November 25, 1994

DISTRIBUTION
See attached sheet

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

SUBJECT: ST. LUCIE UNIT 1 - ISSUANCE OF AMENDMENT RE: REDUCTION OF REACTOR COOLANT SYSTEM DESIGN FLOWRATE (TAC NO. M86064)

Dear Mr. Goldberg:

The Commission has issued the enclosed Amendment No. 130 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated March 19, 1993 and augmented by letter dated August 18, 1994.

This amendment allows a reduction in Reactor Coolant System design flowrate from the current value of 370,000 gpm to 355,000 gpm in Technical Specifications Figure 2.1-1, and Tables 2.2-1 and 3.2-1 and the associated Bases.

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 B. Buckley For)

Jan A. Norris, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-335

Enclosures:
1. Amendment No. 130 to DPR-67
2. Safety Evaluation

cc w/enclosures: See next page

FILENAME - G:\AUTOS\WPDOCS\SL86064.AMD

OFFICE	LA:PDII-2	PM:PDII-2	AD:PDII-2	OGC NLO	
NAME	Dunnington	JNorris	MThadani	M Zobler	
DATE	11/22/94	11/21/94	11/21/94	11/23/94	
COPY	Yes/No	Yes/No	Yes/No	Yes/No	

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St. Lucie Plant
Units 1 and 2

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DATED: November 25, 1994

AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1

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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. 130
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 March 19, 1993 and augmented August 18, 1994, 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. 130, 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

Mohan C Thadani

Mohan C. Thadani, Acting Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 25, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 130

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	Insert Pages
2-2	2-2
2-4	2-4
B 2-1	B 2-1
B 2-2	B 2-2
3/4 2-14	3/4 2-14

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

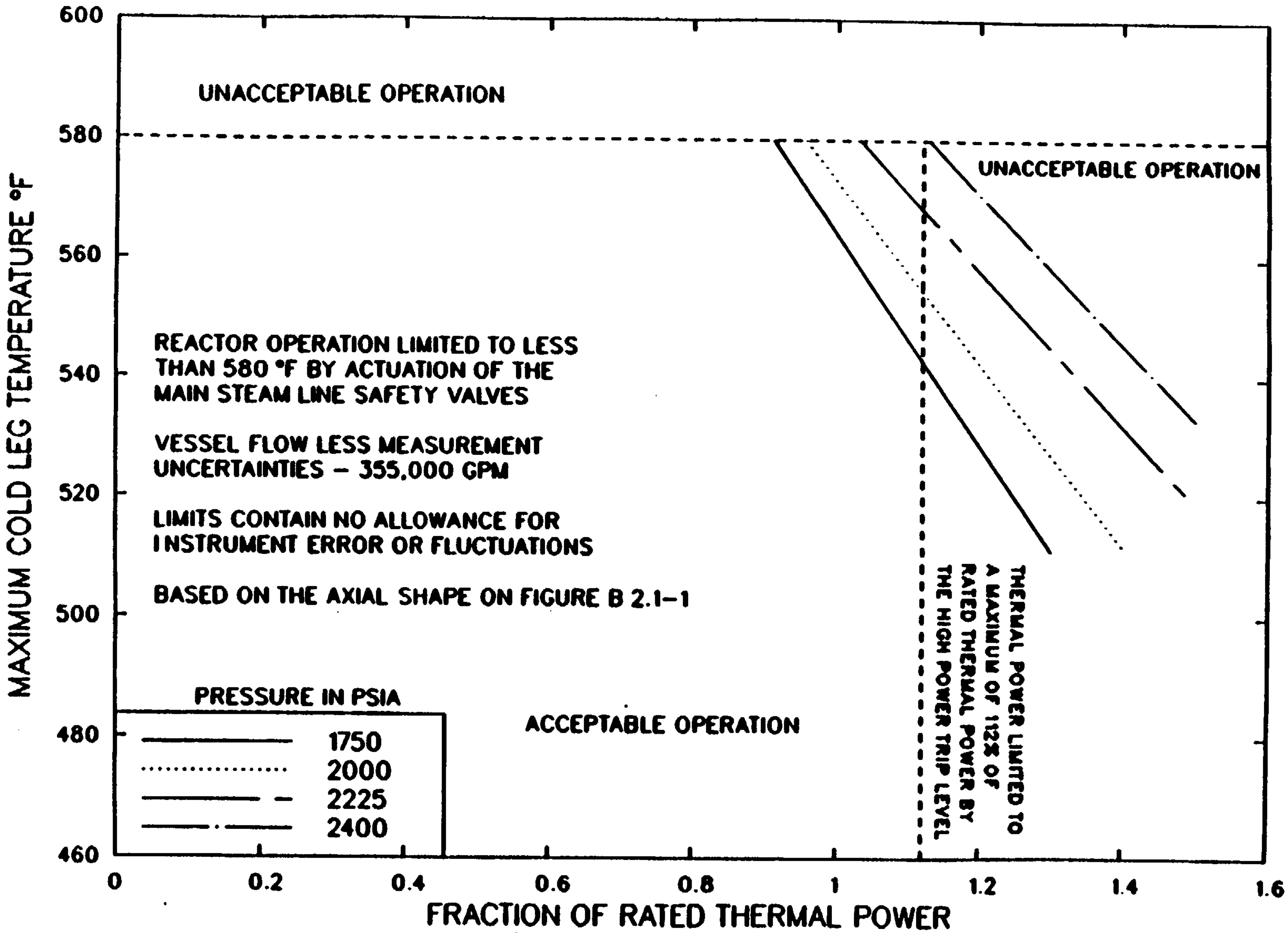

FIGURE 2.1–1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT –
FOUR REACTOR COOLING PUMPS OPERATING

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High (1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, a(a minimum setpoint of 15% of RA ED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	≥ 95% of design reactor coolant flow with 4 pumps operating*	≥ 95% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure - High	≤ 3.3 psig	≤ 3.3 psig
6. Steam Generator Pressure - Low (2)	≥ 600 psia	≥ 600 psia
7. Steam Generator Water Level -Low	≥ 20.5% Water Level - each steam generator	≥ 19.5% Water Level - each steam generator
8. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

*Design reactor coolant flow with 4 pumps operating is 355,000 gpm.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.
9a. Steam Generator Pressure Difference High (1) (logic in TM/LP)	$\leq$ 135 psid	$\leq$ 135 psid
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	$\geq$ 800 psig	$\geq$ 800 psig
11. Rate of Change of Power - High (4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 1% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the Siemens Power Corporation (SPC) XNB correlation. The XNB DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.22 using the XNB DNBR correlation. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than the DNBR limit for the axial shape shown in Figure B 2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.1-1. The area of safe operation is below and to the left of these lines.

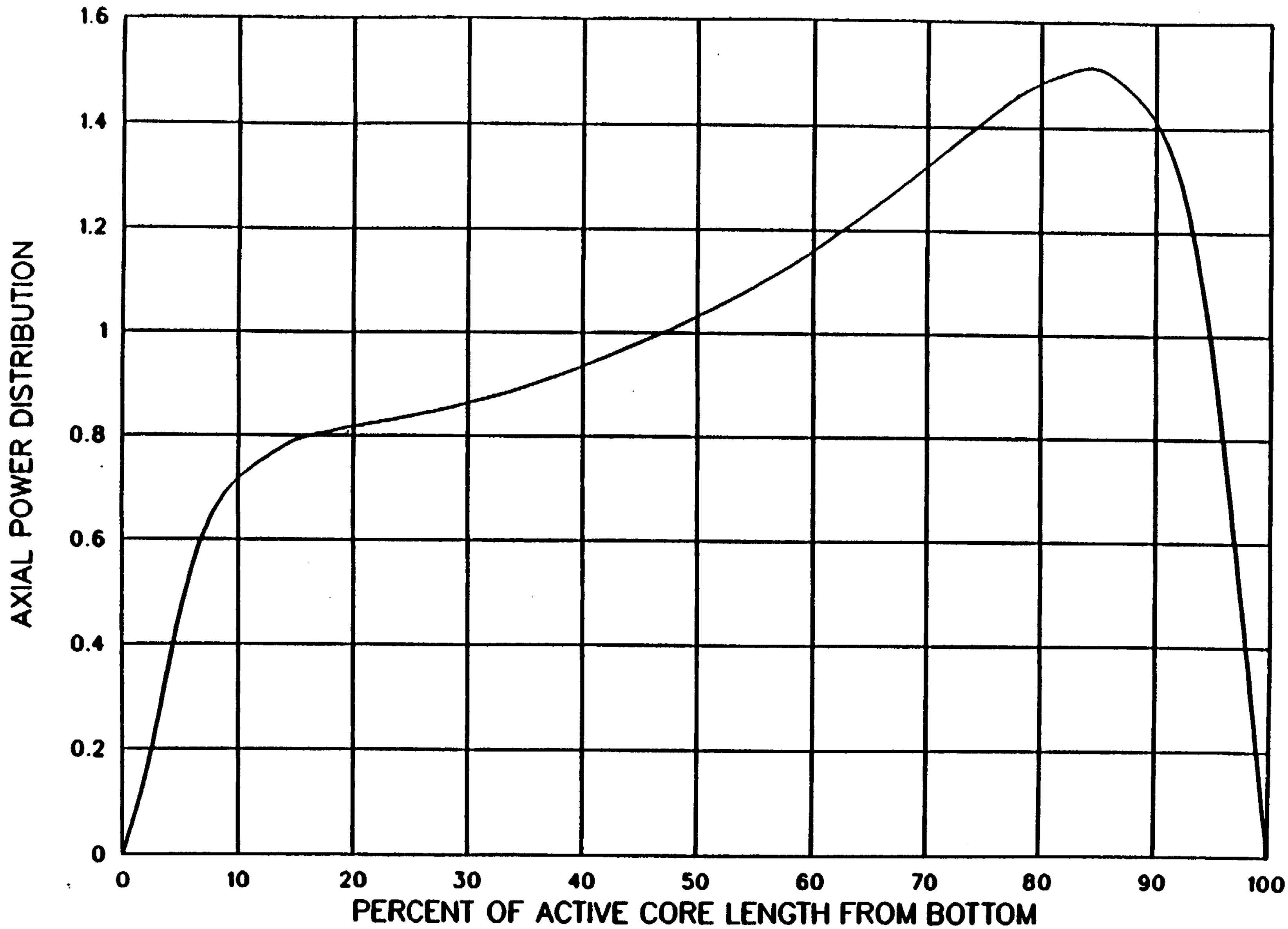

FIGURE B 2.1-1 AXIAL POWER DISTRIBUTION FOR THERMAL MARGIN SAFETY LIMITS

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to $\leq$ 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits by instrument readout at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

TABLE 3.2-1

DNB MARGIN

LIMITS

Parameter	Four Reactor Coolant Pumps Operating
Cold Leg Temperature	$\leq 549°F$
Pressurizer Pressure	$\geq$ 2225 psia*
Reactor Coolant Flow Rate	$\geq$ 355,000 gpm
AXIAL SHAPE INDEX	Figure 3.2-4

*Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 130

TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

By letter dated March 19, 1993 (Reference 1), and supplement dated August 18, 1994, Florida Power and Light (FPL) the licensee for St. Lucie Power Plant (PSL1), proposed changes to the Technical Specifications (TS) for the Unit 1 facility. The supplemental information did not change the initial no significant hazards consideration determination. Specifically, the proposed changes are to TS figure 2.1-1, Tables 2.2-1 and 3.2-1 and the associated TS Bases pages B2-1 and B2-2. These changes are related to reducing the reactor coolant system (RCS) minimum design flow to remove core thermal power from 370,000 gpm to 355,000 gpm to accommodate future steam generator tube plugging (SGTP) beyond the current 15% to up to 25%.

2.0 EVALUATION

The licensee determined that the proposed changes could affect the plant safety analysis in three major ways. First, the reduction in the reactor flow rate can impact the calculated Departure from Nucleate Boiling Ratio (DNBR) for some transients. If Departure from Nucleate Boiling (DNB) occurs, the thermal margin (TM) for the corresponding transient is also reduced. Second, the removal of additional SG tubes from service reduces the primary to secondary heat transfer area in the steam generators. And finally, a reduction in RCS flowrate results in a corresponding increase in RCS average coolant temperature (T_{ave}) which can impact both DNBR-related and loss of primary inventory types of events.

FPL reviewed the events described in their Updated Final Safety Analysis Report, for PSL1, to assess the impact of increasing the SGTP to 25% and reducing the flow rate to 355,000 gpm. The transient events were divided into three categories: (1) reanalysis required, (2) reanalysis not required, and (3) other analyses affected by the flow reduction.

2.1 Transient Events Requiring Reanalysis

The transients that required reanalysis were Loss of External Load, 4-Pump Loss of Reactor Coolant Flow, Seized RCP Rotor, Dropped Control Element Assembly (CEA), and Large and Small Break LOCAs.

2.1.1 Loss of External Load (LOEL)

Loss of external load is the limiting event in the Decrease in Heat Removal by the Secondary System class of events. This event can be impacted by the proposed change in two ways - challenging the DNBR acceptance criteria and the maximum calculated RCS pressure.

Increasing the SGTP and reducing the initial RCS flow will tend to reduce the calculated Minimum Departure from Nucleate Boiling Ratio (MDNBR) for DNB-related events and FPL has concluded, by its reanalysis, that the LOEL event still remains acceptable with respect to DNBR.

The licensee reanalyzed the event to determine the impact on the RCS pressure. This analysis resulted in a peak RCS pressure of 2701 psia, which is within the American Society of Mechanical Engineers (ASME) Code acceptance criteria of 2750 psia. The secondary steam pressure was calculated to be 1022 psia, within the acceptance criteria of 1100 psia.

2.1.2 Decrease in RCS Flow

Two events that result in reduced RCS flow were reanalyzed - Loss of Reactor Coolant Flow (4 Pump Coastdown) and Locked Rotor.

The Loss of Flow (LOF) event is the limiting Anticipated Operational Occurrence (AOO) for MDNBR for this category of transients. The licensee reanalyzed this event using NRC approved methods. The analysis included the event cases with and without the proposed design flow and allowable core power as a function of Axial Shape Index (ASI). The current margin to the acceptance criteria is 12%. The licensee indicated that the proposed changes reduce the margin less than 0.5%.

The locked rotor event was not reanalyzed. Instead, a set of conservative boundary conditions was assumed using the original system calculation as a basis to perform an evaluation. Their analytical approach for locked rotor events is part of the same reload methodology that has been used to evaluate previous operating cycles with Siemens fuel at St. Lucie. The staff, therefore, finds that the approach continues to be applicable for this evaluation.

The proposed rated RCS flow of 355,000 gpm, with an uncertainty of ± 14,000 gpm, was used in the evaluation. The licensee predicted that 1% of the fuel rods will experience DNB. This value remains bounded by the licensing basis radiological consequences analysis assumption of 2.5%.

2.1.3 Dropped Control Element Assembly

The dropped CEA is an AOO that ultimately affects the DNBR by changing the radial core power distribution. This event was reanalyzed using the same NRC approved methods that are used for refueling analysis incorporating the proposed RCS design flow and allowable values of core power as a function of ASI.

The licensee indicated that the reanalysis resulted in a conservative margin of more that 13% to the limiting acceptance criterion of MDNBR 1.22. Therefore, the licensee concluded that the proposed changes would not violate the specified acceptable fuel design limit for DNBR.

2.1.4 Loss-of-Coolant Accident Analyses

In submittals of March 19, 1993, and August 18, 1994, the licensee provided the results of loss-of-coolant accident (LOCA) analyses in support of the proposed St. Lucie Unit 1 reduction in RCS design flow license amendment. These analyses were performed using Siemens Power Corporation approved large break (LB) and small break (SB) LOCA evaluation models (EM).

Large Break LOCA

In the March 19, 1993 submittal, the licensee provided the results of LB LOCA analyses, performed using the Siemens LB LOCA EM as approved by the NRC on July 8, 1986 (Reference 3). The analyses were performed to account for the St. Lucie Unit 1 plant design with average steam generator tube plugging of 25%, with as much as 32% plugging in either steam generator.

The licensee has indicated that calculations support its conclusion that the limiting type and location of large breaks continues to be a double-ended cold leg guillotine (DECLG) rupture. These calculations also confirmed that the limiting coefficient of discharge (C_D) continues to be 0.8. The limiting case assumes a loss of offsite power and the failure of one low pressure safety injection pump. Additional details of the analyses are given in EMF-92-165 (Reference 2), February 9, 1993.

In addition to the assumptions identified above, this case also assumed 102 percent of the St Lucie 1 rated power level of 2700 MWt, an RCS rated design flow of 355,000 gpm, and peaking factors as specified in the St. Lucie Core Operating Limits report.

The calculated peak cladding temperature is 1912°F, the calculated maximum local metal/water reaction is less than 3.0%, and the calculated core-wide metal/water reaction is less than 1%. These results are within the criteria specified in 10 CFR 50.46(b) (1 through 3) of 2200°F, 17%, and 1%, respectively. The results assure that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4). The licensee reported that the time of Emergency Core Cooling System (ECCS) hot leg switchover was determined by analysis to be within 8 hours. This, combined with the St. Lucie ECCS design as approved, assures continued conformance with the long-term cooling requirement of 10 CFR 50.46(b)(5).

SBLOCA Analyses

In an August 18, 1994, submittal the licensee provided the results of a limited spectrum of SBLOCA reanalyses which were performed to account for the St. Lucie Unit 1 plant design with average steam generator tube plugging of 25%, and as much as 32% plugging in either steam generator. The reanalyses were performed using the Siemens SBLOCA Evaluation Model described in XN-NF-82-49, Revision 1, Supplement 1 which has been approved by the NRC (October 3, 1994) for licensing applications and is applicable to the St. Lucie plant.

The licensee has referenced previous studies to support its conclusion that the limiting location of small break continues to be a cold leg rupture. A crossflow sensitivity study was performed to identify the worst crossflow, but a time step sensitivity study was not performed since none of the criteria specified in the NRC October 3, 1994, Safety Evaluation (Reference 4) requiring the study were calculated to occur. The licensee provided the following sensitivity/spectrum analysis cases to identify and quantify the worst case:

a. 0.05 ft^2 cold leg,
b. 0.10 ft^2 cold leg,
c. 0.15 ft^2 cold leg, and
d. 0.20 ft^2 cold leg.

Results of Licensing Basis SBLOCA Analysis

The 0.1 ft^2 case above was identified as the worst SBLOCA case. In addition to the assumptions identified above, this case also assumed 102 percent of a core power level of 2700 MWt, an RCS flow of 355,000 gpm, and peaking factors as specified in the St. Lucie Core Operating Limits report.

The calculated peak cladding temperature is 1846°F, the calculated maximum local metal/water reaction is less than 2.0%, and the calculated core-wide metal/water reaction is less than 1%. These results are within the criteria specified in 10 CFR 50.46(b) (1 through 3) of 2200°F, 17%, and 1%, respectively. The results assure that the core will remain amenable to cooling as required by 10 CFR 50.46(b)(4).

The St. Lucie ECCS design, as approved, assures continued conformance with the long-term cooling requirement of 10 CFR 50.46(b)(5). The results of the analyses of the limiting 0.1 ft^2 SBLOCA are bounded by the results for the limiting LBLOCA.

LOCA Analysis Conclusions

St. Lucie Unit 1 LOCA analyses provided by the licensee in support of the St. Lucie increased steam generator plugging/reduced RCS flow were performed with NRC-approved evaluation models and identify a double-ended cold leg guillotine cold leg break with a discharge coefficient of 0.8, with loss of offsite power and failure of one low pressure safety injection pump, as the limiting LOCA event. The results of the analysis of this event demonstrate conformance with the criteria specified in 10 CFR 50.46(b) and, therefore, the analyses are acceptable.

2.2 Transient Events Not Requiring Reanalysis

Although the following events were not reanalyzed, they were evaluated by the licensee to determine the impact due to the proposed changes. The Uncontrolled CEA Withdrawal event, Inadvertent Opening of Pressurizer Pressure Relief Valves event, and the excess load event - Inadvertent Opening of the Steam Dump and Bypass Control System Valves at Full Power - all remain bounded by the 4 pump LOF event for DNB considerations.

The existing TM/low pressure Trip, the Variable High Power Trip (VHPT), and the Local Power Density Trip Limiting Safety System Settings provide protection against boron dilution events initiated at power. During Modes 2 - 5, the licensee concluded that the calculated times to lose the required shutdown margin is sufficient to accommodate the impact from the proposed changes.

The CEA Ejection Accident, which is a mechanical failure of a control rod drive mechanism pressure housing, results in the rapid reactivity insertion and adverse core power distribution, leading to possible core damage. Since the proposed changes do not impact the nuclear characteristics of the reactor core, the licensee concluded that the existing analysis remains valid.

The proposed changes would reduce the primary to secondary heat transfer rate across the steam generators and lower the initial secondary pressure. The licensee indicated that these changes would, in fact, reduce the effects of steamline break inside and outside of containment and the radiological consequences of inadvertent opening of a steam generator relief or safety valve. Therefore, the current analyses for these events remain valid.

2.3 Other Considerations Evaluated by the Licensee

The licensee also considered the impact of the amendment request on the following non-Chapter 15 analyses:

a. Plant Natural Circulation Capability,
b. Auxiliary Feedwater System High Energy Line Break,
c. Low Temperature Overpressure Protection Analysis,
d. Overpressure Protection Analysis, and
e. Impact on Steam Generator Mechanical Loads.

Based on its evaluation, the licensee determined that the reduced flow rate will not significantly impact the acceptance criteria of the analyses listed above.

3.0 CONCLUSION

The staff has reviewed the licensee's proposal to decrease the RCS flow rate. The proposed changes were included in the reanalyses and evaluation of the applicable Chapter 15 UFSAR transient analyses. Since the reanalyses and evaluations were performed using staff approved methods and the licensee indicated that the results remain within the applicable acceptance criteria, the staff approves the reduction in RCS flow rate to 355,000 gpm.

4.0 REFERENCES

1. Letter from D. A. Sager, FPL to USNRC, "St Lucie Unit 1 Reduction of Reactor Coolant System Design Flow," dated March 19, 1993.

2. EMF-92-165, Siemens Power Corporation, "St. Lucie Unit 1 Chapter 15 Event Review and Analysis for 25% Steam Generator Tube Plugging," dated February 9, 1993.

3. Letter from D. M. Crutchfield (NRC) to G. M. Ward (Siemens Power Corporation), "Safety Evaluation of EXXON Nuclear Company Large Break ECCS Evaluation Model," dated July 8, 1986.

4. Letter from G. M. Holahan (NRC) to R. A. Copeland (Siemens Power Corporation), "Acceptance for Referencing of the Topical Report XN-NF-82-49-P Revision 1, Supplement 1, 'EXXON Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model,' (TAC NO. M83302)," dated October 3, 1994.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

These amendments change 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 (58 FR 25855). Accordingly, these amendments meet 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 these 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: F. Orr
S. Brewer

Date: November 25, 1994