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

December 28, 2000

Mr. T. F. Plunkett
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: ST. LUCIE UNIT 2 - ISSUANCE OF AMENDMENT REGARDING PRESSURE TEMPERATURE LIMITS (TAC NO. MA9532)

Dear Mr. Plunkett:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No.112 to 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 July 19, 2000. It will extend the applicability of the current reactor coolant system pressure/temperature limits and allowed heatup and cooldown rates to 21.7 effective full power years of operation.

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,

Kahtan N. Jabbour

Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 112 to NPF-16
2. Safety Evaluation

cc w/encls: See next page

December 28, 2000

Mr. T. F. Plunkett
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: ST. LUCIE UNIT 2 - ISSUANCE OF AMENDMENT REGARDING PRESSURE TEMPERATURE LIMITS (TAC NO. MA9532)

Dear Mr. Plunkett:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 112 to 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 July 19, 2000. It will extend the applicability of the current reactor coolant system pressure/temperature limits and allowed heatup and cooldown rates to 21.7 effective full power years of operation.

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/

Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 112 to NPF-16
2. Safety Evaluation

cc w/encls: See next page

Distribution:

PUBLIC	GHill (2)
PD II-2 Rdg.	RidsNrrDripRtsb (WBeckner)
RidsNrrDlpmLpdii	LLois
RidsNrrDlpmLpdii2	SSheng
RidsNrrPMKJabbour	RidsRgn2MailCenter
BClayton (Hardcopy)	
RidsOgcRp	
RidsAcrsAcnwMailCenter	

DOCUMENT NAME: G:\PDII-2\St. Lucie\AMDA9532.WPD

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure "E" = Copy with attachment/enclosure "N" = No copy

with change noted.

OFFICE	PDII-2/PM	PDII-2/LA	PDII-2/SC	OGC
NAME	KJabbour	BClayton	RCorreia	C Marco
DATE	12/19/00	12/ /00	12/27/00	12/20/00

OFFICIAL RECORD COPY

cc:

Senior Resident Inspector
St. Lucie Plant
U.S. Nuclear Regulatory Commission
P.O. Box 6090
Jensen Beach, Florida 34957

Joe Myers, Director
Division of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

M. S. Ross, Attorney
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. Douglas Anderson
County Administrator
St. Lucie County
2300 Virginia Avenue
Fort Pierce, Florida 34982

Mr. William A. Passetti, Chief
Department of Health
Bureau of Radiation Control
2020 Capital Circle, SE, Bin #C21
Tallahassee, Florida 32399-1741

Mr. Rajiv S. Kundalkar
Vice President
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Mr. R. G. West
Plant General Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

E. J. Weinkam
Licensing Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Mr. John Gianfrancesco
Manager, Administrative Support
and Special Projects
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. J. A. Stall
Vice President - Nuclear Engineering
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. J. Kammel
Radiological Emergency
Planning Administrator
Department of Public Safety
6000 SE. Tower Drive
Stuart, Florida 34997

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 FACILITY OPERATING LICENSE

Amendment No. 112
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 July 19, 2000, 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, 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 2.C.2 to read as follows:

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 112 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Ronald W. Hernan for

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 28, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 112

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following pages of the Appendix "A" Technical Specifications and Bases 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
XXI	XXI
XXII	XXII
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 4-37a	3/4 3-37a
B 3/4 4-8	B 3/4 4-8
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11

INDEX

LIST OF FIGURES

FIGURE — PAGE

2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT LINES FOUR REACTOR COOLANT PUMPS OPERATING 2-3

2.2-1 LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 1 (FRACTION OF RATED THERMAL POWER VERSUS QR_2) 2-7

2.2-2 LOCAL POWER DENSITY – HIGH TRIP SETPOINT PART 2 (QR_2 VERSUS Y_1) 2-8

2.2-3 THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT PART 1 (Y_1 VERSUS A_1) 2-9

2.2-4 THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1) 2-10

B 2.1-1 AXIAL POWER DISTRIBUTION FOR THERMAL MARGIN SAFETY LIMITS .... B 2-2

3.1-1 MINIMUM BORIC ACID STORAGE TANK VOLUME AS A FUNCTION OF STORED BORIC ACID CONCENTRATION 3/4 1-15

3.1-1a DELETED ....

3.1-2 DELETED ....

3.2-1 DELETED ....

3.2-2 DELETED ....

3.2-3 DELETED ....

4.2-1 DELETED ....

3.2-4 DELETED ....

3.4-1 DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMITS VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1 μCi/GRAM DOSE EQUIVALENT I-131 .... 3/4 4-28

3.4-2 REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITATIONS FOR 21.7 EFPY, HEATUP AND CORE CRITICAL 3/4 4-31a

LIST OF FIGURES (continued)

FIGURE		PAGE
3.4-3	REACTOR CORE SYSTEM PRESSURE – TEMPERATURE LIMITATIONS FOR 21.7 EFPY, COOLDOWN AND INSERVICE TEST	3/4 4-31b
3.4-4	REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITATIONS FOR 21.7 EFPY, MAXIMUM ALLOWABLE COOLDOWN RATES	3/4 4-32
4.7-1	SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST	3/4 7-25
B 3/4.4-1	DELETED	B 3/4 4-10
5.1-1	SITE AREA MAP	5-2
5.6-1a	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT and DECAY TIME, REGION II, 1.3 w/o	5-4B
5.6-1b	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT and DECAY TIME, REGION II, 1.5 w/o	5-4C
5.6-1c	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT and DECAY TIME, REGION I, 1.4 w/o	5-4D
5.6-1d	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT and DECAY TIME, REGION I, 1.82 w/o	5-4E
5.6-1e	REQUIRED FUEL ASSEMBLY BURNUP vs INITIAL ENRICHMENT, REGION I, 2.82 w/o	5-4F
6.2-1	DELETED	6-3
6.2-2	DELETED	6-4

FIGURE 3.4-2
ST. LUCIE-2 P/T LIMITS, 21.7 EFPY
HEATUP AND CORE CRITICAL

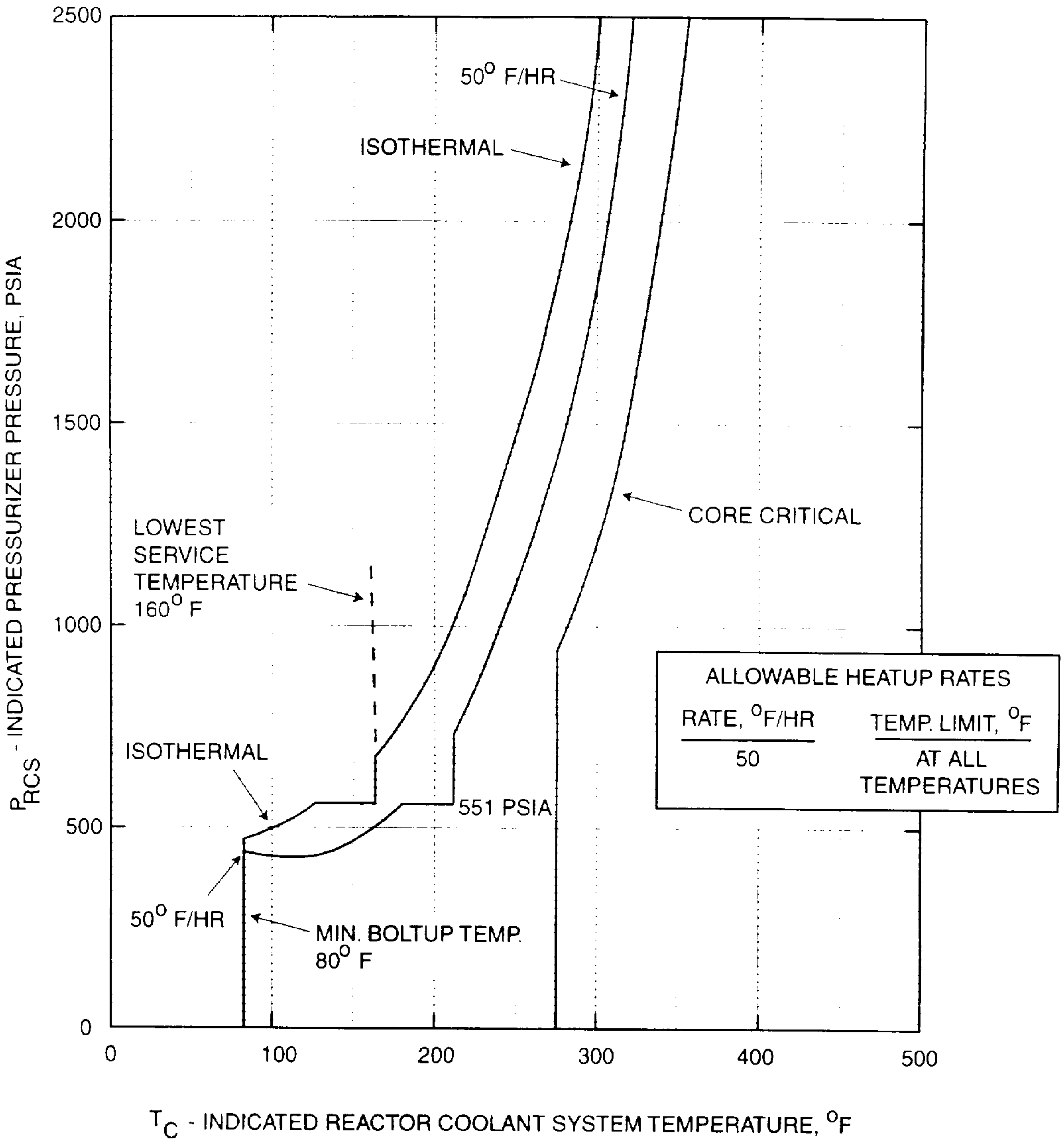


FIGURE 3.4-3
ST. LUCIE-2 P/T LIMITS, 21.7 EFPY
COOLDOWN AND INSERVICE-TEST

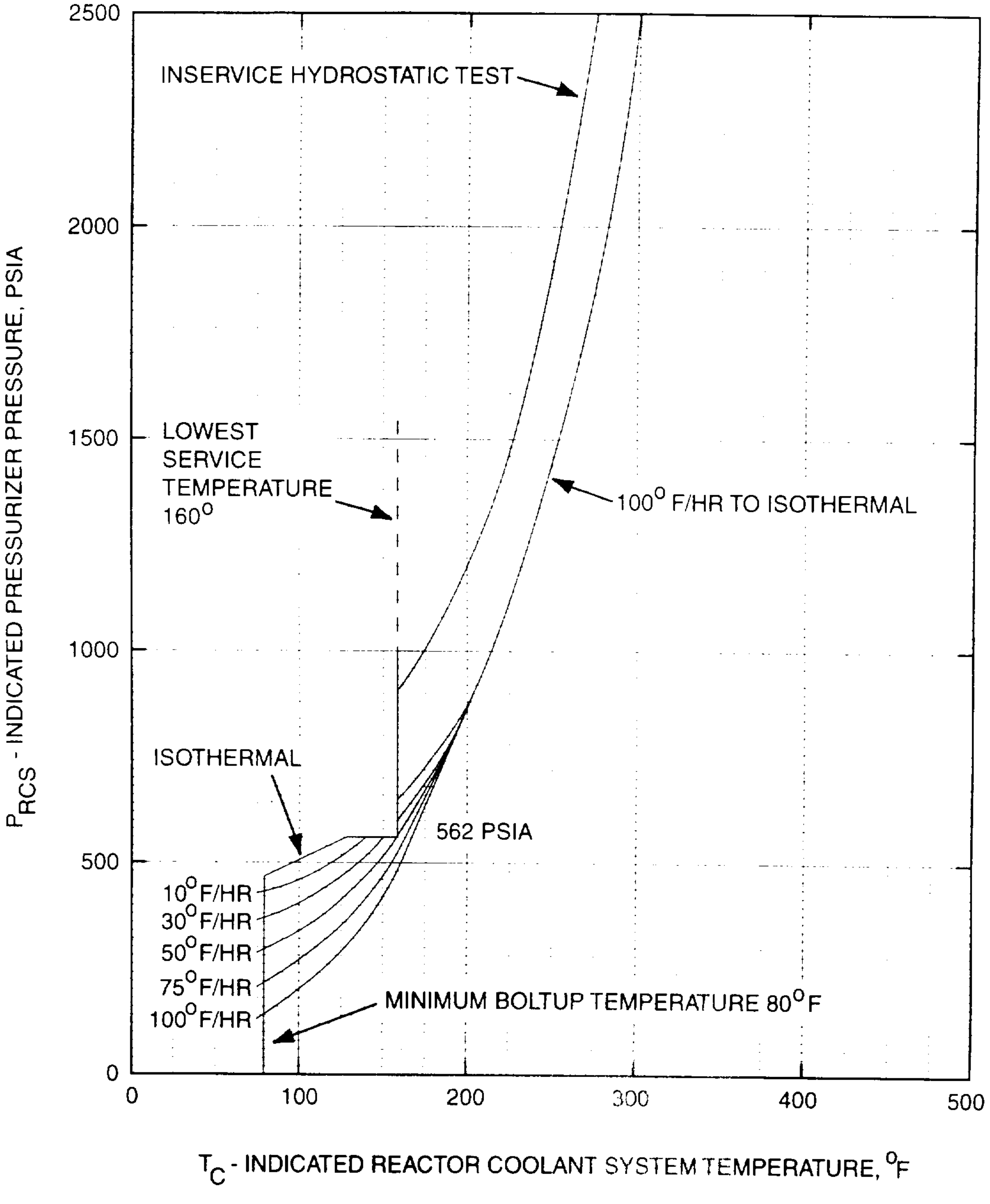
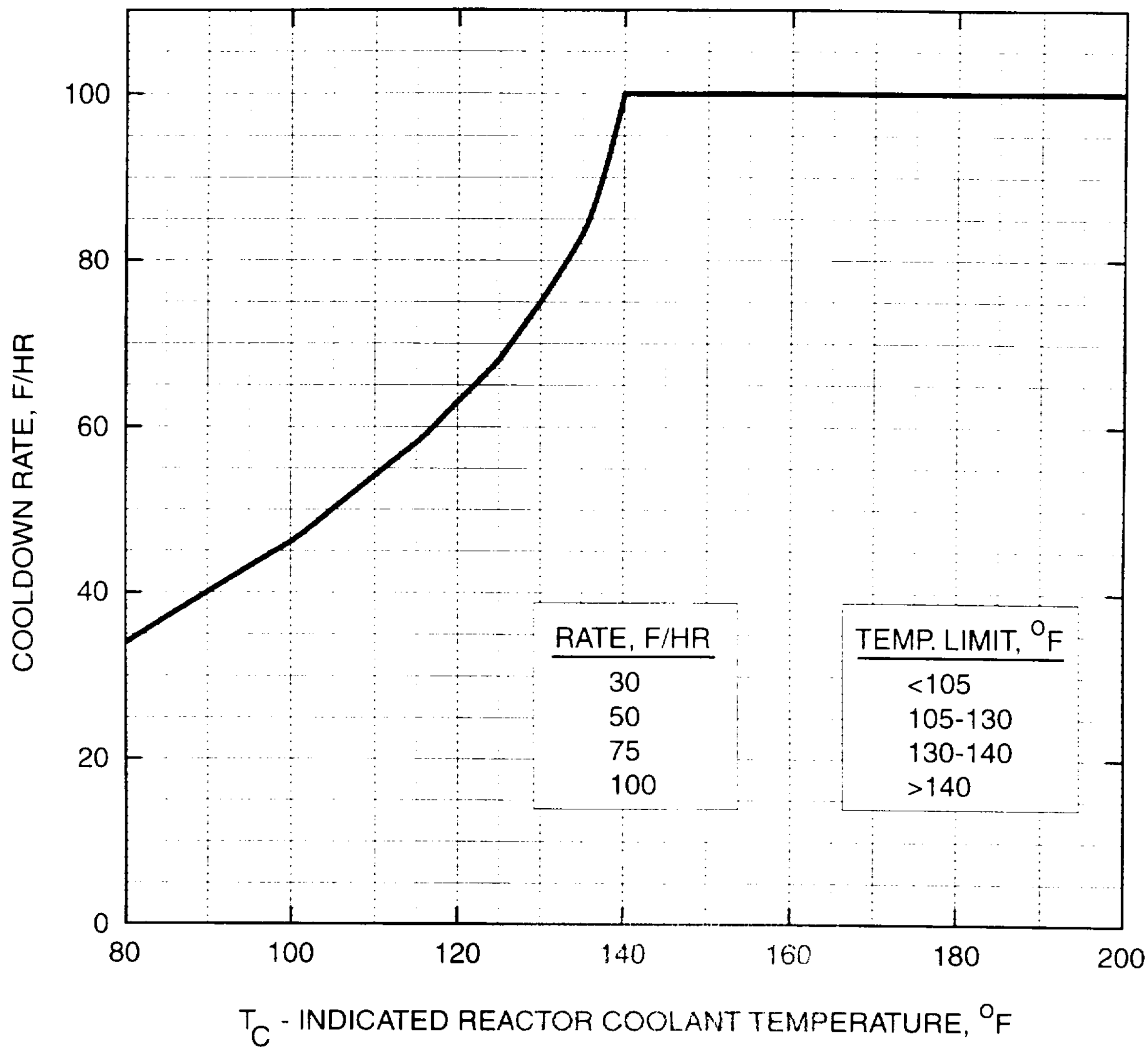


FIGURE 3.4-4
ST. LUCIE-2 P/T LIMITS, 21.7 EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



RATE, F/HR	TEMP. LIMIT, °F
30	<105
50	105-130
75	130-140
100	>140

NOTE: A MAXIMUM COOLDOWN RATE OF 100 F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 140°F.

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
≤ 21.7	≤ 247	≤ 230

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
≤ 21.7	165	165

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron indication damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50 degrees F per hour or cooldown rate of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at 21.7 EFPY, and they include adjustments for pressure differences between the reactor vessel beltline and pressurizer instrument taps.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature can be predicated using a) the initial RT_{NDT}, b) the fluence (E greater than 1 MeV), including appropriate adjustments for neutron attenuation and neutron energy spectrum variations through the wall thickness, c) the copper and nickel contents of the material, and d) the transition temperature shift as recommended by Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or other approved method. The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 include predicted adjustments for this shift in RT_{NDT} at 21.7 EFPY.

DELETED

The actual shift in RT_{NDT} of the vessel materials will be benchmarked periodically during operation, by removing and evaluating, in accordance with 10 CFR 50 Appendix H and ASTM E185, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and the vessel inside radius are essentially identical, the measured transition temperature shift in RT_{NDT} for a set of material samples can be compared to the predications of RT_{NDT} that were used for preparations of the pressure/temperature limits curves. If the measured delta RT_{NDT} values from the surveillance capsule are not conservatively within the measurement uncertainty of the prediction method, then heat up and cooldown curves must be re-evaluated.

The pressure-temperature limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements for Appendix G to 10 CFR 50.

The maximum RT_{NDT} all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 60°F. The Lowest Service Temperature limit line shown on Figures 3.4-2, 3.4-3 and 3.4-4 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, two SDCRVs or an RCS vent opening of greater than 3.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold leg temperatures are less than or equal to the LTOP temperatures. The Low Temperature Overpressure Protection System has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) a safety injection actuation in a water-solid RCS with the pressurizer heaters energized or (2) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 40°F above the RCS cold leg temperatures with the pressurizer water-solid.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated July 19, 2000, Florida Power and Light Company, the licensee, submitted a Technical Specifications (TSs) change request to revise the pressure temperature (P-T) limits for St. Lucie Unit 2. The proposed change involves the revision of the specified effective full power years (EFPY) from 15 EFPY to 21.7 EFPY, while keeping the current P-T limits unchanged. The P-T limits calculations are based on the 1989 American Society of Mechanical Engineers (ASME) Appendix G methodology.

The U.S. Nuclear Regulatory Commission (NRC) has established requirements in Title 10 of the *Code of Federal Regulations* (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. 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) 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. These data are used by the staff as the basis for the staff's review of P-T limit curves and as the basis for the staff's review of pressurized thermal shock assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

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. 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 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 thickness (1/4T) 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 curves are the 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 Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). 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 (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor 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 chemistry factor (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.

2.0 EVALUATION OF FLUENCE FACTOR

The maximum beltline fluence for 15 EFPY was projected to be 1.826E+19 n/cm^2 based on a 24-month fuel cycle. However, since this 24-month fuel cycle with higher flux has never been implemented and its projected fluence has never been accumulated, the licensee decided to revise the EFPY for the same fluence of 1.826E+19 n/cm^2 based on the actual lower flux 18-month cycles and the actual fluence data collected from Cycle 5 to Cycle 10, so that the current P-T limits with a revised EFPY could still be used.

The licensee used the results of the most recently removed and tested Unit 2 surveillance capsule[1] to estimate the time required for the peak inside vessel diameter to reach 1.826×10^{19} n/cm^2 for which the current pressure temperature curves were evaluated. The rate of peak fluence accumulation was estimated at 0.0885×10^{19} n/cm^2/EFPY. This value was derived for the upcoming St. Lucie Unit 2 loadings, and the staff finds it to be conservative.

The intent of the proposed TS change is to utilize the previously estimated fluence values for 15 EFPYs and not reanalyze until this fluence value has been reached. At the end of the current cycle the reactor will have 15.421 EFPYs and the actual fluence will be 1.27×10^{19} n/cm^2. At the rate of 0.0885×10^{19} n/cm^2/EFPY, the reactor will reach 1.826×10^{19} n/cm^2 in an additional 6.28 EFPYs ((1.826 - 1.27)/0.0885 = 6.28 EFPYs), which equates to 21.701 EFPYs (15.421 + 6.28 = 21.701 EFPYs). The staff finds the requested extension to be acceptable because the current pressure temperature curves were evaluated for the fluence value to be attained at the end of the 21.7 EFPYs.

---

[1]WCAP-15404, "Analysis of Capsule 263° from the Florida Power and Light St. Lucie Unit 2 Reactor Vessel Radiation Surveillance Program" dated April 1998.

concluded that "the new 21.7 EFPY

3.0 EVALUATION OF PRESSURE/TEMPERATURE LIMITS

The licensee used the methodology of Appendix G in 10 CFR Part 50 to calculate the current P-T limits for St. Lucie Unit 2. Based on the new fluence projection of 1.27E+19 n/cm^2 at the end of Cycle 12 with 15.421 EFPY and a projected fluence of 8.846E+17 n/cm^2 per EFPY thereafter, the licensee determined that it takes 6.285 EFPY for the limiting beltline material to reach the fluence of 1.826E+19 n/cm^2. The licensee then added this additional 6.285 EFPY to the accumulated 15.421 EFPY at the end of Cycle 12 to arrive at the final EFPY of 21.7 EFPY for the fluence of 1.826E+19 n/cm^2. The "21.7 EFPY" is specified in the proposed P-T limit curves, which are identical to the current P-T limit curves, except for this revised EFPY value. Hence, the ART for the limiting beltline material, which was based on the Chemistry Table of RG 1.99, Rev. 2, remains unchanged for the current P-T limits. In 1998, surveillance data from the second capsule became available, and the licensee evaluated and determined that the limiting plate surveillance data is credible. Considering this new information, the licensee concluded that "the new 21.7 EFPY period of applicability is inherently conservative because the full margin term of 34°F (plate) was used to determine ART for the period of applicability, with no reduction taken as a result of the credible surveillance data."

The NRC staff has reviewed and accepted the revised EFPY and its corresponding fluence. Since the proposed P-T limits only involve the revision of the specified EFPY without changing the P-T limits, the staff has confirmed that using the ART based on the Chemistry Table of RG 1.99, Rev. 2 (licensee's current and proposed approach) is conservative. This is established by demonstrating that the ART based on surveillance data (required by the regulation) is less than the licensee's ART. The staff has evaluated the surveillance data and summarized the results and findings in the attached Table. Based on the information in the Table, the staff concluded that all surveillance data is credible. Further, the staff calculated the ART for the limiting plate using the CF that was derived from the surveillance data and using half of the margin. The ART from this effort is 136.5 (30 + 17 + 87.42x1.0236), less than the licensee's ART of 140 (30 +34 + 74.2x1.0236) based on the Chemistry Table of RG 1.99, Rev. 2. Hence, the licensee's approach is conservative and the P-T limits continue to satisfy Appendix G requirements in 10 CFR Part 50, including the minimum temperature requirement for the closure head flange material during normal operation and inservice leak and hydrostatic testing.

3.1 Reactor Vessel Integrity Database (RVID) Updating

The submittal referred to surveillance data from the W-263 capsule, which was withdrawn in 1998. This information is not available in the RVID and should be included in the next RVID update. The attached Table contains the complete and the most recent information for all surveillance data from the surveillance report WCAP-15040. Some RVID data regarding the surveillance data from the W-83 capsule should also be updated.

4.0 SUMMARY

The staff finds the extension of the EFPY from 15 EFPY to 21.7 EFPY to be acceptable because the current pressure temperature curves were evaluated for the fluence value to be attained at the end of the 21.7 EFPYs. The staff has also determined that the proposed P-T limits for the reactor coolant system for heatup, cooldown, hydrotest, and criticality satisfy the requirements in Appendix G to Section XI of the ASME Code and Appendix G of 10 CFR Part 50 for 21.7 EFPYs for St. Lucie Unit 2. The proposed P-T limits satisfy GL 88-11, because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P-T limit curves may be incorporated into the St. Lucie Unit 2 TSs.

5.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. NRC, the State of Florida does not desire notification of issuance of license amendments.

6.0 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. 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 amendments involve no significant hazards consideration and there has been no public comment on such finding (65 FR 51354). 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 amendments.

7.0 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 Contributors: L. Lois, SRXB, NRR
S. Sheng, EMCB, NRR

Date: December 28, 2000

Attachment: Surveillance Data

TABLE

Documentation and Evaluation of All Surveillance Data Accumulated So Far

	Capsule and lead factor	Fluence (10E19 n/cm.cm)	FF (Fluence Factor)	ΔRT^{ndt} (°F)	FF x ΔRT^{ndt}	FF x FF	Scatter/ ΔRT^{ndt} -CFxFF (°F)	Credibility	USE (ft-lb)
Plate-L	W-83,1.3	0.18	0.5445	45.1	24.55	0.2965	-2.5	Credible	119
		0.00							134
Plate-T	W-83,1.3	0.18	0.5445	29.8	16.23	0.2965	-17.8	Credible	101.5
Plate-T	W-263,1.27	1.24	1.060	103.1	109.29	1.1236	10.4	Credible	79.0
		0.00							103.3
Σ					150.07	1.7166			
Σ(FF x ΔRT^{ndt})/Σ(FF x FF)					87.42 ← CF				
Weld	W-83,1.3	0.18	0.5445	13.8	7.51	0.2965	0.35	Credible	102.6
Weld	W-263,1.27	1.24	1.060	26.0	27.56	1.1236	-0.18	Credible	108.2
		0.00							114.7
Σ					35.07	1.4201			
Σ(FF x ΔRT^{ndt})/Σ(FF x FF)					24.70 ← CF				

Attachment