May 8, 1990

Docket Nos. 50-335
and 50-389

DISTRIBUTION
See attached sheet

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

Dear Mr. Goldberg:

SUBJECT: ISSUANCE OF AMENDMENTS - ADMINISTRATIVE UPDATE FOR ST. LUCIE PLANT, UNIT NOS. 1 AND 2 (TAC NOS. 69322, 69323, 72978 AND 72979)

The Commission has issued the enclosed Amendment Nos. 102 and 45 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 applications dated September 7, 1988 and April 4, 1989, as modified by letters dated February 1, 1990 and April 24, 1990.

These amendments revise the Technical Specifications (TS) in order to (1) achieve consistency throughout the TS, (2) remove outdated and/or fully satisfied material, (3) make minor text changes, (4) correct errors, and (5) delete the specific composition list for the Company Nuclear Review Board (CNRB), replacing it with a statement defining the requisite level of expertise.

In addition, your February 1, 1990 letter withdrew your request to revise the Independent Safety Engineering Group reporting and administrative requirements for St. Lucie Unit 2. The staff has granted your request and therefore this portion of your application was not evaluated.

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. 102 to DPR-67
2. Amendment No. 45 to NPF-16
3. Safety Evaluation

cc w/enclosures:
See next page
*See previous concurrence

OFC	LA:PD22*	PM:PD2	D:PD22	OGC*	LPEB		
NAME	DMiller	JNorris	HBerkow		FAllenspach		
DATE	3/05/90	5/8/90	5/8/90	3/08/90	5/ /90		

DF01
11

DATED: May 8, 1990

AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1
AMENDMENT NO. 45 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-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
G. Hill (8), P1-137
Wanda Jones, P-130A
J. Calvo, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
PD Plant-specific file [Gray File]
M. Sinkule, R-II
Others as required

cc: Plant Service list

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Florida Power & Light Company

St. Lucie Plant

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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. 102
License No. DPR-67

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

   A. The applications for amendments by Florida Power & Light Company, et al. (the licensee), dated September 7, 1988 and April 4, 1989, as modified by letters dated February 1, 1990 and April 24, 1990 comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 102, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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: May 8, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 102

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
B2-4	B2-4
B2-5	B2-5
3/4 3-4	3/4 3-4
3/4 3-11	3/4 3-11
3/4 3-15	3/4 3-15
3/4 3-19	3/4 3-19
3/4 7-44	3/4 7-44
3/4 9-1	3/4 9-1
3/4 9-3	3/4 9-3
3/4 9-5	3/4 9-5
3/4 12-1	3/4 12-1
3/4 12-11	3/4 12-11
3/4 12-12	3/4 12-12
6-9	6-9
6-10	6-10
6-12	6-12

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The Power Level-High trip setpoint is operator adjustable and can be set no higher than 9.61% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 15% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is consistent with the value used in the safety analysis.

Reactor Coolant Flow-Low

The Reactor Coolant Flow - Low trip provides core protection against DNB in the event of a sudden significant decrease in RCS flow. The reactor trip setpoint on low RCS flow is calculated by a

Reactor Coolant Flow-Low (Continued)

relationship between steam generator differential pressure, core inlet temperature, instrument errors and response times. When the calculated RCS flow falls below the trip setpoint an automatic reactor trip signal is initiated. The trip setpoint and allowable values ensure that for a degradation of RCS flow resulting from expected transients, a reactor trip occurs to prevent violation of local power density or DNBR safety limits.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure High trip provides assurance that a reactor trip is initiated concurrently with a safety injection.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 600 psia is sufficiently below the full-load operating point of 800 psig so as not

Steam Generator Pressure-Low (Continued)

to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 22 psi in the accident analyses.

Steam Generator Water Level - Low

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded due to loss of steam generator heat sink. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 10 minutes before auxiliary feedwater is required.

Local Power Density-High

The local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 1% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

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

(e) Deleted

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	9#
6. LOSS OF POWER					
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)					
(1) Undervoltage Device #1	2/Bus	2/Bus	1/Bus	1, 2, 3	12
(2) Undervoltage Device #2	2/Bus	2/Bus	1/Bus	1, 2, 3	12
c. 480 V Emergency Bus Under-voltage (Degraded Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	12
7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	11
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	8
c. SG Level (1A/1B) - Low	4/SG	2/SG	3/SG	1, 2, 3	13#, 14
8. AUXILIARY FEEDWATER ISOLATION					
a. SG 1A - SG 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	13#, 14
b. Feedwater Header SG 1A - SG 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	13#, 14

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1725 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below 685 psig; bypass shall be automatically removed at or above 685 psig.

The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. LOSS OF POWER		
a. (1) 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2900 ± 29 volts with a 1 ± .5 second time delay	2900 ± 29 volts with a 1 ± .5 second time delay
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)		
(1) Undervoltage Device #1	3675 ± 36 volts with a 7 ± 1 minute time delay	3675 ± 36 volts with a 7 ± 1 minute time delay
(2) Undervoltage Device #2	3592 ± 36 volts with a 18 ± 2 second time delay	3592 ± 36 volts with a 18 ± 2 second time delay
c. 480 volts Emergency Bus Undervoltage (Degraded Voltage)	429 + 5-0 volts with a 7 ± 1 second time delay	429 + 5 -0 volts with a 7 ± 1 second time delay
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. SG 1A & 1B Level Low	≥29.0%	≥28.5%
8. AUXILIARY FEEDWATER ISOLATION		
a. Steam Generator ΔP-High	≤275 psid	≤281 psid
b. Feedwater Header High ΔP	≤150.0 psid	≤157.5 psid

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Fan Coolers	Not Applicable
Feedwater Isolation	Not Applicable
Containment Isolation	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIS	
Containment Isolation	Not Applicable
Shield Building Ventilation System	Not Applicable
d. RAS	
Containment Sump Recirculation	Not Applicable
e. MSIS	
Main Steam Isolation	Not Applicable
Feedwater Isolation	Not Applicable
f. AFAS	
Auxiliary Feedwater Actuation	Not Applicable
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0*/19.5**
b. Containment Isolation ***	$\leq$ 30.5*/20.5**
c. Containment Fan Coolers	$\leq$ 30.0*/17.0**
d. Feedwater Isolation	$\leq$ 60.0

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	S	R	M	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)				
(1) Undervoltage Device #1	S	R	M	1, 2, 3
(2) Undervoltage Device #2	S	R	M	1, 2, 3
. 480 V Emergency Bus Undervoltage (Degraded Voltage)	S	R	M	1, 2, 3
7. AUXILIARY FEEDWATER (AFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (A/B) - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M	1, 2, 3
8. AUXILIARY FEEDWATER ISOLATION				
a. SG Level (A/B) - Low and SG Differential Pressure (BtoA/AtoB) - High	N.A.	R	M	1, 2, 3
b. SG Level (A/B) - Low and Feedwater Header Differential Pressure (BtoA/AtoB) - High	N.A.	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The logic circuits shall be tested manually at least once per 31 days.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The fire hose stations shown in Table 3.7-3 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-3 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the station to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the fire hose stations shown in Table 3.7-3 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the station to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Removing the hose for inspection and re-racking, and

   2. Replacement of all degraded gaskets in couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater. (Hoses on exterior hose stations shall be hydrostatic tested once per year.)

TABLE 3.7-3

FIRE HOSE STATIONS

A. Hose Stations (Turbine Building)

1. Operating Floor (northeast corner)
2. Operating Floor (southeast corner)
3. Operating Floor (middle east side)

B. Hose Stations (Reactor Auxiliary Building)

1. 43 ft. level south wall of HVE room
2. 43 ft. level "B" switchgear room by roll-up door
3. 43 ft. level southwest corner of "B" switchgear room near door.
4. 43 ft. level cable spreading room west wall
5. 19.5 ft. level east end of east-west hall
6. 19.5 ft. level middle of east-west hall
7. 19.5 ft. level south end of north-south hall
8. 19.5 ft. level entrance hall on south wall
9. -5 ft. level east end of hall
10. -5 ft. level south wall of hall near MCC 1B2
11. -5 ft. level west end of hall

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling cavity shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1000 pcm conservative allowance for uncertainties, or

b. A boron concentration of $\geq$ 1720 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or its equivalent until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ 1720 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1. The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the refueling cavity shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

*The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two wide range logarithmic neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each wide range logarithmic neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

DECAY TIME

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment penetrations shall be in the following status:

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

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

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

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

REFUELING OPERATIONS

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

REFUELING OPERATIONS

MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane shall be used for movement of CEAs or fuel assemblies and shall be OPERABLE with:

a. A minimum capacity of 2000 pounds, and

b. An overload cut off limit of $\leq$ 3000 pounds.

APPLICABILITY: During movement of CEAs or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane OPERABILITY not satisfied, suspend use of any inoperable manipulator crane from operations involving the movement of CEAs and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 The manipulator crane used for movement of CEAs or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 2500 pounds and demonstrating an automatic load cut off when the crane load exceeds 3000 pounds.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.8, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the confirmed* level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report shall include the methodology for calculating the cumulative potential dose contributions for the calendar year from radionuclides detected in environmental samples and can be determined in accordance with the methodology and parameters in the ODCM. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or broadleaf vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations

*A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis but in any case within 30 days.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.10, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2(500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.7 identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

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*Broad leaf vegetation sampling may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4b shall be followed, including analysis of control samples.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.*

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

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*This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA).

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

FUNCTION

6.5.2.1 The Company Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechnical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The Executive Vice President shall appoint, in writing, a minimum of five members to the CNRB and shall designate from this membership, in writing, a Chairman. The membership shall function to provide independent review and audit in the areas listed in Specification 6.5.2.1. The Chairman shall meet the requirements of ANSI/ANS-3.1-1987, Section 4.7.1. The members of the CNRB shall meet the educational requirements of the ANSI/ANS-3.1-1987, Section 4.7.2, and have at least 5 years of professional level experience in one or more of the fields listed in Specification 6.5.2.1. CNRB members who do not possess the educational requirements of ANSI/ANS-3.1-1987, Section 4.7.2 (up to a maximum of 2 members) shall be evaluated, and have their membership approved and documented, in writing, on a case-by-case basis by the Executive Vice President, considering the alternatives to educational requirements of ANSI/ANS-3.1-1987, Sections 4.1.1 and 4.1.2.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter and as convened by the CNRB Chairman or his designated alternate.

QUORUM

6.5.2.6 The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least a majority of CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The CNRB shall review:

a. The safety evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments completed under the provisions of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meeting minutes of the Facility Review Group.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the CNRB or the Executive Vice President.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

j. The PROCESS CONTROL PROGRAM and implementing procedures for dewatering of radioactive bead resin at least once per 24 months.

AUTHORITY

6.5.2.9 The CNRB shall report to and advise the Executive Vice President on those areas of responsibility specifed in Specifications 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of CNRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each CNRB meeting shall be prepared, approved and forwarded to the Executive Vice President within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 above, shall be prepared, approved and forwarded to the Executive Vice President within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 above, shall be forwarded to the Executive Vice President and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the FRG, and the results of the review shall be submitted to the CNRB, and the Senior Vice President - Nuclear.

6.7 SAFETY LIMIT VIOLATION

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

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Senior Vice President - Nuclear and the CNRB shall be notified within 24 hours.

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

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. 45
License No. NPF-16

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

   A. The applications for amendments by Florida Power & Light Company, et al. (the licensee), dated September 7, 1988 and April 4, 1989, as modified by letters dated February 1, 1990 and April 24, 1990, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, 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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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: May 8, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 45

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
B2-1	B2-1
B2-6	B2-6
3/4 3-41	3/4 3-41
3/4 3-45	3/4 3-45
3/4 3-46	3/4 3-46
3/4 6-1	3/4 6-1
3/4 7-16	3/4 7-16
3/4 7-22	3/4 7-22
3/4 7-32	3/4 7-32
3/4 7-36	3/4 7-36
3/4 7-37	3/4 7-37
3/4 12-1	3/4 12-1
3/4 12-11	3/4 12-11
3/4 12-12	3/4 12-12
6-1	6-1
6-6	6-6
6-10	6-10
6-23	6-23

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.28. This value is derived through a statistical combination of the system parameter probability distribution functions with the CE-1 DNB correlation uncertainty. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.28 for the family of axial shapes and corresponding radial peaks shown in Figure B 2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The Thermal Margin/Low Pressure and Local Power Density Trip Systems, in conjunction with Limiting Conditions for Operation, the Variable Overpower Trip and the Power Dependent Insertion Limits, assure that the Specified Acceptable Fuel Design Limits on DNB and Fuel Centerline Melt are not exceeded during normal operation and design basis Anticipated Operational Occurrences.

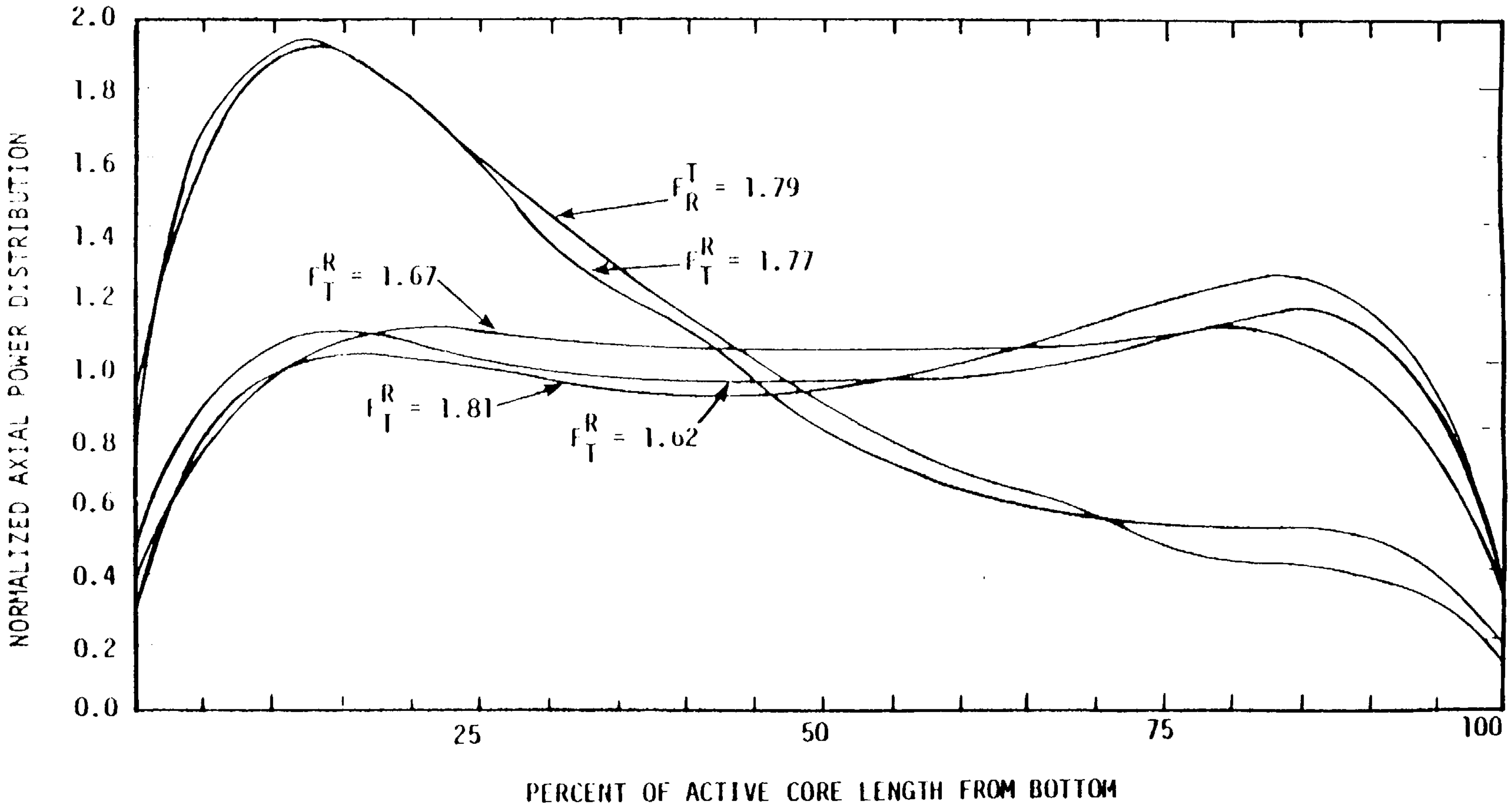

Figure B 2.1-1

Axial power distribution for thermal margin safety limits

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to or concurrently with a safety injection (SIAS). This also provides assurance that a reactor trip is initiated prior to or concurrently with an MSIS.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint of 620 psia is sufficiently below the full load operating point of approximately 885 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of 30 psi in the safety analyses.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before auxiliary feedwater is required. This trip also protects against violation of the specified acceptable fuel design limits (SAFDL) for DNBR, offsite dose and the loss of shutdown margin for asymmetric steam generator transients such as the opening of a main steam safety valve or atmospheric dump valve.

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower excore neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group positiion being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15% power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

BASES

RCP Loss of Component Cooling Water

A loss of component cooling water to the reactor coolant pumps causes a delayed reactor trip. This trip provides protection to the reactor coolant pumps by ensuring that plant operation is not continued without cooling water available. The trip is delayed 10 minutes following a reduction in flow to below the trip setpoint and the trip does not occur if flow is restored before 10 minutes elapses. No credit was taken for this trip in the safety analysis. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protective System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides core protection against DNB in the event of a sudden significant decrease in RCS flow. The reactor trip setpoint on low RCS flow is calculated by a relationship between steam generator differential pressure, core inlet temperature, instrument errors and response times. When the calculated RCS flow falls below the trip setpoint an automatic reactor trip signal is initiated. The trip setpoint and allowable values ensure that for a degradation of RCS flow resulting from expected transients, a reactor trip occurs to prevent violation of local power density or DNBR safety limits.

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a.* With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b.* With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c.** With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

d.** With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and
2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status, and
3. Restore the Channel to OPERABLE status at the next scheduled refueling.

e. The provisions of Specification 3.0.4 are not applicable.

* Action statements do not apply to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

**Action statements apply only to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel will 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

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	SMOKE (x/y)
REACTOR AUXILIARY BUILDING		
ZONE-1A REACTOR AUX. BLDG. EL. 0.50		6/0
ZONE-2A REACTOR AUX. BLDG. EL. 0.50		4/0
ZONE-3A REACTOR AUX. BLDG. EL. 19.50		6/0
ZONE-4A REACTOR AUX. BLDG. EL. 19.50	2/0	5/0
ZONE-5A REACTOR AUX. BLDG. EL. 19.50		8/0
ZONE-6A REACTOR AUX. BLDG. EL. 43.00		5/0
ZONE-7A REACTOR AUX. BLDG. EL. 43.00		7/0
ZONE-8A REACTOR AUX. BLDG. EL. 62.00	1/0	6/0
ZONE-9A REACTOR AUX. BLDG. EL. 43.00		2/0
ZONE-10A REACTOR AUX. BLDG. EL. 43.00		2/0
ZONE-12A ELECT. PEN. ROOM EL. 19.50		3/0
ZONE-1B REACTOR AUX. BLDG. EL. 0.50		6/0
ZONE-2B REACTOR AUX. BLDG. EL. 0.50		5/0
ZONE-3B REACTOR AUX. BLDG. EL. 19.50		6/0
ZONE-4B HSCP-1 REC. AUX. BLDG. EL. 43.00		1/0
ZONE-5B REACTOR AUX. BLDG. EL. 19.50		6/0
ZONE-6B REACTOR AUX. BLDG. EL. 43.00		4/0
ZONE-7B REACTOR AUX. BLDG. EL. 43.00		6/0
ZONE-8B REACTOR AUX. BLDG. EL. 62.00		5/0
ZONE-9B REACTOR AUX. BLDG. EL. 43.00		2/0
ZONE-10B REACTOR AUX. BLDG. EL. 43.00		2/0
ZONE-12B ELECT. PEN. ROOM EL. 19.50		4/0
ZONE-1F FAN ROOM EL. 43.00	0/2	
ZONE-2F CABLE LOFT EL. 19.50	0/26	
ZONE-3F IODINE REMOVAL/WASTE GAS/ HALLWAYS EL. 0.50	0/15	
ZONE-4F B ELECTRICAL PENETRATION ROOM EL. 19.50	0/2	
ZONE-5F A ELECTRICAL PENETRATION ROOM EL. 19.50	0/1	
ZONE-6F CABLE SPREADING ROOM EL. 43.00	0/9	
FUEL HANDLING BUILDING		
ZONE-20A FUEL HANDLING BLDG. EL. 19.50		1/0
ZONE-21A FUEL HANDLING BLDG. EL. 48.00		3/0
ZONE-20B FUEL HANDLING BLDG. EL. 19.50		1/0
ZONE-21B FUEL HANDLING BLDG. EL. 48.00		2/0
DIESEL GENERATOR BUILDING		
ZONE-22A DIESEL GEN. BLDG./D.O. STORAGE TANK	2/2	2/0
ZONE-22B DIESEL GEN. BLDG./D.O. STORAGE TANK	2/2	2/0

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	SMOKE (x/y)
SAFETY RELATED PUMPS		
ZONE-17A COMPONENT COOLING AREA		4/0
ZONE-18A INTAKE COOLING WATER PUMP AREA	1/0	
ZONE-19A STEAM TRESTLE AREA-AUX. FEEDWATER PUMP		2/0
ZONE-17B COMPONENT COOLING AREA		2/0
ZONE-18B INTAKE COOLING WATER PUMP AREA	1/0	
ZONE-19B STEAM TRESTLE AREA-AUX. FEEDWATER PUMP		2/0
TURBINE BUILDING/SWITCHGEAR ROOM		
ZONE-16A TURBINE BLDG. SWITCHGEAR ROOM		3/0
ZONE-16B TURBINE BLDG. SWITCHGEAR ROOM		3/0
CONTAINMENT#		
ZONE-11A ANNULUS		1/0
ZONE-13A REACTOR TUNNEL BELOW EL. 18.00		2/0
ZONE-14A REACTOR EL. 18.00		5/0
ZONE-15A REACTOR EL. 45.00	2/0	4/0
ZONE-11B ANNULUS		1/0
ZONE-13B REACTOR TUNNEL BELOW EL. 18.00		1/0
ZONE-14B REACTOR EL. 18.00		5/0
ZONE-15B REACTOR EL. 45.00	2/0	5/0

*(x/y): x is number of early warning fire detection and notification only instruments.

y is number of actuation of fire suppression systems and early warning notification instruments.

#The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

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 SYSTEMS

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 of than or equal 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

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5.1 The ultimate heat sink shall be OPERABLE with:

a. Cooling water from the Atlantic Ocean providing a water level above -10.5 feet elevation, Mean Low Water, at the plant intake structure, and

b. Two OPERABLE valves in the barrier dam between Big Mud Creek and the intake structure.

APPLICABILITY: At all times.

ACTION:

a. With the water level requirement of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and provide cooling water from Big Mud Creek within the next 12 hours.

b. With one isolation valve in the barrier dam between Big Mud Creek and the intake structure inoperable, restore the inoperable valve to OPERABLE status within 72 hours, or within the next 24 hours, install a temporary flow barrier and open the barrier dam isolation valve. The availability of the onsite equipment capable of removing the barrier shall be verified at least once per 7 days thereafter.

c. With both of the isolation valves in the barrier dam between the intake structure and Big Mud Creek inoperable, within 24 hours, either:

   1. Install both temporary flow barriers and manually open both barrier dam isolation valves. The availability of the onsite equipment capable of removing the barriers shall be verified at least once per 7 days thereafter, or

   2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.5.1.1 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average water level to be within limits.

4.7.5.1.2 The isolation valves in the barrier dam between the intake structure and Big Mud Creek shall be demonstrated OPERABLE at least once per 6 months by cycling each valve through at least one complete cycle of full travel.

PLANTS SYSTEMS

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for the facility site via stoplogs which shall be installed on the southside of the RAB and the southernmost door on east wall whenever a hurricane warning for the plant is posted.

APPLICABILITY: At all times.

ACTION:

With either a Hurricane Watch or a Hurricane Warning issued for the facility site, perform the St. Lucie Plant Beach Survey Procedure pursuant to Surveillance Requirement 4.7.6.1.1 below and ensure the stoplogs are removed from storage and are prepared for installation. The stoplogs shall be installed anytime a hurricane warning is posted.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 The St. Lucie Plant Beach Survey Procedure shall be conducted at least once per year between the dates of May 25 and June 7 and within 30 days following the termination of either a Hurricane Watch or a Hurricane Warning for the facility site. A Special Report containing the results of these surveys shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days following the completion of the survey.

4.7.6.1.2 The St. Lucie Mangrove Photographic Survey Procedure shall be conducted at least once per 12 months and shall be a color infrared photograph(s), or equivalent, of the mangrove area between the facility and the FP&L east property line. The results of these surveys shall be included in the Annual Operating Report for the period in which the survey was completed. This report shall include an evaluation of the facility flood protection if the survey indicates deterioration, either man-made or natural, of this mangrove area.

4.7.6.1.3 Meteorological forecasts shall be obtained from the National Hurricane Center in Miami, Florida at least once per 6 hours during either a Hurricane Watch or a Hurricane Warning.

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All safety-related snubbers shall be OPERABLE.

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

ACTION:

With one or more safety related snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

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

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

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

#The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

c. Refueling Outage Inspections

At least once per 18 months an inspection shall be performed of all safety related snubbers attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; (3) stroking the mechanical snubber through its full range of travel.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visble indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers, irrespective of type, that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.9f. When a fluid port of a hydraulic snubber is found to be uncovered the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as found setting, extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be determined to be OPERABLE by visually verifying the required level of oil for operation for each affected snubber; otherwise declare the snubbers inoperable.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of either: (1) At least 10% of the total of each type of safety related snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9f. an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested or (2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9f. The cumulative number of snubbers of a type tested is denoted by "N." At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all

SURVEILLANCE REQUIREMENTS

4.7.11.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

c. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) in the flow path is in its correct position.

d. At least once per 12 months by performance of a system flush.

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each automatic valve in the flow path actuates to its correct position,

   2. Verifying that each pump develops at least 2350 gpm at a system head of 232 feet,

   3. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   4. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 85 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The following sprinkler systems shall be OPERABLE:

1. Fire Zone 8 - Diesel Generator Building 2A
2. Fire Zone 9 - Diesel Generator Building 2B
3. Fire Zone 19 - RAB East Hallway and Miscellaneous Equipment Areas
4. Fire Zone 20 - RAB East-West Common Hallway
5. Fire Zone 22 - RAB Electrical Penetration Area
6. Fire Zone 23 - RAB Electrical Penetration Area
7. Fire Zone 39 - RAB HVAC Equipment Room
8. Fire Zone 51 - RAB Ceiling and Hallways
9. Fire Zone 52 - Cable Spreading Room

APPLICABILITY: Whenever equipment protected by the sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the above required sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS

c. At least once per 18 months by:

1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

2. Removing the hose for inspection and re-racking, and

3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

d. For all fire hose stations not located in the turbine building, at least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION/ELEVATION		HOSE RACK #
A.	Hose Stations (Turbine Building)	
1.	Operating Floor (northeast corner)	HS-15-4
2.	Operating Floor (southeast corner)	HS-15-10
3.	Operating Floor (middle east side)	HS-15-7
B.	Hose Stations (Reactor Auxiliary Building)	
1.	62 ft level east wall entrance	HS-15-44
2.	62 ft level west wall entrance	HS-15-45
3.	62 ft level west wall entrance to H&V room	HS-15-46
4.	43 ft level S.E. corner cable spreading room	HS-15-36
5.	43 ft level south wall of H&V room	HS-15-37
6.	43 ft level cable spreading room	HS-15-31
7.	43 ft level southwest corner of "B" switchgear room near door	HS-15-42
8.	19.5 ft level east end of east-west hall	HS-15-38
9.	19.5 ft level middle of east-west hall	HS-15-40
10.	19.5 ft level south end of north-south hall	HS-15-33
11.	19.5 ft level entrance hall on south wall	HS-15-34
12.	-0.5 ft level east end of hall	HS-15-41
13.	-0.5 ft level south wall of hall	HS-15-28
14.	-0.5 ft level west end of hall	HS-15-43
C.	Hose Stations (Reactor Containment Building)	
1.	RCB at 23 ft level (near stairway no. 3)	HS-15-47
2.	RCB at 45 ft level (near stairway no. 1)	HS-15-48
3.	RCB at 45 ft level (near stairway no. 2)	HS-15-54
4.	RCB at 62 ft level (near stairway no. 3)	HS-15-49
D.	Hose Station (Fuel Handling Building)	
	62 ft level northwest corner	HS-15-55

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.11.4 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise provide the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings.

3. Performing a flow check of each hydrant to verify its OPERABILITY.

TABLE 3.7-5

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION		HYDRANT NUMBER
RCB	NE	FH#6
RCB	NW	FH#7
CST	N	FH#9
CCWB	E	FH#20
DGB(2A)	E	FH#21
DGB(2B)	SE	FH#22
RAB	S	FH#23
Intake Structure	SE	FH#25
CST	SW	FH#26

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.8, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the confirmed* level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report shall include the methodology for calculating the cumulative potential dose contributions for the calendar year from radionuclides detected in environmental samples and can be determined in accordance with the methodology and parameters in the ODCM. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or broadleaf vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific

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*A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis but in any case within 30 days.

ACTION: (Continued)

locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.9.1.10, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.7.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Pursuant to Specification 6.9.1.7, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

*Broad leaf vegetation sampling may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4b shall be followed, including analysis of control samples.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.*

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.8.

*This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA).

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the control room, a designated individual, shall be responsible for the control room command function. A management directive to this effect, signed by the Senior Vice President - Nuclear , shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Topical Quality Assurance Report and updated in accordance with 10 CFR 50.54(a)(3).

b. The Senior Vice President - Nuclear shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the health physics manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

6.2 ORGANIZATION (Continued)

UNIT STAFF

6.2.2 The unit organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the control room.

c. A health physics technician# shall be on site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed by a licensed operator and supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. The SRO in charge of fuel handling normally supervises from the control room and has the flexibility to directly supervise at either the refueling deck or the spent fuel pool.

e. A site Fire Brigade of at least five members shall be maintained onsite at all times.# The Fire Brigade shall not include the Shift Supervisor, the STA, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modification, on a temporary basis the following guidelines shall be followed:

#The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION
TWO UNITS WITH TWO SEPARATE CONTROL ROOMS

WITH UNIT 1 IN MODE 5 OR 6 OR DEFUELED

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS (SRO)	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	2[b]
STA	1	None

WITH UNIT 1 IN MODE 1, 2, 3 OR 4

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS (SRO)	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	1
STA	1[a]	None

SS - Shift Supervisor with a Senior Reactor Operator's License on Unit 2
SRO - Individual with a Senior Reactor Operator's License on Unit 2
RO - Individual with a Reactor Operator's License on Unit 2
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

a/ Individual may fill the same position on Unit 1
b/ One of the two required individuals may fill the same position on Unit 1.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of five dedicated, full-time members with varied backgrounds and disciplines related to nuclear power plants. No more than two members shall be assigned from any one department. Three or more of the members shall be engineers with a Bachelor's degree in engineering or a related science, with at least 2 years of professional level experience in the nuclear field. Any nondegreed ISEG members will either be licensed as a Reactor Operator or Senior Reactor Operator, or will have been previously licensed as a Reactor Operator or Senior Reactor Operator within the last year at the St. Lucie Plant site; or they will meet the qualifications of a department head as specified in Specification 6.3.1 of the St. Lucie Unit 2 Technical Specifications. The qualifications of each nondegreed candidate for the ISEG shall be approved by the Site Vice President - St. Lucie Plant, prior to joining the group.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of selected plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving plant safety to the Site Vice President - St. Lucie Plant.

AUTHORITY

6.2.3.4 The ISEG is an onsite independent technical review group that reports to the Site Vice President - St. Lucie Plant. The ISEG shall have the authority necessary to perform the functions and responsibilities as delineated above.

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained and a report of the activities forwarded each calendar month to the Site Vice President - St. Lucie Plant.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor function is to provide on shift advisory technical support in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS-3.1-1978 as endorsed by Regulatory Guide 1.8, September 1975 (reissued May 1977), except for the (1) Health Physics Supervisor who shall meet

*Not responsible for sign-off function.

RESPONSIBILITIES (Continued)

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Senior Vice President - Nuclear and to the Company Nuclear Review Board.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and RADWASTE TREATMENT SYSTEMS.

m. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last FRG meeting.

AUTHORITY

6.5.1.7 The Facility Review Group shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under Specifications 6.5.1.6a. through d. and m. above.

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6a. through e. above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Senior Vice President - Nuclear and the Company Nuclear Review Board of disagreement between the FRG and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The Facility Review Group shall maintain written minutes of each FRG meeting that, at a minimum, document the results of all FRG activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Senior Vice President - Nuclear and the Chairman of the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

FUNCTION

6.5.2.1 The Company Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

FUNCTION (Continued)

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The Executive Vice President shall appoint, in writing, a minimum of five members to the CNRB and shall designate from this membership, in writing, a Chairman. The membership shall function to provide independent review and audit in the areas listed in Specification 6.5.2.1. The Chairman shall meet the requirements of ANSI/ANS-3.1-1987, Section 4.7.1. The members of the CNRB shall meet the educational requirements of the ANSI/ANS-3.1-1987, Section 4.7.2, and have at least 5 years of professional level experience in one or more of the fields listed in Specification 6.5.2.1. CNRB members who do not possess the educational requirements of ANSI/ANS-3.1-1987, Section 4.7.2 (up to a maximum of 2 members) shall be evaluated, and have their membership approved and documented, in writing, on a case-by-case basis by the Executive Vice President, considering the alternatives to educational requirements of ANSI/ANS-3.1-1987, Sections 4.1.1 and 4.1.2.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter and as convened by the CNRB Chairman or his designated alternate.

QUORUM

6.5.2.6 The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least a majority of CNRB members including alternates. No more than a minority of the quorom shall have line responsibility for operation of the unit.

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 suport 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 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.102AND45

TO FACILITY OPERATING LICENSE NOS. DPR-67 AND NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By applications dated September 7, 1988 and April 4, 1989, the Florida Power & Light Company (FP&L, the licensee) requested various changes to the Technical Specifications (TS) for the St. Lucie Plant, Unit Nos. 1 and 2. By letter dated February 1, 1990, FP&L modified its April 4, 1989 submittal by (1) revising the proposed general statement for the Company Nuclear Review Board after discussions with the NRC staff, and (2) withdrawing their request to revise the Independent Safety Engineering Group reporting and administrative requirements for St. Lucie Unit 2. In addition, by letter dated April 24, 1990, the licensee supplemented the February 1, 1990 letter in response to the staff's concern regarding the revised CNRB membership. The staff has granted the licensee's request for withdrawal; therefore, this portion of the April 4, 1989 request is not evaluated below. The staff has also determined that the information provided in the February 1, 1990 and the April 24, 1990 letters did not alter the staff's proposed no significant hazards consideration determination as noticed in the Federal Register on May 31, 1989. The staff's evaluation of the other proposed changes follows.

2.0 EVALUATION - OVERVIEW

The proposed TS changes can be grouped into four broad categories. Category 1 changes achieve consistency throughout the TS or between similar TS for both St. Lucie Units 1 and 2. Category 2 changes correct errors in the TS. Category 3 changes delete an outdated and fully satisfied footnote in the TS or a fully satisfied License Condition in the St. Lucie Unit 2 Operating License. The Category 4 change relates to the composition of the Company Nuclear Review Board (CNRB). Below is the staff's evaluation of the changes by category.

2.1 CATEGORY 1 - CHANGES TO ACHIEVE CONSISTENCY WITHIN THE TS OR BETWEEN SIMILAR TS FOR THE TWO ST. LUCIE UNITS

St. Lucie Unit 1 TS

a) Pages B 2-4 and B 2-5, concerning reactor trip setpoints, currently discuss operation in Modes 1 and 2 with only 2 or 3 reactor reactor coolant pumps (RCPs) in operation. A reactor coolant low flow trip provides core protection against Departure for Nucleate Boiling in the event of a sudden significant decrease in reactor coolant system flow. The specified setpoint ensures that a reactor trip occurs to prevent violation of Local Power Density or Departure from Nucleate Boiling Ratio (DNBR) limits under stated conditions. However, St. Lucie Unit 1 TS Limiting Condition for

Operation (LCO) 3.4.1.1 requires both RCPs in both reactor coolant loops be in operation in Modes 1 and 2. Therefore, discussions in the Bases concerning operation with other than 4 pumps in Modes 1 and 2 are inconsistent with LCO 3.4.1.1. For that reason the proposed change, which more clearly reflects the basis for this trip setpoint, is acceptable.

b) Page 3/4 7-44, Table 3.7-3, Fire Hose Stations, Items B-2, B-3 and B-4, are revised to provide more specific descriptions of the hose locations. Since the proposed changes do not change the substance of the table's content and are clarifying in nature, the staff finds them acceptable.

c) On Page 3/4 9-1, in the LIMITING CONDITION FOR OPERATION 3.9.1a, the licensee proposes to change the units of the reactivity uncertainty from "1% delta k/k" to "1000 pcm". By letter dated March 17, 1987, the licensee requested to change the reactivity units from "1% delta k/k" to "1000 pcm" in a number of TS but overlooked the entry on page 3/4 9-1. The request was granted on October 23, 1987 in Amendment No. 86 for St. Lucie No. 1 and Amendment 25 for St. Lucie No. 2. Since the change of the units has been previously approved and the present request only corrects an oversight, the staff finds the change acceptable.

d) On Page 6-12, in RECORDS, the licensee proposes to change "...Section 6.5.2.7..." to "...Specification 6.5.2.7..." to be consistent with other specifications. The proposed change corrects the incorrect nomenclature in the TS and is, therefore, acceptable.

St Lucie Unit No. 2

a) The licensee proposes to change Page B 2-6 to more clearly describe the bases for the reactor trip function and to be consistent with the same proposed revision in 2.1a above for St. Lucie Unit No. 1. As discussed in 2.1a, this change is acceptable.

2.2 CATEGORY 2 - CHANGES TO CORRECT ERRORS

A number of errors and typographical errors exist in the TS of both units. The licensee proposed that these be corrected. The Unit No. 1 errors are contained on pages 3/4 3-19, 3/4 9-3, 3/4 9-5, 3/4 12-1, 3/4 12-11, 3/4 12-12 and 6-12. The Unit No. 2 errors are contained on pages B 2-1, 3/4 3-41, 3/4 3-45, 3/4 3-46, 3/4 6-1, 3/4 7-16, 3/4 7-22, 3/4 7-32, 3/4 7-36, 3/4 7-37, 3/4 12-1, 3/4 12-11, 3/4 12-12, 6-1, 6-6, 6-10 and 6-23. The staff has reviewed each individual error as well as the licensee's rationale for correcting these errors and concludes that these corrections are acceptable.

2.3 CATEGORY 3 - CHANGES TO DELETE AN OUTDATED AND FULLY SATISFIED FOOTNOTE IN THE TS OR A FULLY SATISFIED LICENSE CONDITION IN THE OPERATING LICENSE

St. Lucie Plant, Unit No. 1

a) Page 3/4 3-4, Item (e) states "Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3." Special Test Exception 3.10.3 was deleted by Amendment No. 4, dated April 16, 1976. For that reason, the licensee proposes to delete Item (e), and the staff finds the deletion acceptable.

b) Pages 3/4 3-11, 3/4 3-15 and 3/4 3-19 contain a footnote which states: "*This specification will be effective prior to Cycle 7 restart." This footnote was added to the TS by Amendment No. 58 dated May 3, 1983 to address the criteria and staff positions relating to degraded grid voltage protection. In that amendment, the staff found the licensee's proposed modifications for protecting the Class 1E equipment acceptable. To implement the NRC-approved design changes, the licensee issued Plant Change/Modification 103-184, "Degraded Grid Voltage Design." This modification was completed on December 15, 1985 and St. Lucie Unit No. 1 started Cycle 7 on December 25, 1985. For these reasons, the licensee proposes to delete the footnote and the staff finds the deletion of the footnote acceptable.

St. Lucie Plant, Unit No. 2

a) Page 3/4 7-32 contains a footnote: "*The sprinkler systems shall be completely installed and OPERABLE prior to exceeding 5% of RATED THERMAL POWER." This footnote was a startup condition for Unit 2. St. Lucie Unit No. 2 passed that milestone on June 13, 1983 and the licensee notified the NRC by letter dated June 6, 1983 that this condition had been met. For this reason, the licensee proposes to delete the footnote and the staff finds the deletion acceptable.

b) The licensee also proposed to delete a portion of License Condition 2.C.12. License Condition 2.C.12 was deleted in its entirety by Amendment No. 34 dated September 13, 1988, therefore, no further action is required.

CATEGORY 4 - CHANGE RELATING TO THE COMPOSITION OF CNRB

St. Lucie Plant, Unit Nos. 1 and 2

The licensee proposes to change the description of the Company Nuclear Review Board (CNRB) contained in Specification 6.5.2.2 from the one which describes its members by title of the position they hold to the one which describes its members by technical discipline, level of education and professional experience. The main reason for the change is to avoid processing license amendments each time an organizational change takes place which changes the position titles of the CNRB members.

The change requires the FP&L Executive Vice President to appoint at least five members of CNRB whose qualifications meet the criteria of Section 4.7 of ANSI/ANS 3.1 - 1987 and have at least 5 years of cumulative professional-level experience in one or more of the fields listed in Specification 6.5.2.1. The Chairman of CNRB shall have at least 6 years of professional-level management experience in the nuclear power field.

The proposed change is consistent with recent NRC practice in defining requirements in the TS of a number recently licensed nuclear plants (e.g., Vogtle 1 and 2, Shearon Harris 1). For that reason, the staff finds the proposed change acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These amendments also involve changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, with respect to these items, the 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.

4.0 CONCLUSION

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

Date: May 8, 1990

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