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

January 29, 2009

Mr. Mano Nazar
Senior 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 NO. 2 - ISSUANCE OF AMENDMENT REGARDING PRESSURE VESSEL FLUENCE TO 55 EFFECTIVE FULL-POWER YEARS OF OPERATION (TAC NO. MD8040)

Dear Mr. Nazar:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 154 to Renewed Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your application dated January 23, 2008.

This amendment extends the pressure temperature (PT) limit curves and the low-temperature overpressure protection (LTOP) setpoints for operation to 55 Effective Full-Power Years (EFPYs). The current PT limit curves (and the LTOP setpoints) are applicable to 21.7 EFPYs. The new PT limits and LTOP setpoints will be applicable to 60 calendar years which includes the period until the end of the renewed operating license.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/
Siva P. Lingam

Siva P. Lingam, Project Manager
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 154 to NPF-16
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 154
Renewed License No. NPF-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated January 23, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Renewed Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 3.B to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 154 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: January 29, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 154

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace Page 3 of Renewed Operating License NPF-16 with the attached Page 3.

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
Index XXI	Index XXI
Index XXII	Index XXII
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31a	3/4 4-31a
3/4 4-31b	3/4 4-31b
3/4 4-32	3/4 4-32
3/4 4-35	3/4 4-35
3/4 4-37a	3/4 4-37a

neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required.

D. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

E. Pursuant to the Act and 10 CFR Parts 30, 40, and 70, FPL to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission's regulations: 10 CFR Part 20, Section 30.34 of 10 FR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. <u>Maximum Power Level</u>

FPL is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal).

Commencing with the startup for Cycle 16 and until the Combustion Engineering Model 3410 Steam Generators are replaced, the maximum reactor core power shall not exceed 89 percent of 2700 megawatts (thermal) if:

a. The Reactor Coolant System Flow Rate is less than 335,000 gpm but greater than or equal to 300,000 gpm, or

b. The Reactor Coolant System Flow Rate is greater than or equal to 300,000 gpm AND the percentage of steam generator tubes plugged is greater than 30 percent (2520 tubes/SG) but less than or equal to 42 percent (3532 tubes/SG).

This restriction in maximum reactor core power is based on analyses provided by FPL in submittals dated October 21, 2005 and February 28,2006, and approved by the NRC in Amendment No. 145, which limits the percent of steam generator tubes plugged to a maximum of 42 percent (3532 tubes) in either steam generator and limits the plugging asymmetry between steam generators to a maximum of 600 tubes.

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. **154** are hereby incorporated in the renewed license. FPL shall operate the facility in accordance with the Technical Specifications.

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LIST OF FIGURES

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REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} to less than 200°F within the next 30 hours in accordance with Figure 3.4-3.

SURVEILLANCE REQUIREMENTS

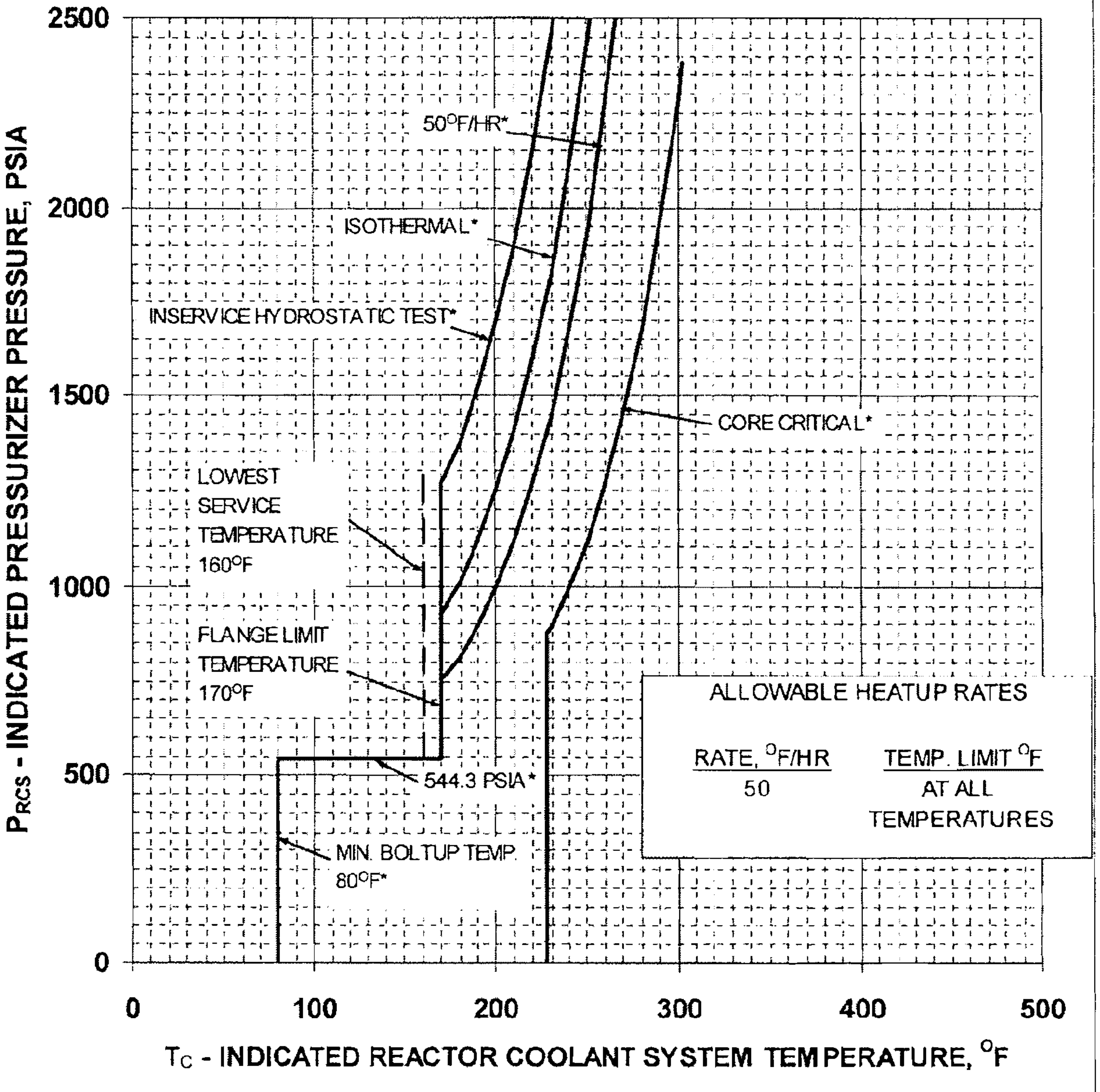
4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

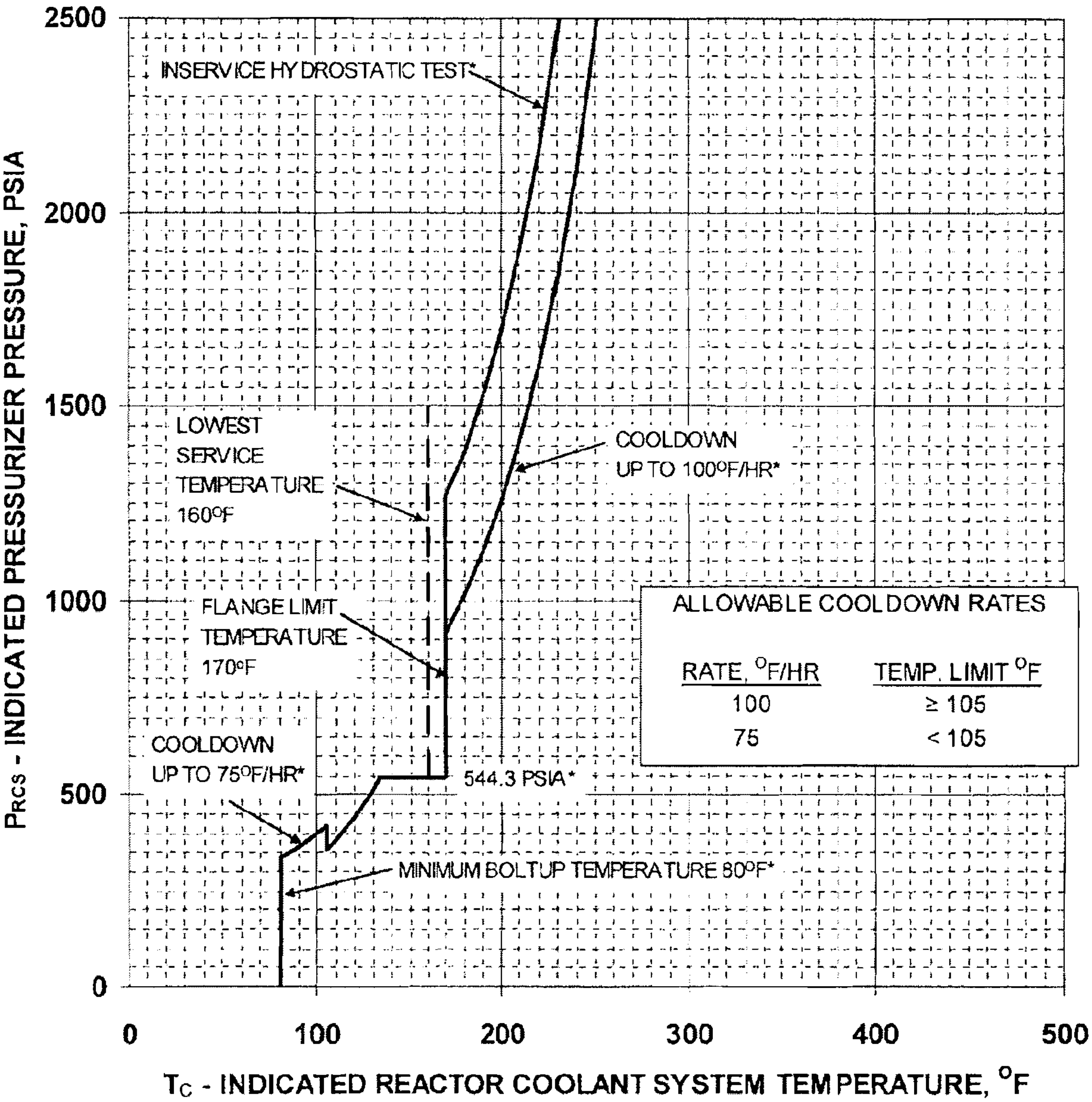
4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

FIGURE 3.4-2
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, HEATUP, CORE CRITICAL, AND INSERVICE TEST



* - Includes Instrument Uncertainty

FIGURE 3.4-3
ST. LUCIE UNIT 2 REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR 55 EFPY, COOLDOWN AND INSERVICE TEST



* - Includes Instrument Uncertainty

DELETED

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 Unless the RCS is depressurized and vented by at least 3.58 square inches, at least one of the following overpressure protection systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of less than or equal to 490 psia and with their associated block valves open. These valves may only be used to satisfy low temperature overpressure protection (LTOP) when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

b. Two shutdown cooling relief valves (SDCRVs) with a lift setting of less than or equal to 350 psia.

c. One PORV with a lift setting of less than or equal to 490 psia and with its associated block valve open in conjunction with the use of one SDCRV with a lift setting of less than or equal to 350 psia. This combination may only be used to satisfy LTOP when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

APPLICABILITY: MODES 4#, 5 and 6.

ACTION:

a. With either a PORV or an SDCRV being used for LTOP inoperable, restore at least two overpressure protection devices to OPERABLE status within 7 days or:

   1. Depressurize and vent the RCS with a minimum vent area of 3.58 square inches within the next 8 hours; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3 within the next 8 hours.

b. With none of the overpressure protection devices being used for LTOP OPERABLE, within the next eight hours either:

   1. Restore at least one overpressure protection device to OPERABLE status or vent the RCS; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

With cold leg temperature within the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, EFPY	Cold Leg Temperature, F° During Heatup	Cold Leg Temperature, F° During Cooldown
≤ 55	≤ 246	≤ 224

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	T_{cold}, F° During Heatup	T_{cold}, F° During Cooldown
≤ 55	80	132

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 154

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated January 23, 2008 [U. S. Nuclear Regulatory Commission (NRC) Agencywide Documents Access and Management System (ADAMS) Accession No. ML080290135], Florida Power & Light Company (FPL), the licensee for St. Lucie Unit 2, submitted a license amendment to change the facility's Technical Specifications (TSs). The proposed amendment would provide new pressure-temperature (P-T) limit curves, low temperature overpressure protection (LTOP) system setpoints, and LTOP system enable temperature (T_{enable}). The licensee revised the P-T limit curves to provide new limits that are valid to 55 effective full power years (EFPY) for St. Lucie Unit 2, which includes the period of extended operation in the renewed operating license.

The proposed P-T limit curves in the licensee's letter dated January 23, 2008, are based on the current P-T limit curves that were approved for 21.7 EFPY for St. Lucie Unit 2, in a NRC letter dated September 5, 2000 [ADAMS Accession No. ML0037475961]. NUREG-1779, "Safety Evaluation Report Related to License Renewal of St. Lucie Nuclear Plant, Units 1 and 2," dated September 2003 [ADAMS Accession No. ML032940205], provided the basis for the NRC staff to approve the extension of the operating period for St. Lucie Nuclear Plant, Units 1 and 2. NUREG-1779 also stated that the licensee needed to request an amendment to the TSs to include revised P-T limit curves prior to the expiration of the most current P-T limits curves approved in the TSs, and that this request should be submitted for NRC review and approval for compliance with the requirements of Title 10, Code of Federal Regulations (10 CFR) Part 50, Appendix G.

2.0 REGULATORY EVALUATION

The NRC has established requirements in 10 CFR Part 50, to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limit curves be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Code. Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2 to review P-T limit curves. RG 1.99, Rev. 2 contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Rev. 1 requested that licensees submit their reactor pressure vessel (RPV) materials property data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1 requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. ASME Code, Section XI, Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 on these stress intensities for hydrostatic testing curves. The flaw postulated in the ASME Code, Section XI, Appendix G has a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the 1/4 thickness (1/4t) and 3/4 thickness (3/4t) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The methodology found in Appendix G to Section XI of the ASME Code requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}) by evaluating material property changes due to neutron radiation. The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}) and a margin term. The ΔRT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

The PT limits must account for neutron embrittlement over the lifetime of the reactor vessel. As such, it is necessary to project the total neutron fluence to which the reactor vessel materials are subject over the extended lifetime of the plant. The NRC has established guidance for performing acceptable calculations of reactor vessel neutron fluence in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

In consideration of the guidance set forth in RG 1.190, General Design Criteria (GDC) 14, 30, and 31 are applicable. GDC 14, "Reactor Coolant Pressure Boundary," requires the design, fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires, among other things, that components comprising the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31, "Fracture

Prevention of Reactor Coolant Pressure Boundary," pertains to the design of the reactor coolant pressure boundary, stating:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

The LTOP system, provided by the power operated relief valves (PORVs) and also by the shutdown cooling relief valves, ensures reactor coolant system (RCS) over pressurization below certain temperatures would be prevented, thus maintaining reactor coolant pressure boundary integrity. The LTOP analysis yields Limiting Conditions for Operation (LCOs) that constitute LTOP system alignments for the period of applicability.

The NRC staff reviewed the LTOP analysis, contained in Chapter 3 of WCAP-16817, using the guidance contained in Branch Technical Position (BTP) 5-2, "Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures," of NUREG-0800. Among other things unaffected by this license amendment request, BTP 5-2 recommends assuming single equipment failures and the analysis of multiple transients to establish the limiting transient.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The licensee determined that the limiting ART that bounds the St. Lucie Unit 2 RPV is from the intermediate shell plate M-605-2, which is controlling at the 1/4t location, and intermediate shell plate M-605-1, which is controlling at the 3/4t location. The critical parameters for the licensee's ART determination for each of these locations are shown in the table below.

Material	Location	Initial RT_{NDT} (°F)	Fluence at Inside Surface (n/cm^2)	Fluence at Location (n/cm^2)	Chemistry Factor [1] (°F)	ΔRT_{NDT} (°F)	Margin [2] (°F)	ART (°F)
Intermediate Shell M-605-2	¼T	10	4.56×10^{19}	2.72×10^{19}	91.5 (table)	116	34	160
Intermediate Shell M-605-1	¾T	30	4.56×10^{19}	9.66×10^{18}	74.2 (table)	73	34	137

(1) Chemistry factors were determined using Regulatory Guide 1.99, Rev. 2, Position 1.1.

(2) The margin term for each ART calculation was based on the establishment of initial material property uncertainty (σ_I) and shift in material property uncertainty (σ_Δ) consistent with the guidance in Regulatory Guide 1.99, Rev. 2.

The TS changes submitted by the licensee in the January 23, 2008, letter include:

- Modified RCS P-T limit curves to include instrument uncertainty and the inservice hydro test line for the new heatup curve.
- New cooldown curve containing 100 °F/hr and 75 °F/hr cooldown rates since the lower rates allow higher pressures than are allowed by the lowest service temperature and flange limitation restriction.
- P-T limit curves extended to 55 EFPY.
- LTOP system setpoints and LTOP system T_{enable} values to reflect the extended period of operation under the renewed license. The PORV lift setpoint has been raised from ≤ 470 pounds-per-square-inch absolute (psia) to ≤ 490 psia.

3.2 Staff's Evaluation

3.2.1 ART Value and P-T Limit Curves

To assess the validity of the licensee's proposed curves, the staff performed an independent assessment of the licensee's submittal. The staff first performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the NRC staff verified that the licensee's limiting material is intermediate shell plate M-605-2 and intermediate shell plate M-605-1 at the 1/4t and 3/4t locations, respectively. The staff's calculated ART values were in good agreement with the licensee's calculated ART values of 160 °F and 137 °F for the 1/4t location and the 3/4t location, respectively.

The staff then evaluated the licensee's P-T limit curves for acceptability by performing independent calculations using the methodologies of Appendix G of Section XI of the ASME Code and 10 CFR Part 50, Appendix G. The licensee stated that the proposed P-T limit curves were based on the methodologies of Appendix G of Section XI of the ASME Code, 1998 Edition with the 2000 Addenda, which utilizes an alternate reference fracture toughness (K_{IC}) curve instead of the K_{Ia} fracture toughness curve for RPV materials in determining the P-T limit curves. NRC Regulatory Issue Summary 2004-04, dated April 5, 2004 [ADAMS Accession No. ML040920323], states that the ASME Section XI, Appendix G, 1998 Edition with the 2000 Addenda may be used without the need for exemption. The use of the K_{IC} fracture toughness curve is appropriate for evaluating the potential for crack initiation without imposing unnecessary conservatism. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the controlled heatup and cooldown process of a RPV.

The proposed P-T limit curves in the January 23, 2008, letter modified the P-T limit curves by including margins to account for control room instrument errors (10 °F) and pressure instrument errors (91.7 pounds-per-square inch differential). The staff also found that the minimum temperature requirements of Table 1 of Appendix G to 10 CFR Part 50 were properly implemented in the P-T limit curves. Therefore, the staff verified that the licensee's proposed P-T limits are in accordance with Appendix G to Section XI of the ASME Code and satisfy the requirements of Appendix G to 10 CFR Part 50. The proposed P-T limit curves also satisfy GL 88-11, because the methodology in RG 1.99, Rev. 2 was used to calculate the ART.

3.2.2 Neutron Fluence

The licensee's submittal outlines the methodology for the calculation of the projection of the ART for the vessel material when given the projected peak vessel fluence, vessel thickness and material of the vessel. The ART is calculated for each element in the beltline region to identify the critical element (highest ART), which is then used to calculate the PT limit curves and the LTOP settings. The general method follows Appendix G to 10 CFR Part 50. The submittal includes the new and proposed technical specification.

The detailed calculations are carried out as described in WCAP-16817-NP, "St. Lucie Unit 2 RCS Pressure and Temperature Limits and Low Temperature Overpressure Protection Report for 55 Effective Full Power Years." A summary of the fluence calculations is in a FPL calculation PSL-BFJF-01-002, "St. Lucie 1 and 2 Vessel Fluence Projections for Life Extension Project." This calculation is a projection of calculations performed by Westinghouse in WCAP-15040, "Analysis of Capsule 263 from FPL St. Lucie Unit 2 Reactor Vessel Radiation Surveillance Program," which documented the analysis of surveillance capsule 263° from the St. Lucie Unit-2, issued on April 1998.

WCAP-15040 was issued in 1998 (i.e., before RG 1.190 was issued). The report states that the calculations adhere to the guidance in RG 1.190, although RG 1.190 was issued in March 2001. However, before the RG was issued the staff circulated a draft guide (DG 1057) that was essentially identical to the RG that Westinghouse applied and the staff accepted as equivalent to RG 1.190.

WCAP-16817-NP (Rev. 2), "St. Lucie Unit 2 RCS Pressure and Temperature Limits and Low Temperature Overpressure Protection Report for 55 Effective Full Power Years," states that the DORT code was used for the neutron transport and dosimetry parts of the calculation using the BUGLE-96 cross sections, derived from the ENDF/B-VI cross section data base. The fuel assembly power distribution was derived from the Westinghouse ANC (advanced nodal code) calculations. Pin-by-pin power distribution was employed. The neutron transport was formulated in forward (r, θ) and (r, z) to obtain relative neutron energy distribution. A series of adjoint fast flux calculations were also performed for the capsule location and several azimuthal vessel locations. The capsule fast flux included dosimetry analysis for the capsule threshold detectors. The fast flux calculations, as described above, adhere to the guidance in RG 1.190 and hence are acceptable.

The licensee's calculations were performed for Unit 2, taking Cycle 12 to be an equilibrium cycle. However, to bound any future cycle variations in fuel management; flux (fluence) projections were augmented by 10 percent. For Unit 2, an additional 5 percent was added to the fluence estimate to account for the full implementation of axial blankets. It is assumed that future core loadings will be low leakage and will not deviate more than a few percentages from the equilibrium cycle. Should larger perturbations occur that would alter the current projections, the licensee must report these changes to the NRC as provided in 10 CFR Part 50.61.

Section 5.2.1 in WCAP-16817-NP (Rev. 2) includes a summary of peak vessel fluence values for Unit 2 for 32, 40, 48 and 60 EFPYs. As described above, the NRC staff reviewed the calculations and determined that they adhere to the guidance in RG 1.190 and, hence, are acceptable. Therefore, the fluence calculations form an acceptable basis for the P-T limits.

3.2.3 Low Temperature Overpressure Protection (LTOP)

The licensee proposed to change the PORV opening setpoints required in TS LCO 3.4.9.3 from ≤ 470 psia to ≤ 490 psia, which accounts for control circuit uncertainty and valve response time. This change is supported by analyses that are consistent with the current licensing basis transient analyses, of mass- and energy-addition transients. While the limiting energy addition transients explicitly analyzed a PORV setpoint of 490 psia, the licensee chose to demonstrate that the mass addition transients are not a function of PORV setpoint in the range of 470 psia to 490 psia. Of course, if the PORV setpoint was analyzed at, for example 610 psia, the NRC staff expects that this transient would progress differently. In the indicated range, however, the mass-addition transients are unaffected by small (i.e., 20 psia) changes in PORV setpoints. To demonstrate this phenomenon, the licensee analyzed three limiting mass addition transients assuming varying PORV setpoints to show that the peak pressure is not a function of the PORV setpoint, but rather the PORV flow area and safety injection cut-in pressure. In the case of both the mass- and energy-addition transients, the licensee demonstrated adequate overpressure limitation with the revised PORV setpoints. Each of the licensee's analysis assumed only one functioning shut down cooling relief valve or one functioning PORV. Because the analyses demonstrate acceptable results and employ assumptions recommended in BTP 5-2, per the regulatory evaluation, the NRC staff accepts the licensee's analyses and concludes that the requested PORV LTOP setpoint change is acceptable.

4.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, NRC, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (73 FR 52418, dated September 9, 2008). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Lambros Lois
Carolyn Fairbanks
Benjamin Parks

Date: January 29, 2009

January 29, 2009

Mr. Mano Nazar
Senior 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 NO. 2 - ISSUANCE OF AMENDMENT REGARDING PRESSURE VESSEL FLUENCE TO 55 EFFECTIVE FULL-POWER YEARS OF OPERATION (TAC NO. MD8040)

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/RA/

Siva P. Lingam, Project Manager
Plant Licensing Branch II-2
Division of Operator Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 154 to NPF-16
2. Safety Evaluation

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