

Mr. J. H. Goldberg  
 President - Nuclear Division  
 Florida Power and Light Company  
 P.O. Box 14000  
 Juno Beach, Florida 33408-0420

October 27, 1995

SUBJECT: ST. LUCIE UNIT 1 - ISSUANCE OF AMENDMENT RE: REACTOR COOLANT  
 SYSTEM PRESSURE/TEMPERATURE LIMITS (TAC NO. M92412)

Dear Mr. Goldberg:

The Commission has issued the enclosed Amendment No. 141 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. This amendment consist of changes to the Technical Specifications in response to your application dated May 17, 1995.

This amendment will extend the applicability of the current Reactor Coolant System (RCS) Pressure/Temperature Limits and maximum allowed RCS heatup and cooldown rates to 23.6 Effective Full Power Years (EFPY) of operation. In addition, administrative changes were proposed for TS 3.1.2.1 (Boration Systems Flow Paths-Shutdown) and TS 3.1.2.3 (Charging Pump-Shutdown) to clarify the conditions for which a High Pressure Safety Injection pump may be used.

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,

Original signed by:

Jan A. Norris, Senior Project Manager  
 Project Directorate II-1  
 Division of Reactor Projects - I/II  
 Office of Nuclear Reactor Regulation

Docket No. 50-335

Enclosures:

1. Amendment No. 141 to DPR-67
2. Safety Evaluation

cc w/enclosures: See next page

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| OFFICE | LA:PDII-1  | PM:PDII-1 | OGC NLO  | D:PDII-1  |  |
| NAME   | Dunnington | JNorris   | M20BCST  | DMatthews |  |
| DATE   | 10/6/95    | 10/17/95  | 10/12/95 | 10/26/95  |  |
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Mr. J. H. Goldberg  
Florida Power and Light Company

St. Lucie Plant

cc:

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DATED: October 27, 1995

AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1

Distribution

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141  
License No. DPR-67

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 May 17, 1995, 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. DPR-67 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) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 141, 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION



David B. Matthews, Director  
Project Directorate II-1  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: October 27, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 141  
TO FACILITY OPERATING LICENSE NO. DPR-67  
DOCKET NO. 50-335

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages

3/4 1-8  
3/4 1-9a  
3/4 1-12  
3/4 4-23a  
3/4 4-23b  
3/4 4-23c

Insert Pages

3/4 1-8  
3/4 1-9a  
3/4 1-12  
3/4 4-23a  
3/4 4-23b  
3/4 4-23c

## REACTIVITY CONTROL SYSTEMS

### MINIMUM TEMPERATURE FOR CRITICALITY

### LIMITING CONDITION FOR OPERATION

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3.1.1.5 The Reactor Coolant System lowest operating loop temperature ( $T_{avg}$ ) shall be  $\geq 515^{\circ}\text{F}$  when the reactor is critical.

APPLICABILITY: MODES 1 and 2#.

#### ACTION:

With a Reactor Coolant System operating loop temperature ( $T_{avg}$ )  $< 515^{\circ}\text{F}$ , restore  $T_{avg}$  to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

### SURVEILLANCE REQUIREMENTS

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4.1.1.5 The Reactor Coolant System temperature ( $T_{avg}$ ) shall be determined to be  $\geq 515^{\circ}\text{F}$ .

- a. Within 15 minutes prior to achieving reactor criticality, and
- b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System temperature ( $T_{avg}$ ) is  $< 525^{\circ}\text{F}$ .

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# With  $K_{eff} \geq 1.0$ .

## **REACTIVITY CONTROL SYSTEMS**

### **3/4.1.2 BORATION SYSTEMS**

#### **FLOW PATHS - SHUTDOWN**

#### **LIMITING CONDITION FOR OPERATION**

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- 3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.
- A flow path from the boric acid makeup tank via either a boric acid pump or a gravity feed connection and any charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a is OPERABLE, or
  - The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump\* to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b is OPERABLE.

**APPLICABILITY:** MODES 5 and 6.

#### **ACTION:**

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

#### **SURVEILLANCE REQUIREMENTS**

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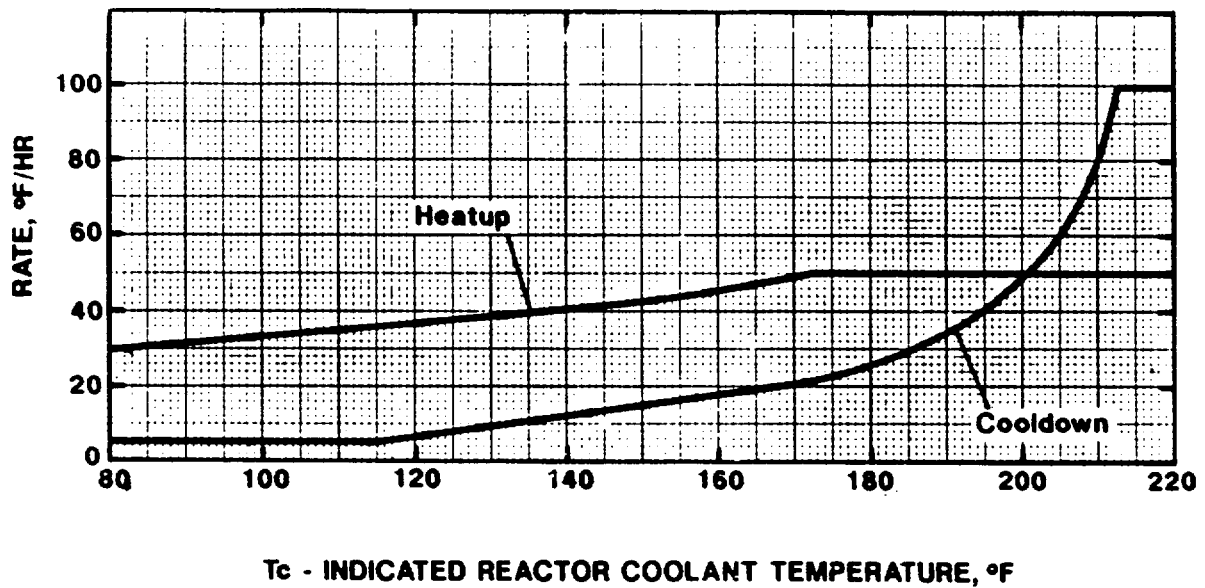
- 4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:
- At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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\* The flow path from the RWT to the RCS via a single HPSI pump shall only be established if: (a) the RCS pressure boundary does not exist, or (b) no charging pumps are operable. In the latter case: 1) all charging pumps shall be disabled; 2) heatup and cooldown rates shall be limited in accordance with Figure 3.1-1b; and 3) at RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

| <b>High Pressure Header</b> | <b>Auxiliary Header</b> |
|-----------------------------|-------------------------|
| HCV-3616                    | HCV-3617                |
| HCV-3626                    | HCV-3627                |
| HCV-3636                    | HCV-3637                |
| HCV-3646                    | HCV-3647                |





**FIGURE 3.1-15**  
**MAXIMUM ALLOWABLE HEATUP AND COOLDOWN RATES,**  
**SINGLE HPSI PUMP IN OPERATION**  
**(23.6 EFY)**

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# **REACTIVITY CONTROL SYSTEMS**

## **CHARGING PUMPS - SHUTDOWN**

### **LIMITING CONDITION FOR OPERATION**

3.1.2.3 At least one charging pump or high pressure safety injection pump\* in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

**APPLICABILITY:** MODES 5 and 6.

#### **ACTION:**

With no charging pump or high pressure safety injection pump\* OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

### **SURVEILLANCE REQUIREMENTS**

4.1.2.3 At least one of the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2571 ft. when tested pursuant to Specification 4.0.5.

\* The flow path from the RWT to the RCS via a single HPSI pump shall be established only if: (a) the RCS pressure boundary does not exist, or (b) no charging pumps are operable. In the latter case: 1) all charging pumps shall be disabled; 2) heatup and cooldown rates shall be limited in accordance with Figure 3.1-1b; and 3) at RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

#### **High Pressure Header**

HCV-3616  
HCV-3626  
HCV-3636  
HCV-3646

#### **Auxiliary Header**

HCV-3617  
HCV-3627  
HCV-3637  
HCV-3647

**FIGURE 3.4-2a**  
**ST. LUCIE UNIT 1 P/T LIMITS, 23.6 EFY**  
**HEATUP AND CORE CRITICAL**

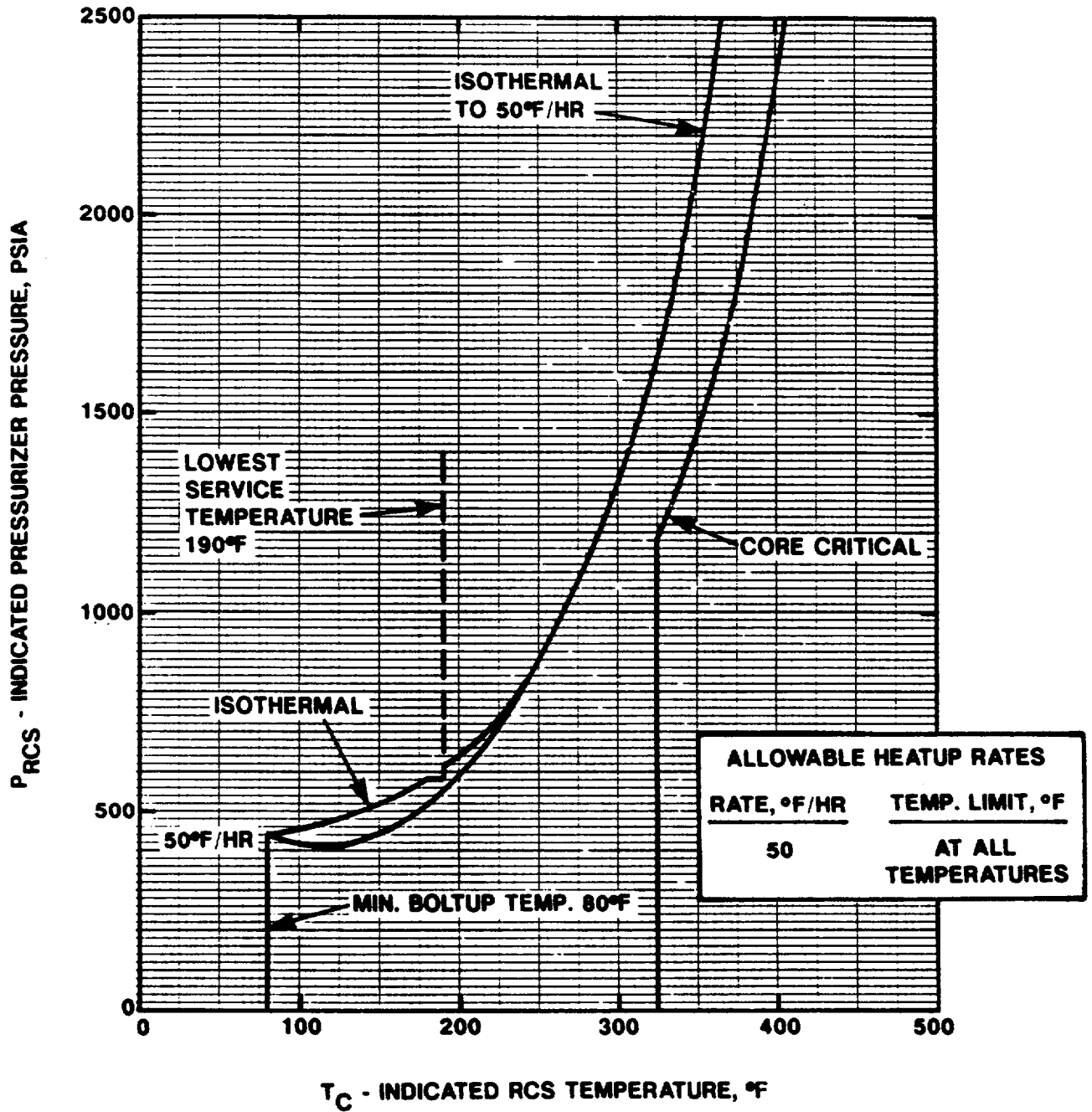
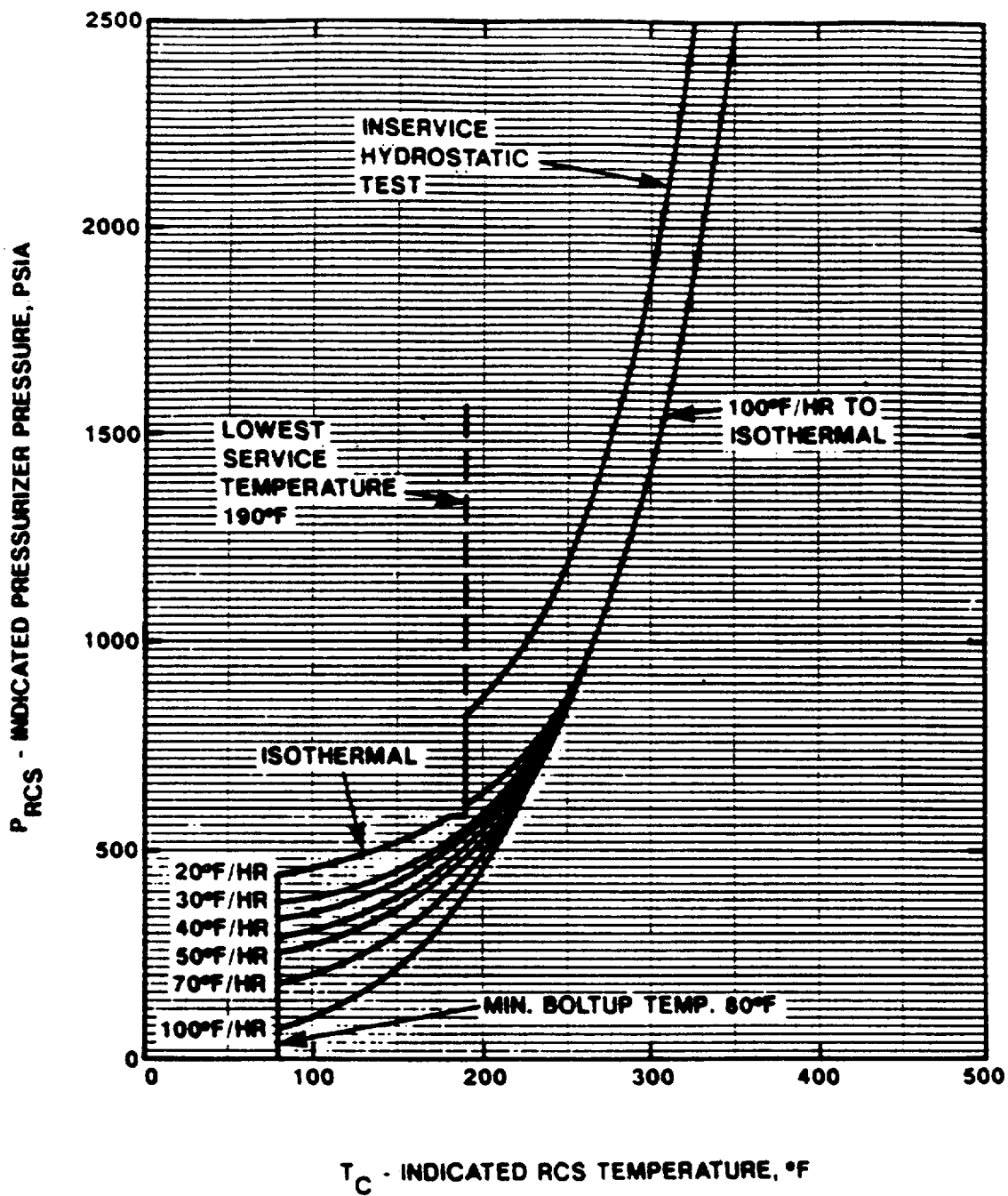
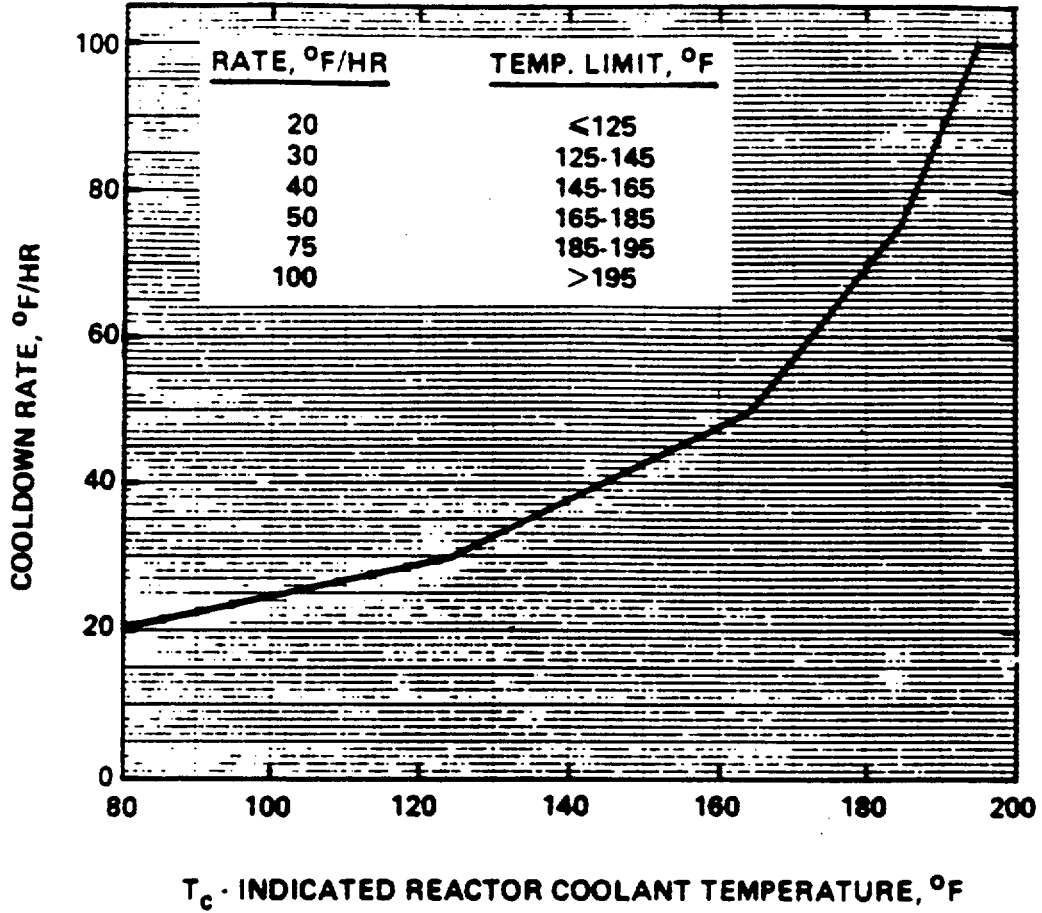


FIGURE 3.4-2b  
 ST. LUCIE UNIT 1 P/T LIMITS, 23.6 EFY  
 COOLDOWN AND INSERVICE TEST



**FIGURE 3.4-3  
ST. LUCIE UNIT 1, 23.6 EPY  
MAXIMUM ALLOWABLE COOLDOWN RATES**



**NOTE: A MAXIMUM COOLDOWN RATE OF  
100°F/HR IS ALLOWED AT ANY  
TEMPERATURE ABOVE 195°F**



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

By letter dated May 17, 1995, the Florida Power and Light Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the St. Lucie 1 Technical Specifications, Section 3.4. This request stems from new information on beltline materials data, which becomes available as a result of Generic Letter (GL) 92-01 review. The proposed P/T limits were requested for 23.6 effective full power years (EFPY), and were developed using Regulatory Guide (RG) 1.99, Revision 2. Extending the applicability of the pressure/temperature limit curves affects Technical Specifications 3.4.9.1 with Figures 3.4-2a, 3.4-2b and 3.4-3, Technical Specifications 3.1.2.1 and 3.1.2.3 with Figure 3.1-1b. Technical Specifications 3.1.2.1 and 3.1.2.3 are also being amended to remove the requirement to verify that two of four valves in the operable high head safety injection flow path are shut with power removed, when in Modes 5 and 6, with a high head safety injection pump in operation and the reactor coolant pressure boundary not established.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and GL 88-11. GL 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper-shelf energy (USE). GL 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

## 2.0 EVALUATION

### 2.1 Neutron Irradiation Embrittlement

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the St. Lucie 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 23.6 EFPY is the lower longitudinal weld (wire Heat No. 305424) with 0.28% copper (Cu), 0.63% nickel (Ni), and an initial  $RT_{ndt}$  of  $-60^{\circ}\text{F}$ . The ART calculated by the staff for this limiting material is  $191.5^{\circ}\text{F}$  at  $1/4T$  ( $T$  = reactor vessel beltline thickness) and  $137.1^{\circ}\text{F}$  at  $3/4T$  at 23.6 EFPY. The corresponding values calculated by the licensee, using Chemistry Factor Table in Section 1.1 of RG 1.99, Rev. 2, are  $191^{\circ}\text{F}$  at  $1/4T$  and  $136^{\circ}\text{F}$  for  $3/4T$ . Both calculations used a neutron fluence of  $1.063\text{E}19$   $\text{n}/\text{cm}^2$  at  $1/4T$  and  $3.78\text{E}18$   $\text{n}/\text{cm}^2$  at  $3/4T$ .

The staff judges that a difference of  $0.5^{\circ}\text{F}$  between the licensee's ART of  $191^{\circ}\text{F}$  and the staff's ART of  $191.5^{\circ}\text{F}$  is negligible. Further, since the licensee back-calculated the fluence in this application based on the ART of  $191^{\circ}\text{F}$ , which is the same as that appeared in the 1989 P/T limits submittal, the proposed P/T limits for heatup, cooldown, and hydrotest are identical to the current P/T limit curves and meet the beltline material requirements in Appendix G of 10 CFR Part 50. The only changes made in Figure 3.4-2a and Figure 3.4-2b by the licensee are EFPY values in the legends associated with these P/T limit curves.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.A.2 of Appendix G states that when the pressure



exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Since the flange reference temperature of 30°F remains the same as that in the 1989 submittal and the proposed higher fluence value does not enter the equation for determining P/T limits for flange materials, the staff has determined that the proposed P/T limits satisfy Section IV.A.2 of Appendix G.

Section IV.A.1 of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The limiting material is the intermediate shell with 76 ft-lb. Using RG 1.99, Rev. 2, the staff calculated that the end of life USE will be 58 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

## 2.2 Use of the High Head Safety Injection Pump

The power-operated relief valves on the pressurizer provide the low temperature overpressure protection (LTOP) for the St. Lucie Unit 1 reactor coolant system. The staff reviewed the LTOP setpoints and documented the review in a safety evaluation dated June 11, 1990 for St. Lucie in support the license amendment No. 104. The review assured that the LTOP setpoints were adequate to prevent a violation of Appendix G heatup and cooldown curves should a reactor coolant system pressure transient occur. The license amendment requested by the May 17, 1995, letter does not change the LTOP or the pressure/temperature limits but rather extends the current limits to 23.6 EFPY. The current LTOP setpoints remain adequate.

The limiting mass addition and energy addition transients have been analyzed. The mass addition transient is based on the maximum possible pumping capacity available. Technical Specifications 3.1.2.1 and 3.1.2.3 have a footnote that requires two of four isolation valves in the operable high pressure safety injection header be verified closed with power removed when a high head safety injection pump is being used for the boration flow path. This requirement assures the amount of water that can be pumped into the RCS is limited while one of these pumps is in operation. The current requirement is that these valves be verified closed with power removed whenever the high head safety injection pump is in service. The amendment requires that the valves be verified closed with power removed only when the high head safety injection pump is in service and when RCS pressure boundary exists. Removing the requirement to verify the two valves are closed with power removed is acceptable because only when RCS pressure boundary is established can a mass addition transient cause an over pressure event. If the RCS pressure boundary is not established, there is no chance of over pressurizing the RCS. The burden of verifying the valves are closed and power is removed is, therefore, not necessary when RCS pressure boundary is not established. The licensee proposed changes to Technical Specifications 3.1.2.1 and 3.1.2.3 are acceptable.

### 3.0 TECHNICAL FINDINGS

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 23.6 EFPY because the proposed limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy GL 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. Hence, the proposed P/T limits may be incorporated into the St. Lucie 1 Technical Specifications.

### 4.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988
2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits
3. May 17, 1995, letter from D. A. Sager (FPL) to USNRC Document Control Desk, subject: Proposed License Amendment, RCS Pressure/Temperature Limits
4. December 5, 1989, letter from J.H. Goldberg (FPL) to USNRC Document Control Desk, subject: Proposed License Amendment, P-T Limits and LTOP Analysis

### 5.0 STATE CONSULTATION

Based upon the written notice of the proposed amendment, the Florida State official had no comments.

### 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 amendment involves 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 (60 FR 32362). 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.

**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 Contributor: S. Sheng

Date: October 27, 1995