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U.S. NUCLEAR REGULATORY COMMISSION

Docket No. 50-335

Dr. Robert E. Uhrig
Vice President
Advanced Systems & Technology
Florida Power & Light Company
P. O. Box 529100
Miami, Florida 33152

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 43 to Facility Operating License No. DPR-67 for St. Lucie Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated July 23, 1981.

The amendment changes the Technical Specifications by adding limits and surveillance requirements for the proposed Asymmetric Steam Generator Transient Protective Trip Function.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by:

Christian C. Nelson, Project Manager
Operating Reactors Branch #3
Division of Licensing

CP
1

Enclosures:
1. Amendment No. 43 to DPR-67
2. Safety Evaluation
3. Notice of Issuance

cc: See next page

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cc:
Robert Lowenstein, Esquire
Lowenstein, Newman, Reis & Alexrad
1025 Connecticut Avenue, N.W.
Washington, D. C. 20036

Norman A. Coll, Esquire
McCarthy, Steel, Hector & Davis
14th Floor, First National Bank Building
Miami Florida 33131

Indian River Junior College Library
3209 Virginia Avenue
Fort Pierce, Florida 33450

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Mr. Weldon B. Lewis
County Administrator
St. Lucie County
2300 Virginia Avenue, Room 104
Fort Pierce, Florida 33450

U.S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Mr. Charles B. Brinkman
Manager - Washington Nuclear Operations
C-E Power Systems
Combustion Engineering, Inc.
4853 Cordell Avenue, Suite A-1
Bethesda, Maryland 20014

Mr. Jack Schreve
Office of the Public Counsel
Room 4, Holland Building
Tallahassee, Florida 32304

Resident Inspector/St. Lucie
Nuclear Power Station
c/o U.S.N.R.C.
P. O. Box 400
Jensen Beach, Florida 33457

cc w/enclosure(s) and incoming
dated: 7/23/81

Bureau of Intergovernmental
Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

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. 43
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 July 23, 1981, 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 applications, the provision 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. 43 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 14, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 43

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.

Pages

2-5
3/4 3-2
3/4 3-6
3/4 3-7
B 2-7
B 2-8

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 585 psig; bypass shall be automatically removed at or above 585 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.

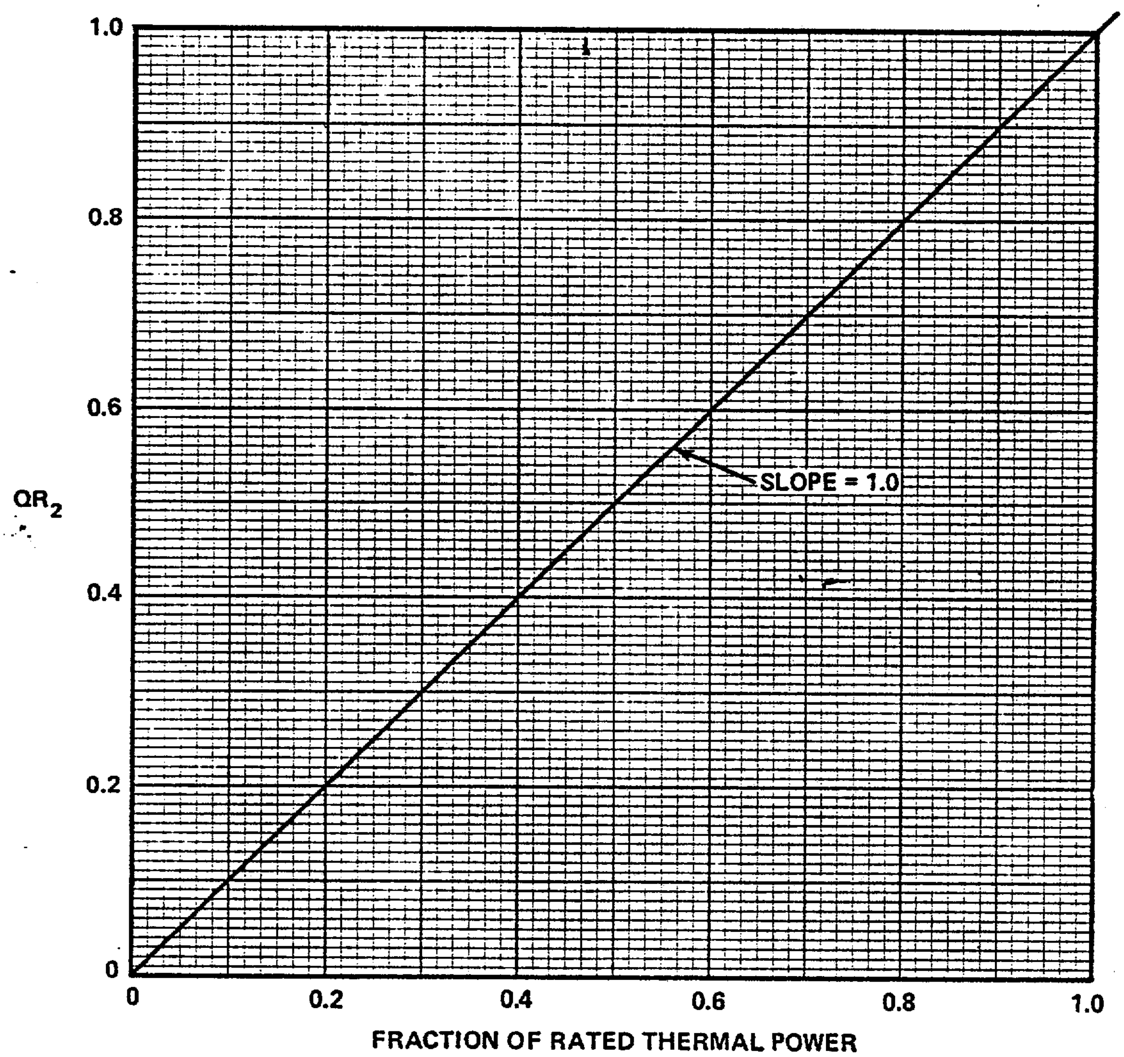

FIGURE 2.2-1

Local Power Density — High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. Power Level - High	4	2(a)	3(f)	1, 2	2#
3. Reactor Coolant Flow - Low	4/SG	2(a)/SG	3/SG	1, 2 (e)	2#
4. Pressurizer Pressure - High	4	2	3	1, 2	2#
5. Containment Pressure - High	4	2	3	1, 2	2#
6. Steam Generator Pressure - Low	4/SG	2(b)/SG	3/SG	1, 2	2#
7. Steam Generator Water Level - Low	4/SG	2/SG	3/SG	1, 2	2#
8. Local Power Density - High	4	2(c)	3	1	2#
9. Thermal Margin/Low Pressure	4	2(a)	3	1, 2 (e)	2#
9a. Steam Generator Pressure Difference - High	4	2(a)	3	1, 2 (e)	2#
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	4	2(c)	3	1	2#

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on than channel provided the other inoperable channel is placed in the tripped condition.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.1.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Level - High	$\leq$ 0.40 seconds*# and $\leq$ 8.0 seconds##
3. Reactor Coolant Flow - Low	$\leq$ 0.65 seconds
4. Pressurizer Pressure - High	$\leq$ 0.90 seconds
5. Containment Pressure - High	$\leq$ 1.40 seconds
6. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
7. Steam Generator Water Level - Low	$\leq$ 0.90 seconds
8. Local Power Density - High	$\leq$ 0.40 seconds*# and $\leq$ 8.0 seconds##
9. Thermal Margin/Low Pressure	$\leq$ 0.90 seconds*# and $\leq$ 8.0 seconds##
9a. Steam Generator Pressure Difference - High	$\leq$ 0.90 seconds
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	Not Applicable
11. Wide Range Logarithmic Neutron Flux Monitor	Not Applicable

*Neutron detectors are exempt from response time testing. Response time shall be measured from detector output or input of first electronic component in channel.

#Response time does not include contribution of RTDs.

##RTD response time only. This value is equivalent to the time interval required for the RTDs output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Power Level - High				
a. Nuclear Power	S	D(2), M(3),Q(5)	M	1, 2
b. ΔT Power	S	D(4), Q	M	1
3. Reactor Coolant Flow - Low	S	R	M	1, 2
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Containment Pressure - High	S	R	M	1, 2
6. Steam Generator Pressure - Low	S	R	M	1, 2
7. Steam Generator Water Level - Low	S	R	M	1, 2
8. Local Power Density - High	S	R	M	1
9. Thermal Margin/Low Pressure	S	R	M	1, 2
9a. Steam Generator Pressure Difference - High	S	R	M	1, 2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	N.A.	N.A.	S/U(1)	N.A.
11. Wide Range Logarithmic Neutron Flux Monitor	S	N.A.	S/U(1)	1, 2, 3, 4, 5 and *
12. Reactor Protection System Logic	N.A.	N.A.	M and S/U(1)	1, 2 and *
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *

TABLE 4.3-1 (Continued)

TABLE NOTATION

* * - With reactor trip breaker closed.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Pwr - ΔT Pwr." During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to ≤ 90% of the maximum allowed THERMAL POWER level with the existing Reactor Coolant Pump combination.

(4) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

(5) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.30.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1875 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with 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.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2°F to compensate for potential temperature measurement uncertainty; and a further allowance of 74 psia to compensate for pressure measurement error and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 74 psia allowance is made up of a 22 psia pressure measurement allowance and a 52 psia time delay allowance.

Asymmetric Steam Generator Transient Protective Trip Function (ASGTPTF)

The ASGTPTF consists of Steam Generator pressure inputs to the TM/LP calculator, which causes a reactor trip when the difference in pressure between the two steam generators exceeds the trip setpoint. The ASGTPTF is designed to provide a reactor trip for those events associated with secondary system malfunctions which result in asymmetric primary loop coolant temperatures. The most limiting event is the loss of load to one steam generator caused by a single main steam isolation valve closure.

The equipment trip setpoint and allowable values are calculated to account for instrument uncertainties, and will ensure a trip at or before reaching the analysis setpoint.

BASES

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuring transient, thus extending the service life of these valves. No credit was taken in the accident 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.

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 accident 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

Introduction

By letter dated July 23, 1981, the Florida Power and Light Company (FPL or the licensee) proposed changes to the Technical Specifications for the St. Lucie Unit 1, regarding the addition of an Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF). The ASGTPTF utilizes steam generator pressure inputs to the Thermal Margin/Low Pressure (TM/LP) calculator, which causes a reactor trip when the differences in pressure between the two steam generators exceeds the trip setpoint (135 psid). The ASGTPTF is designed to provide a reactor trip for those Anticipated Operational Occurrences (AOO) associated with secondary system malfunctions which result in asymmetric primary loop coolant temperatures. The most limiting event is the loss of load to one steam generator (LL/1SG) caused by a single Main Steam Isolation Valve closure. This change was also proposed as part of the licensee's November 14, 1980 application for stretch power which has not yet been approved.

Evaluation

Our evaluation is presented in two parts; (A) Thermal Hydraulic Aspects and (B) Electrical Instrumentation and Control Aspects.

A. Thermal-Hydraulic Aspects

The loss of load to one steam generator causes that steam generator's pressure and temperature to increase to the secondary relief/safety valve setting, while the operable steam generator picks up the lost load. This in turn will lead to a decreased steam generator temperature and pressure, which decreases the primary side temperature and causes an asymmetric reactor temperature distribution. This event is analyzed to assure the DNBR and linear heat generation rates are within design limits.

The St. Lucie 1 ASGTPTF is designed to provide a reactor trip when the secondary steam generator pressures differ by more that 135 psi. The event analyzed is a main steamline isolation valve closure at full power which causes a loss of load to that steam generator.

Assumptions used include the most negative moderator coefficient, a low primary system pressure (2200 psi), an initial core power of 2611 MWt, a high negative flux shape index, and a trip setpoint error of 40 psi in the ASGTPTF. The effect of these conservative assumptions is to decrease the predicted minimum DNBR. The results of the analysis show a minimum DNBR of 1.52, which is above the acceptable DNBR limit of 1.3. In addition the linear heat generation rates are within limits.

Protection against exceeding the DNBR and linear heat generation rate limits during the LL/1SG event is presently provided by the Low Steam Generator Level reactor trip in conjunction with sufficient initial margin maintained by the technical specification Limiting Conditions for Operation (LCOs). The ASGTPTF will result in a reactor trip sooner than the Low Steam Generator Level trip and, hence, will produce a smaller margin degradation during this event. The additional margin gain allows full advantage to be taken of margin recovery programs designed to achieve stretch power or 18 month fuel cycles.

The selection of a trip setpoint is such that adequate protection is provided when all sensor and processing time delays and inaccuracies are taken into account. The nominal setpoint, uncertainties and response time are listed below.

<u>ASYMMETRIC STEAM GENERATOR TRANSIENT PROTECTION TRIP FUNCTION NOMINAL CHARACTERISTICS</u>

Nominal System Accuracy	$\pm$ 35 psi
Analysis Setpoint	+ 175 psid
Nominal Equipment Setpoint	+ 135 psid
Nominal Pretrip Setpoint	+ 100 psid
Nominal System Response Time	$\leq$ .9 seconds

Based on our review of the thermal-hydraulic aspects of the proposed change we conclude that the Asymmetric Steam Generator Transient Protective Trip Function will improve the St. Lucie Unit 1 response to this type of event and is acceptable.

B. <u>Electrical Instrumentation and Control Aspects</u>

The ASGTPTF utilizes existing steam generator pressure sensors and Thermal Margin/Low Pressure (TM/LP) calculators which are used to generate trips as part of the Reactor Protection System (RPS). The TM/LP calculators will be modified to include a bistable with an input of the absolute value of the pressure difference between the two steam generators $|P_{SG1}-P_{SG2}|$. If the difference exceeds a set amount (135 psid), a bias is input to the TM/LP calculation. This will result in a reactor trip. The additional bias input to the TM/LP calculation is the asymmetric factor signal (Fas). The trip signal is preceded by a pretrip alarm (100 psid) to alert the operator of undesirable (abnormal) operating conditions.

The St. Lucie Unit 1 RPS utilizes four vital buses and a two-out-of-four trip logic. There are four channels of steam generator pressure per steam generator and four TM/LP channels. Each of the four channels is powered from a separate vital bus. Each TM/LP channel receives a pressure signal from the corresponding (powered from the same bus) pressure channel of each steam generator. Thus electrical independence between channels is maintained. Loss of power to any TM/LP channel will place that channel in the tripped state.

There are no control of indication functions associated with the trip function, nor is there any interaction with non-safety circuits. The licensee has stated that the ASGTPTF instrumentation conforms to the requirements of IEEE Standard 279-1968 and that the addition of this trip function will in no way degrade or adversely affect the operation of the existing RPS.

A channel functional test will be performed monthly for the ASGTPTF channels. In addition, a channel check (steam generator pressure) is performed during each shift, and the channels are calibrated each refueling outage. The ASGTPTF may be bypassed below 1% of RATED THERMAL POWER. This bypass will be automatically removed when thermal power is $\geq$ 1% of rated. The components being added for the ASGTPTF are of the same type and quality as those being used in the existing RPS.

Based on our review of the electrical, instrumentation, and control aspects of the licensee's proposal, we conclude that the proposed modifications to the St. Lucie Unit 1 RPS regarding the addition of the ASGTPTF and the associated Technical Specification changes are acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 14, 1981

Principal Contributors:

ICSB/DSI: R. Kendall
RSB/DSI: G. Alberthal
CPB/DSI: S. Gupta

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-335

FLORIDA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 43 to Facility Operating License No. DPR-67, issued to Florida Power & Light Company (the licensee), which revised the Technical Specifications for operation of the St. Lucie Plant, Unit No. 1 (the facility), located in St. Lucie County, Florida. The amendment is effective as of the date of issuance.

This amendment changes the Technical Specifications by adding limits and surveillance requirements for the proposed Asymmetric Steam Generator Transient Protective Trip Function.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 23, 1981, (2) Amendment No. 43 to License No. DPR-67, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 14th day of October, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing