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Docket No. 50-335

Florida Power & Light Company
 ATTN: Dr. Robert E. Uhrig
 Vice President
 Nuclear and General Engineering
 Post Office Box 013100
 Miami, Florida 33101

Includer

Gentlemen:

The Commission has issued the enclosed Amendment No. 15 to Facility License No. DPR-67 for the St. Lucie Plant Unit No. 1. The amendment modifies the Technical Specifications to authorize a 48-hour bypass period to perform certain instrumentation tests and maintenance. During the 48-hour period any one of the four Reactor Protection System (RPS) channels and any one of the four Engineered Safety Feature Actuation System (ESFAS) channels may be bypassed to allow testing or maintenance.

The amendment is in response to the proposed change included in your letter dated April 20, 1976. We have completed our review of appropriate aspects of the St. Lucie Unit No. 1 design and cannot grant the indefinite bypass period you requested. However, our modifications to your proposal have been discussed with and agreed to by your staff.

Copies of the related safety evaluation and Notice of Issuance are enclosed.

Sincerely,

Don K. Davis, Acting Chief
 Operating Reactors Branch #2
 Division of Operating Reactors

*6/28/77
 Called FPL (Banker) at 11:45
 and advised him of the issuance of this amendment. EReeves*

Enclosures:

1. Amendment No. 15 to DPR-67
2. Safety Evaluation
3. Notice

SEE PREVIOUS YELLOW FOR CONCURRENCES

*Cont. 1
 60*

cc: (see page 2)

OFFICE ➤	DOR:ORB-2	DOR:ORB-2	DOR:AD/ORs	PER/DOR		
SURNAME ➤	EAREeves:jpf	DKDavis	KRGoller	VStello		
DATE ➤	6/8/77	6/28/77	6/27/77	6/28/77		

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Docket
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Florida Power & Light Company
 ATTN: Dr. Robert E. Uhrig
 Vice President
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Gentlemen:

The Commission has issued the enclosed Amendment No. to Facility License No. DPR-67 for the St. Lucie Plant Unit No. 1. The amendment modifies the Technical Specifications to authorize a 48-hour bypass period to perform certain instrumentation tests and maintenance. During the 48-hour period any one of the four Reactor Protection System (RPS) channels and any one of the four Engineered Safety Feature Activation System (ESFAS) channels may be bypassed to allow testing or maintenance.

The amendment is in response to the proposed change included in your letter dated April 20, 1976. Your proposal was an appeal of the Staff's previous review relating to the RPS and ESFAS design. Our previous review occurred during the operating license review and is recorded in meeting minutes dated March 24, 1976 as item (3). We have completed a re-review of appropriate aspects of the St. Lucie Unit No. 1 design and cannot grant the indefinite bypass period you requested. However, our modifications to your proposal have been discussed with your staff who agreed to accept the specifications consistent with specifications recently approved for Calvert Cliffs Units No. 1 and No. 2, plants of a similar design.

Copies of the related safety evaluation and Notice of Issuance are enclosed.

Sincerely,

Don K. Davis, Acting Chief
 Operating Reactors Branch #2
 Division of Operating Reactors

DOR
 VStello
 6/1/77

Enclosures:

1. Amendment No. DPR-67

2. Safety Evaluation Notice

OFFICE	ST McGough 6/23/77	OELD V. Stello 6/21/77			
SURNAME	ORB#2 E. Reeves:jpf	ORB#2 R. Diggs	ORB#2 D. Davis	DOR D. Eisenhut	DOR K. Goller
DATE	6/8/77	6/7/77	6/1/77	6/6/77	6/1/77



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. 15
License No. DPR-67

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Florida Power & Light Company (the licensee) dated April 20, 1976, 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.

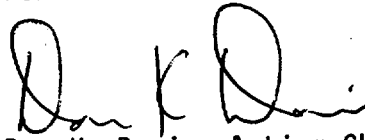
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C(2) of Facility Operating License No. DPR-67 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 15, 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



Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 28, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 15

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

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3/4 3-11
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3/4 3-13

3/4.3 INSTRUMENTATION

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.

ST. LUCIE - UNIT 1

3/4 3-2

Amendment No. 15

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
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#
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	4	2(c)	3	1	2#

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
11. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating-- Rate of Change of Power - High	4	2(d)	3	1, 2 and *	2#
b. Shutdown	4	0	2	3, 4, 5	4
12. Reactor Protection System Logic	2	1	2	1, 2*	5
13. Reactor Trip Breakers	2	1	2	1, 2*	5

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

The provisions of Specification 3.0.4 are not applicable.

- (a) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is \geq 5% of RATED THERMAL POWER.
- (b) Trip may be manually bypassed below 585 psig; bypass shall be automatically removed at or above 585 psig.
- (c) 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.
- (d) Trip may be bypassed below 10^{-4} % and above 15% of RATED THERMAL POWER.
- (e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.
- (f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

- ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.
- ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:
 - a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

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 that channel provided the other inoperable channel is placed in the tripped condition.
- ACTION 4 - 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 5 - 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

<u>FUNCTIONAL UNIT</u>	<u>RESPONSE TIME</u>
1. Manual Reactor Trip	Not Applicable
2. Power Level - High	≤ 0.40 seconds*
3. Reactor Coolant Flow - Low	≤ 0.65 seconds
4. Pressurizer Pressure - High	≤ 0.90 seconds
5. Containment Pressure - High	≤ 1.40 seconds
6. Steam Generator Pressure - Low	≤ 0.90 seconds
7. Steam Generator Water Level - Low	≤ 0.90 seconds
8. Local Power Density - High	≤ 0.40 seconds*
9. Thermal Margin/Low Pressure	≤ 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.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

- a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.
- b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS 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 4.3-2.

4.3.2.1.2 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.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the 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 ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure - High	4	2	3	1, 2, 3	9#
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	9#
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	10
3. CONTAINMENT ISOLATION (CIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure - High	4	2	3	1, 2, 3	9#
c. Containment Radiation - High	4	2	3	1, 2, 3, 4	9#
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3, 4	8
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	9#

ST. LUCIE - UNIT 1

3/4 3-10

Amendment No. 15

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	9#
6. LOSS OF POWER 4.16 kv Emergency Bus Undervoltage (Undervoltage relays)	1/Bus	1/Bus	1/Bus	1, 2, 3	9#

ST. LUCIE - UNIT 1

3/4 3-11

Amendment No. 15

TABLE 3.3-3 (Continued)

TABLE NOTATION

- (a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1725 psia.
- (b) An SIAS signal is first necessary to enable CSAS logic.
- (c) Trip function may be bypassed in this MODE below 585 psig; bypass shall be automatically removed at or above 585 psig.
- # The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

- ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:
 - a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.
 - 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 that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ST. LUCIE - UNIT 1

3/4 3-14

TABLE 3.3-4

ENGINEERED SAFETY FEATURE 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	≤ 5 psig	≤ 5 psig
c. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1600 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 10 psig	≤ 10 psig
3. CONTAINMENT ISOLATION (CIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 5 psig	≤ 5 psig
c. Containment Radiation - High	≤ 10 R/hr	≤ 10 R/hr
4. MAIN STEAM LINE ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 485 psig	≥ 485 psig
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	48 inches above tank bottom	48 inches above tank bottom



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT NO. 1

DOCKET NO. 50-335

INTRODUCTION

By application dated April 20, 1976, Florida Power & Light Company (FPL) requested amendment to the St. Lucie Plant Unit No. 1 license. The amendment would modify the Technical Specifications to allow any one of the four Reactor Protection System (RPS) channels and one of the four Engineered Safety Feature Actuation System (ESFAS) channels to be in the bypassed condition indefinitely. The current specification requires that after one hour the inoperable channel must be in the tripped condition.

DISCUSSION

As the result of the February 18, 1976 meeting, the NRC staff stated that the Technical Specifications dealing with the RPS and ESFAS would require that an inoperable channel be placed in the tripped condition within one hour. Thus, Technical Specifications for St. Lucie Unit No. 1 were issued on March 1, 1976, with the issuance of the facility license. The bases for the staff position were:

- a. The one hour requirement is consistent with the specifications imposed on other pressurized water reactor nuclear steam system suppliers; and
- b. The staff reviewed the plant as a two-out-of-four system, not as a two-out-of-three system with an installed spare.

The staff agreed to re-open the review if FPL provided information sufficient to show that the RPS and ESFAS is a two-out-of-three system with a spare. Also upon completion of the review, if the staff found that the RPS and ESFAS was an acceptable two-out-of-three system with a spare, the Technical Specifications would be appropriately modified. In addition, the staff stated that in order for the system to be an acceptable two-out-of-three system with a spare, the licensee should demonstrate that with any one channel in the bypassed condition, no single failure could cause the RPS or ESFAS to fail to perform its protective function when required.

The staff, in accordance with the previous commitment, re-opened the review of the Technical Specification limitation after receiving the April 20, 1976 FPL letter. Basically, the issue to be resolved may be restated as follows: To take credit for a two-out-of-three logic with an installed spare on the RPS and ESFAS, the licensee must demonstrate that the system satisfies the single failure requirements of IEEE Std 279-1971.

EVALUATION

In August 1976, a member of the NRC staff conducted an onsite review of the actual physical separation features of the RPS and ESFAS at the St. Lucie facility to determine conformance with the licensee's stated criteria and with the requirements of IEEE Std 279-1971. The onsite review included a detailed verification of the location and physical separation of sensors, instrument lines, transmitters, and electric cables routed through conduit and covered cable trays within containment for the following safety-related sensors:

1. Pressurizer Pressure Sensors;
2. Resistance Temperature Detectors;
3. Steam Generator Pressure Sensors;
4. Steam Generator Differential Pressure Sensors;
5. Ex-core Neutron Detectors; and
6. Steam Generator Level Sensors.

The physical independence of containment electrical penetrations and the routing of the four "independent" safety-related channels through the cable spreading room to the control room was also reviewed in detail. In addition, the staff reviewed the routing and location of RPS and ESFAS instrumentation panels and the electric cables observing their potential for damage from postulated breaks in high energy pipe lines and from other hazards.

Based on the onsite visit and the NRC staff review of licensee's submittal of April 20, 1976, the following considerations required evaluation:

1. The licensee indicates that a fourth channel of protective instrumentation was installed as an optional spare at an appreciable cost in order to give the utility added flexibility of operations. Yet no other Combustion Engineering (CE) design has a three channel RPS and ESFAS. In addition, at St. Lucie cables for the four channels are routed in two groups. For example, cables in channels MA and MC are grouped together and likewise, MB and MD are grouped together. The greatest separation distance is maintained between the two groups. Four equally spaced, independent cable groups would have provided greater four channel physical separation and electrical independence.
2. Within containment, a possibility exists that a main steam line break could disable two redundant safety-related steam generator differential pressure transmitters at the 62-foot elevation. In addition, a high energy line break near piping penetrations #1, 2, 3, and 4 could possibly disable redundant RPS and ESFAS cables. Although the transmitter and cable failures may cause the respective channels to fail in the "safe" direction, the staff has not accepted a "fail-safe" design as a design basis. Therefore, high energy line hazards may affect the minimum acceptable redundancy required by IEEE Std 279-1971.
3. The RPS panels at St. Lucie are practically identical to those provided for the Calvert Cliffs Nuclear Plants of Baltimore Gas and Electric Company. A review of these panels was conducted at Calvert Cliffs and the results are applicable to St. Lucie. The wiring terminated at the RPS panel allows an associated circuit to be routed with two protection channel groups. The wiring separation design criteria within the RPS cabinet can allow an associated circuit to be routed with another protection channel. As a result of the existing separation within the RPS panel at Calvert Cliffs, the staff could not conclude that adequate four channel separation was maintained in the RPS cabinets. Because the associated circuits were not distinctively coded and because of inaccessibility to portions of the RPS panels, the staff could not verify these findings at St. Lucie. Therefore, the conclusions of the Calvert Cliffs review were applied to St. Lucie.

4. The RPS and ESFAS channel cables from the transmitters inside containment to the control room are routed in conduit or totally enclosed cable trays. The cable trays and conduit were distinctively marked in most areas. However, flammastic coatings of cables prevented precise cable tray identification. In addition, since the cables could not be individually identified and traced, the actual routing could not be verified, except at the terminations. The licensee provided quality assurance certification showing that the cables were indeed designed to be installed in their respectively marked trays. The staff audited a sample of cable trays for confirmation of the certification. The color code scheme of the cable trays coincided with the description on the quality control sheet. However, at one location at the base of containment there appeared to be an error in the color code scheme of pressurizer pressure channels. Nevertheless, since the cables themselves had not been "electronically traced", the staff could not be conclusively assured that the cables were routed in their appropriate trays. We consider that adequate cable separation is maintained for a two-out-of-four system; however, this cannot be physically verified for a two-out-of-three system with installed spare.

5. During the licensing review for the Calvert Cliffs plant, a potential single failure occurrence (hot short) was discovered that would cause the four scram actuation relays (K1, K2, K3, K4), a one-of-six logic matrix, to de-energize (the four relays are in parallel). A change was made which split the logic matrix in half and powered K1 and K2 from one vital supply and K3 and K4 from a redundant supply. This change corrected the CE design; however, it was later determined by CE that a reassignment of logic power supplies located in different bays of the Reactor Protection System Cabinet was necessary to avoid a spurious reactor trip. As a result of this change, possible damage in one bay of the RPS may be propagated to a redundant bay and/or impact two vital power inverters.

Therefore, a potential exists for a single failure event to impact two channels.

6. A review of the design of the St. Lucie Emergency DC Power system indicates that two emergency DC buses provide power to four inverters, which in turn supply power to the four RPS and ESFAS channels. Since this design does not provide

complete independence to all four channels, an overvoltage condition on one DC bus may be communicated to two channels through their respective inverters and damage ESFAS and RPS instrumentation circuits.

7. One of the associated circuits in the control and safety-related consoles of the control room have instrument cables designated "I". These low level signal cables are fed by current limited power supplies. The licensee indicates that "IA" cables, once in the safety-related cable tray system, would be assigned only to the "IA" safety-related channel. The "IA" cables should never be routed with any other safety-related cable. However, the staff noted that when the "I" cables left their respective channels in the control room, no obvious separation existed. For instance, "IA" and "IB" cables were brought close together in instrument panels. The staff noted that a power supply failure could conceivably compromise redundant associated and possible safety-related instrument cables.
8. The staff considers that a minimum of three ex-core nuclear power detector channels may be required to detect certain transients and accidents, such as rod ejection accidents, to prevent unacceptable core damage. Indefinitely bypassing one of the detector channels has not been considered in combination with other single failures. Therefore, the staff would not allow an indefinite bypass of this parameter.
9. Although the licensee has not requested that the Containment Spray Actuation System and the Recirculation Actuation System be considered for indefinite three-channel operation, four channels of these systems are necessary to satisfy the single failure criterion.

Based on the considerations noted above allowing indefinite bypass of one of the four RPS or ESFAS is not justified. However, the St. Lucie four channel system of safety-related instrumentation does have greater independence than many other two-out-of-four systems and in itself justifies a reasonable outage allowance for testing or maintenance of one channel. Otherwise, required frequent testing with the reactor at power could result in undesirable inadvertent reactor trips. The staff review conducted on these instrumentation systems for the Calvert Cliffs plants and the

St. Lucie plant has been essentially identical. Our review concluded that bypass of one of the four channels may be permitted for test and maintenance purposes for 48 hours. Also, a channel may be placed in trip for an indefinite period of time and while this condition exists, one additional channel may be placed in bypass for a period not to exceed 48 hours for the sole purpose of performing tests and maintenance on that channel.

In addition, the 48 hour bypass period has been approved recently for Calvert Cliffs Units No. 1 and No. 2, plants of a similar design, after a similar NRC staff review. The operability of the RPS and ESFAS instrumentation and modified bypass feature, as approved with this technical specification change, will assure that (1) the RPS and ESFAS trips will occur when the monitored parameter exceeds its setpoint, (2) the 2/3 or 2/4 coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service (bypassed or tripped) for testing or maintenance, and (4) sufficient system functional capability is available from the diverse parameters.

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 the 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 28, 1977

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. 15 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 its date of issuance.

The amendment modified the Technical Specifications to authorize a 48-hour bypass period of any one of the four Reactor Protection System (RPS) channels and any one of the four Engineered Safety Feature Actuation System (ESFAS) channels for testing and maintenance.

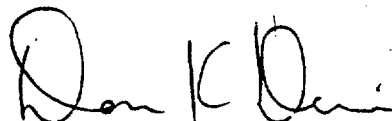
The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 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 April 20, 1976, (2) Amendment No. 15 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 33450. A single 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 Operating Reactors.

Dated at Bethesda, Maryland this 28th day of June, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION



Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors