Docket No. 50-335 SEP 8 1977

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 17 to Facility Operating License No. DPR-67 for St. Lucie Unit No. 1. The amendment consists of changes to the Technical Specifications appended to License No. DPR-67 in response to your applications referred to in the following paragraph.

The amendment:

1. modifies a Limiting Condition of Operation (LCO) to clarify that the containment vacuum relief valve may be operable during refueling operations, in accordance with your application dated August 16, 1976 (L-76-296),

2. corrects Table 3.3-5 - "Engineered Safety Features Response Times" to add the feedwater isolation time requirement and to delete the time response for the "Feedwater Flow Reduction to 5%" for a reactor trip to be consistent with existing plant design, in accordance with your application dated October 29, 1976 (L-76-377),

3. provides new figures ("Reactor Coolant System Pressure Temperature Limitations") to comply with Appendix G of 10 CFR Part 50, in accordance with your application dated March 17, 1977 (L-77-82).

4. modifies the administrative time limit for review and approval of temporary plant procedure changes to be consistent with current NRC requirements, in accordance with your application dated April 4, 1977 (L-77-107),

Const. 1
GD

OFFICE➤						
SURNAME➤						
DATE➤						

5. corrects the number of spent fuel racks to be consistent with the existing plant design, in accordance with your application dated October 19, 1976 (L-76-364),

6. modifies the equation for the power distribution limit by deleting the fuel rod bowing factor of 1.05 consistent with the NSSS vendor analysis, in accordance with your application dated July 7, 1977 (L-77-214), and

7. changes the emergency diesel generator load sequence timing requirement to be consistent with the existing plant design, in accordance with your application dated November 8, 1976 (L-76-386), as supplemented by letters dated May 31 and July 6, 1977.

Our review of your applications resulted in minor modifications to your proposed changes which have been discussed with and agreed to by your staff.

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

Sincerely,

Original signed by
M. Grotenhuis

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

Enclosures:
1. Amendment No. 17 to License No. DPR-67
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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DATE	8/29/77	8/29/77	8/26/77	8/ /77	9/6/77	9/8/77

cc w/enclosures:
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Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

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

Weldon B. Lewis
County Administrator
St. Lucie County
Post Office Box 700
Ft. Pierce, Florida 33450

cc w/enclosures and copy of FP&L filings referred to on page 1 of this letter:
Bureau of Intergovernmental Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

Hamilton Oven, Jr.,
Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
Montgomery Building
2562 Executive Center Circle, E.
Tallahassee, Florida 32301

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. 17
License No. DPR-67

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

   A. The applications for amendment by the Florida Power & Light Company (the licensee) dated August 16, 1976; October 19; October 29; November 8 (as supplemented by letter dated May 31 and July 6, 1977); March 17, 1977; April 4, and July 7, 1977, comply 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. 17, 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

Marshall Grotenhuis

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 8, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 17

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 a vertical line indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

3/4 2-2
3/4 3-16
3/4 3-17
3/4 4-23a
3/4 4-23b
3/4 4-23c
3/4 8-4
3/4 9-4
5-6
6-14

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent limits on the Power Ratio Recorder, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

b. Be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.

b. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the determination made in Specification 4.2.1.3.c, and

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the THERMAL POWER does not exceed the value determined by the following relationship:

$$\frac{L}{17.0} \times M$$

where:

1. L is the maximum allowable linear heat rate as determined from Figure 3.2-1 and is based on the core average burnup at the time of the latest incore flux map.

2. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

   1. Flux peaking augmentation factors as shown in Figure 4.2-1,

   2. A measurement-calculational uncertainty factor of 1.10,

   3. An engineering uncertainty factor of 1.03,

   4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and

   5. A THERMAL POWER measurement uncertainty factor of 1.02.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. LOSS OF POWER		
4.16 kv Emergency Bus Undervoltage (Undervoltage relays)	$\geq$ 3307 volts	$\geq$ 3307 volts

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
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
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0*/19.5**
b. Containment Fan Coolers	$\leq$ 30.0*/17.0**
c. Feedwater Isolation	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME 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

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

**Diesel generator starting and sequence loading delays not included. Offsite power available.

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
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

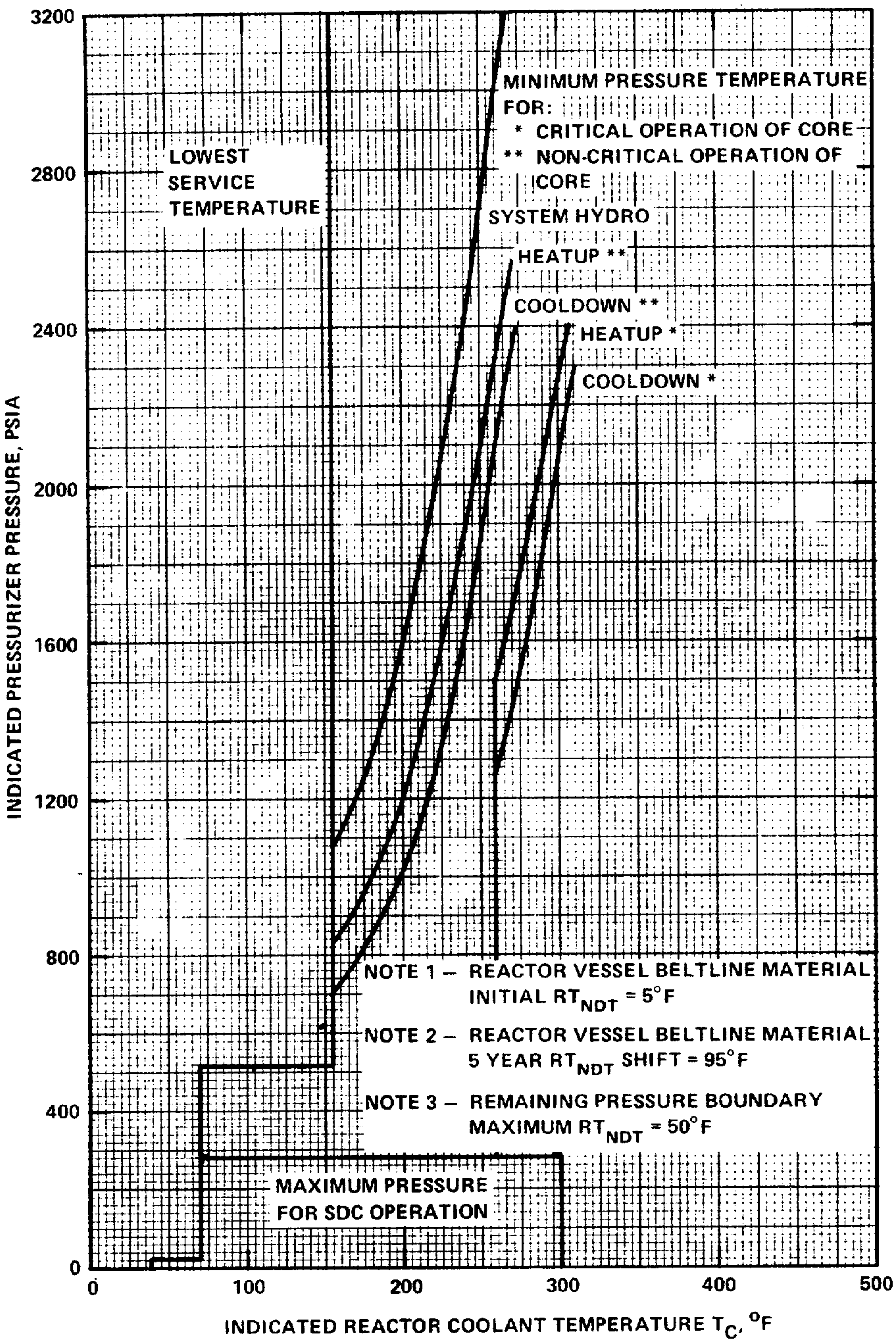

FIGURE 3.4-2a

Reactor Coolant System Pressure Temperature Limitations for up to 5 Years of Full Power Operation

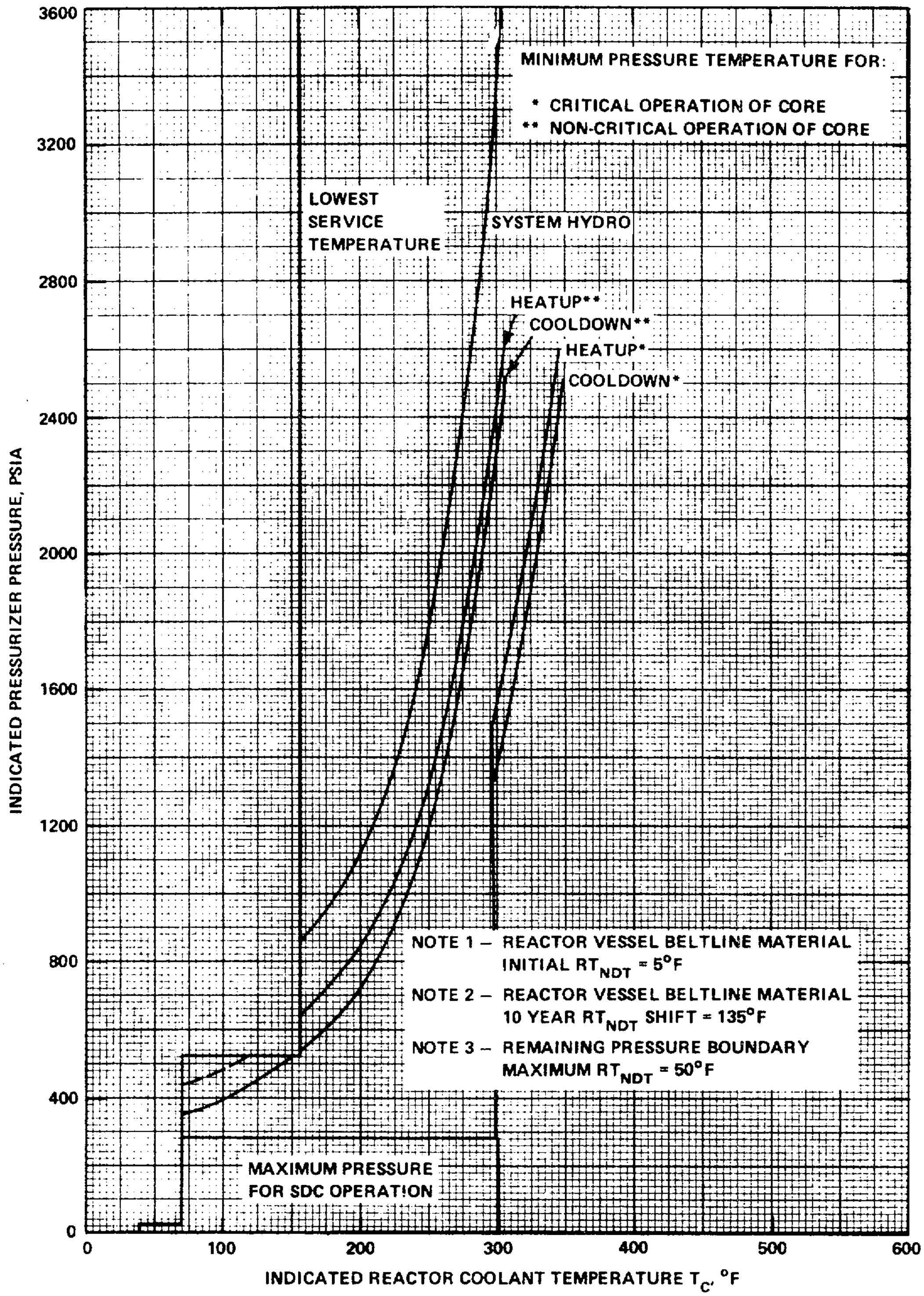

FIGURE 3.4-2b

Reactor Coolant System Pressure Temperature Limitations for up to 10 Years of Full Power Operation

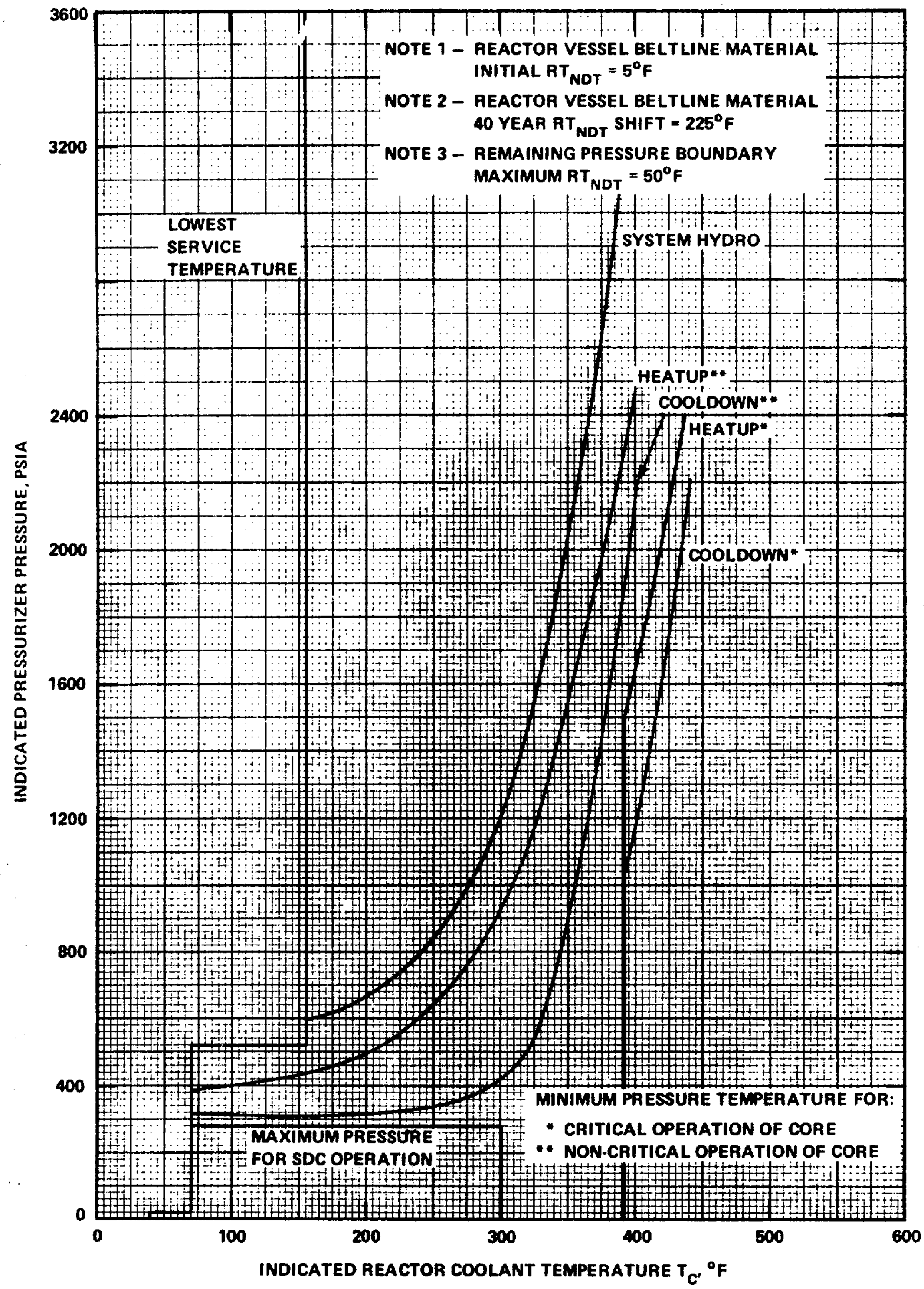

FIGURE 3.4-2c

Reactor Coolant System Pressure Temperature Limitations for up to 40 Years of Full Power Operation

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area	S	R	M	*
b. Containment (CIS)	S	R	M	6
2. PROCESS MONITORS				
b. Containment				
i. Gaseous Activity RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Particulate Activity RCS Leakage Detection	S	R	M	1, 2, 3, & 4

* With fuel in the storage pool or building

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator set shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the engine-mounted fuel tank.

2. Verifying the fuel level in the fuel storage tanks.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the engine-mounted tank.

4. Verifying the diesels start from ambient condition.

5. Verifying the generator is synchronized, loaded to $\geq$ 1300 kw, and operates for $\geq$ 60 minutes.

6. Verifying the diesel generator set is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment.

c. At least once per 18 months during shutdown by:

1. Subjecting the diesels to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a load of $\geq$ 600 hp without tripping.

3. Simulating a loss of offsite power in conjunction with a safety injection actuation signal, and:

a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesels start from ambient condition on the auto-start signal, energize the emergency busses with permanently connected loads, energize

SURVEILLANCE REQUIREMENTS (Continued)

the auto-connected emergency loads through the load sequencing system and operate for $\geq$ 5 minutes while the generator is loaded with the emergency loads.

c) Verifying that on the safety injection actuation signal, all diesel generator trips, except engine overspeed and generator differential, are automatically bypassed.

4. Verifying the diesel generator set operates for $\geq$ 60 minutes while loaded to $\geq$ 3500 kw.

5. Verifying that the auto-connected loads to each diesel generator set do not exceed the 2000 hour rating of 3730 kw.

6. Verifying that the automatic sequence timers are OPERABLE with the interval between each load block within $\pm$ 1 second of its design interval.

d. At least once per 18 months by verifying that each fuel transfer pump transfers fuel from each fuel storage tank to the engine mounted fuel tanks on each diesel via the installed cross connection lines.

4.8.1.1.3 The Class 1E underground cable system shall be demonstrated OPERABLE:

a. Within 30 days after the movement of any loads in excess of 80% of the ground surface design basis load over the cable ducts by pulling a mandrel with a diameter of at least 80% of the duct's inside diameter through a duct exposed to the maximum loading (duct nearest the ground's surface) and verifying that the duct has not been damaged.

b. At least once per 18 months, during shutdown, by:

1. Selecting on a rotating basis at least 3 (one each in the ducts between the diesel generators and the switchgear, between the switchgear and the component cooling water pump motors, and between the switchgear and the intake cooling water pump motors) Class 1E 5000 volt underground cables and megger testing the selected cables at a minimum test

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for a minimum of 72 hours.

APPLICABILITY: During movement of irradicated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration, except as provided in Table 3.6-2 of Specification 3.6.3.1, providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic containment isolation valve, or

3. Be capable of being closed by an OPERABLE containment vacuum relief valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment isolation valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment isolation valves per the applicable portions of Specifications 4.6.3.1.1 and 4.6.3.1.2.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and 8 part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the original design provisions contained in Section 4.2.3.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,100 $\pm$ 180 cubic feet at a nominal T_{avg} of 567°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with a nominal center-to-center distance of 21 inches for new fuel assemblies and 18 inches for spent fuel assemblies placed in the

CRITICALITY (Continued)

storage racks to ensure a K_{eff} equivalent to ≤ 0.95 with the storage pool filled with unborated water. The K_{eff} of ≤ 0.95 includes the conservative assumptions as described in Section 9.1 of the FSAR.

DRAINAGE

5.6.2 The fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 310 fuel assemblies of which the 45 fuel assemblies in the 5 x 5 array and 5 x 4 array nearest the fuel cask compartment shall have decayed for at least 1000 hours.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as seismic Class I in Section 3.2.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 3.7 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower location shall be as shown on Figure 5.1-1.

5.9 COMPONENT CYCLE OR TRANSIENT LIMITS

5.9.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Vice President of Power Resources and to the CNRB within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRG. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the CNRB and the Director of Power Resources within 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the FRG and approved by the Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the FRG and approved by the Plant Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations, shall be in accordance with the Regulatory Position in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

c. Inservice Inspection Program Reviews, Specifications 4.4.10.1 and 4.4.10.2.

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Sealed Source leakage in excess of limits, Specification 4.7.9.1.3.

f. Seismic event analysis, Specification 4.3.3.3.2.

g. Beach survey results, Specification 4.7.6.1.1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 17 TO LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT NO. 1

DOCKET NO. 50-335

INTRODUCTION

By the following applications, Florida Power and Light Company (FPL) requested amendments to St. Lucie Unit No. 1 License No. DPR-67. The amendments would change the Technical Specifications to:

1. modify a Limiting Condition of Operation (LCO) to clarify that the containment vacuum relief valve may be operable, during refueling operations, in accordance with your application dated August 16, 1976 (L-76-296),

2. correct Table 3.3-5 - "Engineered Safety Features Response Times" to add the feedwater isolation time requirement and to delete the time response for the "Feedwater Flow Reduction to 5%" for a reactor trip to be consistent with existing plant design, in accordance with your application dated October 29, 1976 (L-76-377),

3. provide new figures ("Reactor Coolant System Pressure Temperature Limitations") to comply with Appendix G of 10 CFR Part 50, in accordance with your application dated March 17, 1977 (L-77-82),

4. modify the administrative time limit for review and approval of temporary plant procedure changes to be consistent with current NRC requirements, in accordance with your application dated April 4, 1977 (L-77-107),

5. correct the number of spent fuel racks to be consistent with the existing plant design, in accordance with your application dated October 19, 1976 (L-76-364),

6. modify the equation for the power distribution limit by deleting the fuel rod bowing factor of 1.05 consistent with the NSSS vendor analysis, in accordance with your application dated July 7, 1977 (L-77-214), and

7. change the emergency diesel generator load sequence timing requirement to be consistent with the existing plant design, in accordance with your application dated November 8, 1976 (L-76-386), as supplemented by letters dated May 31 and July 6, 1977.

Our review of the applications resulted in minor modifications to FPL's proposed changes which have been discussed with and agreed to by FPL staff.

DISCUSSION AND EVALUATION

1. Operability of the Containment Vacuum Relief Valve During Refueling Operations

   The containment vacuum relief valves are designed to preclude damage to the containment if the external atmospheric pressure exceeds the internal containment pressure. During refueling operations containment penetrations are required to be in a status to preclude leakage of radioactive gas to the outside environment. As a result of a reportable occurrence (LER #76-4) during St. Lucie Unit No. 1 initial fueling operations, FPL proposed a technical specification change for clarification. During initial fueling the containment purge fan had created a partial vacuum inside containment which opened the containment vacuum relief valve. Thus, a literal interpretation of the specification requirements for the containment penetration status would indicate that a direct access path existed from the containment interior atmosphere to the outside atmosphere. However, this was not true as radioactive gas could not exit from the containment interior through the open vacuum relief valve with a negative pressure inside containment. Any flow of air would be into containment from the outside atmosphere.

   We have reviewed the proposed change submitted by FPL letter of August 16, 1976. The proposed change has been modified by adding Specification 3.9.4.c.3, thus requiring an operable status for the containment vacuum relief valve. FPL staff agrees with the change as made in this amendment. Since the requirement on containment penetration closure has not been modified from that previously reviewed and since radioactive material inside containment is still restricted from leakage to the outside environment, the change, as modified, is acceptable.

2. Engineered Safety Features - Feedwater Isolation and Reactor Trip

a. Background

Revision 57 (February 27, 1976) to the St. Lucie Final Safety Analysis Report included changes relating to termination of feedwater flow to the steam generators in the event of a steam line break accident. For this accident, feedwater flow to the steam generators was terminated upon receipt of a low steam generator pressure signal. As a secondary means of isolating feedwater, flow was reduced to 5% by a reactor trip signal. However, with the addition of feedwater pump isolation valves, flow reduction to 5% was no longer necessary for protection against any accident. However, Technical Specification Table 3.3-5 "Engineered Safety Features Response Times" erroneously failed to list "Feedwater Isolation" under item 1.e and 6.b. It also erroneously listed "Feedwater Flow Reduction to 5%" as item 8.a.

b. Evaluation

Our review indicates that the existing technical specifications are in error and did not reflect the final facility design. The licensee has provided backup isolation of feedwater flow by feedwater pump isolation valves which give a more positive isolation of feedwater flow then was provided by the feedwater regulating valves.

Therefore, the proposed technical specification changes would simply delete an unnecessary requirement to measure response time for "Feedwater Flow Reduction to 5%" and add a requirement to assure that response time for the "Feedwater Isolation $\leq$ 60 seconds". Although the changes were considered editorial corrections, the NRC staff evaluated the changes and concluded that Technical Specification Table 3.5-5 changes are acceptable.

3. Pressure Temperature Operating Limitation Curves

On February 15, 1977 the NRC staff notified FPL that the original Reactor Coolant System Pressure Temperature Limitations for St. Lucie Unit No. 1 may not comply with paragraph IV.A.2.b of Appendix G, 10 CFR Part 50. Appendix G, Paragraph IV.A.2.b requires that high stressed regions, such as nozzles, have margins of safety comparable to margins for shells remote from discontinuities.

During the first few years of service life, materials in regions of discontinuities will usually have limiting high stresses. Later in life, materials in the vessel beltline region will generally become more limiting because of radiation damage.

For the FPL proposed operating limit curves, the beltline materials are assumed to have a reference temperature (RT_{NDT}) equal to 100°F at 5 effective full power years (EFPY). Because the maximum initial value of RT_{NDT} of the remaining vessel materials is 50°F, the limiting RT_{NDT} is 50°F higher than the RT_{NDT} of all higher stressed regions. This complies with the requirements of Branch Technical Position MTEB 5-2, which is attached to Standard Review Plan 5.3.2, "Pressure-Temperature Limits," and is an acceptable means of complying with paragraph IV.A.2.b of Appendix G, 10 CFR Part 50.

We have reviewed the proposed operating limits included in FPL's letter dated March 17, 1977, as Figure 3.4-2a for up to 5 EFPY and conclude that Figure 3.4-2a complies with Appendix G, 10 CFR Part 50 and is acceptable. Conformance with Appendix G in establishing safe operation limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitute an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50. FPL also submitted pressure temperature limits for operation to 10 EFPY and to 40 EFPY. These curves appear to be acceptable. However, the NRC staff will defer final evaluation of these curves until additional radiation information is provided by FPL in accordance with requirements of Technical Specification 4.4.9.1.c based on reactor vessel material irradiation surveillance specimen examinations.

4. <u>Administrative Time Limit for Review and Approval of Temporary Procedure Changes</u>

The administrative control section of the Technical Specifications for St. Lucie Unit No. 1 contain requirements relating to procedures used during plant operation. Temporary changes are allowed only if the intent of the procedure is not altered.

Additionally, a review was required within 7 days by the on-site safety review committee and approval is then required by the Plant Manager. Operating experience has shown that 7 days is overly restrictive requiring unnecessary scheduling of the safety review committee. Further, such expedited handling of temporary procedure changes of little safety significance may detract from more important safety considerations. For these reasons and since plants being licensed today are allowed up to 14 days for the same review and approval function, the proposed change to specification 6.8.3.c made in FPL letter dated April 4, 1977, is acceptable.

5. Correct the Number of Spent Fuel Racks

FPL reported that during a review of the St. Lucie Unit No. 1 a conflict was noted in the number of spent fuel storage racks described in the FSAR and Technical Specifications. The design drawing shows 310 storage cells while the text (section 9.1.2.2) states 304 cells. Technical Specification 5.6-3 states the capacity as 304 cells. The spent fuel pool, as constructed, actually contains 310 cells. Thus, FPL evaluated the significance of the conflict and proposed to change the technical specifications to agree with the existing plant design.

We have reviewed the FPL proposal and have evaluated the cooling and reactivity capabilities to assure that the change to the Technical Specifications is purely editorial. Since the change in Technical Specification does not involve a modification to the facility and resulted from an editorial error in the FSAR which was carried over into the Technical Specification, we find the proposed change acceptable.

6. Power Distribution Limit Equation

If the excore detectors are used to monitor the core power distribution, Technical Specification 4.2.1.3.c will require that a verification be done every 31 days to ensure the thermal power does not exceed the value determined by the following relationship:

$$\frac{LM}{17.00}$$

where L is the maximum allowable linear heat rate and M is the maximum allowable thermal power level. To obtain this proposed formula FPL has removed a fuel rod bowing factor of 1.05 from the denominator of the formula. The basis for removal of the rod bowing factor was included in a Combustion Engineering (CE) letter, A. E. Scherer to D. F. Ross "Fuel and Poison Rod Bowing Effects in

Combustion Engineering Fuel" dated July 16, 1976. In the submittal CE showed that the uncertainty factors which are presently applied to the CE design for 14x14 fuel are sufficiently large to account for the effects of rod bowing. These uncertainty factors are the 8% factor applied for nuclear power distribution measurement uncertainty and the 3% engineering factor.

We have reviewed the generic rod bowing information submitted by CE and conclude that the uncertainty factors which are presently included in the safety analysis for St. Lucie Unit No. 1, which are described above, are sufficient to account for bowing effects. A similar change has been approved for Calvert Cliffs.

Therefore, FPL's proposal to delete the rod bowing factor of 1.05 from the equation shown in Technical Specification 4.2.1.3.c and from 4.2.1.4.b.6 are acceptable.

7. <u>Emergency Diesel Generator Load Sequencing</u>

a. <u>Background</u>

FPL in letters of November 8, 1976, and May 31 and July 6, 1977 requested a change to Technical Specification 4.8.1.1.2.c.6. The specification is a surveillance requirement for the automatic sequencers, which initiate the timed sequence loading of the emergency diesel generators. FPL proposed that the provision for verification of timer limits, which now requires an accuracy within ± 10% of setpoint limits, be changed to ± 1 second of setpoint. Their reason for the change is that verification of the timer setpoint accuracy for time periods less than one second cannot be assured.

b. <u>Evaluation</u>

The NRC staff considers that the licensee has misunderstood the intent of the surveillance requirement. The present provision in the Technical Specifications for St. Lucie Unit No. 1 is as follows:

"Verifying that the automatic sequence timers are OPERABLE with each load sequence time within ± 10% of its required value".

The licensee's proposed change to the specification is as follows:

"Verifying that the automatic sequence timers are OPERABLE with each load sequence time within $\pm$ 1 second of its required value".

The licensee has interpreted the existing specifications, i.e., $\pm$ 10% of the required value as follows: a load that is required to be sequenced at 30 seconds would be acceptably sequenced if the time of sequence was between 27 and 33 seconds, i.e., $\pm$ 10% of its required value. This interpretation is not what was intended by the NRC staff. The objectives for diesel generator load sequencing are presented in Regulatory Guide (RG) 1.9. Position 4 of the guide presents the voltage and frequency values and relates them to the sequence time interval (delta-T). The objective of this provision of the guide, i.e., verification of automatic sequencer timers, is to assure an adequate time interval (delta-T) between application of loads to permit the diesel generator voltage and frequency to recover to acceptable levels. To ensure this, the NRC staff considers it necessary to verify that each sequential setpoint "interval" is not changed by more than $\pm$ 10%. Therefore, referring back to this example of a load sequenced at 30 seconds, it would be necessary to determine the $\pm$ 10% delta-T value, by knowing the load sequence time before and the load sequence time after the 30 second setpoint. The $\pm$ 10% delta-T value when using a typical 3 seconds delta-T between loads, allows a variation of $\pm$ 0.3 seconds and not the 3 seconds as would be the case using the licensee's interpretation.

The licensee has supported the proposed change by preoperational load test results from both emergency diesel generators A and B. The test results demonstrate that the diesel generator's voltage and frequency return to normal within acceptable limits as provided for in RG 1.9. In fact, the frequency and voltage recover to acceptable levels for the worst case sequential loading in less than 22 percent of a sequential time interval. The test results show the voltage decreasing to 92.4 percent of nominal (4160 volts) and returning to nominal voltage in about 0.66 seconds for the largest sequenced load (575 kW) at three seconds after the diesel generator breaker closed. The most severe frequency decrease was to about 59.25 hertz and returned to normal (60 hertz) in about 0.6 seconds. These data more than satisfy the requirements of RG 1.9 which would allow the voltage and frequency to decrease to 75 and 95 percent, respectively, of nominal and return to within 10 and 2 percent of normal, respectively, within 40 percent of load sequence interval (three seconds). The data support the licensee's contention that verifying that the sequence timers are operable within + 1 second of each load sequence time interval will not result in overloading of the diesel generators and will satisfy the provisions of RG 1.9. Therefore, we conclude the proposed change to Technical Specification 4.8.1.1.2.c.6 is acceptable when modified to prevent future misinterpretations. FPL has agreed to this change.

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: September 8, 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. 17 to Facility Operating License No. DPR-67, issued to Florida Power & Light Company (the licensee), which revised the license and its appended 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 changed the Technical Specifications for the facility to: (1) clarify that the containment vacuum relief valve may be operable during refueling, (2) add a feedwater isolation response time requirement and delete the response time for the "Feedwater Flow Reduction to 5%" for a reactor trip, (3) provide new figures ("Reactor Coolant System Pressure Temperature Limitations") to comply with Appendix G of 10 CFR 50, (4) modify the administrative time allowed for review and approval of temporary plant procedure changes to be consistent with current NRC requirements, (5) correct the number of spent fuel racks to be consistent with the existing plant design, (6) modify the equation for the power distribution limit by deleting the fuel rod bowing factor of 1.05 consistent with the NSSS vendor analysis, and (7) change the emergency diesel generator load sequence timing requirement to be consistent with the existing plant design.

The applications for the amendment comply 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 applications for amendment dated August 16, 1976; October 19, October 29; November 8 (as supplemented by letters dated May 31 and July 6, 1977); March 17, 1977; April 14 and July 7, 1977, (2) Amendment No. 17 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 8th day of September, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall Grotenhuis

Marshall Grotenhuis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors