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

January 25, 2001

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

SUBJECT: ST. LUCIE UNIT 1 - ISSUANCE OF AMENDMENT REGARDING CYCLE 17 RELOAD (TAC NO. MA9531)

Dear Mr. Plunkett:

The Commission has issued the enclosed Amendment No. 171 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. This amendment consists of a revision to the Technical Specification (TS) in response to your application dated July 19, 2000. This amendment revises the license: (1) to implement Siemens Power Corporation (SPC) high thermal performance fuel assembly design in Cycle 17, (2) relocate shutdown margin requirements in Modes 1 to 5 to the Core Operating Limits Report (COLR), (3) update the COLR methodologies listed in the TS Section 6.9.1.11, and (4) request relief from the SPC fuel assembly reconstitution restrictions for peripheral low power fuel assemblies. Additionally, administrative changes are proposed to the boron concentration specifications related to the boration requirements.

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,

Kahtan N. Jabbour

Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-335

Enclosures:
1. Amendment No. 171 to DPR-67
2. Safety Evaluation

cc w/enclosures: See next page

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FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
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 July 19, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 171 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 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 25, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 171

TO FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
XV	XV
B 2-1	B 2-1
B 2-3	B 2-3
3/4 1-1	3/4 1-1
3/4 1-3	3/4 1-3
3/4 1-10	3/4 1-10
3/4 1-18	3/4 1-18
3/4 9-1	3/4 9-1
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 2-2	B 3/4 2-2
6-19	6-19
6-19b	6-19b
	6-19c

INDEX

ADMINISTRATIVE CONTROLS

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 measured parameter during operation and therefore THERMAL POWER, Reactor Coolant Temperature and Pressure have been related to DNB using a DNB correlation developed to predict the Critical Heat Flux (CHF) for DNB. The CHF is the heat flux at a particular core location that would cause DNB. The ratio of the CHF to the actual local heat flux at a particular core location is called the DNB Ratio (DNBR) and is indicative of the margin to DNB.

The minimum allowed value of the DNBR during steady state operation, normal operational transients, and anticipated transients is the DNBR limit from the appropriate DNB correlation. The DNBR limit corresponds to a 95% probability at a 95% confidence level that DNB will not occur at a particular core location, providing appropriate margin to DNB for all operating conditions. In a core with fuel assemblies of different designs (mixed core), there may be more than one DNB correlation and associated DNBR limit that defines DNB for the core.

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 DNBR limit corresponding to the Siemens Power Corporation (SPC) XNB DNB correlation is not violated for the following conditions:

1. reactor coolant inlet temperatures less than or equal to 580°F,
2. THERMAL POWER less than or equal to 112%,
3. reactor coolant vessel flow of 365,000 gpm, and
4. the axial power shape shown on Figure B2.1-1.

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.

SAFETY LIMITS

BASES

The reactor protective system in combination with the Limiting Conditions for Operation is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities. Specific verification of the DNBR limit with an appropriate DNB correlation ensures that the Reactor Core Safety Limit is satisfied.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings are designed to ANSI B 31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

3/4.1 REACTIVITY CONTROL SYSTEMS

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 within the limits specified in the COLR.

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

ACTION:

With the SHUTDOWN MARGIN not within limits immediately initiate and continue boration at ≥ 40 gpm of 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 within the COLR limits:

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.

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* See Special Test Exception 3.10.1.

\# With $K_{eff} \geq 1.0$.

\## With $K_{eff} < 1.0$.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - $T_{avg} \leq 200$ °F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

Within the limits specified in the COLR, and in addition with the Reactor Coolant System drained below the hot leg centerline, one charging pump shall be rendered inoperable.*

APPLICABILITY: MODE 5.

ACTION:

If the SHUTDOWN MARGIN requirements cannot be met, immediately initiate and continue boration at ≥ 40 gpm of greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN requirements of Specification 3.1.1.2 shall be determined:

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 immovable or 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. At least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2.b and by verifying at least one charging pump is rendered inoperable.*

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* Breaker racked-out.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

OR

At least two of the following three boron injection flow paths shall be OPERABLE:

a. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

b. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

c. The flow path from the refueling water storage tank, via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.2 at 200°F; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following four borated water sources shall be OPERABLE:

a. Boric Acid Makeup Tank 1A in accordance with Figure 3.1-1.

b. Boric Acid Makeup Tank 1B in accordance with Figure 3.1-1.

c. Boric Acid Makeup Tanks 1A and 1B with a minimum combined contained borated water volume in accordance with Figure 3.1-1.

d. The refueling water tank with:

1. A minimum contained volume of 401,800 gallons of water,

2. A minimum boron concentration of 1720 ppm,

3. A maximum solution temperature of 100°F,

4. A minimum solution temperature of 55°F when in MODES 1 and 2, and

5. A minimum solution temperature of 40°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to the requirements of Specification 3.1.1.2 at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 At least two borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water source,

3/4.9 REFUELING OPERATIONS

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 within the limit specified in the COLR.

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 greater than or equal to 1720 ppm boron or its equivalent to restore boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration limit 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.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg}, at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN as specified in the COLR for Specification 3.1.1.1 is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transient resulting from a boron dilution event with a partially drained Reactor Coolant System requires a SHUTDOWN MARGIN as specified in the COLR for Specification 3.1.1.2 and restrictions on charging pump operation to provide adequate protection. This SHUTDOWN MARGIN is 1000 pcm conservative for Mode 5 operation with total RCS volume present, however LCO 3.1.1.2 is written conservatively for simplicity.

3/4.1.1.3 BORATION DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 11,400 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limiting values of the MTC ensure that the assumptions for the MTC used in the accident and transient analyses remain valid through each fuel cycle. Determination of MTC at the specified conditions ensures that the maximum positive and/or negative values of the MTC will not exceed the limiting values.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions corresponding to the requirements of Specification 3.1.1.2 after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions. This requirement can be met for a range of boric acid concentrations in the Boric Acid Makeup Tanks (BAMTs) and Refueling Water Tank (RWT). This range is bounded by 5400 gallons of 3.5 weight percent (6119 ppm boron) boric acid from the BAMTs and 17,000 gallons of 1720 ppm borated water from the RWT to 8700 gallons of 2.5 weight percent (4371 ppm boron) boric acid from the BAMTs and 13,000 gallons of 1720 ppm borated water from the RWT. A minimum of 45,000 gallons of 1720 ppm boron is required from the RWT if it is to be used to borate the RCS alone.

The requirements for a minimum contained volume of 401,800 gallons of borated water in the refueling water tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the 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 $(1+T_q)$ be multiplied by the calculated value of F_r to determine F_r^T is applicable only when F_r is calculated with a non-full core power distribution analysis. 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_r.

The surveillance requirements for verifying that F_r^T and T_q are within their limits provide assurance that the actual values of F_r^T and T_q do not exceed the assumed values. Verifying 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 accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than or equal to the DNBR limit 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.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.11 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater Than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Full Length CEA Position – Misalignment > 15 inches
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.3	Total Integrated Radial Peaking Factor – F_r^T
Specification 3.2.5	DNB Parameters
Specification 3.9.1	Refueling Operations – Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the following documents or any approved Revisions and Supplements thereto:

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary)

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. XN-75-27(A) and Supplements 1 through 5, [also issued as XN-NF-75-27(A)], "Exxon Nuclear Neutronic(s) Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Inc. / Advanced Nuclear Fuels Corporation, Report and Supplement 1 dated April 1977, Supplement 2 dated December 1980, Supplement 3 dated September 1981 (P), Supplement 4 dated December 1986 (P), and Supplement 5 dated February 1987 (P)

4. ANF-84-73(P)(A) Revision 5, Appendix B, & Supplements 1 and 2, "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," Advanced Nuclear Fuels Corporation, October 1990

5. XN-NF-82-21(P)(A) Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Inc., September 1983

6. a) ANF-84-93(P)(A) and Supplement 1, [also issued as XN-NF-84-93(P)(A)], "Steamline Break Methodology for PWRs," Advanced Nuclear Fuels Corporation, March 1989

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

11. d) XN-NF-85-16(P)(A) Volume 1, and Supplements 1, 2 and 3; Volume 2, Revision 1 and Supplement 1, "PWR 17x17 Fuel Cooling Test Program," Advanced Nuclear Fuels Corporation, February 1990

    e) XN-NF-85-105(P)(A) and Supplement 1, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Advanced Nuclear Fuels Corporation, January 1990.

    f) EMF-2087(P)(A) Revision 0, "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," Siemens Power Corporation, June 1999.

12. XN-NF-82-06(P)(A) Revision 1, and Supplements 2, 4 and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, Inc., October 1986

13. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, December 1991

14. XN-NF-85-92 (P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, Inc., November 1986

15. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992

16. XN-NF-507(P)(A), Supplements 1 and 2, "ENC Setpoint Methodology for C. E. Reactors: Statistical Setpoint Methodology," Exxon Nuclear Company, Inc., September 1986

17. EMF-92-116(P)(A), Revision 0, "Generic Mechanial Design Criteria for PWR Fuel Design," Siemens Power Corporation, February 1999.

18. EMF-92-153(P)(A) and Supplement 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, March 1994.

19. EMF-96-029(P)(A) Volumes 1 and 2, "Reactor Analysis System for PWRs Volume 1 – Methodology Description, Volume 2 – Benchmarking Results," Siemens Power Corporation, January 1997.

20. EMF-1961(P), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, December 1998.

CORE OPERATING LIMITS REPORT (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 171TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

By letter dated July 19, 2000 (Ref. 1), Florida Power & Light Company (FPL) requested changes to the St. Lucie Unit 1 (SL1) Technical Specifications (TS) and the reload evaluation process to be implemented for SL1. The proposed amendment would revise the SL1 license to: (i) implement Siemens Power Corporation (SPC) high thermal performance (HTP) fuel assembly design in Cycle 17, (ii) relocate shutdown margin (SDM) requirements in Modes 1 to 5 to the Core Operating Limits Report (COLR), and (iii) update the COLR methodologies listed in TS 6.9.11. The proposed amendment also requests relief from the SPC fuel assembly reconstitution restrictions for peripheral low power fuel assemblies. TS surveillance requirements would also be changed to be consistent with the proposed licensing amendment. In addition, the licensee requested administrative changes to the boron concentration specifications related to TS 3.1.1.1, 3.1.1.2, and 3.9.1.

The staff's evaluation of the proposed updated methodology and the TS changes is presented below.

2.0 EVALUATION

2.1 Implementation of SPC HTP fuel assembly and affected TS.

The basis for the thermal limit lines are provided in Figure 2.1-1 of Reference 2. The text in the bases for the thermal limits lines is modified to reflect the departure from nucleate boiling (DNB) correlation used in generating these limits lines. Since SL1 could have both the non-vane fuel assemblies and the HTP fuel design in the core, the licensee chose the more conservative limit lines, which are based on the Siemen's XNB DNB correlation. The DNB correlation for the non-vane fuel, that is the XNB DNB correlation, is a more conservative fuel correlation because the DNB thermal margin lines are more limiting than those for the HTP fuel.

Typically, the DNB ratio (DNBR) is defined by the particular type of correlation used in the analysis. Specifically, the XNB correlation DNBR limit will be different from the HTP correlation DNBR limit. No specific value for the DNBR limit is stated in the basis of TS 2.1.1 because the licensee will use the appropriate U.S. Nuclear Regulatory Commission (NRC) approved DNB correlation and corresponding DNBR limit in their cycle specific analysis to ensure that the thermal margin DNBR limit is not violated for any of the anticipated combinations of transient conditions initiated within the limiting conditions for operation in combination with the reactor protection systems. Use of the appropriate NRC approved DNB correlation is specified in TS 6.9.1.11.

BASES 2.1.1 incorrectly refers to Table 2.1-1 as specifying the high power level trip setpoints. The correct table is Table 2.2-1. An administrative change would correct the Bases to refer to Table 2.2-1.

2.2 Relocation of the Shutdown Margin (SDM) to the COLR

2.2.1 TS 3/4.1.1.1: Shutdown Margin - T_{avg} Greater Than 200 °F (Mode 1 through 4)

The licensee proposed to relocate the shutdown margin limit in this TS to the COLR. Because the scram worth and power distribution may vary substantially from cycle-to-cycle, the shutdown margin requirement may also be cycle-dependent. Moving the shutdown margin limits to the COLR provides the flexibility to optimize the SDM requirements based on cycle specific fuel management and design considerations, such as scram worth, burnable absorber loadings, soluble boron level, etc. The proposed relocation to the COLR would obviate the need for license amendments due to cycle related changes to the SDM. Changes to the SDM limit would be evaluated in accordance with the provisions of Title 10, *Code of Federal Regulations* (10 CFR), Section 50.59. Since there is no change to the shutdown margin requirements due to this proposed amendment, and TS 6.9.1.11 requires core operating limits to meet all applicable limits of the safety analysis and to use analytical methods which have been reviewed and approved by the NRC, the proposed relocation is acceptable.

2.2.2 TS3/4.1.1.2: Shutdown Margin - T_{avg} Less Than Or Equal To 200 °F (Mode 5)

The licensee proposed to relocate the shutdown margin limit in this TS to the COLR. The shutdown margin requirement for T_{avg} less than or equal to 200 °F may also vary from cycle-to-cycle based on cycle-specific fuel management and design considerations. The proposed change would allow for accommodating cycle-to-cycle variations in shutdown margin requirements without the need for a license amendment. Changes to the SDM limit would be evaluated in accordance with the provisions of 10 CFR 50.59. Since there is no change to the shutdown margin requirements due to this proposed amendment, and TS 6.9.1.11 requires core operating limits to meet all applicable limits of the safety analysis and to use analytical methods which have been reviewed and approved by the NRC, the proposed relocation is acceptable.

2.2.3 Reactivity Control Systems and Boration Requirements

The reference to shutdown margin limits greater than or equal to 2000 pcm in the Action Statements of TSs 3.1.1.1, 3.1.1.2, and 3.1.2.2, would be changed to refer to the COLR limit. This change is necessary to be consistent with the change to relocate the SDM limit to the COLR and is, therefore, acceptable.

The proposed change to specify boration requirements to read "greater than or equal to 1720 ppm" is consistent with the requirements for borated water sources specified in TS 3.1.2.8. Therefore, this change is acceptable.

2.3 TS 6.9.1.11 Update: Core Operating Limits Report (COLR)

The shutdown margin specifications (TS 3.1.1.1 and 3.1.1.2) proposed for relocation to the COLR would be added to the list of COLR specification limits listed in TS 6.9.1.11.b. In addition, the list of NRC-approved analytical methods that can be used to determine the COLR parameters would be expanded to include reports recently approved by the NRC, including EMF-1961(P)(A), Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," which was approved by the NRC on July 12, 2000. The proposed changes are acceptable.

2.4 Fuel Assembly Reconstitution

St. Lucie Unit 1 has, in the past, experienced limited fuel failures of fuel assemblies located on the periphery of the core and adjacent to the guide tubes. Peripheral assemblies are typically low powered assemblies and, thus, not limiting from a safety point of view. When these peripheral fuel rods fail, they may be replaced by inert Zircaloy clad rods. The Zircaloy cladding houses stainless steel pellets. The licensee is proposing to replace failed fuel rods with a limited number (eight per assembly) of solid stainless rods instead of Zircaloy clad stainless steel pellets rods (Ref. 2). The proposed relief from the fuel assembly reconstitution restrictions would allow replacement of the fuel rods with inert solid steel rods near the guide tube locations, which were found to be the most susceptible to fretting failures.

The solid stainless steel rods will be manufactured from common materials already used in SPC fuel assemblies. SPC performed an evaluation for FPL and concluded that these solid steel rods are equivalent to the inert rods described in the topical report and are therefore acceptable for replacement of failed fuel rods. This evaluation included the assembly-specific safety, mechanical, and neutronic evaluations described in the NRC-approved reference report (ANF-90-082(P)(A)) methodology. Based on the SPC evaluation, the staff agrees with the proposed use of up to eight solid stainless steel rods per peripheral fuel assembly as substitutes for failed fuel rods.

2.5 Summary

The staff has reviewed the licensee's safety analyses to support the proposed TS changes for operation of Fuel Cycle 17 and future cycles at the SL1 plant. Based on this review, as described above, the staff concludes that the proposed TS changes and supporting safety analyses are acceptable, including the use of a maximum of eight (8) solid stainless steel replacement rods per peripheral fuel assembly as substitutes for failed fuel rods. The limit of 26 percent on the total number of inert replacement rods per assembly (including a maximum of eight solid stainless steel rods) will remain unchanged, consistent with the SPC NRC approved fuel reconstitution methodology.

3.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

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 (65 FR 48748). 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.

6.0 REFERENCES

1. Letter from Rajiv S. Kundalkar, Vice President, FPL, to NRC, Proposed License Amendment Cycle 17 for St. Lucie Unit 1, dated July 19, 2000.

2. Letter from Rajiv S. Kundalkar, Vice President, FPL, to NRC, Proposed License Amendment Cycle 17, Reload Supplement for St. Lucie Unit 1, dated November 6, 2000.

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Date: January 25, 2001

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