September 22, 1997

Mr. T. F. Plunkett
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: ST. LUCIE UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: ADMINISTRATIVE UPDATE (TAC NOS. M98980 AND M98981)

Dear Mr. Plunkett:

The Commission has issued the enclosed Amendment Nos. 152 and 89 to Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated May 29, 1997, regarding removing outdated material, incorporating minor changes in text, making editorial corrections, and resolving other inconsistencies in the Unit 1 and 2 TS.

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 L.Wiens

L. A. Wiens, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-335
and 50-389

Enclosures:
1. Amendment No. 152 to DPR-67
2. Amendment No. 89 to NPF-16
3. Safety Evaluation

cc w/enclosures: See next page

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DATE	08/14/97		08/11/97		09/9/97		08/16/97		08/22/97	

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 22, 1997

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President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

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Mr. T. F. Plunkett
Florida Power and Light Company

ST. LUCIE PLANT

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 152
License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company, (the licensee), dated May 29, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

9710020159

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 152, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 22, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 152

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
VII	VII
B 2-6	B 2-6
B 2-8	B 2-8
3/4 1-1	3/4 1-1
3/4 1-4	3/4 1-4
3/4 1-8	3/4 1-8
3/4 1-12	3/4 1-12
3/4 1-21	3/4 1-21
3/4 1-22	3/4 1-22
3/4 2-9	3/4 2-9
3/4 4-21	3/4 4-21
3/4 6-3	3/4 6-3
3/4 6-20	3/4 6-20
3/4 6-26	3/4 6-26
3/4 9-4	3/4 9-4
3/4 9-6	3/4 9-6
3/4 10-2	3/4 10-2
3/4 11-14	3/4 11-14
B 3/4 1-3	B 3/4 1-3
B 3/4 4-15	B 3/4 4-15
B 3/4 9-1	B 3/4 9-1
5-4	5-4

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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 sufficient time for any operator action to initiate auxiliary feedwater before reactor coolant system subcooling is lost.

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.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident 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.

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. The trip is not credited in any design basis accident evaluated in UFSAR Chapter 15; however, the trip is considered in the safety analysis in that the presence of this trip function precluded the need for specific analyses of other events initiated from subcritical conditions.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} ≥ 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be ≥ 3600 pcm.

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

ACTION:

With the SHUTDOWN MARGIN < 3600 pcm, immediately initiate and continue boration at ≥ 40 gpm of 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be ≥ 3600 pcm:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2## at least once during CEA withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

---

* See Special Test Exception 3.10.1.
# With Keff ≥ 1.0.
With Keff < 1.0.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant to the reactor pressure vessel < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be determined to be ≥ 3000 gpm within one hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm to the reactor pressure vessel.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from the boric acid makeup tank via either a boric acid pump or a gravity feed connection and any charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump* to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. A least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

---

* The flow path from the RWT to the RCS via a single HPSI pump shall only be established if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case: 1) all charging pumps shall be disabled; 2) heatup and cooldown rates shall be limited in accordance with Figure 3.1-1b; and 3) at RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

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

CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

* The flow path from the RWT to the RCS via a single HPSI pump shall be established only if: (a) the RCS pressure boundary does not exist, or (b) RCS pressure boundary integrity exists and no charging pumps are operable. In the latter case: 1) all charging pumps shall be disabled; 2) heatup and cooldown rates shall be limited in accordance with Figure 3.1-1b; and 3) at RCS temperatures below 115°F, any two of the following valves in the operable HPSI header shall be verified closed and have their power removed:

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

LIMITING CONDITION FOR OPERATION (continued)

2. Declared inoperable and satisfy SHUTDOWN MARGIN requirements of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided all of the following conditions are met:

   a) Within 1 hour, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours.

e. With one full length CEA misaligned from any other CEA in its group by 15 or more inches, operation in MODES 1 and 2 may continue provided that the misaligned CEA is positioned within 7.5 inches of other CEAs in its group in accordance with the time constraints shown in COLR Figure 3.1-1a.

f. With one full-length CEA misaligned from any other CEA in its group by 15 or more inches beyond the time constraints shown in COLR Figure 3.1-1a, reduce power to ≤ 70% of RATED THERMAL POWER prior to completing ACTION f.1 or f.2.

1. Restore the CEA to OPERABLE status within its specified alignment requirements, or

2. Declare the CEA inoperable and satisfy the SHUTDOWN MARGIN requirements of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

   a) Within 1 hour, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

LIMITING CONDITION FOR OPERATION (continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours.

g. With more than one full length CEA inoperable or misaligned from any other CEA in its group by 15 inches (indicated position) or more, be in HOT STANDBY within 6 hours.

h. With one full-length CEA inoperable due to causes other than addressed by ACTION a above, and inserted beyond the long term steady state insertion limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length CEA shall be determined to be within 7.5 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Block Circuit are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full length CEA not fully inserted shall be determined to be OPERABLE by inserting it at least 7.5 inches at least once per 92 days.

4.1.3.1.3 The CEA Block Circuit shall be demonstrated OPERABLE at least once per 92 days by a functional test which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 7.5 inches (indicated position).

4.1.3.1.4 The CEA Block Circuit shall be demonstrated OPERABLE by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2:

*a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 92 days, and

b. At least once per 6 months.

---

* The licensee shall be excepted from compliance during the startup test program for an entry into MODE 2 from MODE 3 made in association with a measurement of power defect.

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T not within limits, within 6 hours either:

a. Be in at least HOT STANDBY, or

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

---

* See Special Test Exception 3.10.2.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2a, 3.4-2b and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

APPLICABILITY: At all times. *#

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} to less than 200°F within the following 30 hours in accordance with Figures 3.4-2b and 3.4-3.

* When the flow path from the RWT to the RCS via a single HPSI pump is established per 3.1.2.1 or 3.1.2.3 and RCS pressure boundary integrity exists, the heatup and cooldown rates shall be established in accordance with Fig. 3.1-1b.

During hydrostatic testing operations above system design pressure, a maximum temperature change in any one hour period shall be limited to 5°F.

Pages 3/4 6-4 through 3/4 6-9 have been DELETED.

Page 3/4 6-10 is the next valid page.

Pages 3/4 6-21 through 3/4 6-22 have been DELETED.

Page 3/4 6-23 is the next valid page.

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.1 The containment vessel to annulus vacuum relief valves shall be OPERABLE with an actuation setpoint of 2.25 ± 0.25 inches Water Gauge differential.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.5.1 No additional Surveillance Requirements other than those required by Specification 4.0.5 and at least once per 3 years verify that the vacuum relief valves open fully within 8 seconds at 2.25 ± 0.25 inches Water Gauge differential.

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 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 except for valves that are open on an intermittent basis under administrative control, 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 of containment isolation valves per the applicable portions of Specifications 4.6.3.1.1 and 4.6.3.1.2.

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 ≤ 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.

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 before the crane load exceeds 3000 pounds.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.3, or 3.2.4 are suspended.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 The concentration of oxygen in the waste gas decay tank shall be determined to be within the above limits by continuously monitoring the waste gases in the on service waste gas decay tank.

4.11.2.5.2 With the oxygen concentration in the on service waste gas decay tank greater than 2% by volume as determined by Specification 4.11.2.5.1, the concentration of hydrogen in the waste gas decay tank shall be determined to be within the above limits by gas partitioner sample at least once per 24 hours.

3/4.1.2 BORATION SYSTEMS (Continued)

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

The restrictions associated with the establishing of the flow path from the RWT to the RCS via a single HPSI pump provide assurance that 10 CFR 50 Appendix G pressure/temperature limits will not be exceeded in the case of any inadvertent pressure transient due to a mass addition to the RCS. If RCS pressure boundary integrity does not exist as defined in Specification 1.16, these restrictions are not required. Additionally, a limit on the maximum number of operable HPSI pumps is not necessary when the pressurizer manway cover or the reactor vessel head is removed.

3/4.1.3 MOVABLE CONTROL ASSEMBLES

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

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

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

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

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

3/4.4.13 POWER OPERATED RELIEF VALVES and 3/4.4.14 REACTOR COOLANT PUMP - STARTING

The low temperature overpressure protection system (LTOP) is designed to prevent RCS overpressurization above the 10 CFR 50 Appendix G operating limit curves (Figures 3.4-2a and 3.4-2b) at RCS temperatures at or below 304°F during heatup and 281°F during cooldown. The LTOP system is based on the use of the pressurizer power-operated relief valves (PORVs) and the implementation of administrative and operational controls.

The PORVs aligned to the RCS with the low pressure setpoints of 350 and 530 psia, restrictions on RCP starts, limitations on heatup and cooldown rates, and disabling of non-essential components provide assurance that Appendix G P/T limits will not be exceeded during normal operation or design basis overpressurization events due to mass or energy addition to the RCS. The LTOP system APPLICABILITY, ACTIONS, and SURVEILLANCE REQUIREMENTS are consistent with the resolution of Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," pursuant to Generic Letter 90-06.

3/4.4.15 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The redundancy design of the Reactor Coolant System vent systems serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.9.1 BORON CONCENTRATION

The limitation on minimum boron concentration ensures that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volumes having direct access to the reactor vessel. The limitation on K_{eff} is sufficient to prevent reactor criticality with all full length rods (shutdown and regulating) fully withdrawn.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the wide range logarithmic range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

In accordance with Generic Letter 91-08, Removal of Component Lists from the Technical Specifications, the opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements of the cranes used for movement of fuel assemblies ensures that : 1) each crane has sufficient load capacity to lift a fuel element, and 2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet

b. Annulus nominal volume = 543,000 cubic feet

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

d. Nominal inside diameter = 148 feet

e. Cylinder wall minimum thickness = 3 feet

f. Dome minimum thickness = 2.5 feet

g. Dome inside radius - 112 feet

DESIGN PRESSURE AND TEMPERATURE

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

PENETRATIONS

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

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of between 134.1 and 136.7 inches. Individual fuel assemblies shall contain fuel rods of the same nominal active fuel length. Fuel assemblies shall be limited to those designs that have been analyzed using NRC approved methodology and shown by tests or analyses to comply with fuel design and safety criteria. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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. 89
License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated May 29, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 2.C.2 to read as follows:

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 89 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: September 22, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 89

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
XVIII	XVIII
XIX	XIX
B 2-1	B 2-1
B 2-4	B 2-4
B 2-5	B 2-5
B 2-6	B 2-6
3/4 1-1	3/4 1-1
3/4 1-19	3/4 1-19
3/4 2-14	3/4 2-14
3/4 3-26	3/4 3-26
3/4 6-3	3/4 6-3
3/4 6-21	3/4 6-21
3/4 6-26	3/4 6-26
3/4 8-7	3/4 8-7
3/4 9-6	3/4 9-6
3/4 11-14	3/4 11-14
B 3/4 2-2	B 3/4 2-2

ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

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 the DNB-SAFDL of 1.28 in conjunction with the Extended Statistical Combination of Uncertainties (ESCU). 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 conservative loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the DNB-SAFDL is not violated 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 2.1-1. 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.

Variable Power Level - High

A Reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions which are too rapid to be protected by a Pressurizer Pressure - High or Thermal Margin/Low Pressure Trip.

The Variable 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 15.0% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the safety analysis.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer 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 at less than or equal to 2375 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than the DNB-SAFDL of 1.28, in conjunction with ESCU methodology.

The trip is initiated whenever the Reactor Coolant System pressure signal drops below either 1900 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with 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.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time, measurement uncertainties and processing error. The allowances include: a variable (power dependent) allowance to compensate for potential power measurement error; an allowance to compensate for potential temperature measurement uncertainity; an allowance to compensate for pressure measurement error; and an allowance to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit.

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 sufficient time for any operator action to initiate auxiliary feedwater before reactor coolant system subcooling is lost. 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 position 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.

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. The trip is not credited in any design basis accident evaluated in UFSAR Chapter 15; however, the trip is considered in the safety analysis in that the presence of this trip function precluded the need for specific analyses of other events initiated from subcritical conditions.

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.

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 5000 pcm.

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

ACTION:

With the SHUTDOWN MARGIN less than 5000 pcm, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5000 pcm:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be untrippable, the above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODE 1 or MODE 2 with Keff greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

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* See Special Test Exception 3.10.1.

ACTION: (Continued)

e. With one full-length CEA misaligned from any other CEA in its group by more than 15 inches beyond the time constraints shown in Figure 3.1-1a, reduce power to ≤ 70% of RATED THERMAL POWER prior to completing ACTION e.1 or e.2.

1. Restore the CEA to OPERABLE status within its specified alignment requirements, or

2. Declare the CEA inoperable and satisfy SHUTDOWN MARGIN requirement of Specification 3.1.1.1. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:*

   a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.0 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours.

f. With one or more full-length CEA(s) misaligned from any other CEAs in its group by more than 7.0 inches but less than or equal to 15 inches, operation in MODES 1 and 2 may continue, provided that within 1 hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

   a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.0 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

   b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within the next 6 hours .

g. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, and inserted beyond the Long Term Steady State Insertion Limits but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

---

* If the pre-misalignment ASI was more negative than -0.15, reduce power to ≤ 70% of RATED THERMAL POWER or 70% of the THERMAL POWER level prior to the misalignment, whichever is less, prior to completing ACTION e.2.a) and e.2.b).

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits shown on Table 3.2-2.

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to ≤ 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-2 shall be verified to be within their limits by instrument readout at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement* at least once per 18 months.

* Not required to be performed until THERMAL POWER is ≥ 90% of RATED THERMAL POWER.

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
PROCESS MONITORS (Continued)					
c. Noble Gas Effluent Monitors					
i. Reactor Auxiliary Building Exhaust System (Plant Vent Low Range Monitor)	1	1, 2, 3 & 4	***	10^{-7} - 10^{-2} µCi/cc	27
ii. Reactor Auxiliary Building Exhaust System (Plant Vent High Range Monitor)	1	1, 2, 3 & 4	***	10^{-2} - 10^{5} µCi/cc	27
iii. Steam Generator Blowdown Treatment Facility Building Exhaust System	1	1, 2, 3 & 4	***	10^{-7} - 10^{-2} µCi/cc	27
iv. Steam Safety Valve Discharge#	1/steam header	1, 2, 3 & 4	***	10^{-1} - 10^{3} µCi/cc	27
v. Atmospheric Steam Dump Valve Discharge#	1/steam header	1, 2, 3 & 4	***	10^{-1} - 10^{3} µCi/cc	27
vi. ECCS Exhaust	1/train	1, 2, 3 & 4	***	10^{-7} - 10^{5} µCi/cc	27

*** The Alarm/Trip Setpoints are determined and set in accordance with the requirements of the Offsite Dose Calculation Manual.
# The steam safety valve discharge monitor and the atmospheric steam dump valve discharge monitor are the same monitor.

Pages 3/4 6-4 through 3/4 6-8 have been DELETED.

Page 3/4 6-9 is the next valid page.

Pages 3/4 6-22 through 3/4 6-23 have been DELETED .

Page 3/4 6-24 is the next valid page.

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

9. Verifying the diesel generator's capability to:

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

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

   c) Be restored to its standby status.

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

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

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

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

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane shall be used for movement of fuel assemblies, with or without CEAs, and shall be OPERABLE with:

a. A minimum capacity of 2000 pounds, and

b. An overload cut off limit of less than or equal to 3000 pounds.

APPLICABILITY: During movement of fuel assemblies, with or without CEAs, 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 .

SURVEILLANCE REQUIREMENTS

4.9.6 The manipulator crane used for movement of fuel assemblies, with or without CEAs, 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 2000 pounds and demonstrating an automatic load cut off before the crane load exceeds 3000 pounds.

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 The concentration of oxygen in the waste gas decay tank shall be determined to be within the above limits by continuously monitoring the waste gases in the on service waste gas decay tank.

4.11.2.5.2 With the oxygen concentration in the on service waste gas decay tank greater than 2% by volume as determined by Specification 4.11.2.5.1, the concentration of hydrogen in the waste gas decay tank shall be determined to be within the above limits by gas partitioner sample at least once per 24 hours.

assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid.

An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The requirement that the measured value of T_q be multiplied by the calculated values of F_r and F_{xy} to determine F_r^T and F_{xy}^T is applicable only when F_r and F_{xy} are calculated with a non-full core power distribution analysis code. When monitoring a reactor core power distribution, F_r or F_{xy} with a full core power distribution analysis code the azimuthal tilt is explicitly accounted for as part of the radial power distribution used to calculate F_{xy} and F_r.

The Surveillance Requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}, F_r and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and safety analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of ≥ 1.28 in conjunction with ESCU methodology throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12-hour basis.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 152 AND 89

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

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By letter dated May 29, 1997, Florida Power and Light Company (the licensee) submitted a request for changes to the St. Lucie Plant, Units 1 and 2 Technical Specifications (TS). The requested changes are intended to improve consistency throughout the TS and their related Bases by removing outdated material, incorporating minor changes in text, making editorial corrections, and resolving other inconsistencies identified by the licensee.

2.0 DISCUSSION AND EVALUATION

2.1 St. Lucie Unit 1

2.1.1 Changes to the following Unit 1 TS were proposed to correct typographical errors.

(1) On page VII, in the SECTION column of this index, the first section number was revised from "3/7.7.2" to read "3/4.7.2."

(2) Specification 3.1.1.3, BORON DILUTION, the improper spelling "concentation" was corrected to read "concentration."

(3) On page 3/4 2-9, in the listing of amendment numbers at the bottom of the page, superseded Amendment No. "~~63~~" was changed to read, "~~65~~." This typographical error was introduced when the page was re-typed for Amendment 150.

(4) On page 3/4 6-3, the text stating "Pages 3/4 6-3 through ... have been deleted" was changed to read "Pages 3/4 6-4 through ... have been deleted." This typographical error was introduced in Amendment 149. Page 3/4 6-3 has not been deleted from the TS.

(5) On page 3/4 6-20, the text stating "Pages 3/4 6-20 through ... have been deleted" was changed to read "Pages 3/4 6-21 through ... have been deleted." This error was introduced in Amendment 149. Page 3/4 6-20 has not been deleted from the TS.

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(6) On page 5-4, under the heading FUEL ASSEMBLIES, the second paragraph number "5.3.2" was changed to "5.3.1.1." This error was introduced in Amendment 44.

The staff reviewed these proposed changes and confirmed they were corrections to typographical errors. Therefore, these changes are acceptable.

2.1.2 Changes to the following TS were proposed to delete outdated references.

(1) Specification 3.10.2, SPECIAL TEST EXCEPTIONS, and in the associated ACTION statement, reference to Specification "3.1.3.2" is deleted. Specification 3.1.3.2, PART LENGTH CEA INSERTION LIMITS, was removed from the TS by Amendment 27.

(2) Specification 3.11.2.5 , EXPLOSIVE GAS MIXTURE, Surveillance 4.11.2.5.1, was changed to delete reference to Table 3.3-13 of Specification 3.3.3.10. Specification 3.3.3.10, EXPLOSIVE GAS MONITORING SYSTEM, and associated Table 3.3-13 were deleted from the TS by Amendment 147.

The staff reviewed these proposed changes and confirmed they were administrative changes which deleted outdated references. Therefore, these changes are acceptable.

2.1.3 Changes to the following TS were proposed for purpose of clarification and consistency with other TS.

(1) Specification 3/4.1.1, BORATION CONTROL, SHUTDOWN MARGIN-$T_{avg} > 200$ °F, Surveillance 4.1.1.1.1.a, was revised to clarify the conditions of immovable and untrippable consistent with these same conditions addressed in ACTION 3.1.3.1.a of Specification 3/4.1.3, MOVABLE CONTROL ASSEMBLIES/FULL LENGTH CEA POSITION, and makes clear that the specified shutdown margin (SDM) need not be increased by an additional amount corresponding to the withdrawn worth of the affected control element assembly (CEA)(s) if the affected CEA(s) is fully inserted. The proposed changes are administrative changes that make the TS consistent with other TS and clarify the conditions of the ACTION statement and are acceptable.

(2) Specification 3/4.1.2, BORATION SYSTEMS, page 3/4 1-8, FLOW PATHS-SHUTDOWN and page 3/4 1-12, CHARGING PUMPS-SHUTDOWN, condition (b) of Footnote * was revised to clarify that the additional restrictions on charging pumps, heatup and cooldown rates, and high pressure safety injection (HPSI) header isolation valves specified in the footnote are only required if reactor coolant system (RCS) pressure boundary integrity exists. The proposed change is an administrative change that clarifies the conditions under which the footnote applies, and is acceptable.

(3) Specification 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, FULL LENGTH CEA POSITION, ACTION 3.1.3.1.d was revised to place the requirement to be in at least hot standby within 6 hours if the action statement is not met

at the end of the statement. Repositioning the text removes an ambiguity from the existing construction in that the required unit transition to hot standby applies to failure to complete ACTION d as a whole rather than only failure to complete part d.2.b. In addition, adding the word "next" makes clear that the 6 hour action completion time begins upon failure to satisfy ACTION 3.1.3.1.d. The proposed change is an administrative change that clarifies when and under what conditions the 6 hour action requirement applies and is acceptable.

(4) Specification 3/4.1.3, MOVABLE CONTROL ASSEMBLIES, FULL LENGTH CEA POSITION, ACTION 3.1.3.1.f was revised to add the requirement to be in at least hot standby within 6 hours if the action statement is not met at the end of the statement. Adding the proposed statement ensures that, upon failure to comply with ACTION 3.1.3.1.f, actions will be taken to place the unit in a mode for which the limiting condition for operation (LCO) no longer applies, and without a need to invoke Specification 3.0.3. The action and completion time are consistent with the requirements of NUREG-1432, Revision 1, "Standard Technical Specifications Combustion Engineering Plants," and with the other action requirements of this specification. Therefore, this change is acceptable.

(5) Specification 3/4.4.9, PRESSURE/TEMPERATURE LIMITS, REACTOR COOLANT SYSTEM, Footnote * was revised to clarify that the limit on heatup and cooldown rates per Figure 3.1-1b apply only for the condition in which a flow path from the refueling water tank (RWT) to the RCS via an HPSI pump is established and pressure boundary integrity exists. If pressure boundary integrity does not exist as defined in TS 1.16, then Figure 3.1-1b does not apply. The revision to the footnote conforms with the requirements provided in TS 3.1.2.1 and 3.1.2.3 for using the referenced Figure 3.1-1b, removes the existing ambiguity and improves consistency within the facility TS. Therefore, this change is acceptable.

(6) Specification 3/4.6.5, VACUUM RELIEF VALVES, was revised to remove the words "less than or equal to" in conjunction with "± 0.25," which created an ambiguous syntax. Florida Power and Light Company (FPL) reviewed the vacuum relief valve setpoint calculations, and confirmed that 2.25 ± 0.25 inches Water Gauge differential meets the nominal design values assumed in the analysis for inadvertent containment spray actuation. The proposed revision will remove the erroneous phrase introduced in Amendment 90, and make the Specification consistent with the associated Surveillance 4.6.5.1. Since the vacuum relief valve setpoint meets the analyzed value, and the change removes the ambiguous syntax, this change is acceptable.

(7) Specification 3.9.4.c, REFUELING OPERATIONS, CONTAINMENT PENETRATIONS, was revised to remove reference to Table 3.6-2, CONTAINMENT ISOLATION VALVES, of Specification 3.6.3.1, which was deleted from the facility TS in accordance with Generic Letter (GL) 91-08 by Amendment 149. The referenced Table 3.6-2 contained a provision (footnote *) for opening certain normally closed valves on an intermittent basis under administrative control. It was recognized in GL 91-08 that the list of

containment isolation valves typically includes footnotes that modify the TS requirements for these valves and also address operational considerations for specific valves that may be opened on an intermittent basis under administrative control. GL 91-08 states, "Such notes must be incorporated into the associated LCO so that they will remain in effect when the table containing these footnotes is removed from the TS." The proposed revision to TS 3.9.4 deletes the outdated reference to Table 3.6-2, incorporates the associated provision for intermittent operation of valves under administrative control that was contained in footnote * to that table, and is consistent with GL 91-08. Therefore, this change is acceptable.

(8) Specification 3.9.6, REFUELING OPERATIONS, MANIPULATOR CRANE OPERABILITY, was revised to delete the statement that Specification 3.0.3 is not applicable and clarifies the manipulator crane overload cut off limit value. The ACTION statement reference to Specification 3.0.3 is outdated and redundant, since TS 3.9.6 is only applicable during refueling operations and Specification 3.0.3, by definition, is not applicable in this mode. TS 3.9.6 requires the manipulator crane overload cut off limit to be ≤ 3000 pounds. The existing Surveillance requires a demonstration of the load cut off "when" the crane load exceeds 3000 pounds. The Surveillance was changed to require a demonstration of the load cut off before the load exceeds 3000 pounds. The proposed revision makes Surveillance 4.9.6 consistent with the LCO and is acceptable.

The staff has reviewed these proposed changes and confirmed that the purpose of these changes were for clarification and consistency with other TS. These changes were found to be acceptable.

2.1.4 The following revisions will update the TS Bases as indicated.

(1) On page B 2-6, the bases for Steam Generator Water Level-Low was changed to delete the reference to providing a margin of more than 10 minutes before auxiliary feedwater is required to provide sufficient time for operator action to initiate auxiliary feedwater before reactor coolant system subcooling is lost. The revision more accurately reflects the acceptance criteria associated with this reactor trip setpoint which requires that the reactor coolant remain subcooled for the limiting loss of feedwater transients.

(2) On page B 2-8, the bases for Rate of Change of Power-High was changed to clarify that the Rate of Change of Power-High trip is considered in the safety analysis in that the presence of this trip function precluded the need for specific analyses of other events initiated from subcritical conditions. The revision more accurately describes the relation between this reactor protection system trip function and the plant safety analyses.

(3) On page B 3/4 1-3, the bases for 3/4.1.2 BORATION SYSTEMS was changed to clarify that the restrictions do not apply when RCS pressure boundary integrity does not exist as defined in Specification 1.16. The revision

provides additional clarification in support of the footnotes associated with TS 3.1.2.1 and 3.1.2.3 and is consistent with the St. Lucie Unit 1 low temperature over pressure (LTOP) analysis. In addition, LTOP events with either the pressurizer manway cover or the reactor vessel head removed are not credible.

(4) On page B 3/4 9-1, the bases for 3/4.9.4 CONTAINMENT PENETRATIONS, was changed to add the considerations to be included under the administrative controls required by the TS to open, on an intermittent basis, locked or sealed closed containment isolation valves.

(5) On page B 3/4 4-15, the bases for 3/4.4.13 POWER OPERATED RELIEF VALVES and 3/4.4.14 REACTOR COOLANT PUMP-STARTING, reference to "10 CFR Appendix G" is revised to read, "10 CFR 50, Appendix G."

The staff has reviewed these changes to the TS bases and finds them to be acceptable.

2.2 St. Lucie Unit 2

2.2.1 Changes to the following Unit 2 TS were proposed to correct typographical and minor administrative errors.

(1) Surveillance 4.1.1.2.e.9, the word "that" is deleted to correct a grammatical error.

(2) On page 3/4 6-3, the text stating "Pages 3/4 6-3 through ... have been deleted" is changed to read "Pages 3/4 6-4 through ... have been deleted." This typographical error was introduced in Amendment 88. Page 3/4 6-3 has not been deleted from the TS.

(3) On page 3/4 6-21, the text stating "Pages 3/4 6-21 through ... have been deleted" is changed to read "Pages 3/4 6-22 through ... have been deleted." This error was introduced in Amendment 88. Page 3/4 6-21 has not been deleted from the TS.

The staff reviewed these proposed changes and confirmed they were corrections to typographical errors. Therefore, these changes are acceptable.

2.2.2 Changes to the following TS were proposed to make the TS consistent with previously approved amendments.

(1) On page XVIII, section "6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP," and associated subsections is deleted and the subsequent section is renumbered to read, " 6.2.3 SHIFT TECHNICAL ADVISOR." The indicated sections were changed by Amendment 69.

(2) On page XIX, section 6.5.2, "COMPANY NUCLEAR REVIEW BOARD," was changed to add subsection "TECHNICAL REVIEW RESPONSIBILITIES .... 6-12." This subsection was added to the TS by Amendment 69.

(3) Specification 3.11.2.5, EXPLOSIVE GAS MIXTURE, Surveillance 4.11.2.5.1, was changed to delete reference to Table 3.3-13 of Specification 3.3.3.10.

Specification 3.3.3.10, EXPLOSIVE GAS MONITORING SYSTEM, and associated Table 3.3-13 were deleted from the TS by Amendment 86.

The staff reviewed these proposed changes and confirmed they were administrative changes which deleted outdated references and improved consistency within the TS. Therefore, these changes are acceptable.

2.2.3 The following minor changes were proposed for the purpose of clarification, and to assure consistent implementation of the stated requirements.

(1) On page 3/4 3-26, TABLE 3.3-6, RADIATION MONITORING INSTRUMENTATION, the required minimum channels operable for process monitor c.vi, ECCS Exhaust, was changed from "1" to "1/Train." The emergency core cooling system (ECCS) area ventilation system is designed with two independent trains and the revision will minimize the potential for an unmonitored noble gas effluent release. The revision is consistent with the format shown for the corresponding monitors in the Unit 1 TS and is a more restrictive requirement. Therefore, the staff finds this change acceptable.

(2) Specification 3/4.1.1, BORATION CONTROL, SHUTDOWN MARGIN-$T_{avg} > 200$ °F, Surveillance 4.1.1.1.1.a, was revised to clarify the conditions of immovable and untrippable consistent with these same conditions addressed in ACTION 3.1.3.1.a of Specification 3/4.1.3, MOVABLE CONTROL ASSEMBLIES CEA POSITION, and makes clear that the specified SDM need not be increased by an additional amount corresponding to the withdrawn worth of the affected CEA(s) if the affected CEA(s) is fully inserted. The proposed changes are administrative changes that make the TS consistent with other TS and clarified the conditions of the ACTION statement and are acceptable.

(3) Specification 3/4.1.3, MOVABLE CONTROL ASSEMBLIES CEA POSITION, ACTION 3.1.3.1.e was revised to add the requirement to be in at least hot standby within 6 hours if the action statement is not met at the end of the statement. Adding the proposed statement ensures that, upon failure to comply with ACTION 3.1.3.1.e, actions will be taken to place the unit in a mode for which the LCO no longer applies, and without a need to invoke Specification 3.0.3. ACTION 3.1.3.1.f was revised to add the word "next" to the statement to be in at least hot standby within 6 hours. Adding the word "next" makes clear that the 6 hour action completion time begins upon failure to satisfy ACTION 3.1.3.1.e. The proposed change is an administrative change that clarifies when and under what conditions the 6 hour action requirement applies. The action and completion time are consistent with the requirements of NUREG-1432, Revision 1, and with the other action requirements of this specification. Therefore, this change is acceptable.

(4) Specification 3/4.2.5, DNB PARAMETERS, Surveillance 4.2.5.2, was revised to add an asterisk to refer to a footnote which clarifies that the surveillance is not required until thermal power is greater than or equal to 90% of rated thermal power. The proposed footnote is necessary to allow measurement of the flow rate at normal operating conditions in Mode 1 since the surveillance cannot be performed in Mode 2 or below, and will not yield accurate results if performed below 90% of rated thermal power. This change clarifies the conditions under which the surveillance is performed, is consistent with corresponding Surveillance 3.4.1.4 of NUREG-1432, Revision 1, and is acceptable.

(5) Specification 3/4.6.5, VACUUM RELIEF VALVES, was revised to remove the words "less than or equal to" in conjunction with "± 0.25," which created an ambiguous syntax. FPL reviewed the vacuum relief valve setpoint calculations, and confirmed that 9.85 ± 0.35 inches Water Gauge differential meets the nominal design values assumed in the analysis for inadvertent containment spray actuation. The proposed revision will remove the erroneous phrase introduced in Amendment 60. Since the vacuum relief valve setpoint meets the analyzed value, and the change removes the ambiguous syntax, this change is acceptable.

(6) Specification 3.9.6, REFUELING OPERATIONS, MANIPULATOR CRANE, was revised to clarify the manipulator crane overload cut off limit value. TS 3.9.6 requires the manipulator crane overload cut off limit to be ≤ 3000 pounds. The existing Surveillance requires a demonstration of the load cut off "when" the crane load exceeds 3000 pounds. The proposed revision makes Surveillance 4.9.6 consistent with the LCO.

The staff has reviewed these proposed changes and confirmed that the purpose of these changes were for clarification and consistency with other TS. These changes were found to be acceptable.

2.2.4 The following revisions will update the TS Bases as indicated.

(1) On pages B 2-1, 2.1.1 REACTOR CORE, B 2-4, Thermal Margin/Low Pressure, and B 3/4 2-2, 3/4.2.5 DNB PARAMETERS, the bases were changed to revise the numerical value of the Departure from Nucleate Boiling - Specified Acceptable Fuel Design Limits (DNB-SAFDL) from 1.20 to 1.28. The revision updates the reference Departure from Nucleate Boiling Ratio (DNBR) limit value that is currently used in the Extended Statistical Combination of Uncertainties (ESCU) cycle specific setpoint analyses for St. Lucie, Unit 2. The reference DNBR limit value of 1.28 in conjunction with ESCU corresponds to a 95% probability at a 95% confidence level that DNB will not occur during steady state operation, normal operational transients, and anticipated transients. Also, the bases for Thermal Margin/Low Pressure was changed to delete the numerical values associated with the safety margin allowances. The Thermal Margin/Low Pressure (TM/LP) reactor trip setpoints are verified as part of the cycle-specific setpoint analyses performed for each fuel reload, which is evaluated under 10 CFR 50.59. Individual parameter values may be adjusted, within limits, to optimize analytical and operating margins provided by the setpoints. Removing the numerical

values associated with the equipment response time measurement uncertainties and processing errors will preclude the need for unnecessary Bases revisions as a result of minor adjustments that can be made to such values for reload core designs.

(2) On page B 2-5, the bases for Steam Generator Water Level-Low was changed to delete the reference to provide a margin of more than 10 minutes before auxiliary feedwater is required to provide sufficient time for operator action to initiate auxiliary feedwater before reactor coolant system subcooling is lost. The revision more accurately reflects the acceptance criteria associated with this reactor trip setpoint which requires that the reactor coolant remain subcooled for the limiting loss of feedwater transients.

(3) On page B 2-6, the bases for Rate of Change of Power-High was changed to clarify that the Rate of Change of Power-High trip is considered in the safety analysis in that the presence of this trip function precluded the need for specific analyses of other events initiated from subcritical conditions. The revision more accurately describes the relation between this reactor protection system trip function and the plant safety analyses.

The staff has reviewed these changes to the TS bases and finds them acceptable.

3.0 STATE CONSULTATION

Based upon the written notice of the proposed amendments, the Florida State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has 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 the amendments involve no significant hazards consideration and there has been no public comment on such finding (62 FR 40849). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: L. Wiens

Date: September 22, 1997