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

July 9, 1996

SUBJECT: ST. LUCIE UNIT 1 - ISSUANCE OF AMENDMENT RE: THERMAL MARGIN AND REACTOR COOLANT SYSTEM FLOW LIMITS (TAC NO. M95472)

Dear Mr. Plunkett:

The Commission has issued the enclosed Amendment No.145 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. This amendment consist of changes to the Technical Specifications in response to your application dated June 1, 1996.

This amendment revises the St. Lucie Unit 1 Technical Specifications to reflect reduced Reactor Coolant system flows resulting from an increased percentage of plugged steam generator tubes.

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

Leonard A. Wiens, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-335

Enclosures:
1. Amendment No. 145 to DPR-67
2. Safety Evaluation

Distribution
Docket File
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PDII-3 Reading
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OGC
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D Verelli, RII
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DATE	6/26/96	6/27/96	7/8/96	7/9/96	
COPY	Yes/No	Yes/No	Yes/No	Yes/No	

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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. 145
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 June 1, 1996, 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. 145, 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: July 9, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 145

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
2-1	2-1
2-2	2-2
Fig. 2.1-1	Fig. 2.1-1
2-4	2-4
3/4 2-14	3/4 2-14
5-5	5-5

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

*For Cycle 14 operation beyond 7000 EFPH, THERMAL POWER shall not exceed 90% of 2700 Megawatts (thermal).

Amendment No. 145

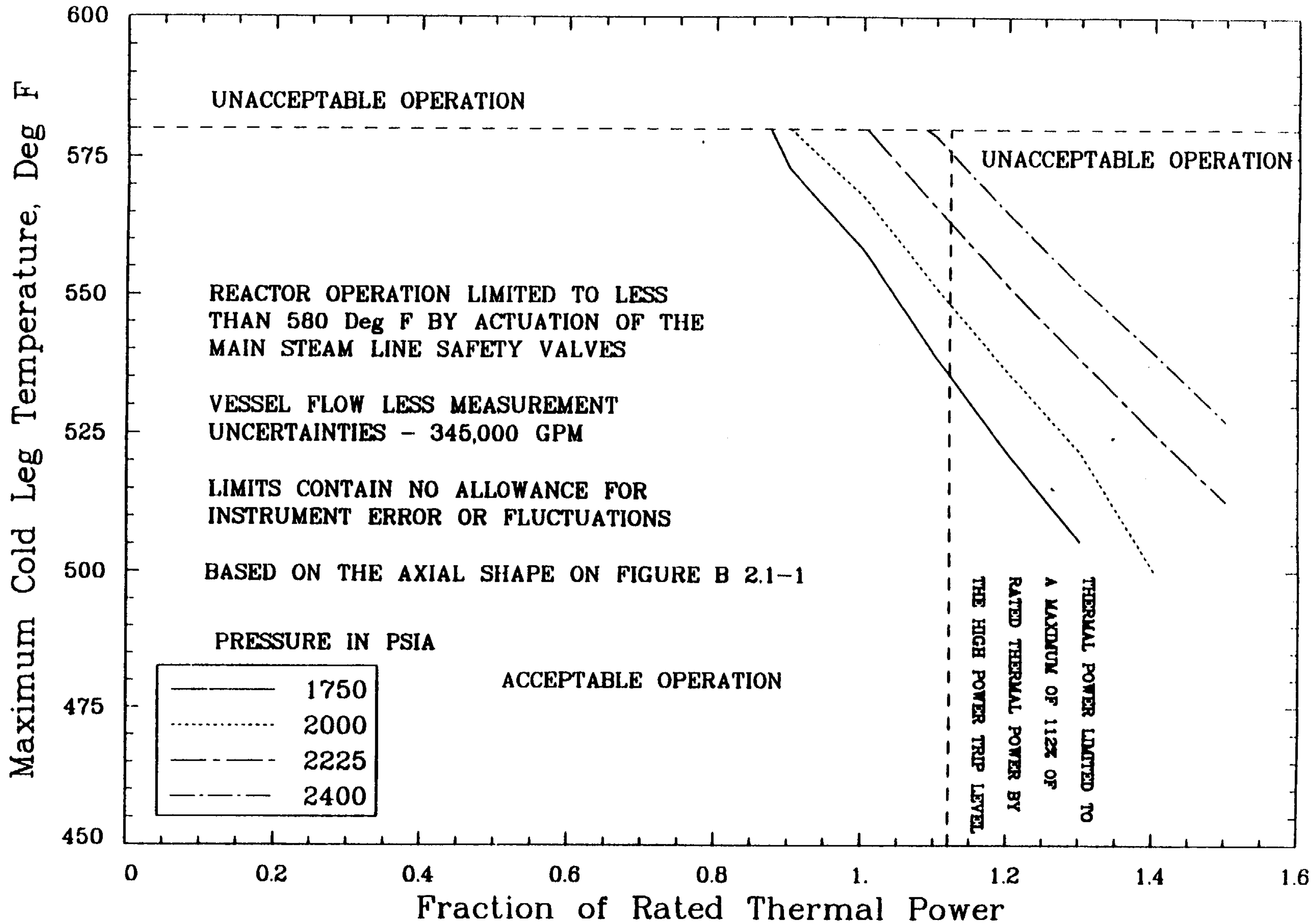

FIGURE 2.1-1: REACTOR CORE THERMAL MARGIN SAFETY LIMIT - FOUR REACTOR COOLING PUMPS OPERATING

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	≥ 93% of design reactor coolant flow with 4 pumps operating*	≥ 93% 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 345,000 gpm.

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	≥ 345,000 gpm
AXIAL SHAPE INDEX	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 ± 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. 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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

1.0 INTRODUCTION

By letter dated June 1, 1996 (Ref. 1), the Florida Power & Light Company (FPL) requested changes to the Technical Specifications (TSs) for the St. Lucie Unit 1 plant. Based on safety analyses assumptions of 30% (average) of all steam generator tubes removed from service, the amendment proposes the following changes:

(1) the design reactor coolant system (RCS) flow rate is reduced from 355,000 gpm to 345,000 gpm,

(2) the reactor core thermal margin safety limits shown in Figure 2.1-1 are revised,

(3) the reactor coolant system total water and steam volume described in the design features is modified,

(4) the Limiting Safety System Setting (LSSS) for the reactor coolant low flow trip function is reduced from 95% to 93% of design reactor coolant flow, and

(5) TS 2.1.1 is modified to limit reactor power to 90% rated thermal power for Cycle 14 operation exceeding mid-cycle fuel burn up conditions.

The proposed changes to the TSs and Bases are as follows:

a. Page 2-1, Specification 2.1.1, REACTOR CORE: an asterisk is inserted following THERMAL POWER, and also the following footnote is presented:

   * For Cycle 14 operation beyond 7000 EFPH, THERMAL POWER shall not exceed 90% of 2,700 Megawatts (thermal).

b. Page 2-2, FIGURE 2.2-1, Reactor Core Thermal Margin Safety Limit-Four Reactor Cooling Pumps Operating: This figure is replaced in its entirety with the revised FIGURE 2.1-1.

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The "vessel flow less measurement uncertainties" is changed from 355,000 gpm to 345,000 gpm. The thermal limit lines have been revised to reflect the reduced flow.

c. Page 2-4, TABLE 2.2-1, Reactor Protective Instrumentation Trip Setpoint Limits:

(1) For Reactor Coolant Flow-low FUNCTIONAL UNIT 3, the TRIP SETPOINT and ALLOWABLE VALUES has been changed from 95% of design reactor coolant flow with 4 pumps operating* to 93% of design reactor coolant flow with 4 pumps operating*.

(2) In Footnote *, the design reactor coolant flow with 4 pumps operating has been changed from 355,000 gpm to 345,000 gpm.

d. Page 3/4 2-14, TABLE 3.2-1, DNB MARGIN: the Reactor Coolant Flow Rate has been changed from 355,000 gpm to 345,000 gpm.

e. Page 5-5, DESIGN FEATURES, Specification 5.4.2: the description of the reactor coolant system VOLUME has been modified as an administrative change to read:

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

2.0 <u>BACKGROUND</u>

The current safety analyses for St. Lucie Unit 1 assume a minimum design RCS flow rate of 355,000 gpm and an average 25% (± 7%) of all steam generator tubes plugged (SGTP). It is estimated that in the current refueling outage the number of steam generator tubes that will be removed from service (currently in excess of 2,000) will likely exceed the 25% (average) limit. To conservatively accommodate the larger number of plugged SG tubes, FPL proposed to change the TS to reflect the safety analysis assumption of 345,000 gpm minimum RCS design flow rate (based on 30% average of all steam generator tubes plugged), and proposed a change in the Reactor Protective System RCS Low Flow Limiting Safety System Setting from 95% to 93% of design reactor coolant flow.

The licensee has stated that the proposed changes affect the plant safety analyses in the following manner.

a. A reduction in RCS flow rate has an adverse effect on the calculated Departure from Nucleate Boiling Ratio (DNBR). DNBR is a direct indication of available thermal margin, and a reduction in the calculated minimum DNBR indicates that thermal margin for the corresponding transient has been reduced.

b. A reduction in the value of the low flow trip setpoint will result in a lower reactor core flow rate at the time of reactor trip, and can thereby impact the calculated minimum DNBR for certain transients.

c. A reduction in RCS flow rate results in a corresponding increase in RCS average coolant temperature (T_{ave}). A higher T_{ave} can impact both DNBR-related and loss of primary inventory types of transients.

d. The removal of additional steam generator tubes from service (plugging) reduces the primary to secondary heat transfer area in the steam generators. This effect is most relevant to transients involving a sudden reduction in the heat removal capability of the secondary plant. In addition, a reduction of initial RCS inventory due to significant steam generator tube plugging (SGTP) can affect the results of boron dilution events, as well as the depth of core uncovery and calculated peak containment pressure resulting from loss of coolant accidents.

3.0 EVALUATION

The events in the St. Lucie Unit 1 Chapter 15 Updated Final Safety Analysis Report (UFSAR) were reviewed for Cycle 14 by FPL and Siemens Power Corporation-Nuclear Division (SPC) to assess the impact of an increase in SGTP to 30% ± 7%, a reduced minimum Technical Specification RCS flow of 345,000 gpm, and a reduced reactor coolant flow trip setpoint of 93 ±3 % of design flow (345,000 gpm). The licensee indicated that NRC approved computer codes (References 2, 3 and 4) were used for the new supporting safety analyses. The events identified that required reanalysis are: Loss of External Load (15.2.1), Loss of Normal Feedwater (15.2.7), Loss of Forced Reactor Coolant Flow (15.3.1), CEA Misoperation (Dropped CEA Only) (15.4.3), Decrease of Boron Concentration (15.4.6), and Small Break LOCA (15.6.5). All other events are either bounded by another event in the same category or are bounded by existing analyses of record.

3.1 UFSAR Chapter 15 events were reviewed in the following categories.

a. Decrease in Secondary Side Heat Removal (15.2)

(1) Loss of External Load (LOEL) (15.2.7)

LOEL was identified as the limiting transient within this event category and reanalyzed to examine the impact of the proposed changes on the calculated maximum primary and secondary pressures. Results of the reanalysis for this event indicated the calculated peak primary pressure to be 2,714 psia, below the limiting criteria of 110% of design pressure (2,750 psia). Secondary system pressure was determined to be 1,031 psia, which is less than the 1,100 psia secondary side acceptance criteria.

Therefore, it is concluded that increased SGTP and the associated reduction in RCS flow, has no adverse impact.

b. Decrease in Reactor Coolant System Flow Rate

Events within this category of transients are initiated by a malfunction of the Reactor Coolant Pumps (RCP) with the resultant decrease in coolant flow causing a degradation of the calculated DNBR (closer to the limit of 1.22). Two events in this category are impacted by the proposed reduction in design RCS flow and low flow trip setpoint: Loss of Reactor Coolant Flow (LOF) and Seized RCP Rotor.

(1) Loss of Reactor Coolant Flow (15.2.5)

The Loss of Flow (LOF) transient was evaluated with the initial conditions modified to include the proposed changes. The objective of this evaluation was to determine whether the existing DNB-LCO (TS 3/4.2.5), in conjunction with the RPS Low Flow Trip, will prevent the DNBR limit of 1.22 from being violated. Results of this evaluation show a reduction in the minimum power margin from 6.8% to 1.9 % of rated power. The available margin confirms that the minimum DNBR is greater than its limit value of 1.22. Or, equivalently, the LOF event initiated within the existing DNB LCO constraints will not result in violation of the Specified Acceptable Fuel Design Limit (SAFDL) for DNBR. Therefore, we find this to be acceptable.

(2) Seized RCP Rotor (15.3.4)

The seized rotor accident is assumed to be initiated by an instantaneous seizure of one of the reactor coolant pump shafts. The margin available in this analysis, due to excess conservatism in the reactor power and Radial Peaking Factor (Fr), has been determined to nearly offset the effects of the decreased coolant flow and reduced low flow trip setpoint, resulting in a net power penalty of 0.57%. The small decrease in DNBR associated with the 0.57% power penalty will not cause the fuel rod failures to increase from the present value of 1% to more than the 2.5% value used in the radiological analysis. Therefore, it is concluded that the impact of increased SGTP, reduced RCS flow, and reduced low flow trip setpoint on the fuel failure rate resulting from the Seized RCP Rotor accident is acceptable since the radiological consequence of this accident is bounded by the current analysis.

c. Reactivity and Power Distribution Anomalies

The events in this category are not impacted by the change in low flow trip setpoint except that the dropped CEA transient requires evaluation due to the reduced RCS flow.

(1) Dropped CEA (15.2.3)

The result of the evaluation performed, after accounting for the proposed changes, show a reduction in the minimum power margin from 8.0% to 4.6% of rated power. Based on the available margin, it is concluded that the occurrence of a CEA drop event, after implementation of the proposed changes, will not result in violation of the DNBR SAFDL, provided the transient is initiated within the constraints of the DNB-LCO. We therefore find this acceptable.

(2) Uncontrolled CEA Withdrawal (15.2.1)

Both the uncontrolled CEA withdrawal from low power and the CEA withdrawal initiated from high power conditions are events analyzed against DNBR criteria. The proposed reduction in RCS flow is expected to affect the DNB-related events in a similar manner. Therefore, the CEA withdrawal event will continue to remain bounded by the Loss of Flow (LOF) transient. Since the LOF analysis results were found to be acceptable, it is concluded that the uncontrolled CEA withdrawal will not result in violation of the DNBR SAFDL, when initiated from within the DNB-LCO.

(3) Boron Dilution Event (15.2.4)

Protection against violation of SAFDL's for boron dilution events initiated at power is provided by the existing Thermal Margin/Low Pressure (TM/LP) trip, the Variable High Power Trip (VHPT) and the Local Power Density (LPD) LSSS.

Increased SG tube plugging will result in a small change in RCS fluid volume (~1.28%). This in turn will impact the time to criticality determined in the boron dilution event analyses. The reference analyses for dilution events initiated from hot standby or hot/cold shutdown conditions at St. Lucie Unit 1, show that margin exists to the acceptance criteria in the time to criticality. Since Mode 6 only considers the mass inventory in the reactor vessel, the increase in SGTP does not affect Mode 6.

For Modes 2 to 4, the decrease in the RCS inventory was calculated to reduce the time to criticality from 72.02 minutes to 71.1 minutes. This time is greater than the acceptance criteria of 15 minutes. The time to criticality for Mode 5 is reduced from 20.54 minutes to 20.3 minutes, relative to the criteria of 15 minutes. The boron dilution event results are, therefore, acceptable for the proposed changes.

(4) CEA Ejection Accidents (15.4.5)

A control rod ejection accident is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a CEA and its drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion and an adverse core power distribution, which may result in localized fuel damage.

In Section 15.4.5 of the existing UFSAR licensing design basis, predictions of fuel failure are based on fuel centerline melt criteria (deposited energy in the fuel rod), not on DNBR criteria. Therefore, a reduction in RCS flow proposed here, will not significantly impact the results of this event with respect to core damage or offsite radiological dose consequences.

d. Decrease in Reactor Coolant Inventory Events

(1) Large Break LOCA (LBLOCA) (15.4.1)

This event was evaluated to account for the impact of the proposed changes on the peak cladding temperature. A maximum rescinder density of 1.1% is used in the licensing analysis compared to the as-built rescinder density for Cycle 14 of 0.81%. The use of an as-built rescinder density is estimated to result in a reduction in initial fuel average temperature of 34°F for the case of fuel stored energy near the beginning-of-cycle (BOC) and at least 16°F for the case of fuel stored energy representing the middle-of-cycle (MOC). This amount of conservatism in the fuel stored energy represents a significant conservatism in PCT, and is acceptable as the analysis of record would continue to remain bounding and meet the requirements of 10 CFR 50.46(b).

(2) Small Break LOCA (SBLOCA) (15.3.1)

The Small Break LOCA event was evaluated for the impact of the reduced primary system flow and the increased SG tube plugging level (30% average). The results of this event are influenced more by changes to the top-peaked axial profiles. A review of Cycle 14 axial profiles showed that the maximum peak power elevation corresponding to the MOC was lower than that used in the analysis of record. The average burn up of the MOC axial profiles was 7000 EFPH. The conservatism in the analysis due to this axial profile, up to this burn up, will offset any adverse effects due to the increased tube plugging and decreased RCS flow.

To operate beyond 7000 EFPH, the licensee conducted an evaluation to determine at what power they could operate to satisfy the SBLOCA criteria. Since SBLOCA is sensitive to core power, a reduction in power will affect peak fuel cladding temperatures during a SBLOCA, thus ensuring 10 CFR 50.46 conformity beyond 7000 EFPH for cycle 14. The licensee is proposing to reduce power from 100% to 90% of rated power beyond 7000 EFPH. This will provide sufficient margin to offset any adverse effects of the proposed changes. The staff agrees with the derating of power for the St. Lucie's Cycle 14 operation beyond 7000 EFPH.

(3) Inadvertent Opening of Pressurizer Pressure Relief Valves (15.2.12)

This event is bounded by the Loss of Flow event. Since the proposed changes will not affect the relative behavior of DNBR between the two transients, this event will continue to remain bounded by the Loss of Flow event.

The inadvertent PORV opening is also one of the transients used in the determination of the limiting pressure bias term in the TM/LP equation. This bias term is dependent on the maximum rate of change of DNBR experienced during the event, which for this case, is directly dependent on the rate of depressurization. Since the proposed changes do not affect the depressurization rate in this transient, it is concluded that there is no impact on the existing TM/LP pressure bias.

(4) Steam Generator Tube Rupture (SGTR) (15.4.4)

The existing analyses have concluded that the associated radiological release is primarily dependent on the break flow rate and the corresponding primary-to-secondary mass transfer during the event. The radiological releases were determined to be a small fraction of 10 CFR 100 limits. The differential pressure across the steam generator tubes determines if flow through the break is choked or not. The existing analysis or record examined the bounding case where break flow was choked before reactor scram. After reactor scram, the transient response is governed by the opening of steam dump and bypass valves.

The postulated increase in SGTP level will result in reduced secondary side operating pressure at St. Lucie Unit 1. This change could result in slightly longer times of choked flow for an actual SGTR. However, the analysis of record assumes choked flow conditions during the period of interest before reactor scram, and therefore will remain bounding. It is, therefore, concluded that the proposed changes will not alter the system response and the resultant potential offsite dose consequences for the SGTR event.

e. Increase in Heat Removal by the Secondary System (15.2.11)

Events in this category are evaluated by calculating the increase in primary system cooling due to the particular event initiator.

(1) Excess Load (15.2.11)

In the UFSAR three events with different initiators are postulated for the excess load event: 1) malfunction of the generator load limiter, 2) opening of the steam dump and bypass valves at power due to turbine trip permissive failure and 3) opening of the steam dump and bypass valves at hot standby due to controller malfunction. The limiting sub-event is the inadvertent opening of all the steam dump and bypass system valves at full power. This scenario would cause an approximate 43.4% increase in steam mass flow rate, resulting in a decrease in reactor coolant temperature and pressure. Under these conditions a negative moderator temperature coefficient of reactivity will cause an increase in core power. The High Power Level and TM/LP trips provide primary protection to prevent exceeding the DNBR limit during the full power excess load event.

In Section 15.2.11.3 of the UFSAR design licensing basis, this excess load event has been determined to be bounded by the Loss of Coolant Flow event for DNB considerations and none of the proposed changes will significantly impact the relative DNBR behavior in these two transients. Therefore, no reanalysis of this event was required.

(2) Steam System Piping Failures (Inside/Outside Containment) (15.4.6)

Steam System Piping Failure events are analyzed to ensure that any fuel failures which might occur are limited to a small percentage of the fuel in the core. These analyses are used to determine whether fuel failures would result from violation of either the DNBR or fuel centerline melt SAFDL's.

The primary system cooldown following a limiting steam system piping failure initiated with increased steam generator tube plugging and reduced RCS flow will be bounded by (no more severe than) the existing analysis. The reduced primary to secondary heat transfer rate across the steam generator and the lower initial secondary pressure both contribute to make this a more benign event. These effects ensure that the existing analysis of record for steam system piping failures will remain bounding and potential off-site dose consequences remain unchanged.

(3) Inadvertent Opening of a Steam Generator Relief (Atmospheric Dump) Valve (15.2.11)

This event is normally evaluated to assess radiological consequences. Radiological releases caused by this event will be less severe and less likely to occur after implementation of the proposed changes because of the lower initial secondary side pressure resulting from the increased steam generator tube plugging level. The analysis of record assumes conservative Technical Specification limits for the primary to secondary leak which remains unchanged. Therefore, the existing analysis of record will remain bounding for this event.

3.2 Impact of the Proposed Changes on Relevant Setpoint Analyses (15.6.5)

The impact of the proposed changes on relevant setpoint analyses was evaluated and verified to be acceptable. The setpoint analyses included the Reactor Protection System (RPS) Local Power Density (LPD) LSSS, LPD Limiting Condition for Operation (LCO), TM/LP LSSS, and the DNB LCO for allowable core power as a function of Axial Shape Index (ASI).

3.3 Other selected UFSAR analyses that were evaluated included the following.

a. Plant Natural Circulation Capability (Appendix 5C)

FPL examined the increased tube plugging to determine if any adverse impact on natural circulation cooling capability would result. FPL determined that the cooldown rate was dominated by operation of the secondary safety valves, and that increased SGTP had no adverse impact. We have reviewed the licensee's evaluation and agree with their assessment and have concluded that the proposed changes will not prevent the occurrence of natural circulation.

b. Peak Containment Pressurization Following LBLOCA or Steam System Piping Failure (6.2)

Large Break LOCA and Steam Pipe Break inside of Containment analyses of record were evaluated by FPL to determine if the reduced RCS flow and/or increased tube plugging level would cause the containment design pressure value to be exceeded.

For the LBLOCA event inside containment, the reduction in primary system fluid volume available for blowdown, a higher resistance to blowdown, and less secondary to primary heat transfer completely offset the effects from a slight increase in system energy due to the higher initial RCS T_{ave}. The peak pressure in the analysis of record remains bounding.

Steam Piping Failures inside containment were also examined and it was concluded that, after allowing for the proposed changes, no compromise of the pressure limits on containment analysis would result. Increased tube plugging will result in a small increase in the total secondary side mass inventory, but the overall energy stored in the fluid (and eventually released to containment during this event) is not increased. In addition, the lower initial secondary pressure will allow less blowdown (from the intact SG) prior to Main Steam Isolation Signal (MSIS).

c. Auxiliary Feedwater System (AFW) High Energy Line Break (10.5.3)

The analysis for this event was evaluated with respect to the increased average primary coolant temperature. It was determined that an additional 637 lbm of inventory would be boiled off from the secondary side reducing the dryout time from 650 seconds to 611.7 seconds. No credit was taken for the increased initial secondary side mass inventory (due to an increase in water density from a decrease in secondary side temperature). From the analysis it was concluded that the acceptance criteria of more than 10 minutes (600 seconds) time for operator action to initiate AFW flow to avoid dryout is satisfied with increased steam generator tube plugging and reduced RCS flow.

d. Low Temperature Overpressure Protection (LTOP) Analysis (Appendix 5Bi, 5.2.2.6)

The existing LTOP analysis was evaluated to determine whether the postulated increase in steam generator tube plugging would impact the consequences of starting a RCP with the plant secondary side at a higher temperature than the primary.

Only a change in the RCP heat output or in the initial condition of primary to secondary ΔT could change the energy deposited in the primary system, and hence, the peak pressure. Therefore, increasing the steam generator average tube plugging to 30% has no adverse impact on the pressure spike caused by starting a RCP pump under low temperature conditions.

e. Overpressure Protection Analysis (Appendix 5A)

The impact of the proposed increase in steam generator tube plugging and reduced RCS flow on the licensing analysis for the Loss of External Load event was previously discussed (Section 3.1.a). Since that analysis confirmed compliance with the pressurization criteria, it indirectly verified the continued validity of the main steam safety valve sizing analysis of reference. Therefore, it is concluded that the proposed changes do not require an increase in main steam safety valve capacity to satisfy the overpressurization criteria.

f. Impact on Steam Generator Mechanical Loads (5.5)

The steam generator inlet temperature corresponding to 345,000 gpm RCS design flow (with 30% tube plugging) is calculated to be less than the acceptable value of 604°F. The temperature value of 604°F was supported by the previous SG mechanical load calculations performed for the 25% ±7% asymmetry tube plugging case. Therefore, there is no adverse impact on any acceptance criteria for the tube sheet and steam generator tube bundle, and sufficient margin to stress limits will remain available.

3.4 Summary

FPL has performed the relevant UFSAR Chapter 15 safety analyses for the requested changes of reduced RCS flow and increased SGTP. These changes have been found to be acceptable including the derate to 90% rated thermal power for operation beyond 7000 EFPH in Cycle 14.

The reactor core thermal margin safety limits given in TS Figure 2.1-1 have been adjusted to account for the proposed value of design flow, and define the areas of safe operation in terms of thermal power, RCS pressure, and cold leg temperature for which the DNBR is no less than the MDNBR limit. The minimum DNBR limit for steady state operation, normal operational transients, and anticipated transients remains unchanged from the existing, approved value of 1.22.

The validity of Reactor Protective Instrumentation settings and trip functions in conjunction with related Limiting Conditions for Operation has been verified to provide assurance that reactor core design limits are not exceeded for the proposed change in RCS design flow.

The potential radiological consequences determined in the analyses of record, and which demonstrate compliance with 10 CFR 100 acceptance criteria, remain bounding for operation with the reduced RCS flow and increased SGTP.

4.0 STATE CONSULTATION

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

5.0 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 (61 FR 29140). 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.

6.0 CONCLUSIONS

Based on the evaluation in Section 3.0 above, the staff concludes that the licensee's proposed revision to the Technical Specifications for the St. Lucie Unit 1 plant to allow a reduction in the required minimum RCS flow rate and RCS Low-flow rate trip are acceptable.

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

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7.0 REFERENCES

1. Letter from W. H. Bohlke, Florida Power & Light Company (FPL), to USNRC, June 1, 1996.

2. XN-NF-81-58(A), Revision 2, and Supplements 1 through, "RODEX fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Revision 2 and Supplements 1 and 2 dated March 1984, Revision 2, Supplements 3 and 4 dated June 1990.

3. XN-NF-74-5(A) Supplements 1 through 6 and Revision 2, "Description of the Exxon Nuclear Plant Transient Simulation Model Pressurized Water Reactors (PTS-PWR)," Exxon Nuclear Company, October 1986.

4. XN-NF-75-21(A) Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation," Exxon Nuclear Company, January 1986.