

September 24, 1987

Docket No. 50-389

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Mr. C. O. Woody
Group Vice President
Nuclear Energy
Florida Power & Light Company
P. O. Box 14000
Juno Beach, Florida 33408

Dear Mr. Woody:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 64091)

The Commission has issued the enclosed Amendment No. 23 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 December 2, 1986, as supplemented February 3, 1987.

This amendment reduces the steam generator water level setpoints for reactor trip and auxiliary feedwater initiation. The water level for reactor trip is reduced from 39.5% to 20.5%. The water level for auxiliary feedwater initiation is reduced from 20.6% to 19.0%.

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

Sincerely,

/s/

E. G. Tourigny, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No.23 to NPF-16
2. Safety Evaluation

cc w/enclosures:

See next page

LA:PD22
DMiller
8/24/87

PM:PD22
ETourigny:hc
8/26/87

*msd
RSB 4/19/87*
SICM
Joyce
8/24/87
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OGC: LKY
S H Lewis
8/15/87

D:PD22
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8/23/87

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St. Lucie Plant

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

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. 23
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 December 2, 1986, as supplemented February 3, 1987, 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.

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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. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 23, 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, 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 24, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 23
TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

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

2-4
3/4 3-18
B2-5

Insert Pages

2-4
3/4 3-18
B2-5

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to or concurrently with a safety injection (SIAS). This also provides assurance that a reactor trip is initiated prior to or concurrently with an MSIS.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint of 620 psia is sufficiently below the full load operating point of approximately 885 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of 30 psi in the safety analyses.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required. This trip also protects against violation of the specified acceptable fuel design limits (SAFDL) for DNBR, offsite dose and the loss of shutdown margin for asymmetric steam generator transients such as the opening of a main steam safety valve or atmospheric dump valve.

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower excore neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15% power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

RCP Loss of Component Cooling Water

A loss of component cooling water to the reactor coolant pumps causes a delayed reactor trip. This trip provides protection to the reactor coolant pumps by ensuring that plant operation is not continued without cooling water available. The trip is delayed 10 minutes following a reduction in flow to below the trip setpoint and the trip does not occur if flow is restored before 10 minutes elapses. No credit was taken for this trip in the safety analysis. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protective System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 3.5 psig	≤ 3.6 psig
c. Pressurizer Pressure - Low	≥ 1736 psia	≥ 1728 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 5.40 psig	≤ 5.50 psig
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Safety Injection (SIAS)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 3.5 psig	≤ 3.6 psig
d. Containment Radiation - High	≤ 10 R/hr	≤ 10 R/hr
e. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM LINE ISOLATION		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 600 psia	≥ 567 psia
c. Containment Pressure - High	≤ 3.5 psig	≤ 3.6 psig
d. Automatic Actuation Logic	Not Applicable	Not Applicable

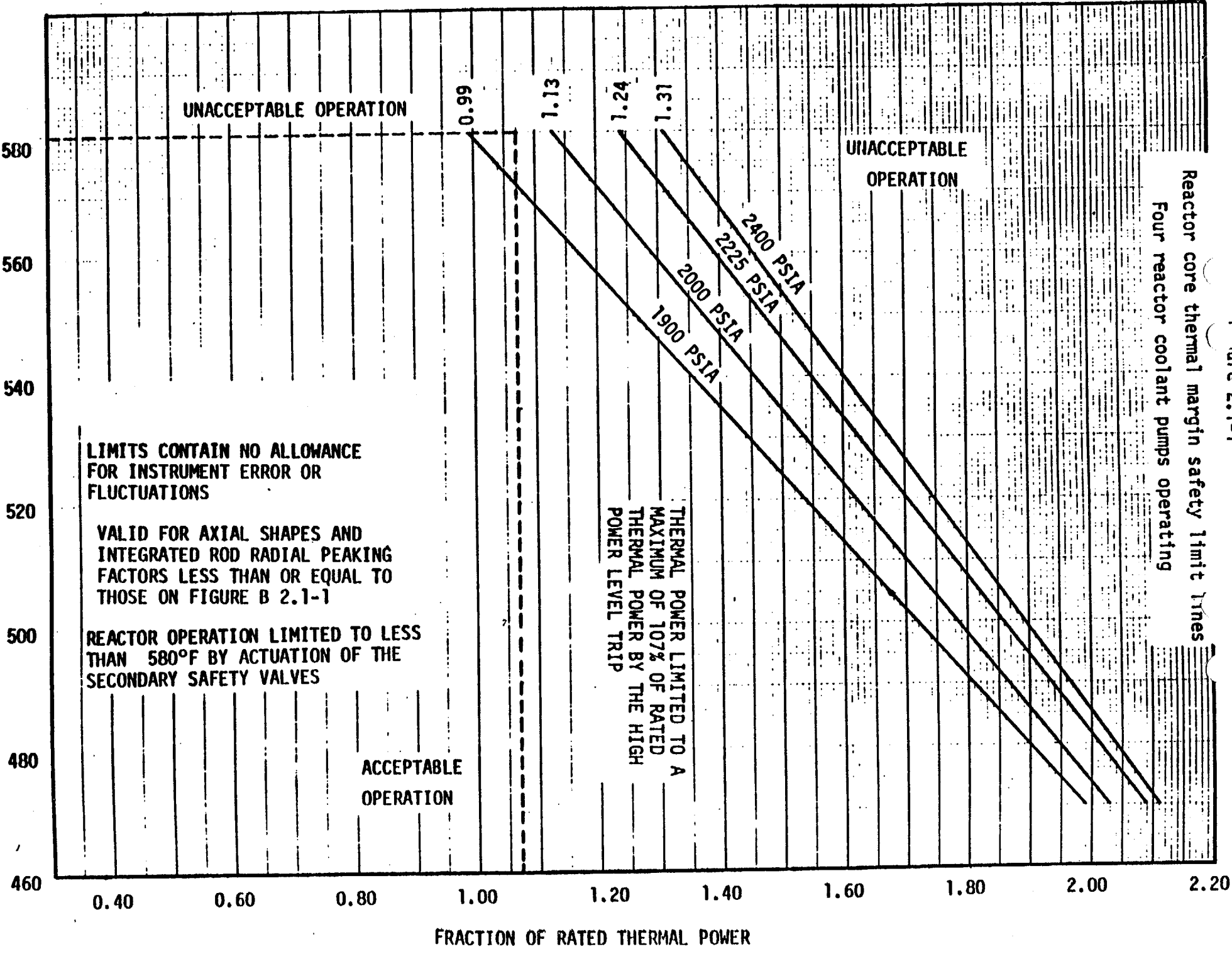
TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

<u>FUNCTIONAL UNIT</u>	<u>TRIP VALUE</u>	<u>ALLOWABLE VALUES</u>
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank - Low	5.67 feet above tank bottom	4.62 feet to 6.24 feet above tank bottom
c. Automatic Actuation Logic	Not Applicable	Not Applicable
6. LOSS OF POWER		
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3120 volts	≥ 3120 volts
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	≥ 360 volts	≥ 360 volts
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≥ 3848 volts with a 10-second time delay	≥ 3848 volts with a 10-second time delay
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	≥ 432 volts	≥ 432 volts
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Steam Generator ΔP -High	≤ 180.0 psid	≤ 187.5 psid
d. SG 2A&2B Level Low	$\geq 19.0\%$	$\geq 18.0\%$
e. Feedwater Header High ΔP	≤ 100.0 psid	≤ 107.5 psid

Reactor core thermal margin safety limit lines
Four reactor coolant pumps operating

Amendment No. 8
2-3
MAXIMUM COLD LEG TEMPERATURE (°F)



UNACCEPTABLE OPERATION

UNACCEPTABLE OPERATION

LIMITS CONTAIN NO ALLOWANCE FOR INSTRUMENT ERROR OR FLUCTUATIONS

VALID FOR AXIAL SHAPES AND INTEGRATED ROD RADIAL PEAKING FACTORS LESS THAN OR EQUAL TO THOSE ON FIGURE B 2.1-1

REACTOR OPERATION LIMITED TO LESS THAN 580°F BY ACTUATION OF THE SECONDARY SAFETY VALVES

ACCEPTABLE OPERATION

THERMAL POWER LIMITED TO A MAXIMUM OF 107% OF RATED THERMAL POWER BY THE HIGH POWER LEVEL TRIP

FRACTION OF RATED THERMAL POWER

0.99

1.13

1.24

1.31

2400 PSIA

2225 PSIA

2000 PSIA

1900 PSIA

Amendment No. 8

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

<u>FUNCTIONAL UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Variable 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 RATED THERMAL POWER and a maximum of < 107.0% of RATED THERMAL POWER.
3. Pressurizer Pressure - High	≤ 2370 psia	≤ 2374 psia
4. Thermal Margin/Low Pressure Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.
5. Containment Pressure - High	≤ 3.0 psig	≤ 3.1 psig
6. Steam Generator Pressure - Low	≥ 626.0 psia (2)	≥ 621.0 psia (2)
7. Steam Generator Pressure ⁽¹⁾ Difference - High (Logic in TM/LP Trip Unit)	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level - Low	≥ 20.5% (3)	≥ 19.5% (3)



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 23
TO FACILITY OPERATING LICENSE NO. NPF-16
FLORIDA POWER & LIGHT COMPANY, ET AL.
ST. LUCIE PLANT, UNIT NO. 2
DOCKET NO. 50-389

INTRODUCTION

By letter dated December 2, 1986, the Florida Power and Light Company (the licensee) requested technical specification changes for the St. Lucie Plant, Unit No. 2. The proposed changes would reduce the steam generator water level setpoints for reactor trip and auxiliary feedwater initiation. The staff published in the Federal Register on February 11, 1987 an initial determination that the proposed amendment does not involve a significant hazards consideration. The initial determination was based upon the licensee's December 2, 1986 submittal. By letter dated February 3, 1987, the licensee provided additional information. The additional information provided by the licensee does not change the staff's initial determination of no significant hazards consideration for the proposed amendment.

EVALUATION

The staff requested that its contractor, EG&G Idaho, review and evaluate the licensee's proposed changes. The contractor's report is attached. It concludes that the changes are acceptable. The staff has reviewed the contractor's report and agrees with the evaluation and conclusion of the report. Thus, the proposed technical specification changes are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the 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 the amendment.

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.

Date: September 24, 1987

Principal Contributor:

E. Tourigny

Attachment:

EG&G Report

REVIEW OF FLORIDA POWER AND LIGHT COMPANY'S
PROPOSED TECHNICAL SPECIFICATION REVISION
FOR LOW LEVEL STEAM GENERATOR SETPOINTS
FOR ST. LUCIE, UNIT 2

I. BACKGROUND

By letter L-86-452, dated December 2, 1986, Florida Power & Light Company (FPL) proposed to revise Technical Specification 2.2.1, Reactor Protection Instrumentation (RPS) and 3/4.3.2, Engineered Safety Features Actuation System (ESFAS) Instrumentation and the associated Bases of the St. Lucie, Unit 2, Technical Specifications. Additional information was provided by letter L-87-39, dated February 3, 1987.

The proposed amendment revises Technical Specification 2.2.1, Reactor Protection Instrumentation (RPS), and 3/4.3.2, Engineered Safety Features Actuation System (ESFAS) Instrumentation and the associated Bases. The proposed change lowers the RPS steam generator low level trip setpoint from $\geq 39.5\%$ narrow range (NR) (allowable value $\geq 39.1\%$ NR) to $\geq 20.5\%$ NR (allowable value $\geq 19.5\%$ NR). The Auxiliary Feedwater Actuation Signal (AFAS) steam generator low level trip setpoint is lowered from its current value of $\geq 20.6\%$ NR (allowable value $\geq 20.0\%$ NR) to $\geq 19.0\%$ NR (allowable value $\geq 18.0\%$ NR).

The existing Technical Specification setpoints correspond to a Cycle 1 analysis assumption of steam generator low level RPS setpoint of 30% narrow range (NR) for the most limiting event, Loss of Feedwater, and 5% NR setpoint for all accidents.

II. DISCUSSION

During Cycle 1, credit was not taken for an installed safety grade Auxiliary Feedwater Actuation Signal (AFAS), thus necessitating the 30% narrow range setpoint. Uncertainties and instrument error are reflected in the 9.5% difference between 30% and 39.5%.

During Cycle 2 and 3, taking credit for the AFAS, the analytical Chapter 15 Accident Analysis setpoint value of 5% NR for Reactor Trip insures that the Reactor Coolant System (RCS) pressure limit of 110% of design pressure is not exceeded. The 5% NR setpoint value is also used for AFAS in the Chapter 15 Accident Analysis.

The margin to allow for instrumentation uncertainties which include inherent process instrumentation errors, equipment response time, instrument drift and environmental concerns is 14.5% for reactor trip and 13% for AFAS. This compares with the 9.5% margin used for Cycle 1.

III. CONCLUSION

The proposed technical specification change is acceptable, as the Chapter 15 Accident analysis is based on a low steam Generator Trip and AFAS actuation at 5% NR, and there is reasonable assurance that an acceptable margin exists between the proposed values and the 5% analytical value.