December 6, 1989

Docket No. 50-335

DISTRIBUTION
See attached sheet

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

Dear Mr. Goldberg:

SUBJECT: ST. LUCIE UNIT 1 - ISSUANCE OF AMENDMENT RE: REACTOR VESSEL MATERIAL SURVEILLANCE (TAC NO. 75045)

The Commission has issued the enclosed Amendment No. 100 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated October 2, 1989.

This amendment changes the St. Lucie, Unit 1, Technical Specifications to revise the reactor vessel material surveillance capsule removal schedule. The revised capsule removal schedule is consistent with the recommendations of ASTM E 185-82 as required by Appendix H to 10 CFR Part 50.

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

Sincerely,

Original signed by

Jan A. Norris, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 100 to DPR-67
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDII-2*	PM:PDII-2*	PM:PDII-2	D:PDII-2	OGC*
DMiller	SHoffman:jd	JNorris	HBerkow	
11/17/89	11 / 20/89	~~11/20/89~~ 12/5/89	12/5/89	11/22/89

*See sheet for previous concurrences

DF01
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Mr. J. H. Goldberg
Florida Power & Light Company

St. Lucie Plant

cc:
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Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
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DATED: December 6, 1989

AMENDMENT NO. 100 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1

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cc: Plant Service list

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. 100
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 October 2, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 100, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amenament is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 6, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 100

TO FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
3/4 4-24	3/4 4-24
B 3/4 4-7	B 3/4 4-7

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

Specimen Location on Vessel Wall	Lead Factor[2]	Approximate Removal Schedule (EFPY)	Predicted Fluence (n/cm^2)
97°[1]	1.54	4.67	5.5×10^{18}
104°	1.02	10	8.78×10^{18}
284°	1.02	18	1.58×10^{19}[3]
263°	1.54	21	2.78×10^{19}
277°	1.54	32	4.24×10^{19}
83°	1.54	Standby	---

NOTES

1) Information for this capsule is actual
2) Ratio of capsule fluence divided by the fluence at the controlling weld
3) Approximate end of life 1/4T fluence

The heatup and cooldown limit curves (Figures 3.4-2a and 3.4-2b) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50°F/hr and for any cooldown rate of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the applicable service period.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be calculated based upon the fluence. The heatup and cooldown limit curves shown on Figures 3.4-2a and 3.4-2b include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82, reactor vessel material surveillance specimens installed near the inside wall of the reactor vessel in the core area. The capsules are scheduled for removal at times that correspond to key accumulated fluence levels within the vessel through the end of life. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, measured ΔRT_{NDT} for surveillance samples can be applied with confidence to the corresponding material in the reactor vessel wall. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2a and 3.4-2b for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements for Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been estimated to be 90°F. The Lowest Service Temperature limit line shown on Figures 3.4-2a and 3.4-2b is based upon this RT_{NDT} since Article NB-2332 of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	NI %	P %	NDTT F	50 FT-LB/35 MIL TEMP F LONG[1]	50 FT-LB/35 MIL TEMP F TRANS[1,2]	RTNDT[4] F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS[3]
Vessel Flange Forging	C-1-1	A508Cl.2	-	-	.008	+20	+70	+90	+30	133	86
Bottom Head Plate	C-10-1	A533BCl.1	-	-	.010	-40	+42	+62	+2	120	78
Bottom Head Plate	C-9-2	A533BCl.1	-	-	.011	-40	-18	+2	-40	146	95
Bottom Head Plate	C-9-3	A533BCl.1	-	-	.013	-70	-20	0	-60	148	96
Bottom Head Plate	C-9-1	A533BCl.1	-	-	.011	-30	+10	+30	-30	138	90
Inlet Nozzle	C-4-3	A508Cl.2	-	-	.005	0	0	+20	0	111	72
Inlet Nozzle	C-4-2	A508Cl.2	-	-	.004	0	+20	+40	0	146	95
Inlet Nozzle	C-4-1	A508Cl.2	-	-	.005	+10	-25	-5	10	144	94
Inlet Nozzle	C-4-4	A508Cl.2	-	-	.004	0	+10	+30	0	139	90
Inlet Nozzle Ext.	C-16-3	A508Cl.2	-	-	.001	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-2	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-1	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90
Inlet Nozzle Ext.	C-16-4	A508Cl.2	-	-	.011	+10	+52	+72	+12	139	90

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 100

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 October 2, 1989, Florida Power & Light Company (FPL) submitted a request to amend the St. Lucie, Unit 1, Technical Specifications Table 4.4-5 "Reactor Vessel Material Irradiation Surveillance Schedule" and the related Bases 3/4.4.9 to revise the reactor vessel material surveillance capsule withdrawal schedule to be consistent with the recommendations of ASTM E 185-82. ASTM E 185-82 recommends a minimum number of capsules for removal and testing based on the predicted transition temperature shift at the vessel inside surface. The current surveillance program in use at Unit 1 is based on ASTM E 185-73 which was the edition in effect when the reactor vessel was purchased. Appendix H to 10 CFR Part 50 states that the requirements of ASTM E 185-82 must be met, to the extent practical, for capsule withdrawals after July 26, 1983. This amendment revises the schedule for removal of surveillance capsules to comply with the recommendations of ASTM E 185-82. The proposed schedule is based on the projected accumulated neutron fluence as permitted by ASTM E 185-82.

EVALUATION

The NRC staff has reviewed the requested amendment and finds that the proposed withdrawal schedule is consistent with the recommendations of ASTM E 185-82 with the exception of the schedule for removal of the second capsule. The proposed schedule for removal of the second capsule is approximately one to two effective full power years earlier than that recommended by ASTM E 185-82.

The justification for the early removal of the second capsule is as follows. FPL indicated in a teleconference on October 17, 1989, that the capsules at St. Lucie Unit 1 have small lead factors resulting in later capsule withdrawals. The lead factor is determined by the ratio of the neutron fluence at the surveillance capsule to the neutron fluence at the reactor vessel inside surface. With a high lead factor, surveillance capsules will demonstrate a shift in transition temperature well in advance of the shift in the transition temperature of the material in the reactor vessel. High lead factors result in earlier removal of samples and earlier validation of the predictive methodology used to calculate the end of life reactor vessel material conditions. The St. Lucie, Unit 1, lead factors are smaller due to the attachment of the surveillance capsules to the inner radius of the reactor vessel. This location results in lower accumulated neutron fluence and the later withdrawal of surveillance capsules per the recommendations of ASTM E 185-82. Thus, few capsule data would be available early in plant life. Although the first capsule test data at St. Lucie, Unit 1, were consistent with the predictive model in Regulatory Guide 1.99, Revision 2, the licensee prefers to examine the second capsule earlier than recommended by ASTM E 185-82 to further confirm the application of the predictive model at the plant. The staff concurs with the licensee's position.

The revised material surveillance capsule removal schedule in this amendment will result in better predictions of reactor vessel material conditions and conforms with the recommendations of ASTM E 185-82 per the requirements of Appendix H to 10 CFR Part 50. Based on the staff's evaluation, the staff finds the proposed amendment acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 6, 1989

Principal Contributor:
S. Lee