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RECEIVED OCT 27 1981 U.S. NUCLEAR REGULATORY COMMISSION

Docket No. 50-335 ✓

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

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 44 to Facility Operating License No. DPR-67 for St. Lucie Unit No. 1. This amendment consists of changes to your Technical Specifications in response to your applications dated May 11 and July 23, 1981.

The amendment changes the Technical Specifications by clarifying the test requirements for snubbers and by adding the requirement that mechanical snubbers be tested. Also the requirement that NRC approved sleeves be used in control element assembly guide tubes was added to the Technical Specification as a Design Feature.

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

Sincerely,

Original signed by:

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

cp
1

Enclosures:
1. Amendment No. 44 to DPR-67
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

note typo on p. 2 SER
as to form of amendment
+ F.R. notice only

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SURNAME	PMKreutzer	CNelson/pn	RAClark	TMNovak	W.Paton	
DATE	10/2/81	10/2/81	10/2/81	10/5/81	10/9/81	

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Florida Power & Light Company

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Jensen Beach, Florida 33457

cc w/enclosure(s) and incoming
dated: 5/11/81, 7/23/81

Bureau of Intergovernmental
Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. DPR-67

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

   A. The applications for amendment by Florida Power & Light Company (the licensee) dated May 11 and July 23, 1981, 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 applications, the provision of the Act, and the rules and regulations of the Commission;

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

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

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

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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. 44, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 14, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 44

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.

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

BASES

3/4 7.10 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10 All snubbers listed in Tables 3.7-2a and 3.7-2b shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.10 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be performed after four months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Tables 3.7-2a and 3.7-2b. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	123 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection may be lengthened one step the first time and two steps thereafter if no inoperable snubbers are found.

#The provisions of Specification 4.0.2 are not applicable.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and/or (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.10.d or 4.7.10.e, as applicable.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample* (10% of the snubbers listed in Tables 3.7-2a and 3.7-2b) shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.10.d or 4.7.10.e, an additional 10% of that type of snubber shall be functionally tested. Functional testing shall continue until no additional snubbers are found inoperable or all snubbers listed in Tables 3.7-2a and 3.7-2b have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers.

Snubbers identified in Tables 3.7-2a and 3.7-2b as "Especially Difficult to Remove" or in "High Exposure Zones During Shutdown" shall also be included in the representative sample.** Tables 3.7-2a and 3.7-2b may be used jointly or separately as the basis for the sampling plan.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers shall not result in additional functional testing due to failure.

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*The requirements of this section for functionally testing mechanical snubbers may be waived until startup following the fifth refueling outage for Unit 1.

**Permanent or other exemptions from the functional testing for individual snubbers in these categories may be granted by the Commission only if justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.

2. Activation (restraining action) is achieved in both tension and compression.

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designed service life is based shall be maintained as required by Specification 6.10.2.m.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber listed in Tables 3.7-2a and 3.7-2b shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded by more than 10% prior to the next scheduled snubber service life review. If the indicated service life will be exceeded by more than 10% prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. The results of the reevaluation may be used to justify a change to the service life of the snubber. This reevaluation, replacement or reconditioning shall be indicated in the records.

TABLE 3.7-2a
SAFETY RELATED HYDRAULIC SNUBBERS*

FPL LOCATION NO.	MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
001	SS-1 1A	MS, Steam Gen. 1A, Evel. 62′	I	No	Yes
002	SS-2 1A	MS, Steam Gen. 1A, Evel. 62′	I	No	Yes
003	SS-3 1A	MS, Steam Gen. 1A, Evel. 62′	I	No	Yes
004	SS-4 1A	MS, Steam Gen. 1A, Evel. 62′	I	No	Yes
005	SS-5 1A	MS, Steam Gen. 1A, Evel. 62′	I	No	Yes
006	SS-6 1A	MS, Steam Gen. 1A, Evel. 62′	I	No	Yes
007	SS-7 1A	MS, Steam Gen. 1A, Evel. 62′	I	No	Yes
008	SS-8 1A	MS, Steam Gen. 1A, Evel. 62′	I	No	Yes
009	SS-1 1B	MS, Steam Gen. 1B, Elev. 62′	I	No	Yes
010	SS-2 1B	MS, Steam Gen. 1B, Elev. 62′	I	No	Yes
011	SS-3 1B	MS, Steam Gen. 1B, Elev. 62′	I	No	Yes
012	SS-4 1B	MS, Steam Gen. 1B, Elev. 62′	I	No	Yes
013	SS-5 1B	MS, Steam Gen. 1B, Elev. 62′	I	No	Yes
014	SS-6 1B	MS, Steam Gen. 1B, Elev. 62′	I	No	Yes
015	SS-7 1B	MS, Steam Gen. 1B, Elev. 62′	I	No	Yes
016	SS-8 1B	MS, Steam Gen. 1B, Elev. 62′	I	No	Yes
017	1A1	RC, RCP Motor 1A1, Elev. 57′	I	No	No
018	1A2	RC, RCP Motor 1A1, Elev. 57′	I	No	No
019	1B1	RC, RCP Motor 1A1, Elev. 57′	I	No	No
020	1B2	RC, RCP Motor 1A1, Elev. 57′	I	No	No

TABLE 3.7-2a (CONTINUED)

SAFETY RELATED HYDRAULIC SNUBBERS*

FPL LOCATION NO.	MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
033	MS 649-319	MS, Reactor Bldg. Elev. 82′	A	No	No
034	MS 548-5	MS, Reactor Bldg. Elev. 82′	A	No	No
035	MS 1076-3164	MS, M.S. Trestle, Elev. 62′	A	No	No
036	MS 649-314	MS, Reactor Bldg. Elev. 55′	I	No	No
037	MS 649-314	MS, Reactor Bldg. Elev. 82′	I	No	No
038	MS 649-310	MS, Reactor Bldg. Elev. 50′	I	No	No
039	MS 649-304A	MS, Reactor Bldg. Elev. 30′	A	No	Yes
040	MS 548-9	MS, Reactor Bldg. Elev. 50′	I	No	Yes
041	MS 548-9	MS, Reactor Bldg. Elev. 50′	I	No	Yes
042	BF 549-7	BF, Reactor Bldg. Elev. 40′	I	No	No
043	BF 549-7	BF, Reactor Bldg. Elev. 40′	I	No	No
044	BF 549-8	BF, Reactor Bldg. Elev. 40′	I	No	Yes
047	BF 549-17	BF, Reactor Bldg. Elev. 36′	A	No	Yes
052	BF 549-17	BF, Reactor Bldg. Elev. 36′	A	No	No
053	SI 968210	SI, Reactor Bldg. Elev. 16′	I	No	No
058	SI 969 1216	SI, Reactor Bldg. Elev. 18′	A	No	No
061	MS 549-11	SI, Reactor Bldg. Elev. 18′	A	No	No
066	MS 549-11	SI, Reactor Bldg. Elev. 20′	I	No	No
073	SI 972-6240	SI, Reactor Bldg. Elev. 16′	I	No	No
074	SI 973-240	SI, Reactor Bldg. Elev. 18′	A	No	No
076	SI 973-6224	SI, Reactor Bldg. Elev. 18′	A	No	No
077	SI 868 64	SI, RAB, Elev. 4′	A	No	No
079	SI 868-163	SI, RAB, Elev. 4′	A	No	No
080	SI 868 410	SI, RAB, Elev. 4′	A	No	No
081	SI 676-67	SI, RAB, Elev. 4′	A	No	No
082	SI 676 67	SI, RAB, Elev. 4′	A	No	No
083	SI 676-105	SI, RAB, Elev. 4′	A	No	No

TABLE 3.7-2a (CONTINUED)

SAFETY RELATED HYDRAULIC SNUBBERS*

FPL LOCATION NO.	MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
084	SI 676-105	SI, RAB, Elev. 4′	A	No	No
086	SI 676 129	SI, RAB, Elev. 4′	A	No	No
087	SI 676-2481	SI, RAB, Elev. 24′	A	No	No
110	SI 676 247	SI, RAB, Elev. 30′	A	No	No
111	SI 676-2475A	SI, RAB, Elev. 30′	A	No	No
112	SI 676 4505	SI, RAB, Elev. 7′	A	No	No
114	SI 972-6240	SI, RAB, Elev. 4′	A	No	No
091	SPS-417	Pressurizer Spray, Reactor Bldg. Elev. 50′	I	No	No
090	SPS-27	Pressurizer Spray, Reactor Bldg. Elev. 50′	I	No	No
092	SPS-467	Pressurizer Spray, Reactor Bldg. Elev. 80′	A	No	No
093	SPS-777	Pressurizer Spray, Reactor Bldg. Elev. 80′	A	No	No

TABLE 3.7-2a (CONTINUED)
SAFETY RELATED HYDRAULIC SNUBBERS*

FPL LOCATION NO.	MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION (A or I)	ACCESSIBLE OR INACCESSIBLE ("A" or "I")	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE
096	CC-1865-9	CC, Reactor Bldg, Elev. 25′	A	No	No
088	CC-1899-48	CC, Reactor Bldg, Elev. 25′	A	No	No
089	CC-1852-6241	CC, Reactor Bldg, Elev. 25′	A	No	No
101	CC-17-1	CC, RAB, Elev. 20′	A	No	No
102	MS-649-313	CC, RAB, Elev. 26′	A	No	No
104	CC-21-1	CC, RAB, Elev. 20′	A	No	No
103	BF-549-7	CC, RAB, Elev. 26′	A	No	No
105	CC-23-2	CC, RAB, Elev. 26′	A	No	No
106	CH-3-40	CH, RAB, Elev. 34′	A	No	No
107	CH-3-75	CH, RAB, Elev. 23′	A	No	No
108	MS-649-313	MS, Reactor Bldg, Elev. 80′	I	No	No
109	MS-649-313	MS, Reactor Bldg, Elev. 80′	I	No	No
097	MS-649-314	MS, Reactor Bldg, Elev. 80′	I	No	No
099	MS-649-314	MS, Reactor Bldg, Elev. 80′	I	No	No

*Snubbers may be added to or removed from safety related systems without prior License Amendment to Table 3.7-2a provided a revision to Table 3.7-2a is included with the next License Amendment request.

**Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-2a is included with the next License Amendment request.

TABLE 3.7-2b

SAFETY RELATED MECHANICAL SNUBBERS*

FPL LOCATION NO.	MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION (A or I)	ACCESSIBLE OR INACCESSIBLE (Yes or No)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE
021	RC 005-34A	RC, Reactor Bldg, Elev. 68′	A	NO	NO
022	RC 005-34B	RC, Reactor Bldg, Elev. 68′	A	NO	NO
023	RC 005-36	RC, Reactor Bldg, Elev. 68′	A	NO	NO
025	RC 005-12B	RC, Reactor Bldg, Elev. 80′	A	NO	NO
026	RC 005-12B	RC, Reactor Bldg, Elev. 80′	A	NO	NO
024	RC 005-12A	RC, Reactor Bldg, Elev. 80′	A	NO	NO
028	RC 005-55C	RC, Reactor Bldg, Elev. 80′	A	NO	NO
027	RC 005-55B	RC, Reactor Bldg, Elev. 80′	A	NO	NO
029	RC 005-62A	RC, Reactor Bldg, Elev. 80′	A	NO	NO
030	RC 005-89	RC, Reactor Bldg, Elev. 80′	A	NO	NO
031	RC 005-90	RC, Reactor Bldg, Elev. 80′	A	NO	NO
032	RC 005-98	RC, Reactor Bldg, Elev. 80′	A	NO	NO
045	BF 549-11	BF, Reactor Bldg. Elev. 50′	I	NO	NO
046	BF 549-11	BF, Reactor Bldg. Elev. 50′	I	NO	NO
048	BF 661-407	BF, Reactor Bldg. Elev. 40′	I	NO	NO
049	BF 6610407	BF, Reactor Bldg. Elev. 40′	I	NO	NO
050	BF 661-416	BF, Reactor Bldg. Elev. 50′	I	NO	NO
051	BF 661-416	BF, Reactor Bldg. Elev. 50′	I	NO	NO
054	SI 968-565	SI, Reactor Bldg. Elev. 25′	A	NO	NO
055	SI 989 1205	SI, Reactor Bldg. Elev. 30′	A	NO	NO
056	SI 968-1207	SI, Reactor Bldg, Elev. 18′	A	NO	NO
057	SI 969-1190	SI, Reactor Bldg, Elev. 20′	I	NO	NO
059	SI 969-6193	SI, Reactor Bldg, Elev. 18′	A	NO	NO
060	SI 969 6195	SI, Reactor Bldg, Elev. 18′	A	NO	NO
062	SI 969-6201	SI, Reactor Bldg, Elev. 18′	A	NO	NO
063	SI 696-6217	SI, Reactor Bldg, Elev. 18′	A	NO	NO
064	SI-696-6217	SI, Reactor Bldg, Elev. 18′	A	NO	NO
065	SI-970-1210	SI, Reactor Bldg, Elev. 33′	I	NO	NO

TABLE 3.7-2b (CONTINUED)
SAFETY RELATED MECHANICAL SNUBBERS*

FPL LOCATION NO.	MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
067	SI 970-1251	SI, Reactor Bldg, Elev. 20′	A	NO	NO
068	SI 971-6	SI, Reactor Bldg, Elev. 20′	I	NO	NO
069	SI 971-1229	SI, Reactor Bldg, Elev. 20′	I	NO	NO
070	SI 971-6229	SI, Reactor Bldg, Elev. 20′	I	NO	NO
071	SI 971-6236	SI, Reactor Bldg, Elev. 20′	I	NO	NO
072	SI 972-1243	SI, Reactor Bldg, Elev. 25′	A	NO	NO
075	SI 973-6219	SI, Reactor Bldg, Elev. 36′	I	NO	NO
078	SI 868 111	SI, RAB, Elev. 4′	A	NO	NO
095	SI 676-127	SI, RAB, Elev. 4′	A	NO	NO
113	SI 971-6236	SI, RAB, Elev. 4′	A	NO	NO
094	CS-832-118	CS, Reactor Bldg, Elev. 125′	A	NO	YES
085	CS-878 115	CS, Reactor Bldg, Elev. 18′	A	No	YES
098	CC-1899-2208	CC, Reactor Bldg, Elev. 59′	A	NO	NO
100	CC-1865-2207	CC, Reactor Bldg, Elev. 59′	A	NO	NO

TABLE 3.7-2b (CONTINUED)
SAFETY RELATED MECHANICAL SNUBBERS*

FPL LOCATION NO.	MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
115	RC164-11	RC RCB Elev. 36'	I	NO	NO
116	RC162-11	RC RCB Elev. 35'	I	NO	NO
117	RC-1-221A	RC RCB Elev. 31'	I	NO	NO
118	RC-1-124B	RC RCB Elev. 19'	I	NO	NO
119	RC162-11	RC RCB Elev. 35'	I	NO	NO
120	RC-1-192A	RC RCB Elev. 19'	I	NO	NO
121	RC-5-475	RC RCB Elev. 77'	I	NO	NO
122	CH-129-99	CH RCB Elev. 33'	I	NO	NO
123	RC-217-5	RC RCB Elev. 75'	I	NO	NO
124	CH-65-54C	CH RAB Elev. 21'	I	NO	NO
125	CH-142-9	CH RCB Elev. 20'	I	NO	NO
126	CH143-30C	CH RCB Elev. 17'	I	NO	NO
127	RC-215-9A	RC RCB Elev. 73'	I	NO	NO
128	MSI-22-3A	MS RCB Elev. 57'	I	NO	NO
129	RC-1-124C	RC RCB Elev. 19'	I	NO	NO
130	MSI-20-3A	MS RCB Elev. 58'	I	NO	NO
131	MSI-16-3A	MS RCB Elev. 57'	I	NO	NO
132	CH-125-35B	CH RCB Elev. 30'	I	NO	NO
133	MSI-14-3A	MS RCB Elev. 52'	I	NO	NO
134	CH-129-339	CH RCB Elev. 32'	I	NO	NO
135	RC-220-105	RC RCB Elev. 72	I	NO	NO
136	RC-217-5	RC RCB Elev. 75'	I	NO	NO
137	CH 129-339	CH RCB Elev. 32'	I	NO	NO
138	RC 114-129	RC RCB Elev. 68'	I	NO	NO
139	RC 221-162	RC RCB Elev. 68'	I	NO	NO
140	CH-187-38A	CH RCB Elev. 24'	I	NO	NO
141	CH-143-26C	CH RCB Elev. 17'	I	NO	NO
142	CH-141-74	CH RCB Elev. 24'	I	NO	NO
143	B-2-H1	B RCB Elev. 38'	I	NO	NO
144	RC 128-99	RC RCB Elev. 67'	I	NO	NO
145	CH-143-34C	CH RCB Elev. 17'	I	NO	NO
146	RC-1-25C	RC RCB Elev. 31'	I	NO	NO
147	CH-141-36=C	CH RCB Elev. 32'	I	NO	NO

TABLE 3.7-2b (CONTINUED)

SAFETY RELATED MECHANICAL SNUBBERS*

FPL LOCATION NO.	MARK. NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
148	RC-1-124A	RC RCB Elev. 17'	I	NO	NO
149	MSI-10-3A	MS RCB Elev. 57'	I	NO	NO
150	MSI-8-3A	MS RCB Elev. 57'	I	NO	NO
151	CH 142-17	CH RCB Elev. 21'	I	NO	NO
152	RC-163-11	RC RCB Elev. 35'	I	NO	NO
153	RC 165-11	RC RCB Elev. 35'	I	NO	NO
154	MSI-18-3A	MS RCB Elev. 57'	I	NO	NO
155	RC 128-99	RC RCB Elev. 68'	I	NO	NO
156	MSI-12-3A	MS RCB Elev. 57'	I	NO	NO
157	RC 219-6B	RC RCB Elev. 71'	I	NO	NO
158	CH 142-18	CH RCB Elev. 21'	I	NO	NO
159	RC 163-11	RC RCB Elev. 35'	I	NO	NO
160	CH 142-18	CH RCB Elev. 21'	I	NO	NO
161	RC 165-11	RC RCB Elev. 35'	I	NO	NO
162	RC-217-5	RC RCB Elev. 75'	I	NO	NO
163	SI-69-58	SI RCB Elev. 62'	I	NO	NO
164	SI-39-6	SI RAB Elev. 12'	I	NO	NO
165	RC 221-148	RC RCB Elev. 72'	I	NO	NO
166	RC 164-11	RC RCB Elev. 36'	I	NO	NO
167	CH-67-81	CH RAB Elev. 21'	I	NO	NO
168	RC 114-129	RC RCB Elev. 68'	I	NO	NO
169	RC-44-26	RC RCB Elev. 40'	I	NO	NO
170	RC 220-112	RC RCB Elev. 71'	I	NO	NO
171	RC 218-26	RC RCB Elev. 76'	I	NO	NO
172	RC 44-11	RC RCB Elev. 35'	I	NO	NO
173	CH 67-81	CH RAB Elev. 21'	I	NO	NO
174	CH-65-54A	CH RAB Elev. 12'	I	NO	NO
175	RC 220-112	RC RCB Elev. 72'	I	NO	NO
176	MSI-2-HI	MS RCB Elev. 78'	I	NO	NO
177	RC 222-43	RC RCB Elev. 70'	I	NO	NO
178	MSI-4-HI	MS RCB Elev. 78'	I	NO	NO

TABLE 3.7-2b (CONTINUED)
SAFETY RELATED MECHANICAL SNUBBERS*

FPL LOCATION NO.	MARK NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
180	RC 220-114	RC RCB Elev. 71′	I	NO	NO
181	RC 215-9B	RC RCB Elev. 74′	I	NO	NO
182	RC 219-6A	RC RCB Elev. 73′	I	NO	NO
182	MSI-3-H1	MS RCB Elev. 78′	I	NO	NO
184	RC 218-26	RC RCB Elev. 76′	I	NO	NO
185	CH-64-45A	CH RAB Elev. 30′	I	NO	NO
186	B-1-H3A	B RCB Elev. 38′	I	NO	NO
187	SI-69 60B	SI RCB Elev. 57′	I	NO	NO
188	RC 150-H7	RC RCB Elev. 50′	I	NO	NO
189	SI-69 60	SI RCB Elev. 57′	I	NO	NO
190	CH-141-44A	CH RCB Elev. 32′	I	NO	NO
191	CH-141-44A	CH RCB Elev. 32′	I	NO	NO
192	B-3-795	B RAB Elev. 36′	I	NO	NO
193	CH 130-66	CH RAB Elev. 26′	I	NO	NO
194	MSH-7B	MS Steam Trestle Elev. 38′	A	NO	NO
195	MPR-200-250	MP RCB Elev. 34′	I	NO	NO
196	MSH-7A	Sm Stream Trestle Elev. 38′	A	NO	NO
197	MPR-201-16	MP RCB Elev. 35′	I	NO	NO
198	MPR 200-20	MP RCB Elev. 35′	I	NO	NO
199	B-21-311	B-RAB Elev. 36′	A	NO	NO
200	B-21-3110	B-RAB Elev. 35′	A	NO	NO
201	B-21-3240	B-RAB Elev. 38′	A	NO	NO
202	CC-1899-2200	CC-RCB Elev. 57′	I	NO	NO

*Snubbers may be added to or removed from safety related systems without prior License Amendment to Table 3.7-2b provided a revision to Table 3.7-2b is included with the next License Amendment request.

**Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-2b is included with the next License Amendment request.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent, and maintaining the chlorine concentration within acceptable limits during and following a chlorine accident. This limitation is consistent with the requirements of General Design Criteria 10 of Appendix "A", 10 CFR 50.

3/4.7.8 ECCS AREA VENTILATION SYSTEM

The OPERABILITY of the ECCS area ventilation system ensures that radioactive materials leaking from the ECCS equipment following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the probable leakage from the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Quantities of interest to this specification which are exempt from the leakage testing are consistent with the criteria of 10 CFR Parts 30.11-20 and 70.19. Leakage from sources excluded from the requirements of this specification is not likely to represent more than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

3/4.7.10 SNUBBERS

All snubbers are required to be OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubber that may be generically susceptible and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

In cases where the cause of failure has been identified, additional snubbers having a high probability for the same type failure or that are being used in the same application that caused the failure shall be tested. This requirement increases the probability of locating inoperable snubbers without testing 100% of the snubbers.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. ...). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

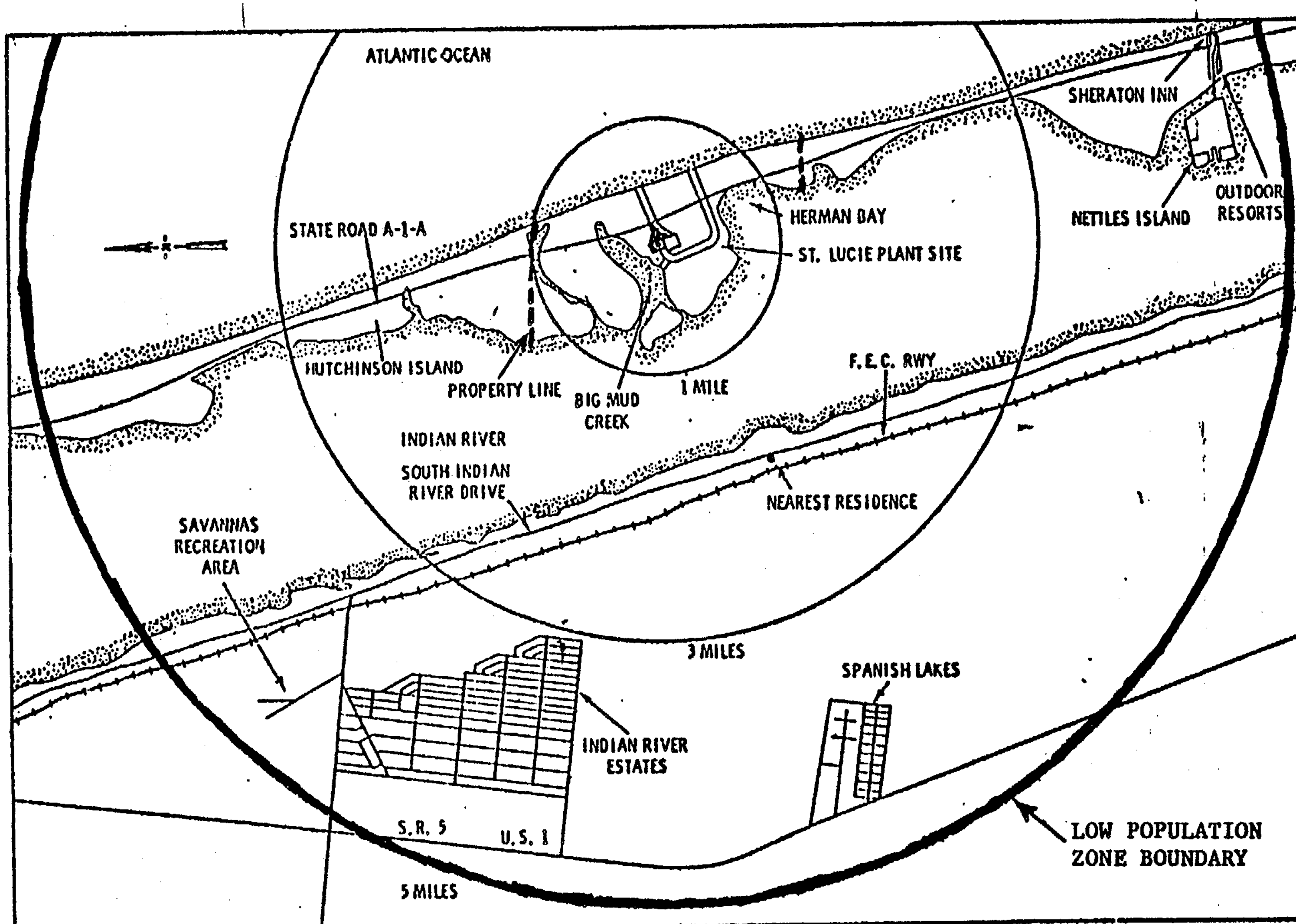
In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service.

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.12 PENETRATION FIRE BARRIERS

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the barriers are not intact, routine fire watch patrols in conjunction with OPERABLE fire detection instrumentation or a continuous fire watch are required to be maintained in the vicinity of the affected barrier until the barrier is restored to intact status.



LOW POPULATION ZONE

FIGURE 5.1-2

5.2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet.

b. Annulus nominal volume = 543,000 cubic feet.

c. Nominal outside height (measured from top of foundation base to the top of the dome) = 230.5 feet.

d. Nominal inside diameter = 148 feet.

e. Cylinder wall minimum thickness = 3 feet.

f. Dome minimum thickness = 2.5 feet.

g. Dome inside radius = 112 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment vessel is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

PENETRATIONS

5.2.3 Penetrations through the containment structure are designed and shall be maintained in accordance with the original design provisions contained in Sections 3.8.2.1.10 and 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 2250 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

5.3.2 Except for special test as authorized by the NRC, all fuel assemblies under control element assemnlies shall be sleeved with a sleeve design previously approved by the NRC.

h. Core Barrel Movement, Specifications 3.4.11 and 4.4.11.

i. Fire Detection Instrumentation, Specification 3.3.3.7.

j. Fire Suppression Systems, Specification 3.7.11.1.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those facility components identified in Table 5.9-1.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the FRG and the CNRB.

l. Records of Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of the service lives of all hydraulic and mechanical snubbers listed on Tables 3.7-2a and 3.7-2b including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. A High Radiation Area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.12.1.a above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

Introduction

We have evaluated the following Florida Power & Light Company (FPL and the licensee) proposed changes to the St. Lucie Unit 1 Technical Specification:

1. May 11, 1981 FPL submittal regarding hydraulic and mechanical snubbers.

2. July 23, 1981 FPL submittal regarding control element assembly (CEA) guide tube sleeves.

Our evaluations are presented below.

Snubbers

Numerous discoveries of inoperative snubbers from 1973 to 1975 resulted in snubber surveillance requirements being placed in the Technical Specifications for operating reactors. Several deficiencies were identified after the original requirements were in force for a few years. These deficiencies are:

1. Mechanical snubbers were not included in these requirements.

2. The rated capacity of snubbers was used as a limit to the inservice test requirement.

3. NRC approval was necessary for the acceptance of seal materials.

4. Inservice test requirements were not clearly defined.

5. In-place inservice testing was not permitted.

Since mechanical snubbers were not subject to any surveillance requirements, some licensees believed that mechanical snubbers were preferred by the NRC. Many licensees used mechanical snubbers as original equipment and may others requested approval to replace their hydraulic snubbers with mechanical ones to simplify or avoid an inservice surveillance program. This is contradictory to the intent of snubber Technical Specifications since, for an unsurveyed mechanical snubber, the most likely failure is permanent lock-up. This failure mode can be harmful to the associated system piping during normal plant operation.

During the period 1973-1975, when the first hydraulic snubber surveillance requirements were drafted, a compromise was made to limit the testing of snubbers to those with rated capacity of not more than 50,000 lbs. This was because of the available capacity of the test equipment and the requirement to test some parameters at the snubber rated load. Since then, greater equipment capacity and a better understanding of parametric correlation have been developed. To maintain this arbitrary 50,000 lb. limit could mean an unnecessary compromise to plant safety.

The original hydraulic snubber problems started with leaking seals. Most seal material of the 1973 vintage could not withstand the temperature and irradiation environments. Ethylene propylene was the first material that could offer a reasonable service life for the seals. In order to discourage the use of unproven material for the seals, the words "NRC approved material" were used in the Technical Specifications. As a result we were asked to approve different seal material on many occasions. Since the basis for that approval was not defined, our reviews were hampered and the development of better seal materials by the industry was actually discouraged.

The acceptance criteria in the earlier version of the testing requirements were not well defined and resulted in non-uniform interpretations and implementation. This resulted in problems in inspecting the conduct of snubber surveillance. In some cases, snubbers were tested without reference to acceptance criteria.

Testing of snubbers was usually accomplised by removing snubbers from their installed positions, mounting them on a testing rig, conducting the test, removing them from the rig, and reinstalling them in the working position. Many snubbers were damaged during removal and reinstallation, defeating the purpose for conducting the tests. Methods and equipment have been developed which allow in-place tests of snubbers. Taking advantage of these developments could result in minimizing the damage to snubbers caused by removal and reinstallation plus possible savings of time and cost.

As a result of these deficiencies we prepared revised model Technical Specifications regarding snubbers and sent them to FPL on November 20, 1980. The revised model technical specifications correct the deficiencies discussed above in the following manner:

1. Mechanical snubbers are now included in the surveillance program.

2. No arbitrary snubber capacity is used as a limit to the inservice test requirements.

3. Seal material no longer requires NRC approval. A monitoring program is used to assure that snubbers are functioning within their service life.

4. Inservice test requirements for snubbers are more clearly defined.

5. In-place inservice testing is permitted.

By letter dated May 11, 1981, the licensee proposed a change to the snubber Technical Specifications which will put in effect the improvements listed above. The licensee's proposal did omit the requirement to test snubbers of the same design as one which, based on the licensee's evaluation, fails due to a manufacturer or design deficiency. The licensee's staff has agreed to include this requirement as it appears in the model Technical Specifications. We have determined that, with this addition, the licensee's proposed snubber Technical Specifications are acceptable.

CEA Guide Tube Sleeves

By letter dated July 6, 1981 we informed FPL that we accepted NRC approved CEA guide tube sleeves as a solution to the wear problem. Also we requested that FPL propose a change to the Design Features (Section 5) of the St. Lucie Unit 1 Technical Specifications which would require NRC approved sleeves as part of the design of fuel assemblies located under CEA's.

By letter dated July 23, 1981 FPL proposed the requested change and we find this change acceptable.

Conclusion

Environmental Consideration

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

Conclusion

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

Date: October 14, 1981

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-335

FLORIDA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 44 to Facility Operating License No. DPR-67, issued to Florida Power & Light Company (the licensee), which revised the Technical Specifications for operation of the St. Lucie Plant, Unit No. 1 (the facility), located in St. Lucie County, Florida. The amendment is effective as of the date of issuance.

This amendment changes the Technical Specifications by clarifying the test requirements for snubbers and by adding the requirement that mechanical snubbers be tested. Also the requirement that NRC approved sleeves be used in control element assembly guide tubes was added to the Technical Specification as a Design Feature.

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 May 11 and July 23, 1981, (2) Amendment No. 44 to License No. DPR-67, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

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