December 24, 1992

Docket Nos. 50-335
and 50-389

DISTRIBUTION
See attached sheet

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

Dear Mr. Goldberg:

SUBJECT: ST. LUCIE UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADMINISTRATIVE CHANGES (TAC NOS. M84381 AND M84382)

The Commission has issued the enclosed Amendment Nos. 118 and 60 to Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated August 21, 1992.

These amendments make administrative changes to the St. Lucie Unit 1 and Unit 2 Technical Specifications and achieve consistency throughout the Technical Specifications by removing outdated material, making minor text changes, and correcting errors.

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. 118 to DPR-67
2. Amendment No. 60 to NPF-16
3. Safety Evaluation

cc w/enclosures:
See next page

Document Name - SL84381.AMD

OFC	:LA:PDII-2	:PM:PDII-2	:D:PDII-2	:OGC w/change to SE	:	:
NAME	:D. Miller	:J. Norris	:H. Berkow	:CPW	:	:
DATE	:10/29/92	:10/29/92	:10/29/92	:11/4/92	:	:

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PDR ADOCK 05000335
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Mr. J. H. Goldberg
Florida Power and Light Company

St. Lucie Plant

cc:
Jack Shreve, Public Counsel
Office of the Public Counsel
c/o The Florida Legislature
111 West Madison Avenue, Room 812
Tallahassee, Florida 32399-1400

Senior Resident Inspector
St. Lucie Plant
U.S. Nuclear Regulatory Commission
7585 S. Hwy A1A
Jensen Beach, Florida 33457

Mr. Robert G. Nave, Director
Emergency Management
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

Harold F. Reis, Esq.
Newman & Holtzinger
1615 L Street, N.W.
Washington, DC 20036

John T. Butler, Esq.
Steel, Hector and Davis
4000 Southeast Financial Center
Miami, Florida 33131-2398

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 34982

Mr. James V. Chisolm, County
Administrator
St. Lucie County
2300 Virginia Avenue
Fort Pierce, Florida 34982

Mr. Charles B. Brinkman, Manager
Washington Nuclear Operations
ABB Combustion Engineering, Nuclear Power
12300 Twinbrook Parkway, Suite 330
Rockville, Maryland 20852

Mr. Jacob Daniel Nash
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

Regional Administrator, RII
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, Georgia 30323

Mr. R. E. Grazio
Director, Nuclear Licensing
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

DATED: December 24, 1992

AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1
AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. NPF-16 - ST. LUCIE, UNIT 2

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
J. Norris
OGC
D. Hagan, 3302 MNBB
G. Hill (8), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
ACRS (10)
OPA
OC/LFMB
M. Sinkule, R-II

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. 118
License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated August 21, 1992, 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.

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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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 118, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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 24, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 118

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
III of the Index	III of the Index
3/4 0-3	3/4 0-3
3/4 1-12	3/4 1-12
3/4 3-41	3/4 3-41
3/4 4-1b	3/4 4-1b
3/4 4-8	3/4 4-8
3/4 4-14	3/4 4-14
3/4 6-5	3/4 6-5
3/4 6-20	3/4 6-20
3/4 6-21	3/4 6-21
3/4 8-6b	3/4 8-6b
3/4 11-2	3/4 11-2
3/4 11-10	3/4 11-10
3/4 12-10	3/4 12-10
B 3/4 1-4	B 3/4 1-4
B 3/4 7-5	B 3/4 7-5
B 3/4 9-3	B 3/4 9-3
6-24	6-24

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or high pressure safety injection pump* in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump* OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least one of the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2571 ft. when tested pursuant to Specification 4.0.5.

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*The flow path from the RWT to the RCS via a single HPSI pump shall be established only if: (a) the RCS pressure boundary does not exist, or (b) no charging pumps are operable. In this case, all charging pumps shall be disabled and heatup and cooldown rates shall be limited in accordance with Fig. 3.1-1b.

At RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

High Pressure Header	Auxiliary Header
HCV-3616	HCV-3617
HCV-3626	HCV-3627
HCV-3636	HCV-3637
HCV-3646	HCV-3647

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. Actions per Table 3.3-11.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Pressurizer Water Level	3	1	1
2. Auxiliary Feedwater Flow Rate	1/pump	1/pump	1
3. RCS Subcooling Margin Monitor	2	1	1
4. PORV Position Indicator Acoustic Flow Monitor	1/valve	1/valve	2
5. PORV Block Valve Position Indicator	1/valve	1/valve	2
6. Safety Valve Position Indicator	1/valve	1/valve	3
7. Incore thermocouples	4/core quadrant	2/core quadrant	1
8. Containment Sump Water Level (Narrow Range)	1*	1*	4, 5
9. Containment Sump Water Level (Wide Range)	2	1	4, 5
10. Reactor Vessel Level Monitoring System	2**	1**	4, 5
11. Containment Pressure	2	1	1

*The non-safety grade containment sump water level instrument may be substituted.

**Definition of OPERABLE: A channel is composed of eight (8) sensors in a probe, of which four (4) sensors must be OPERABLE.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one reactor coolant or shutdown cooling loop shall be in operation.*

a. Reactor Coolant Loop A and its associated steam generator and at least one associated reactor coolant pump,

b. Reactor Coolant Loop B and its associated steam generator and at least one associated reactor coolant pump,

c. Shutdown Cooling Loop A,

d. Shutdown Cooling Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required reactor coolant or shutdown cooling loops OPERABLE, within one (1) hour initiate corrective action to return the required loops to OPERABLE status. If the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 30 hours.

b. With no reactor coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and within one (1) hour initiate corrective action to return the required coolant loop to operation.

*All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% of narrow range indication at least once per 12 hours.

4.4.1.3.3 At least one reactor coolant or shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions.

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections $\geq$20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a special report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a special report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This special report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

c. With the containment sump level and flow monitoring system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

a. Containment atmosphere gaseous and particulate monitoring systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies specified in Table 4.3-3, and

b. Reactor cavity sump level and flow monitoring system-performance of CHANNEL CALIBRATION TEST at least once per 18 months.

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through steam generators,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and Reactor Coolant System Pressure Isolation Valve leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. (Note, however, that this may lead to ACTION requirements for systems involved.) Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity at least once per 12 hours.

TABLE 3.6-1

CONTAINMENT LEAKAGE PATHS

Penetration	System	Valve Tag Number	Location to Containment	Service	Test Type*
7	Makeup Water	Gate (I-MV-15-1) Check (I-V-15328)	Outside Inside	Primary Makeup Water	Bypass
8	Station Air	Globe (I-V-18-794) Globe (I-V-18-796)	Outside Outside	Station Air Supply	Bypass
9	Instrument Air	Gate (I-MV-18-1) Check (I-V-18195)	Outside Inside	Instrument Air Supply	Bypass
10	Containment Purge	Butterfly (I-FCV-25-4) Butterfly (I-FCV-25-5)	Inside Outside	Containment Purge Exhaust	Type C
11	Containment Purge	Butterfly (I-FCV-25-3) Butterfly (I-FCV-25-2)	Inside Outside	Containment Purge Supply	Type C
14	Waste Management	Globe (V-6741) Check (V-6779)	Outside Outside	Nitrogen supply to SI Tanks	Bypass
23	Component Cooling	Butterfly (I-HCV 14-7) Butterfly (I-HCV-14-1)	Outside Outside	RC Pump CW Supply	Bypass
24	Component Cooling	Butterfly (I-HCV-14-6) Butterfly (I-HCV-14-2)	Outside Outside	RC Pump CW Return	Bypass
25	Fuel Transfer Tube	Double Gasket Flange	Inside	Fuel Transfer	Bypass
26	CVCS	Globe (V-2515) Globe (V-2516)	Inside Inside	Letdown Line	Bypass
28	Sampling	Globe (V-5200) Globe (V-5203) Globe (I-FCV-03-1E) Globe (I-FCV-03-1F)	Outside Outside Outside Outside	Reactor Coolant Sample SI Tank Sample SI Tank Sample	Bypass Bypass

TABLE 3.6-1 (Continued)

Penetration	System	Valve Tag Number	Location to Containment	Service	Test Type*
29	Sampling	Globe (V-5202) Globe (V-5205)	Outside Outside	Pressurizer Steam Space Sample	Bypass
29	Sampling	Globe (V-5201) Globe (V-5204)	Outside Outside	Pressurizer Surge Line Sample	Bypass
31	Waste Management	Gate (V-6554) Gate (V-6555)	Outside Outside	Containment Vent Header	Bypass
41	Safety Injection Tank Test Lines	Gate (V-3463) Gate (I-V-07009)	Outside Outside	Safety Injection Tank Fill and Sampling	Bypass
42	Waste Management	Gate (I-LCV-07-11A) Gate (I-LCV-07-11B)	Outside Outside	Reactor Cavity Sump Pump Discharge	Bypass
43	Waste Management	Gate (V-6301) Gate (V-6302)	Outside Outside	Reactor Drain Tank Pump Suction	Bypass
44	CVCS	Gate (V-2505) Gate (I-SE-01-1)	Outside Inside	RC Pump Controlled Bleedoff	Bypass
46	Fuel Pool Cleanup	Gate (I-V-07-206) Gate (I-V-07-189)	Outside Inside	Refueling Cavity Purification Flow Inlet	Bypass
47	Fuel Pool Cleanup	Gate (I-V-07-170) Gate (I-V-07-188)	Outside Inside	Refueling Cavity Purification Flow Outlet	Bypass
48a	Sampling	Globe (I-FSE-27-1, 2, 3, 4) Globe (I-FSE-27-8)	Inside Outside	H_2 Sampling	Type C

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-2 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Containment Isolation test signal, and/or SIAS test signal, each isolation valve actuates to its isolation position.

4.6.3.1.3 The isolation time of each power operated or automatic valve of Table 3.6-2 shall be determined to be within its limits when tested pursuant to Specification 4.0.5.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

Valve Tag Number		Penetration Number	Function	Testable During Plant Operation	Isolation Time (Sec)
A.	CONTAINMENT ISOLATION				
1.	I-FCV-25-4,5	10	Containment purge air exhaust, CIS	No	5
2.	I-FCV-25-2,3	11	Containment purge supply, CIS	No	5
3.	I-MV-15-1	7	Primary makeup water, CIS	Yes	19
4.	I-MV-18-1	9	Instrument air supply, CIS	No	28
5.	V-6741	14	Nitrogen supply to safety injection tanks, CIS	Yes	5
6.	I-HCV-14-1 & 7	23	Reactor coolant pump cooling water supply, SIAS	No	5
7.	I-HCV-14-6 & 2	24	Reactor coolant pump cooling water return, SIAS	No	5
8.	V-2515,2516	26	Letdown line, CIS, SIAS	No	5
9.	V-5200,5203	28	Reactor coolant sample, CIS	Yes	5
10.	V-5201,5204	29	Pressurizer surge line sample, CIS	Yes	5
11.	V-5202,5205	29	Pressurizer steam space sample, CIS	Yes	5
12.	V-6554,6555	31	Containment vent header, CIS	Yes	5
13.	I-LCV-07-11A,11B	42	Reactor cavity sump pump discharge, CIS	Yes	10
14.	V-6301,6302	43	Reactor drain tank pump suction, CIS	Yes	5
15.	V-2505	44	Reactor coolant pump controlled bleedoff, CIS	No	5
16.	I-SE-01-1	44	Reactor coolant pump controlled bleedoff, CIS	No	5

TABLE 3.6-2 (Continued)

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Isolation Time (Sec)
B. MANUAL OR REMOTE MANUAL				
1. I-V-18-794 I-V-18-796	8	Station air supply, Manual	Yes	NA
2. I-V-25-11,12	56	Hydrogen purge outside air make-up, Manual (NC)	Yes	NA
3. I-V-25-13,14, 15,16	57 & 58	Hydrogen purge exhaust, Manual (NC)	Yes	NA
4. V-3463	41	Safety injection tank test line, Manual (NC)	Yes	NA*
5. I-V-07009	41	Safety injection tank test line, Manual (NC)	Yes	NA*
6. V-07206, V-07189	46	Refueling cavity purification flow inlet, Manual (NC)	Yes	NA
7. V-07170, V-07188	47	Refueling cavity purification flow outlet, Manual (NC)	Yes	NA
8. I-FSE-27-1,2,3, 4,8,11	48a & 48c	Hydrogen sampling line, Remote manual	Yes	NA*
9. I-FSE-27-5,6,7, 9,10	51a & 51c	Hydrogen sampling line, Remote manual	Yes	NA*

TABLE 3.6-2 (Continued)

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Isolation Time (Sec)
10. I-FCV-26-1 & 2	52a	Radiation monitoring	Yes	NA
11. I-FCV-26-3 & 4	52b	Radiation monitoring	Yes	NA
12. I-FCV-26-5 & 6	52c	Radiation monitoring, return	Yes	NA
13. I-V00140 I-V00143	52d	ILRT test tap	Yes	NA
14. I-V00139 I-V00144	52e	ILRT test tap	Yes	NA
15. I-V00101	54	ILRT pressure connection	Yes	NA
16. I-FCV-03-1E & 1F	28	SI Tank Sample	Yes	NA**

NA - Manual Valve-Isolation time not applicable.
* May be opened on an intermittent basis under administrative control.
** Normally closed valves - Isolation time not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying the diesel generator operates for at least 24 hours****. During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3960 kW# and during the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3300 to 3500 kW#. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

7. Verifying that the auto-connected loads do not exceed the 2000-hour rating of 3730 kW.

8. Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power.

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

9. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

10. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the engine-mounted tanks of each diesel via the installed cross connection lines.

11. Verifying that the automatic load sequence timers are operable with the interval between each load block within ± 1 second of its design interval.

f. At least once per ten years or after any modification which could affect diesel generator independence by starting**** the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to approximately 900 rpm in less than or equal to 10 seconds.

#This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

****This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

SURVEILLANCE REQUIREMENTS (Continued)

g. At least once per ten years by:

1. Draining each fuel storage tank, removing the accumulated sediment and cleaning the tank using an appropriate cleaning compound, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to USAS B31.7 Class 3 requirements at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.2. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

4.8.1.1.4 The Class 1E underground cable system shall be demonstrated OPERABLE 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.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1. Also, results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1.

4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance with Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[c]	P Each Batch	P Each Batch	Principal Gamma Emitters[e]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[dg]	Daily	4/M Composite	Principal Gamma Emitters[e]	5×10^{-7}
			I-131	1×10^{-6}
	D Grab Sample Daily	4/M Composite	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Daily	M Composite	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Daily	Q Composite	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
C. Settling Basin[f]	W Grab Sample	W	Principal Gamma Emitters[e]	5×10^{-7}
			I-131	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

c. Samples shall be changed at least 4 times a month and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

d. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

[b]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

[c]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

[d]LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

[e]An equilibrium mixture of the parent and daughter isotopes which corresponds to 15 pCi/ℓ of the parent isotope.

3/4.1.2 BORATION SYSTEMS (Continued)

The boron addition capability after the plant has been placed in MODES 5 and 6 requires either 3650 gallons of 2.5 to 3.5 weight percent boric acid solution (4371 to 6119 ppm boron) from the boric acid tanks or 11,900 gallons of 1720 ppm borated water from the refueling water tank to makeup for contraction of the primary coolant that could occur if the temperature is lowered from 200°F to 140°F.

The restrictions associated with the establishing of the flow path from the RWT to the RCS via a single HPSI pump provide assurance that Appendix G pressure/temperature limits will not be exceeded in the case of any inadvertent pressure transient due to a mass addition to the RCS.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment ($\geq$ 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of the CEA requires a prompt realignment of the misaligned CEA.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements brings the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors, and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

The requirement to reduce power in certain time limits, depending upon the previous F_r^t, is to eliminate a potential nonconservatism for situations when a CEA has been declared inoperable. A worst case analysis has shown that a DNBR SAFDL violation may occur during the second hour after the CEA misalignment if this requirement is not met. This potential DNBR SAFDL violation is eliminated by limiting the time operation is permitted at FULL POWER before power reductions are required. These reductions will be necessary once the deviated CEA has been declared inoperable. The time allowed to continue operation at a reduced power level can be permitted for the following reasons:

1. The margin calculations that support the Technical Specifications are based on a steady-state radial peak of $F_r^t > 1.70$.
2. When the actual $F_r^t \leq 1.70$, significant additional margin exists.
3. This additional margin can be credited to offset the increase in F_r^t with time that can occur following a CEA misalignment.
4. This increase in F_r^t is caused by xenon redistribution.
5. The present analysis can support allowing a misalignment to exist for up to 60 minutes without correction, if the initial $F_r^t \leq 1.67$.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

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. This limitation is consistent with the requirements of General Design Criteria 19 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 safety related 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.

3/4.9.12 FUEL POOL VENTILATION SYSTEM-FUEL STORAGE

The limitations on the fuel handling building ventilation system ensures that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.13 SPENT FUEL CASK CRANE

The maximum load which may be handled by the spent fuel cask crane is limited to a loaded single element cask which is equivalent to approximately 25 tons. This restriction is provided to ensure the structural integrity of the spent fuel pool in the event of a dropped cask accident. Structural damage caused by dropping a load in excess of a loaded single element cask could cause keakage from the spent fuel pool in excess of the maximum makeup capability.

3/4.9.14 DECAY TIME - STORAGE POOL

The minimum requirements for decay of the irradiated fuel assemblies in the entire spent fuel storage pool prior to movement of the spent fuel cask into the fuel cask compartment ensure that sufficient time has elapsed to allow radioactive decay of the fission products. The decay time of 1180 hours is based upon one-third of a core placed in the spent fuel pool each year during refueling until the pool is filled. The decay time of 1490 hours is based upon one-third of a core being placed in the spent fuel pool each year during refueling following which an entire core is placed in the pool to fill it. The cask drop analysis assumes that all of the irradiated fuel in the filled pool (7 2/3 cores) is ruptured and follows Regulatory Guide 1.25 methodology, except that a Radial Peaking Factor of 1.0 is applied to all irradiated assemblies.

6.13 PROCESS CONTROL PROGRAM (PCP)

Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information.

   b. A determination that the change did not reduce the overall conformance of the dewatered bead resin to existing criteria for radioactive wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Facility Review Group. The discussion of each shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected maximum exposures to individuals in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period prior to when the changes are to be made;

   g. An estimate of the exposure to plant operating personnel as a result of the change; and

   h. Documentation of the fact that the change was reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

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

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 60
License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated August 21, 1992, 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. NPF-16 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. 60 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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 24, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 60

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

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
2-4	2-4
2-5	2-5
3/4 2-9	3/4 2-9
3/4 3-2	3/4 3-2
3/4 3-13	3/4 3-13
3/4 6-2	3/4 6-2
3/4 6-26	3/4 6-26
3/4 8-7	3/4 8-7
3/4 9-9	3/4 9-9
3/4 11-10	3/4 11-10
3/4 12-10	3/4 12-10
6-23	6-23

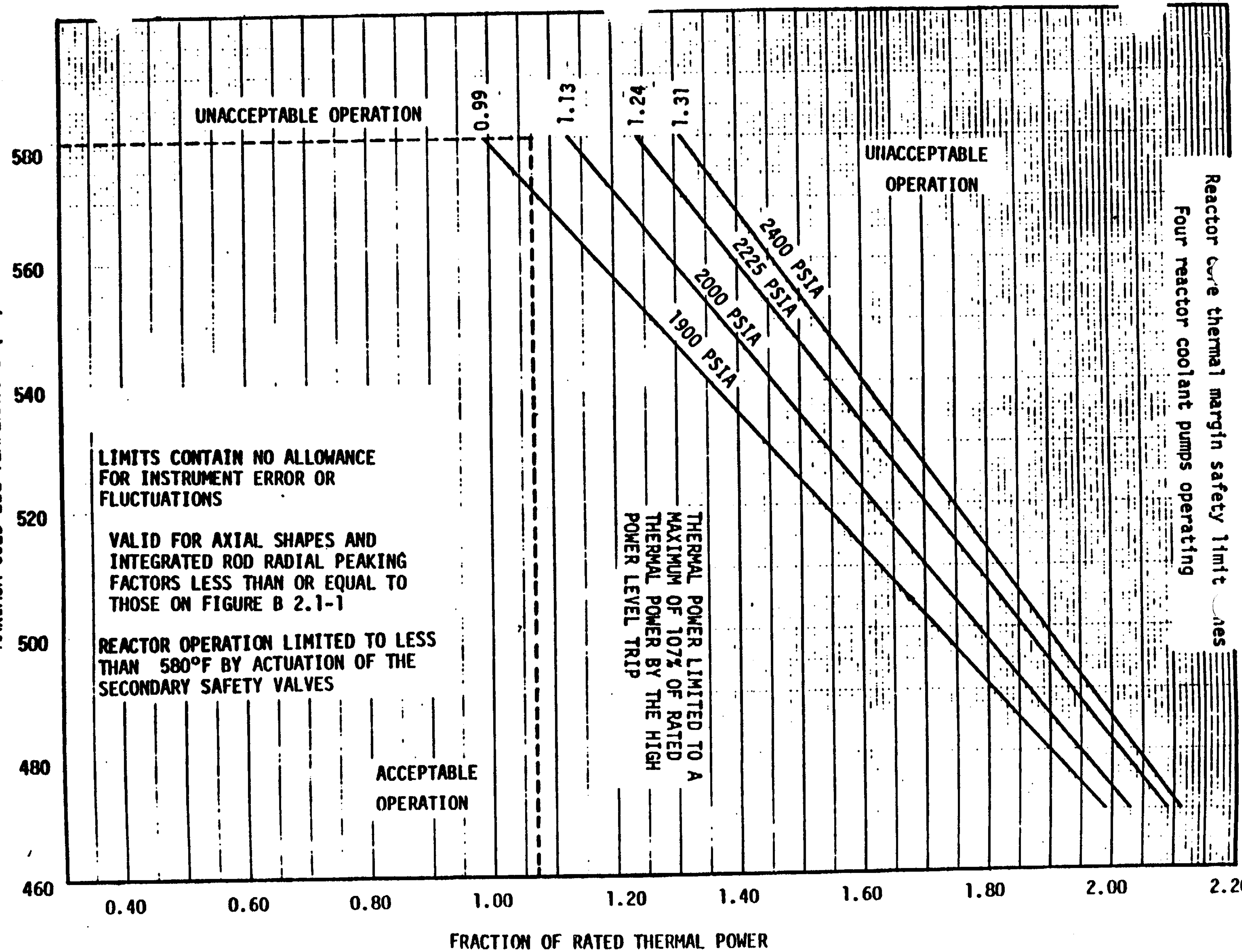

Figure 2.1-1

Reactor core thermal margin safety limit lines

Four reactor coolant pumps operating

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Variable Power Level - High (1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Pressurizer Pressure - High	≤ 2370 psia	≤ 2374 psia
4. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.
5. Containment Pressure - High	≤ 3.0 psig	≤ 3.1 psig
6. Steam Generator Pressure - Low	≥ 626.0 psia (2)	≥ 621.0 psia (2)
7. Steam Generator Pressure (1) Difference - High (Logic in TM/LP Trip Unit)	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level - Low	≥ 20.5% (3)	≥ 19.5% (3)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High (5) Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps-Low	≥ 636 gpm**	≥ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power - High (4)	≤ 2.49 decades per minute	≤ 2.49 decades per minute
14. Reactor Coolant Flow - Low (1)	≥ 95.4% of design Reactor Coolant flow with four pumps operating*	≥ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure - Low (5)	≥ 800 psig	≥ 800 psig

*Design reactor coolant flow with four pumps operating is 363,000 gpm.
**10-minute time delay after relay actuation.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER during testing pursuant to Special Test Exception 3.10.3; bypass shall be automatically removed when the THERMAL POWER is greater than or equal to 0.5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(3) % of the narrow range steam generator level indication.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$% or $\leq$ 15% of RATED THERMAL POWER.

(5) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 15% of RATED THERMAL POWER.

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T shall be limited to ≤ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With $F_r^T > 1.70$, within 6 hours either:

a. Be in at least HOT STANDBY, or

b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3 and withdraw the full-length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6. The THERMAL POWER limit determined from Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ when F_r is calculated with a non-full core power distribution analysis code and shall be calculated as $F_r^T = F_r$ when calculations are performed with a full core power distribution analysis code. F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70% of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within 4 hours if the AZIMUTHAL POWER TILT (T_q) is > 0.03.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing reactor coolant pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is made using a non-full core power distribution analysis code. The value of T_q used to determine F_r^T in this case shall be the measured value of T_q.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

4.3.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

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

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	4	2	4	1, 2	1
	4	2	4	3*, 4*, 5*	5
2. Variable Power Level - High	4	2(a)(d)	3	1, 2	2#
3. Pressurizer Pressure - High	4	2	3	1, 2	2#
4. Thermal Margin/Low Pressure	4	2(a)(d)	3	1, 2	2#
5. Containment Pressure - High	4	2	3	1, 2	2#
6. Steam Generator Pressure - Low	4/SG	2/SG(b)	3/SG	1, 2	2#
7. Steam Generator Pressure Difference - High	4	2(a)(d)	3	1, 2	2#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#
9. Local Power Density - High	4	2(c)(d)	3	1	2#
10. Loss of Component Cooling Water to Reactor Coolant Pumps	4	2	3	1, 2	2#
11. Reactor Protection System Logic	4	2	3	1, 2	2#
				3*, 4*, 5*	5
12. Reactor Trip Breakers	4	2(f)	4	1, 2	4
				3*, 4*, 5*	5
13. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating - Rate of Change of Power - High	4	2(e)(g)	3	1, 2	2#
b. Shutdown	4	0	2	3, 4, 5	3
14. Reactor Coolant Flow - Low	4/SG	2/SG(a)(d)	3/SG	1, 2	2#
15. Loss of Load (Turbine Hydraulic Fluid Pressure - Low)	4	2(c)	3	1	2#

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3	16
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	13*, 14
c. Containment Pressure - High	4	2	3	1, 2, 3	13*, 14
d. Automatic Actuation Logic	2	1	2	1, 2, 3	12
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	17
c. Automatic Actuation Logic	2	1	2	1, 2, 3	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. LOSS OF POWER (LOV)					
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	1/Bus	1/Bus	1/Bus	1, 2, 3	12
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	2/Bus	2/Bus	2/Bus	1, 2, 3	12
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17
7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	15
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	12
c. SG Level (2A/2B) - Low	4/SG	2/SG	3/SG	1, 2, 3	13*, 14
8. AUXILIARY FEEDWATER ISOLATION					
a. SG 2A-- SG 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	13*, 14
b. Feedwater Header SG 2A - SG 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	13*, 14

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1*, 2*, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a, 41.8 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

*In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 41.8 psig, or

2. Less than or equal to L_t, 0.35 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 20.9 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

c. A combined bypass leakage rate of less than or equal to 0.12 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.12 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than or equal to 0.60 L_a, and the bypass leakage rate to less than or equal to 0.12 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS - W

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 kW.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.).

3. Verifying the integrity of the heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5 The primary containment vessel to annulus vacuum relief valves shall be OPERABLE with an actuation setpoint of less than or equal to 9.85 ± 0.35 inches water gauge.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one primary containment vessel to annulus vacuum relief valve inoperable, restore the valve to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.5 No additional Surveillance Requirements other than those required by Specification 4.0.5.

SURVEILLANCE REQUIREMENTS (Continued

c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

7. Verifying the diesel generator operates for at least 24 hours.**** During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3985 kW# and during the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3450 to 3685 kW#. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2e.4.b.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3935 kW.

9. Verifying that the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power.

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the engine-mounted tanks of each diesel via the installed cross connection lines.

#This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

****This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

12. Verifying that the automatic load sequence timers are operable with the interval between each load block within ± 1 second of its design interval.

f. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting**** the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to approximately 900 rpm in less than or equal to 10 seconds.

g. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

4.8.1.1.4 The Class 1E underground cable system shall be demonstrated OPERABLE 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.

****This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation.

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, within 1 hour initiate corrective action to return the required loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and within 1 hour initiate corrective action to return the required shutdown cooling loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

3/4.9.9 CONTAINMENT ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment isolation system shall be OPERABLE.

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

ACTION:

With the containment isolation system inoperable, close each of the containment penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

c. Samples shall be changed at least 4 times a month and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

d. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

[b]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

[c]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

[d]LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

[e]An equilibrium mixture of the parent and daughter isotopes which corresponds to 15 pCi/ℓ of the parent isotope.

6.13 PROCESS CONTROL PROGRAM (PCP)

Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the dewatered bead resin to existing criteria for radioactive wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change wail not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Facility Review Group. The discussion of each shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

   g. An estimate of the exposure to plant operating personnel as a result of the change; and

   h. Documentation of the fact that the change was reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

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*Licensees may chose to submit the information called for in this Specification as part of the annual FSAR update.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 118 AND 60

TO FACILITY OPERATING LICENSE NO. DPR-67 AND NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By letter dated August 21, 1992, Florida Power & Light Company (FPL, the licensee) requested various changes to the Technical Specifications (TS) for the St. Lucie Plant, Unit Nos. 1 and 2. The licensee proposed to make administrative changes and achieve consistency throughout the TS by removing outdated material, making minor text changes and correcting errors.

2.0 EVALUATION

St. Lucie Unit 1 TS only

a. The licensee proposed the following minor changes in text:

(1) On page III change "pump" to "pumps".

(2) On page 3/4 0-3 add a period after the word "Requirements".

(3) On page 3/4 1-12 change "PUMP" to "PUMPS".

(4) On page 3/4 3-41 add a period after the word "OPERABLE".

(5) On page 3/4 4-1b replace a period with a comma after the word "pump".

(6) On page 3/4 4-14 add a title "SURVEILLANCE REQUIREMENTS" before section 4.4.6.2.

(7) On page 3/4 8-6b change "accummulated" to "accumulated"

(8) On page 3/4 11-2 delete a dash from the page number.

(9) On page 3/4 11-10 change "measureable" to "measurable".

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(10) On page 3/4 12-10 change "Specification 6.9.1.11" to "Specification 6.9.1.8".

(11) On page B 3/4 1-4 add subscript "r" to the term "F" with superscript "t".

(12) On page B 3/4 7-5 change "Criteria 10" to "Criteria 19"

(13) On page B 3/4 9-3 change the word "exceed" to "excess" and the word "insure" to "ensure".

(14) On page 6-24 replace a period with a comma after the word "systems".

a. The staff has reviewed the proposed changes and determined that they correct grammar, spelling, syntax and improve the consistency of the TS format. For that reason the staff finds the proposed changes acceptable.

b. On page 3/4 4-8 the licensee proposed to change Surveillance Requirement 4.4.5.5a to read, "Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2". In addition, to change Surveillance Requirement 4.4.5.5b to read, "The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection."

These changes do not alter the intent of these reporting requirements. They only make them consistent with the corresponding reporting requirements of Unit 2 TS. For that reason, the staff finds these changes acceptable.

c. On pages 3/4 6-5, 3/4 6-20 and 3/4 6-21, the licensee proposed to delete new valve numbers and any reference to them. These new valve numbers were approved by Amendment 95 issued on June 17, 1988 in anticipation of physical modification to the plant. Subsequently, the licensee determined that the modification was not justified economically and cancelled the modification. The proposed change returns the TS to its original form which reflects the existing valves.

In support of this change, the licensee provided the following explanation, "FPL is returning to the Technical Specifications that existed prior to the proposed modification described in FPL letter L-87-123. The modificattion was planned as an enhancement to the Breathing Air System to save time and provide operational flexibility during a refueling outage. The penetration is isolated during normal operation. It has been determined that this modification is economically unjustified. No detrimental effects to the Breathing Air System will result and the system will remain as originally designed."

The staff finds the above explanation reasonable and the changes acceptable.

St. Lucie Unit 2 TS only

a. The licensee proposed the following minor changes in text:

(1) On page 2-4 add reference note "(1)" after the word "pressure" in Table 2.2-1 part 4.

(2) On page 2-5 add reference note "(1)" after the word "low" in Table 2.2-1 part 14.

(3) On page 3/4 2-9 replace a period with a comma after the word "loading."

(4) On page 3/4 3-2 add reference note "(a)" after the "2" and before "(d)" in Table 3.3-1 part 4, and reference note "(a)" after the "2/SG" in Table 3.3-1 part 14.

(5) On page 3/4 3-13 close parenthesis ")" after the word "buttons" in Table 3.3-3 part 4a.

(6) On page 3/4 6-2 delete the word "of" and add the word "to" in item 3.6.1.2c.

(7) On page 3/4 6-26 replace the words "set points" with the word "setpoints" in item 3.6.5.

(8) On page 3/4 8-7 delete the close parenthesis ")" at the end of paragraph 7.

(9) On page 3/4 9-9 delete the comma "," after the words "Amendment No. 48" at the bottom of the page.

(10) On page 3/4 11-10 replace the word "measureable" with the word "measurable" in item e.

(11) On page 3/4 replace referenced Specification "6.9.1.11" with "6.9.1.8."

(12) On page 6-23 in item 6.13.2.1a replace the word "suport" with "support."

The staff has reviewed the proposed changes and determined that they correct grammar, spelling, syntax and improve the consistency of the TS format. For that reason the staff finds the proposed changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commissions regulations the Florida State Official was notified of the proposed issuance of the amendments. The State Official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change administrative procedures. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (57 FR 47130). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. A. Norris

Date: December 24, 1992