EPU conditions.) Evaluations completed for the EPU determined that the EPU will not result in an increase in the best estimate flow in the reactor coolant system. Because the flow will not change for EPU conditions, there will be no impact on the wear of the components of the RVI.

For potential irradiation embrittlement, sufficient fluence is required, but stress and temperature do not influence this degradation mechanism.

Calculations were performed for stress relaxation, based on the fluence screening criteria provided by MRP-175. No specific threaded structural fastener will lose enough preload to potentially lose mechanical closure integrity.

Loss of mechanical closure integrity, including loss of preload, applies to core support threaded structural fasteners (bolting). All RVI threaded structural fasteners have already been identified in LRA Table 3.1-1 as being susceptible to loss of mechanical closure integrity. The chemistry limits remain unchanged for EPU conditions and will not adversely affect material conditions in the primary system. Changes in stress or temperature are not expected to change how bolting is managed during the license extension period.

Void swelling was assessed using the screening criteria of MRP-175 (608°F and 1.3×10^{22} n/cm^2 (E > 1.0 MeV) for the RVI components. The highest temperature in the internals and the highest fluence on the core shroud plates were compared (as if they occur in the same position) for a conservative calculation of potential void swelling, using the equation provided by MRP-175.

b. A comparison of the design projections of gamma heating to the projected amount of gamma heating of the RVI components under EPU conditions was not performed. Thermal stresses due to gamma heating of the RVI components were calculated under EPU conditions. The thermal stresses due to gamma heating based on EPU conditions, where applicable, were included in the stress analyses performed for the RVI components. As discussed in sub-section 2.2.3.2.5 of Reference 1, all RVI components met the acceptance criteria for stress and temperature, demonstrating that the gamma heating of the RVI components was acceptable.

c. Compared to components listed in LRA Table 3.1-1 as susceptible to the various age-related degradation mechanisms, additional components may be susceptible to degradation under EPU conditions. A fluence and temperature map was used, but beyond the core shroud area, the fluence was extrapolated from the core shroud model (using the decreasing fluence values at increasing distances from the mid-plane core shroud plate interior surface). The fuel alignment plate, CEA shroud assemblies, and the upper guide structure support plate may be susceptible to irradiation embrittlement in addition to discussion of other degradation mechanisms in the LRA tables. Cracking of these components was previously identified in the LRA. Irradiation embrittlement may result in decreases in fracture toughness of the fuel alignment plate, CEA shroud assemblies, and the upper guide structure support plate; FPL will update License Renewal documentation to reflect this change. Other components potentially susceptible to the various degradation mechanisms had been previously mentioned in LRA Table 3.1-1.

<u>CVIB-6</u>

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.

Response

The material degradation mechanisms that are potentially affected by changes in plant operating conditions and environments due to the EPU were evaluated with regard to the various components in the RVI.

All RVI threaded structural fasteners (bolting) have already been identified in LRA Table 3.1-1 as being susceptible to loss of mechanical closure integrity. With respect to SCC the chemistry limits remain unchanged for EPU conditions and will not adversely affect material conditions in the primary system. Changes in stress or temperature are not expected to change how bolting is managed during the license extension period. With respect to stress relaxation, the minimal changes in temperature and fluence due to the EPU are not expected to change how bolting is managed during the period of extended license. The aging degradation mechanisms that could cause a loss of closure integrity were identified and discussed in the assessment of the EPU conditions at St. Lucie Unit 1.

Loss of material due to wear is a flow dependent phenomenon. (Wear is also dependent on thermal expansion and movement due to differential thermal expansion during heatup and cooldown, which are not affected by the change to EPU conditions.) Evaluations completed for the EPU determined that the EPU will not result in an increase in the best estimate flow in the reactor coolant system. Because the flow will not change for EPU conditions, there will be no impact on the wear of the components of the RVI.

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. ML 103560429).

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. ML1 03560511).

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. ML 110660300).

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).