August 18, 1999

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 REACTOR COOLANT SYSTEM MINIMUM FLOW AND TECHNICAL SPECIFICATION UPDATE (TAC NO. MA4358)

Dear Mr. Plunkett:

The Commission has issued the enclosed Amendment No. 163 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated November 22, 1998.

This amendment revises the reactor thermal margin safety limit lines and flow rates stated in the St. Lucie, Unit 1, TS. The amendment also updates the reference for dose conversion factors used in Dose Equivalent Iodine-131 calculations, makes administrative changes to the criticality analysis uncertainty described in TS 5.6.1.a.1, updates the analytical methods used in determining core operating limits listed in TS 6.9.1.11, and revises the TS Bases for the steam generator pressure-low trip setpoint.

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:

William C. Gleaves, 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. 163 to DPR-67
2. Safety Evaluation

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

August 18, 1999

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

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William C. Gleaves, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
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Docket No. 50-335

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1. Amendment No. 163 to DPR-67
2. Safety Evaluation

cc w/encls: See next page

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. 163
License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company, et al., (the licensee), dated November 22, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 163, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Sheri R. Peterson, 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: August 18, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. **163**

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.

Remove Pages	Insert Pages
1-3	1-3
2-2	2-2
2-4	2-4
B 2-5	B 2-5
3/4 2-14	3/4 2-14
5-5	5-5
6-19	6-19
6-19a	6-19a
n/a	6-19b

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ICRP-30, Supplement to Part 1, pages 192-212, Tables entitled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity(Sv/Bq)."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES REPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

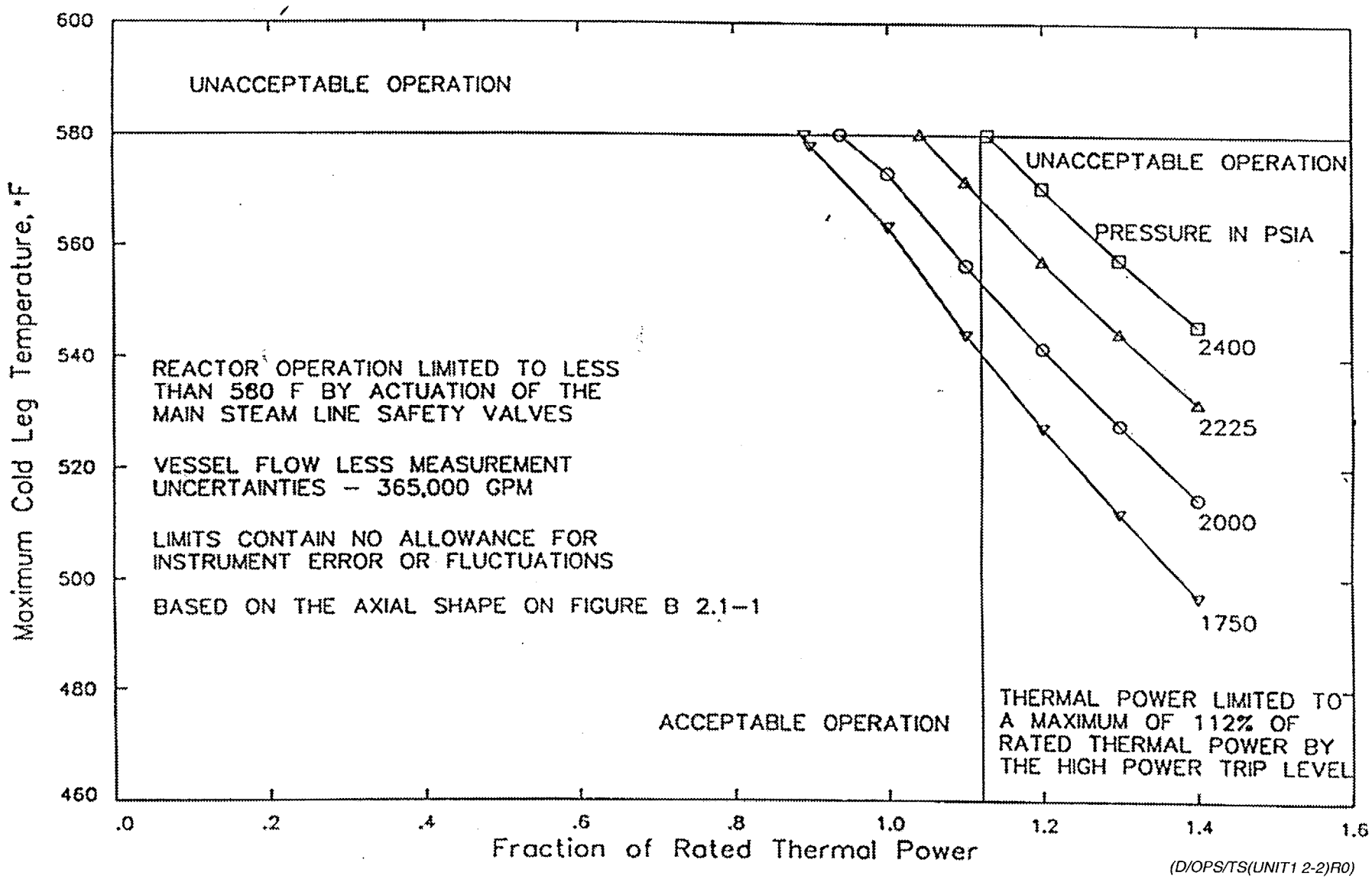

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level – High (1) Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of < 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow – Low (1) Four Reactor Coolant Pumps Operating	≥ 95% of design reactor coolant flow with 4 pumps operating *	≥ 95% of design reactor coolant flow with 4 pumps operating *
4. Pressurizer Pressure – High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure – High	≤ 3.3 psig	≤ 3.3 psig
6. Steam Generator Pressure – Low (2)	≥ 600 psia	≥ 600 psia
7. Steam Generator Water Level – Low	≥ 20.5% Water Level – each steam generator	≥ 19.5% Water Level – each steam generator
8. Local Power Density – High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

* Design reactor coolant flow with 4 pumps operating is 365,000 gpm.

Reactor Coolant Flow-Low (continued)

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

Pressurizer Pressure-High

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

Containment Pressure-High

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

Steam Generator Pressure-Low

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

TABLE 3.2-1

DNB MARGIN

LIMITS

Parameter	Four Reactor Coolant Pumps Operating
Cold Leg Temperature	≤ 549°F
Pressurizer Pressure	≥ 2225 psia *
Reactor Coolant Flow Rate	≥ 365,000 gpm
AXIAL SHAPE INDEX	COLR Figure 3.2-4

* Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and no part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the original design provisions contained in Section 4.2.3.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,100 $\pm$ 180 cubic feet at a nominal T_{avg} of 567°F, when not accounting for steam generator tube plugging.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.a The spent fuel storage racks are designed and shall be maintained with:

1. k_{eff} less than or equal to 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

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 3.1.1.4	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

CORE OPERATING LIMITS REPORT (continued)

6. b) EMF-84-093(P)(A) Revision 1, "Steam Line Break Methodology for PWRs," Siemens Power Corporation, February 1999 (This document is a Revision to ANF-84-93)

7. XN-75-32(P)(A) Supplements 1 through 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, Inc., October 1983.

8. Siemens Power Corporation Small Break LOCA methodology as defined by:

   a) XN-NF-82-49(P)(A) Revision 1, "Exxon Nuclear Company Evaluation Model EXEM PWR Small Break Model," Advanced Nuclear Fuels Corporation, April 1989

   b) XN-NF-82-49(P)(A) Revision 1 Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model," Siemens Power Corporation, December 1994

9. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, Inc., October 1983

10. XN-NF-621(P)(A) Revision 1," Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, Inc., September 1983

11. EXEM PWR Large Break LOCA Evaluation Model as defined by:

   a) 1. XN-NF-82-20(P)(A) Revision 1 Supplement 2, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Company, Inc., February 1985

      2. XN-NF-82-20(P)(A) Revision 1 and Supplements 1, 3 and 4, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates," Advanced Nuclear Fuels Corporation, January 1990.

   b) XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, " Exxon Nuclear Company, Inc., November 1982

   c) 1. XN-NF-81-58(P)(A) Revision 2, and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Inc., March 1984

      2. ANF-81-58(P)(A) Revision 2 Supplement 3, and Supplement 4, "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," Advanced Nuclear Fuels Corporation, June 1990

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

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. 163 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

In a letter dated November 22, 1998, Florida Power & Light Company (FPL) requested changes to the St. Lucie, Unit 1, Technical Specifications (TS) in order to recover the analysis margin lost as a result of changes to the Reactor Coolant System (RCS) flow and low-flow trip setpoint limits that were previously made to reflect steam generator (SG) tube plugging over more than 20 years of plant operation. The SGs were replaced during the 1997 refueling outage for the current operating cycle (Cycle 15) and resulted in an increase in RCS flow compared to previous values with SG tube plugging.

Specifically, the Thermal Margin Safety Limit Lines of TS Figure 2.1-1 would be revised to reflect an increase in the value of design minimum RCS flow from 345,000 gpm to 365,000 gpm. The flow rates stated in Tables 2.2-1 and 3.2-1 would be changed accordingly. The reactor protective instrumentation Reactor Coolant Flow-Low trip setpoint limits would be increased from 93 percent to 95 percent of design flow.

In addition, the following TS changes are also proposed. TS 5.6.1.a.1, "Design Features, Fuel Storage Criticality," would be revised to delete the numerical value of criticality analysis uncertainty and reference the value described in Section 9.1 of the Updated Final Safety Analysis Report (UFSAR). The acceptable analytical methods used for determining core operating limits listed in TS 6.9.1.11," Administrative Controls, Core Operating Limits Report (COLR)," would be updated. The Limiting Safety System Settings in Bases Section 2.2 for the Steam Generator Pressure-Low trip would be revised to delete the numerical value of steam pressure described as the full load operating point.

The specific change that the Nuclear Regulatory Commission (NRC) staff addressed was a proposed revision to TS 1.10 to change the reference for thyroid dose conversion factors used in Dose Equivalent Iodine-131 calculations from those listed in Table III of TID-14844 (Reference 1) to those listed in the International Commission on Radiological Protection (ICRP) Publication 30 (Reference 2).

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2.0 EVALUATION

2.1 TS Figure 2.1-1: Reactor Core Thermal Margin Safety Limit - Four Reactor Coolant Pumps Operating

The current Thermal Margin Safety Limit Lines and the notation of Figure 2.1-1 reflect a design minimum RCS flow of 345,000 gpm with 4 pumps operating based on tube plugging in the previous SGs which have been replaced. FPL has proposed to revise these to include the effects of the proposed design reactor coolant flow of 365,000 gpm with four pumps operating. The associated TS Bases Figure B2.1-1 remains unchanged.

Based on the RCS flow of 407,000 gpm measured during the current Cycle 15 with the replacement SGs, the expected flow with an assumed value of 15 percent SG tube plugging would be in excess of 380,000 gpm. Therefore, the proposed increase in minimum design flow to 365,000 gpm would provide sufficient margin to TS flow compliance with 15 percent SG tube plugging after accounting for flow measurement uncertainties. The new limits were obtained using approved methodology of the current fuel vendor, Siemens Power Corporation (SPC). FPL has evaluated the impact of the proposed change to RCS design flow on applicable plant analyses. The NRC staff has reviewed these evaluations, as discussed below, and has determined that the proposed increase in RCS design flow to 365,000 gpm would not adversely affect the safety analyses conclusions supporting operation of St. Lucie, Unit 1. Therefore, the proposed increase in RCS design flow is acceptable.

2.2 TS Table 2.2-1: Reactor Protective Instrumentation Trip Setpoint Limits and TS Table 3.2-1: DNB Margin Limits

FPL has proposed to revise Table 2.2-1 to change the Reactor Coolant Flow-Low trip setpoint and allowable value from 93 percent of design reactor coolant flow with four pumps operating to 95 percent of design reactor coolant flow with four pumps operating. The footnote would be changed to reflect the new design flow with four pumps operating of 365,000 gpm. In addition, Table 3.2-1 of TS 3.2.5 would be revised to change the Reactor Coolant Flow Rate from >345,000 gpm to >365,000 gpm.

When the RCS flow falls below the trip setpoint, an automatic reactor trip signal is initiated to ensure that, for a degradation of RCS flow resulting from expected transients, local power density or departure from nucleate boiling ratio (DNBR) safety limits are not violated. The increase in value of the Reactor Coolant Flow-Low trip setpoint has a beneficial impact for those transients which rely on this trip since the trip would occur at a higher core flow rate leading to an earlier reactor trip. Therefore, the proposed trip setpoint increase is acceptable.

2.3 TS 1.10: Dose Equivalent I-131

In the definition of Dose Equivalent I-131, the reference for the thyroid dose conversion factors is changed from, "Table III of TID-14844, 'Calculation of Distance Factors for Power and Test Reactor Sites.'" to "ICRP-30, Supplement to Part 1, pages 192-212, Tables entitled, 'Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity (Sv/Bq)'."

The licensee's justification for making use of ICRP-30 thyroid dose conversion factors in place of those from TID-14844 is that the values in ICRP-30 are more recent and incorporate the considerable advances in the state of knowledge of limits for intakes of radionuclides. Also, the ICRP-30 dose conversion factors are consistent with Federal Guidance Report 11 (Reference 3) and the recommendations of Environmental Protection Agency 1987 guidance. The staff has generally accepted the use of ICRP-30 dose conversion factors, and such use is consistent with current industry standards. The licensee stated that the analyses of record for dose consequences use thyroid dose conversion factors from TID-14844. Since TID-14844 thyroid dose conversion factors give conservative dose results as compared to those in ICRP-30, the current dose analyses of record remain bounding and no reanalysis is necessary. The licensee may and should use ICRP-30 thyroid dose conversion factors for future reanalyses. The staff finds the licensee's proposed change to TS 1.10 and the use of the ICRP-30 thyroid dose conversion factors reflect current industry standards and account for advances in the state of knowledge of limits for intakes of radionuclides. Therefore, the licensee's proposed change to TS 1.10 is acceptable.

2.4 TS 5.6: Fuel Storage

TS 5.6.1.a.1 states that the spent fuel storage racks are designed and shall be maintained with a k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 0.0065 Δk for uncertainties. FPL has proposed to delete the numerical value of 0.0065 Δk, in TS 5.6.1.a.1, for the uncertainty allowance and reference Section 9.1 of the UFSAR for a description of these uncertainties. The proposed TS 5.6 change affects a design feature. The change does not affect a safety limit, limiting safety system setting, or limiting control setting as defined in 10 CFR 50.36. Its removal from the TS does not affect a design feature, which, if altered or modified, would have a significant effect on safety.

The fuel storage criticality analysis uncertainties are dependent on the methods and assumptions used in the criticality analysis and are delineated in detail in Table 9.1-12 of the UFSAR. Any changes in these uncertainties due to revisions to the criticality analysis would be reflected in the UFSAR. This proposed change is also consistent with the Standard TS for Combustion Engineering Plants (NUREG-1432). Based on the previous evaluation, the proposed change is acceptable.

2.5 TS 6.9.1.11: Core Operating Limits Report (COLR)

FPL has proposed to update the analytical methods listed in TS 6.9.1.11.b. The analytical methods listed in TS 6.9.1.11.b are approved methodologies used in the safety analyses performed for determining the St. Lucie, Unit 1, core operating limits that are documented in the COLR. The proposed modifications to the list would make it consistent with proper identification of the fuel vendor (SPC) topical reports and would improve the clarity and accuracy of report identification. Therefore, these proposed modifications are merely administrative in nature and are acceptable. Two of the proposed references had not been approved by the NRC staff when these proposed modifications were submitted with this amendment request. One of the proposed references, EMF-84-093(P), Revision 1, subsequently has been approved by the NRC staff in a letter from F. Akstulewicz (NRC) to J. F. Mallay (SPC), "Acceptance for Referencing of Topical Report EMF-84-093(P), Revision 1,

'Steam Line Break Methodology for PWRs'," dated February 16, 1999, and the accepted version, EMF-84-093(P)(A), Revision 1, which incorporates the acceptance letter, the staff safety evaluation, and an "A" (designating accepted) after the report identification symbol, was published by SPC in February 1999. Another proposed reference, EMF-2087(P), Revision 0, dated August 1998, has recently been approved by the NRC staff in a letter from C. Carpenter (NRC) to J. F. Mallay (SPC), "Acceptance for Referencing of Topical Report EMF-2087(P), 'SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications'," dated June 15, 1999. Therefore, these two reports are acceptable references for inclusion in TS 6.9.1.11.b.

2.6 Bases 2.2: Limiting Safety System Setting

The Bases for "Steam Generator Pressure-Low" would be revised to delete the specific numerical value of 800 psig presently described as the "full-load operating point."

The present full-load operating point following replacement of the steam generators is approximately 850 psig. However, the numerical value of the full-load steam pressure in the Bases is merely for descriptive purposes and has no bearing on the design requirement to have this setpoint sufficiently below the full-load operating point so as not to interfere with normal operation. Therefore, the proposed change is editorial and therefore is acceptable.

2.7 Plant Safety Analyses

The current plant UFSAR Chapter 15 safety analyses of record use the design RCS flow of 345,000 gpm and the Reactor Coolant Flow-Low trip setpoint of 93 percent of design reactor coolant flow. SPC performed a safety evaluation of these analyses to assess the impact of an increase in the RCS flow to 365,000 gpm and an increase in the Reactor Coolant Flow-Low trip setpoint value to 95 percent of design reactor coolant flow.

For the design basis events, the increase in the design RCS flow rate has an insignificant effect on fuel centerline melt and shutdown margin and has a beneficial impact on DNBR considerations since it tends to increase the calculated DNBR. The proposed changes, therefore, would not adversely affect fuel failures and subsequent dose consequences for events that result in fuel failures. The proposed RCS flow changes were also found to have an insignificant impact on the pressure rise for those events that are analyzed for the overpressurization of primary and secondary systems. The decrease in core average temperature due to the increase in core flow rate would tend to increase the rate of depressurization in the small break loss of coolant accident (LOCA), resulting in a decrease in peak cladding temperature (PCT). Although the decreased temperature would cause a small change in the blowdown characteristics of a large break LOCA, the small effects of these changes would have an insignificant impact on the PCT. Therefore, the small break and large break LOCA analyses of record would remain bounding for the proposed changes.

The proposed flow increase would only impact the heat addition part of the low temperature overpressure protection (LTOP) analysis due to its effect on the heat transfer rate from the secondary side to the primary side of the steam generators. However, the proposed flow would remain within the range of currently allowed minimum and maximum RCS flows for the operation of the plant. Therefore, the LTOP analysis would remain bounded by the analysis of record for the proposed RCS flow increase. In addition, since an increase in RCS flow tends

to increase the resulting DNBR, the current DNB-Limiting Condition for Operation (LCO) and DNB-Limiting Safety System Settings (LSSS) setpoint analyses would remain bounding for the case with increased RCS flow and low flow trip setpoint values. The increases in design flow and low flow trip setpoint have no impact on the Local Power Density (LPD)-LCO and the LPD-LSSS.

In summary, the staff has reviewed the evaluation of the St. Lucie, Unit 1, Chapter 15 safety analysis to assess the impact of the proposed increase in the RCS flow rate to 365,000 gpm and the proposed increase in the Reactor Coolant Flow-Low trip setpoint value to 95 percent of design reactor coolant flow. Based on our review, we conclude that the proposed changes are acceptable and that reanalysis of the current plant safety analyses is not required.

3.0 STATE CONSULTATION

By letter dated March 8, 1991, Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, informed Deborah A. Miller, Licensing Assistant, U.S. NRC, that the State of Florida does not desire notification of issuance of license amendments. Thus, the 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 (64 FR 6696). 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.

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