Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

February 28, 2003

L-2003-042
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: St. Lucie Unit 2
Docket No. 50-389
Proposed License Amendment
Reduce the Minimum Reactor Coolant System Flow

By letter L-2002-196 dated October 15, 2002, Florida Power & Light Company (FPL) requested to amend Facility Operating License NPF-16 for St. Lucie Unit 2. FPL proposed to modify the St. Lucie Unit 2 Technical Specification (TS) Table 3.2-2 and a footnote to Table 2.2-1 to reduce the design reactor coolant system (RCS) flow rate from 363,000 gpm to 355,000 gpm. The proposed flow reduction was intended to accommodate an increase in the steam generator tube plugging (SGTP) level to a maximum of 1250 tubes per steam generator (SG). The current safety analyses support a SGTP level of 1250 tubes/SG (~15%). The analyses, however, are based on the current TS RCS flow limit of 363,000 gpm. The flow rate corresponding to 15% SG tube plugging level is estimated to be lower than 363,000 gpm, when considering the effects of flow measurement uncertainties.

In a January 10, 2003 conference call, FPL and the NRC staff discussed staff requests for additional information. On January 30, 2003, the NRC issued a request for additional information (RAI) with the FPL response estimated by March 3, 2003. Attachment 1 provides the FPL response to the RAI. Attachment 2 provides a replacement marked up TS page 2-5 and a revised proposed smooth TS page 2-5.

In accordance with 10 CFR 50.91 (b)(1), a copy of the proposed amendment supplement is being forwarded to the State Designee for the State of Florida.

Approval of the proposed license amendment remains requested by April 2003 to support potential steam generator tube plugging during the spring 2003 St. Lucie Unit 2 refueling outage (SL2-14). Please issue the amendment to be effective on the date of

A001

issuance and to be implemented prior to startup for Cycle 14. Please contact George Madden at 772-467-7155 if there are any questions about this submittal.

Very truly yours,

Donald E. Jernigan
Vice President
St. Lucie Plant

DEJ/GRM

Attachments

cc: Mr. William A. Passetti, Florida Department of Health

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

Donald E. Jernigan being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute the document on behalf of said Licensee.

Donald E. Jernigan

STATE OF FLORIDA

COUNTY OF ST LUCIE

Sworn to and subscribed before me

this 28 day of Feb., 2003
by Donald E. Jernigan, who is personally known to me.

Leslie J. Whitwell
Name of Notary Public - State of Florida

Leslie J. Whitwell
MY COMMISSION # DD020212 EXPIRES
May 12, 2005
BONDED THRU TROY FAIN INSURANCE, INC.

(Print, type or stamp Commissioned Name of Notary Public)

St. Lucie Unit 2
Response to the Request for Additional Information
Reduction in Minimum Reactor Coolant System Flow Amendment

Request 1:

In Attachment 4, on pages 2, 4, and 5 specify replacement of "DNB-SAFDL of 1.28" with "appropriate correlation limit for DNB-SAFDL." Given that there are two departure from Nucleate Boiling (DNB) correlations that can be used, how do you choose which limit to use and where is that detailed?

Response 1:

The correlation limits for the two DNB correlations, namely CE-1 and ABB-NV, which define the minimum DNBR value not to be violated during steady state operation, normal operational transients and anticipated operational transients, as approved by the NRC are 1.19 and 1.13, respectively. Both these correlations are approved for use for St. Lucie Unit 2 fuel and are included in the COLR methodology listed in Technical Specification 6.9.1.11, items 16 & 17 and item 55, respectively. The design limits in conjunction with Extended Statistical Combination of Uncertainties (ESCU) methodology (CEN-348(B)-P-A, Supplement-1-P-A, COLR methodology item no. 42), as applied to the reload analysis, are 1.28 for CE-1 DNB correlation and 1.21 for ABB-NV DNB correlation. The decision to use a particular DNB correlation is made on a case by case basis, typically as part of the reload analysis. The decision is usually made based on the need for additional margin and plant configuration changes such as increased steam generator tube plugging. The use of the appropriate correlation limit is controlled through the reload analysis process and assures compliance to the correlation topical report safety evaluation reports (SER) and the NRC approved application methodology SERs.

Request 1 a:

Is it true that there is now only GUARDIAN fuel in the core (Section 3.1 Attachment 6 Page 3)?

Response 1 a:

GUARDIAN fuel was first introduced in the St. Lucie Unit 2 core for Cycle 11. All fresh fuel loaded in the core since Cycle 11 is of GUARDIAN design. The current operating Cycle 13 has mainly GUARDIAN fuel in the core, except for nine low power burned fuel assemblies of non-GUARDIAN design. Cycle 14 will contain all GUARDIAN fuel except for one non-GUARDIAN center assembly.

Request 1 b:

Section 2 of Attachment 6 on page 2 states that the "conclusions of this evaluation remain applicable to all fuel designs currently used in the St. Lucie Unit 2 core." Do you need to use two different correlations?

Response 1 b:

No, it is not required to use two different DNB correlations for the fuel designs used at St. Lucie Unit 2. However, both the CE-1 and ABB-NV DNB correlations are applicable to and are approved for use for the fuel designs used at St. Lucie Unit 2. The GUARDIAN fuel design differs from the non-GUARDIAN fuel only in the design of the bottom grid, which is below the active fuel. The grids in the active fuel region are all essentially the same for the two fuel designs. As such, either of the two DNB correlations, CE-1 and ABB-NV, is acceptable for GUARDIAN and non-GUARDIAN fuel designs, and these correlations have been previously approved by the NRC for use in the St. Lucie Unit 2 analysis.

Request 1 c:

Are you using CE-1 and ABB-NV or McBeth? Are these approved for GUARDIAN fuel?

Response 1 c:

Although ABB-NV is approved for use, as listed in COLR methodology item 55 of Technical Specification 6.9.1.11, St. Lucie Unit 2 reload analysis currently uses the CE-1 correlation for DNB calculations, except for the post-trip steam line break analysis, which uses MacBeth correlation as documented and approved in Response 1 References 1 and 2. To gain additional DNB margin, as needed based on plant configuration changes such as increased steam generator tube plugging, FPL will transition to the ABB-NV DNB correlation for reload analysis.

References

1. Letter from J. A. Stall (FPL) to NRC, "Proposed License Amendment Cycle 12 Reload Process Improvement," L-98-308, December 18, 1998.

2. NRC Safety Evaluation Report, "St. Lucie, Unit 2 - Correction to Withdrawal of Amendment Request and Issuance of Revised Bases (TAC No. MA5619) and Correction to Safety Evaluation Issued as Part of a License Amendment Regarding the Cycle 12 Reload Process Improvement (TAC NO. MA4523)," March 1, 2000. (NRC approval of Reference 1 above)

Request 2:

Section 2.0 of Attachment 6 on page 2 states that the flow reduction results in "...small coolant and fuel rod temperature increases..." Table 3.1-1 of Attachment 6 on page 4 includes the maximum clad surface temperature. Please explain why this did not increase with reduced flow.

Response 2:

Although the average film temperature difference and average core enthalpy rise presented in Table 3.1-1 of Response 2 Reference 1 show small increases due to the proposed RCS flow reduction, the maximum clad surface temperature does not. The maximum clad surface temperature is a calculated value based on the Jens-Lottes correlation (Response 2 Reference 2); and is comprised of the coolant saturation temperature corresponding to the operating pressure, plus a small temperature difference between the coolant and the clad surface which is a function of the operating pressure and heat flux. The reduction in RCS flow did not change the operating pressure or the heat flux, therefore this parameter is unchanged.

<u>References</u>

1) Letter from D. E. Jernigan (FPL) to NRC, "St. Lucie Unit 2 Docket No. 50-389 Proposed License Amendment Reduce the Minimum Reactor Coolant System Flow," L-2002-196, October 15, 2002.
2) Jens, W. H. and Lottes, P. A., Argonne National Laboratory Report, ANL-4627, May 1, 1951.

Request 3:

Please clarify the meaning of the plot in Attachment 4 on page 3. Is this the axial and radial peak distribution or only the axial power distribution?

Response 3:

The curves in Attachment 4 of Response 3 Reference 1, on page 3, denote the axial power distributions with corresponding radial peaks (shown on the curves) for each of the two axial power shapes. These curves are representative axial power distributions used in the generation of the Technical Specifications limit lines of Figure 2.1-1, as described in the bases to Technical Specification 2.1.1.

<u>Reference</u>

1. Letter from D. E. Jernigan (FPL) to NRC, "St. Lucie Unit 2 Docket No. 50-389 Proposed License Amendment Reduce the Minimum Reactor Coolant System Flow," L-2002-196, October 15, 2002.

Request 4:

Demonstrate that the appropriate criteria will be met for all non-loss-of-coolant accident events. Please quantify results in your demonstration. Include a statement of the methodology used and whether there were any deviations from the methodology and treatment of uncertainties as previously approved by the staff. Please confirm that any changes in the use of the methodology continues to comply with any staff requirements for its use.

Response 4:

1.0 Information on Non-LOCA Safety Analysis Methodology

The discussion in Sections 1.1 and 1.2 below identifies where non-LOCA methods are described within the Updated Final Safety Analysis Report (UFSAR) and identifies the existing topical reports. This information is unaffected by the proposed license amendment for reduced RCS flow. The evaluation of reduced RCS flow relied on the UFSAR analyses and identified certain event specific analysis sensitivities and conservatisms to trade off against the 8000 gpm decrease in minimum reactor vessel flow. In a few cases as required, limiting transient analysis cases were repeated explicitly modeling the reduced RCS flow but employing the same codes and approved methods as the existing UFSAR. Reference numbers in this Response 4 refer to the References list in Response 4 Section 5.0.

For reduced RCS flow, all events were reevaluated or reanalyzed to assure that they meet their respective criterion as further discussed below, except for the Partial Loss of Forced Reactor Coolant Flow (UFSAR Section 15.3.2.2.5), CVCS Malfunction (Inadvertent Boron Dilution) (UFSAR Section 15.4.2.3.9), and the Inadvertent Operation of the ECCS During Power Operation (UFSAR Section 15.5.3.2.3) as the current analyses of these events are not impacted by the RCS flow reduction or bounded by other Chapter 15 design basis events (DBE).

1.1 Methods of Analysis as Described in UFSAR Section 15.0.1.10.2

The Design Basis Events considered in the safety analyses are listed in UFSAR Table 15.0.12a. These events are categorized into two groups: Moderate Frequency Events (referred to herein as Anticipated Operational Occurrences (AOO)) and Postulated Accidents. The DBEs were evaluated with respect to four criteria: Offsite Dose, Reactor Coolant System Pressure, Fuel Performance and Loss of Shutdown Margin. UFSAR Tables 15.0-13a, 15.0.14a, 15.0.15a, and 15.0.16a present the list of events analyzed for each criterion. All events had been previously reanalyzed or reevaluated in Reference 3, and supplemented in Reference 15, for a bounding reload analysis to assure that they meet their respective criterion at a reactor thermal power rating of 2700 MWt. The DBEs chosen for analysis for each criterion are the limiting events with respect to that criterion. In addition, the CVCS Malfunction Event (UFSAR Section 15.5.3.2.2) has been analyzed to ensure that the operator has sufficient time to terminate the event before the pressurizer fills solid. These criteria and the analysis methodology for all the events remain unchanged for the case of proposed RCS flow reduction.

1.2 Prior NRC Review and Approval of Methods of Analysis

UFSAR Section 15.0.1.10.2 summarizes the event categorization, methodology, and acceptance criteria submitted to the NRC in References 2 and 15 and approved in the References 4 and 16, respectively. Page 5 of the Reference 4 SER provided the following safety evaluation.

> "The non-LOCA safety analysis was performed using core physics and plant parameters that are anticipated to be bounding values for future cycles. The DBEs were categorized into moderate frequency events, or AOOs, and postulated accidents. The DBEs were evaluated with respect to one or more of the following criteria: offsite dose, reactor coolant system (RCS) pressure, fuel performance, and loss of shutdown margin. The DBEs chosen for analysis for each criterion are the limiting events with respect to that criterion."

This prior NRC review included each of these events and concluded that appropriate criteria were applied to each event in the Reference 2 submittal. The following was stated on page 4 of the Reference 4 NRC SER concerning the analysis codes and methods used. (Note that reference numbers have been changed to match the references of Section 5.0 below.)

> "The analytical methods used for the transient analyses to support the reload application and TS changes are normally reviewed on a generic basis. The methods for the transient analyses include the following computer codes:
>
> CESEC: The CESEC code (Reference [5]) provides a simulation of the system response and calculates system parameters such as core power, RCS flow, primary and secondary temperatures and pressures during a transient. The code was previously approved by the NRC for licensing applications.
>
> TORC and CETOP-D: TORC (References [6] and [7]) is used to simulate three-dimension fluid conditions within the reactor core. Results from TORC, including the core radial distribution of the relative channel axial flow, are used to calibrate CETOP-D (Reference [7]). With transient core heat flux and thermal-hydraulic conditions from CESEC as input, CETOP-D calculates the DNBR using approved critical heat flux correlations. Both TORC and CETOP-D have been reviewed and approved by the NRC for use in the design basis analysis for licensing applications.

STRIKIN-II: The STRIKIN-II code is used in the analyses of the CEA ejection event (Reference [8]). This code calculates the clad and fuel temperatures for an average or hot fuel rod. STRIKIN-II was previously approved by the NRC for the licensing calculations.

HERMITE: The HERMITE code (Reference [9]) is used to determine the reactor core response during the total loss of reactor coolant pump (RCP) flow event. The application of HERMITE to simulate the total loss of RCP flow event was previously approved by the NRC for similar CE plants. HERMITE/TORC is used to calculate the negative reactivity feedback during the post-trip steamline break (SLB) conditions. The application of HERMITE/TORC for the negative reactivity calculations is consistent with the method used in the original SLB analysis for the Cycle 1 core. Both HERMITE and TORC were previously approved by the NRC for the licensing applications.

Since the methods used in the non-LOCA transient analyses were previously approved by the NRC and the design parameters for the SL2 reactor core are within the applicable ranges of the approved methods, the staff concludes that the application of the previously approved methods is acceptable for transient analyses to support the SL2 reload applications."

"...For transient analyses other than the SLB post-trip analysis, the licensee applied the extended statistical combination of uncertainty (ESCU) methodology (Reference [10]) and the CE-1 critical heat flux (CHF) correlation (References [11] and [12]) to calculate DNBRs with the safety limit DNBR of 1.28. The 1.28 value incorporates all applicable penalties, including rod bow. As previously described in [Reference 4,] Section 2.2.1, the rod bow penalty of 1.2 percent on minimum DNBR was calculated for burnup up to 31,700 MWD/MTU. Because of lower radial power peaks in fuel assemblies and rods at burnup above 31,700 MWD/MTU, sufficient margin exists to offset the rod bow penalty at these higher burnup. For the post-trip return-to-power portion of the SLB analysis, the licensee calculated DNBRs using the modified McBeth CHF correlation with the safety limit DNBR of 1.30. Since the ESCU methodology was previously approved by the NRC for the SL2 licensing calculations, and the use of the modified McBeth correlation in the SLB post trip analysis is consistent with that used in the existing SLB analysis, the staff concludes that the use of the approved ESCU methodology and CHF correlations with the associated safety DNBR limits to assess the fuel failure in the SL2 transient analyses are acceptable."

This prior NRC review included the protection required against AOOs and accidents in the plant setpoints. The following was stated in the SER concerning the setpoint methods used.

"2.2.3 Reclassification of Anticipated Operational Occurrences (AOOs) from Protection via Reactor Protection System (RPS) Trips to RPS and/or Limiting Conditions of Operation (LCOs).

The analysis of the uncontrolled CEA withdrawal (CEAW) event at power will be modified such that a combination of RPS trips and initial thermal margin are used to assure that SAFDLs are not violated.

In the current CEAW analysis the change in integrated radial peak is an additive adjustment to the final overpower margin. In the revised analysis, the impact of the change in integrated radial peak will be explicitly calculated in place of the additive adjustment to the final overpower margin. In addition, the delta-T power trip will be credited in determining the most adverse case. The following, other non-loss-of-coolant accident (LOCA) design basis events (DBEs), were also analyzed such that a combination of RPS trips and initial thermal margin are used to assure that SAFDLs would not be violated: (1) decrease in feedwater temperature, (2) increase in feedwater flow, (3) increased main steam flow, and (4) CEAW from subcritical or low power. This is a change from prior analysis and was done to optimize the thermal margin operating space. As discussed below, the results indicate that all the safety analysis acceptance criteria are met and therefore, these revised analyses are acceptable."

Initial thermal margin and LHR margin credited in the non-LOCA safety analysis is preserved in the COLR limits as documented in the approved licensing topical report CENPD-199-P, Revision 1-P-A, Reference 13, which was cited in the Reference 2 submittal.

2.0 Analysis Criteria And Quantification of Effect of Reduced RCS Flow

This section provides a brief summary and the impact of the reduced RCS flow on the results of the non-LOCA transient analysis.

Events Analyzed for Offsite Dose

Cycle specific analyses are performed to confirm that the cycle specific fuel failures do not exceed the level of fuel failures assumed in the design basis dose consequences for the limiting fuel failure events. This confirmation assures that the dose consequences in the UFSAR remain bounding.

These fuel failure events are:

- pre-trip steam line break,
- post-trip steam line break, and
- seized rotor.

The dose consequences for the non-fuel-failure limiting events are not sensitive to the reduction in RCS flow and thus are not changed. These limiting events are:

- feedwater line break,
- small primary line break, and
- steam generator tube rupture.

Full Power Events Analyzed for DNBR

The events in this category are analyzed to calculate the required overpower margin (ROPM) which is preserved in the COLR limits as documented in Reference 13. Cycle specific analysis verifies that the plant setpoints reserve more margin than the transient analysis ROPM. Typically, the limiting ROPM is based on the Steam System Piping Failures (Pretrip Steam Line Break) Event. This verification, per Reference 13 methodology, assures the validity of the fuel failure calculation for the Pretrip Steam Line Break Event and assures that the DNBR limit is not violated for AOOs

Table 2-1
Full Power Events Analyzed for DNBR [1]

Events	Impact of Reduced RCS Flow on Required Overpower Margin
1. Increased Main Steam Flow	ROPM decreased < 0.1%
2. Steam System Piping Failures (pretrip)	ROPM increased < 1.6%
3. Total Loss Forced Reactor Coolant Flow	ROPM increased < 0.1%
4. Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	No change to ROPM; new flow used to verify fuel failure is bounded by UFSAR
5. Pressurizer Pressure Decrease Events	No change to transient input to TM/LP setpoint
6. Asymmetric Steam Generator Events	No change to ROPM
8. Uncontrolled CEA Withdrawal at Power	ROPM increased < 0.3%
9. CEA Drop Event	No change to ROPM

[1] For some events, the impact results from the net effect of reduced flow and other parameter changes as described for the respective events described in Section 5 of Attachment 6 to Reference 1.

Cycle specific analysis for St. Lucie Unit 2 Cycle 14 has verified that the plant setpoints reserve more margin than the transient analysis ROPM by at least 5%.

Full Power Events Analyzed for LHR Margin / CTM

The events in this category are analyzed to calculate the LHR margin which is preserved in the COLR limits as documented in Reference 13. Cycle specific analysis verifies that the plant setpoints reserve more margin than the transient analysis LHR margin. Typically, the limiting CTM event is the Control Element Assembly Ejection Event. This verification, per Reference 13 methodology, assures that the accident consequences reported in the UFSAR are not exceeded.

Table 2-2
Full Power Events Analyzed for LHR Margin / CTM Criterion [1]

Events	Impact of Reduced RCS Flow on LHR Margin
1. Increased Main Steam Flow	Negligible effect on LHR margin
Events	**Impact of Reduced RCS Flow on CTM Criterion**
2a. Steam System Piping Failures (pretrip)	No change; CTM limit met
2b. Steam System Piping Failures (post-trip)	No change to fuel failure calculation
3. Asymmetric Steam Generator Events	Non limiting / bounded by CEA Withdrawal
4. Uncontrolled CEA Withdrawal at Power	Cycle specific verification
5. CEA Drop Event	Cycle specific verification
6. Control Element Assembly Ejection	23 degrees F margin to CTM reduced, about 15 degrees F CTM margin remains

[1] For some events, the impact results from the net effect of reduced flow and other parameter changes as described for the respective events described in Section 5 of Attachment 6 to Reference 1.

Cycle specific analysis for St. Lucie Unit 2 Cycle 14 has verified that the plant setpoints reserve more margin than the transient analysis LHR margin by at least 10%.

Zero Power Events Analyzed for DNBR and CTM

Zero power events listed in Table 2-3 are analyzed to assure the validity of the DNBR values used in the cycle specific fuel failure calculations and to assure that the DNBR limit is not violated for AOOs.

Zero power events listed in Table 2-4 are analyzed to support cycle specific fuel failure calculations and to assure that the CTM limit is not exceeded for AOOs or the CEA Ejection Event.

Table 2-3
Zero Power Events Analyzed for DNBR [1]

Events	Impact of Reduced RCS Flow on DNBR
1. Steam System Piping Failures (post-trip)	No change to fuel failure calculation. Decrease in MacBeth DNBR < 0.01
2. Uncontrolled CEA Withdrawal from a Subcritical Condition or Low Power	DNBR increased by 0.1 due to use of a more realistic axial power shape

Table 2-4
Zero Power Events Analyzed for CTM Criterion [1]

Events	Impact of Reduced RCS Flow on CTM Criterion
1. Steam System Piping Failures (post-trip)	No change to fuel failure calculation
2. Uncontrolled CEA Withdrawal from a Subcritical Condition or Low Power	Max enthalpy increased by 7 cal/gm and temperature well below CTM limit
3. Control Element Assembly Ejection	Totally insensitive to flow per Reference 8

[1] For some events, the impact results from the net effect of reduced flow and other parameter changes as described for the respective events described in Section 5 of Attachment 6 to Reference 1.

Events Analyzed for Peak RCS Pressure Criteria and Shutdown Margin

Events listed in Table 2-5 are analyzed to assure the RCS pressure safety limit (2750 psia) is not exceeded for AOOs or for the low probability Feedwater Line Break Event. The peak RCS pressure criterion for very low probability events is 3000 psia.

Events listed in Table 2-6 are analyzed to assure that the consequences reported in the UFSAR are not exceeded.

Table 2-5
Events Analyzed for Peak RCS Pressure Criteria [1]

Safety Limit 2750 psia (3000 psia for Very Low Probability Events)	
Events	**Impact of Reduced RCS Flow on Peak RCS Pressure**
1. Loss of Condenser Vacuum (LOCV)	Peak pressure increased 1.8 psi to 2747.1 psia
2. Uncontrolled CEA Withdrawal at Power	Peak pressure increased to 2507 psia
3. Feedwater Line Break Events	
Low Probability (Small Feedwater Line Breaks)	Peak pressure increased but 'bounded by LOCV'
Very Low Probability (Feedwater Line Breaks w/LOAC)	Peak pressure increased from 2847 psia, but the increase is insignificant compared to the 153 psi margin to the limit

Table 2-6
Events Analyzed for Shutdown Margin [1]

Events	**Impact of Reduced RCS Flow on Shutdown Margin**
1. Inadvertent Opening of a Steam Generator Safety Valve or ADV	Totally insensitive to flow
2. Steam System Piping Failures (post-trip)	Maximum post-trip reactivity decreased from 0.4407 %$\Delta\rho$ to 0.4373 %$\Delta\rho$

[1] For some events, the impact results from the net effect of reduced flow and other parameter changes as described for the respective events described in Section 5 of Attachment 6 to Reference 1.

3.0 Discussion of SER Compliance for Codes and Methods

The current analyses for all UFSAR events, as described earlier, are performed consistent with the requirements of applicable approved methodology SERs, and these analyses have been previously approved in References 4 and 16. The input changes for the RCS flow reduction work are the following:

- Minimum RCS flow reduced from 363,000 gpm to 355,000 gpm.
- RCS core inlet temperature reduced from 550^{0}F to 549^{0}F (Tech Spec value).
- RCS temperature uncertainty, analysis value, reduced from 4^{0}F to 3^{0}F.

Additionally, the UFSAR analysis of Pressurizer Pressure Decrease Events had used a conservative TM/LP trip setpoint as compared to the analysis setpoint value in the

UFSAR Table 15.0-18c, after accounting for uncertainties. This conservatism, documented in the footnote to the UFSAR Table 15.0-18c, is removed in the reduced RCS flow analysis as stated in Section 5.6.1 of Attachment 6 of Reference 1.

None of these changes affects the SER compliance consideration for any of the methodologies applicable to UFSAR analysis. Also, there is no change to any of the methodologies used in support of the proposed flow reduction. Thus, it is concluded that all the event analyses/evaluations performed remain in compliance with the staff requirements for the applicable methodology SERs.

4.0 Analysis Assumptions and Treatment of Uncertainties

Reference 1 Attachment 6, Table 5.0-2 presents the key parameters assumed in the transient analysis. The values are exactly the same as in UFSAR Table 15.0-17a, except for the Minimum Reactor Vessel Flow Rate and the Maximum Steady State Coolant Temperature. The Minimum RCS Vessel Flow Rate is decreased from 377,500* / 363,000 gpm to 369,500* / 355,000 gpm and the Maximum Steady State Coolant Temperature (including uncertainties) is decreased from 553°F to 552°F. A value of 554°F was however supported by all current UFSAR non-LOCA transient analyses, except for the steam generator tube rupture analysis.

Reference 1, Attachment 6 Table 5.0-3 presents the Reactor Protection System (RPS) and Engineering Safety Features Actuation System (ESFAS) Instrumentation trip setpoints and delay times. The values are exactly the same as in the UFSAR Table 15.0-18c, except for the footnote 3, which is deleted. This footnote 3 to the UFSAR Table 15.0-18c reflected the conservatism in the TM/LP trip setpoint value used in the analysis of Pressurizer Pressure Decrease Events. This conservatism has been removed.

To further offset the effect of decreased minimum reactor vessel flow, the Maximum Steady State Coolant Temperature value (nominal) at hot full power was decreased from 550°F to 549°F consistent with the Technical Specifications requirements. The Maximum Steady State Coolant Temperature uncertainty value used in the transient analysis was decreased from a conservative value of 4°F to 3°F over the entire power range. The value of 3°F bounds the calculated RCS temperature uncertainty based on the instrumentation uncertainties. The HFP core coolant inlet temperature range used in the analysis is thus 546°F to 552°F with consideration of uncertainties.

There are no changes to any other uncertainty value as used in the current UFSAR analyses. The technique used to account for the uncertainties statistically in the thermal margin LCO limits is described in, Reference 14, Appendix C. This approach was presented in Reference 3, Section 8.0.1.1 and approved in the Reference 4 NRC SER.

5.0 References

1. Letter from D. E. Jernigan (FPL) to NRC, "St. Lucie Unit 2 Docket No. 50-389 Proposed License Amendment Reduce the Minimum Reactor Coolant System Flow," L-2002-196, October 15, 2002.

2. Letter from J. A. Stall (FPL) to NRC, "Proposed License Amendment Cycle 12 Reload Process Improvement," L-98-308, December 18, 1998.

3. Attachment 2 of Enclosure to Reference 2, "St. Lucie Unit 2 Safety Analysis Report for Initial Application of PAC and NPAC."

4. NRC Safety Evaluation Report, "St. Lucie, Unit 2 - Correction to Withdrawal of Amendment Request and Issuance of Revised Bases (TAC No. MA5619) and Correction to Safety Evaluation Issued as Part of a License Amendment Regarding the Cycle 12 Reload Process Improvement (TAC NO. MA4523)," dated March 1, 2000 (NRC approval of Reference 2).

5. NRC Safety Evaluation Report, "CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," enclosure to letter from C.O. Thomas (NRC) to A. E. Scherer (CE), April 3, 1984.

6. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin for a Reactor Core," April 1986.

7. CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods," June 1981.

8. CENPD-190-A, "CEA Ejection, C-E Method for Control Element Assembly Ejection," July 1976.

9. CENPD-188-A, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients," July 1976.

10. CEN-348(B)-P-A Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January 1997.

11. CENPD-162-P-A, "Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," September 1976.

12. CEN-207-P-A, "Critical Heat Flux Correlation for CE Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Power Distribution," December 1984.

13. CENPD-199-P, Rev. 1-P-A, "CE Setpoint Methodology, C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986.

14. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for St. Lucie Unit 1," February 1980.

15. Letter from R. S. Kundalkar (FPL) to NRC, "Proposed License Amendment – Unreviewed Safety Question Revised Post-Trip Steam Line Break Analysis," L-2000-224, November 28, 2000.

16. NRC Safety Evaluation Report, "St. Lucie Plant, Unit No. 2 – Issuance of Amendment Regarding Revised Post-Trip Steam Line Break Analysis (TAC No. MB0616)," dated June 19, 2001 (NRC approval of Reference 15).

Attachment 2

Replacement mark up of TS page 2-5

Replacement smooth TS page 2-5

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High [5] Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps-Low	≥ 636 gpm**	≥ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power - High[4]	≤ 2.49 decades per minute	≤ 2.49 decades per minute
14. Reactor Coolant Flow - Low[1]	≥ 95.4% of design Reactor Coolant flow with four pumps operating*	≥ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure - Low[5]	≥ 800 psig	≥ 800 psig

*Design reactor coolant flow with four pumps operating is ~~363,000~~ 355,000 gpm.
**10-minute time delay after relay actuation.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density – High(5) Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	≥ 636 gpm**	≥ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power – High(4)	≤ 2.49 decades per minute	≤ 2.49 decades per minute
14. Reactor Coolant Flow – Low(1)	≥ 95.4% of design Reactor Coolant flow with four pumps operating*	≥ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low(5)	≥ 800 psig	≥ 800 psig

* Design reactor coolant flow with four pumps operating is 355,000 gpm.

** 10-minute time delay after relay actuation.