Docket No. 50-335

September 11, 1990

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

Dear Mr. Goldberg:

SUBJECT: ST. LUCIE UNIT 1 - ISSUANCE OF AMENDMENT RE: AUTOMATIC FEEDWATER ACTUATION SYSTEM SETPOINTS (TAC NO. 76172)

The Commission has issued the enclosed Amendment No. 105 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 9, 1990.

This amendment revises Technical Specifications 2.2-1, Reactor Trip Setpoints, and 3/4.3.2, Engineered Safety Feature Actuation System Instrumentation. The changes lower the Reactor Protection System Steam Generator level-low trip setpoint from ≧ 37.0% narrow range to ≧ 20.5% narrow range. The Auxiliary Feedwater Actuation System setpoint for the steam generator level-low trip is lowered from its current value of ≧ 29.0% narrow range to ≧ 19.0% narrow range. The changes also reduce the Auxiliary Feedwater System response time on low steam generator level. Additionally, the changes revise the allowable values for steam generator and feedwater header high differential pressure for auxiliary feedwater initiation.

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-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 105 to DPR-67
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDII-2	PM:PDII-2	D:PDII-2	OGC
DMiller	JNorris	HBerkow	BMB
8/27/90	8/27/90	8/27/90	8/24/90

Subject to correction of typos

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DATED: September 11, 1990

AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 105
License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company, (the licensee) dated March 9, 1990, 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. 105 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, 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: September 11, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 105

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
2-4	2-4
3/4 3-15	3/4 3-15
3/4 3-17	3/4 3-17

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, and a minimum setpoint of 15% of RA 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 370,000 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.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. LOSS OF POWER		
a. (1) 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2900 ± 29 volts with a 1 ± .5 second time delay	2900 ± 29 volts with a 1 ± .5 second time delay
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)		
(1) Undervoltage Device #1	3675 ± 36 volts with a 7 ± 1 minute time delay	3675 ± 36 volts with a 7 ± 1 minute time delay
(2) Undervoltage Device #2	3592 ± 36 volts with a 18 ± 2 second time delay	3592 ± 36 volts with a 18 ± 2 second time delay
c. 480 volts Emergency Bus Undervoltage (Degraded Voltage)	429 + 5-0 volts with a 7 ± 1 second time delay	429 + 5 -0 volts with a 7 ± 1 second time delay
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. SG 1A & 1B Level Low	≥19.0%	≥18.0%
8. AUXILIARY FEEDWATER ISOLATION		
a. Steam Generator ΔP-High	≤275 psid	89.2 to 281 psid
b. Feedwater Header High ΔP	≤150.0 psid	56.0 to 157.5 psid

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Fan Coolers	Not Applicable
Feedwater Isolation	Not Applicable
Containment Isolation	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIS	
Containment Isolation	Not Applicable
Shield Building Ventilation System	Not Applicable
d. RAS	
Containment Sump Recirculation	Not Applicable
e. MSIS	
Main Steam Isolation	Not Applicable
Feedwater Isolation	Not Applicable
f. AFAS	
Auxiliary Feedwater Actuation	Not Applicable
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0*/19.5**
b. Containment Isolation ***	$\leq$ 30.5*/20.5**
c. Containment Fan Coolers	$\leq$ 30.0*/17.0**
d. Feedwater Isolation	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIMES IN SECONDS
3. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 30.0*/19.5**
b. Containment Isolation***	≤ 30.5*/20.5**
c. Shield Building Ventilation System	≤ 30.0*/14.0**
d. Containment Fan Coolers	≤ 30.0*/17.0**
e. Feedwater Isolation	≤ 60.0
4. Containment Pressure -- High-High	
a. Containment Spray	≤ 30.0*/18.5**
5. Containment Radiation-High	
a. Containment Isolation***	≤ 30.5*/20.5**
b. Shield Building Ventilation System	≤ 30.0*/14.0**
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	≤ 6.9
b. Feedwater Isolation	≤ 60.0
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	≤ 91.5
8. Steam Generator Level-Low	
a. Auxiliary Feedwater	≥ 205**, ≤ 305 *

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

**Diesel generator starting and sequence loading delays <u>not</u> included. Offsite power available.

***Not applicable to containment isolation valve I-MV-18-1.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure -- High - High	S	R	M	1, 2, 3
c, Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
3. CONTAINMENT ISOLATION (CIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Containment Radiation - High	S	R	M	1, 2, 3, 4
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
e. SIAS	N.A.	N.A.	R	N.A.
4. MAIN STEAM LINE ISOLATION (MSIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Steam Generator Pressure - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
5. CONTAINMENT SUMP RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 105

TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

By letter dated March 9, 1990 (Ref. 1), Florida Power and Light Company (FPL or the licensee) proposed to amend the Technical Specifications (TS) for St. Lucie Unit 1. The proposed amendment revises TS 2.2-1, Reactor Trip Setpoints, and TS 3/4.3.2, Engineered Safety Feature Actuation System Instrumentation. The changes would lower both the Reactor Protection System (RPS) steam generator level-low trip and the Auxiliary Feedwater Actuation System (AFAS) setpoint for the steam generator level-low trip. The changes would also reduce the Auxiliary Feedwater System (AFWS) response time on low steam generator level, and revise the allowable values for steam generator and feedwater header high differential pressure for AFW initiation.

Similar changes have already been approved by the staff for St. Lucie Unit 2 by Amendment No. 23 dated September 24, 1987 (Ref. 2) and Amendment No. 28 dated March 22, 1988 (Ref. 3). The proposed changes to St. Lucie Unit 1 will provide setpoints which are consistent with St. Lucie Unit 2.

2.0 EVALUATION

The RPS provides protection against operation with insufficient secondary liquid inventory. This is accomplished by the steam generator level-low reactor trip. The AFWS is designed to supply feedwater to the steam generators in order to maintain water inventory. The AFAS responds to decreasing steam generator coolant inventory and provides an actuation signal to the AFWS at the steam generator level-low setpoint. The AFWS response time permits the operators to assess post-trip plant conditions prior to the automatic actuation of the AFWS.

By lowering the steam generator level-low reactor trip setpoint and the AFAS setpoint, larger fluctuations in steam generator water level can be accommodated without resulting in a reactor trip or actuation of AFWS. This will result in a reduction of unnecessary reactor trips and AFWS actuations. To ensure that adequate steam generator inventory is available during design basis transients, the upper limit for AFWS response time is also being reduced.

The licensee has also proposed to change the AFW isolation setpoint allowable values. The function of the AFW isolation trip for steam generator and feedwater header differential pressure is to isolate AFW flow to a faulted steam generator in the event of a steam line break or a feedwater line break. Placing a lower bound on the allowable values of steam generator and feedwater header high differential pressure will provide additional assurance that the AFAS isolation logic properly identifies the faulted steam generator.

The licensee has evaluated the above changes to assess their impact on the existing safety analyses. A review of Final Safety Analysis Report (FSAR) Chapter 15 events for St. Lucie Unit 1 was performed in order to determine which events required reanalysis as a result of the proposed changes. No accident analyses were identified that take credit for the AFW isolation feature. The only category of events shown to be impacted by the steam generator level setpoints is the "Decrease in Heat Removal by the Secondary System." In terms of challenging the adequacy of steam generator liquid inventory, loss of normal feedwater flow (LONF) is the most limiting event in the category. The steam generator water level-low trip is the controlling trip for this event. This event is affected by the proposed changes, therefore, the event was reanalyzed with respect to minimum steam generator inventory requirements. The proposed setpoint changes were found to be adequate to maintain a secondary heat sink throughout the LONF event, and thus ensure that the limits for DNBR are not exceeded. The results of a sensitivity calculation, in which no AFW was supplied to the steam generator, showed that dryout would occur in 700 seconds. The proposed AFWS response time of 305 seconds is well within the 700 seconds available to verify automatic AFW pump start or to manually initiate AFW.

The impact on the design bases of the AFWS, FSAR Section 10.5, was also evaluated. The two most limiting transients are the loss of main feedwater (LMFW) concurrent with an AFWS high energy line break (HELB) and a loss of offsite AC power concurrent with an AFWS HELB. These transients were reevaluated based on the proposed setpoint changes. LMFW concurrent with AFWS HELB with offsite power available is the most limiting case, since the reactor coolant pumps (RCPs) are available to add additional heat to the RCS. The results of the analysis of the limiting case demonstrate that the AFWS is adequate to maintain a secondary heat sink for up to 30 minutes following the steam generator level-low trip, at which time it would be required to trip one RCP per coolant loop to recover steam generator levels, as indicated in the EOPs.

Based on the results of the reanalysis of Chapters 10 and 15 of the FSAR, it can be concluded that the proposed changes do not have an adverse effect on the existing safety analysis for St. Lucie Unit 1, particularly, the licensing bases defined in the FSAR and the AFWS design bases in the FSAR.

2.1 Technical Specification Changes

The TS changes associated with the proposal are as follows:

1. TS 2.2.1, "Reactor Trip Setpoints," Table 2.2-1, is revised to reduce the steam generator water level-low trip setpoint from ≧ 37.0% narrow range (NR) with an allowable value of ≧ 37.0% NR to a setpoint of ≧ 20.5% NR with an allowable value of ≧ 19.5% NR.

2. TS 3.3.2.1, "ESF Actuation System Instrumentation," Table 3.3-4, is revised to reduce the AFAS trip steam generator low level from ≧ 29.0% NR with an allowable value of ≧ 28.5% NR to a setpoint of ≧ 19.0% NR with an allowable value of ≧ 18.0% NR.

3. Table 3.3-4 is also revised to add lower limits to the allowable values of steam generator differential pressure-high trip and feedwater differential pressure-high trip. This will yield a steam generator differential pressure-high trip allowable value range of 89.2 psid to 281 psid, and a feedwater header differential pressure-high trip allowable value range of 56.0 psid to 157.5 psid. The upper allowable values will remain unchanged.

4. Table 3.3-5 is revised to reduce the AFWS response time on steam generator low level limit from ≦ 600 seconds to ≦ 305 seconds. The lower limit remains unchanged at ≧ 205 seconds.

3.0 TECHNICAL FINDING

Based on the staff evaluation in Section 2.0 above, the staff concludes that the proposed TS concerning low steam generator level reactor trip and AFAS setpoints are acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that this 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. 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.

5.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter from J. H. Goldberg (FPL) to USNRC, dated March 9, 1990.

2. Amendment No. 23 to Facility Operating License No. NPF-16 for St. Lucie Unit 2 dated September 24, 1987.

3. Amendment No. 28 to Facility Operating License No. NPF-16 for St. Lucie Unit 2 dated March 22, 1988.

Date: September 11, 1990

Principal Contributor:
A. E. Almond