FPL

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

October 15, 2002

L-2002-196
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

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating License NPF-16 for St. Lucie Unit 2. FPL proposes 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 is 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.

Attachment 1 is a description of the proposed changes and the supporting Safety Analysis. Attachment 2 is the Determination of No Significant Hazards and Environmental Considerations. Attachment 3 provides marked up copies of the proposed Technical Specification changes. Attachment 4 provides copies of the proposed changes to the TS Bases. Attachment 5 provides copies of the retyped TS pages. Attachment 6 provides additional details of the evaluations performed to support the proposed change.

The proposed amendment has been reviewed by the St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board. In accordance with 10 CFR 50.91 (b)(1), a copy of the proposed amendment is being forwarded to the State Designee for the State of Florida.

Approval of this proposed license amendment is requested by April 2003 to support potential steam generator tube plugging during the spring 2003 St. Lucie

A001

Unit 2 refueling outage (SL2-14). Please issue the amendment to be effective on the date of 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/

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 15 day of Oct., 2002
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)

ATTACHMENT 1

SAFETY ANALYSIS

1. Introduction

Florida Power and Light Company (FPL) proposes 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 is 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.

The basis Figure B2.1-1, for representative axial shapes and radial peaks, is revised so that the conditions for the thermal margin safety limit lines (TMSLL) of TS Figure 2.1-1, as specified in the basis, continue to remain valid for the case of reduced RCS flow. Additionally, the numerical value for the departure from nucleate boiling-specified acceptable fuel design limit (DNB-SAFDL) is deleted from the Bases for TS 2.1.1, TS 2.2.1, and TS 3/4.2.5. This limit value is dependent on the DNB correlation used and is different for the two correlations that could be currently used for St. Lucie Unit 2.

RCS flow is an important parameter in the safety analysis, particularly for those events that challenge the departure from nucleate boiling ratio (DNBR) limits. The analysis performed by Westinghouse, to evaluate the impact of the proposed changes, has demonstrated that the safety analysis will continue to meet all applicable acceptance criteria for the operations of the plant with the proposed flow reduction.

2. Description of Proposed Changes

2.1. Technical Specification 3.2.5 DNB Parameters

Table 3.2-2 of this Technical Specification is revised to reduce the reactor coolant system flow rate from ≥363,000 gpm to ≥355,000 gpm.

2.2. Technical Specification 2.2.1 Reactor Trip Setpoints

A footnote to Table 2.2-1 of this Technical Specification is revised to change the design reactor coolant system flow with four pumps operating from 363,000 gpm to 355,000 gpm, consistent with the change to TS Table 3.2-2.

2.3. Basis to Technical Specification 2.1.1 Reactor Core

The basis Figure B2.1-1 is revised to include the effects of the proposed reduction in reactor coolant system design flow rate to 355,000 gpm, so that the conditions specified in the basis continue to remain valid for the thermal margin safety limit lines (TMSLL) of TS Figure 2.1-1 for the case of reduced RCS flow. Also, the wording "DNB-SAFDL of 1.28" is replaced with "appropriate correlation limit for DNB-SAFDL."

2.4. Bases to Technical Specifications 2.2.1 (Reactor Trip Setpoints) and 3/4.2.5 DNB Parameters)

The change to the basis to TS 2.2.1 for thermal margin/low pressure trip includes replacing "DNB-SAFDL of 1.28" with "appropriate correlation limit for DNB-SAFDL." Similarly, the basis to TS 3/4.2.5 is revised to replace "DNBR of ≥ 1.28" with "DNBR of greater than or equal to the appropriate correlation limit for DNB-SAFDL."

3. Basis for Proposed Change

3.1. Technical Specification 3.2.5 DNB Parameters

The basis for the flow reduction is to allow a SGTP level of up to 15% maximum in each of the two steam generators. The RCS flow is estimated to be less than 363,000 gpm for 15% SGTP after accounting for the effects of flow measurement uncertainties. A value of 355,000 gpm for RCS flow is expected to provide adequate margin to allow up to 15% SGTP.

The change to the RCS flow proposed by this license amendment affects the plant safety analysis in the following manner:

3.1.1. A reduction in core flow rate affects the calculated DNBR. This parameter is a direct indication of available thermal margin and is sensitive to changes in core flow rate (a reduction in the minimum DNBR indicates that thermal margin for the corresponding transient has been reduced).

3.1.2. The reduction in RCS flow rate results in a corresponding increase in RCS average coolant temperature (T_{ave}). This becomes a factor both for DNBR limiting events and loss of primary inventory-type transients.

The evaluation of the effects on safety is described in Section 4, which concludes that the safety analyses would continue to meet all the applicable acceptance criteria for operation of the plant with the proposed reduction in the

RCS flow. In addition, this reduction in the RCS flow requirement will not have an adverse impact on plant operation.

3.2. Technical Specification 2.2.1 Reactor Trip Setpoints

A revision to the footnote to TS Table 2.2-1 is made so that this footnote remains consistent with the amended RCS flow specification of 355,000 gpm in TS Table 3.2-2. This footnote in TS Table 2.2-1 refers to the low flow trip setpoint, whose value in terms of "percent of design RCS flow rate" remains unchanged. The impact of this change on the safety analysis is addressed in Section 4 as part of the evaluation for flow reduction from 363,000 gpm to 355,000 gpm.

3.3. Basis to Technical Specification 2.1.1 Reactor Core

The basis Figure B2.1-1 defines the axial shapes and radial peaks, which along with thermal power, RCS pressure, RCS flow, inlet temperature and DNB correlation, determine the curves of TS Figure 2.1-1 for which DNB-SAFDL is not violated. Revised radial peak values for this figure were calculated to reflect the decreased margin associated with the proposed reduction in the RCS flow from 363,000 gpm to 355,000 gpm. Consistent with the TS basis, Limiting Conditions of Operation (LCO) and reactor protection system (RPS) trips are verified to provide adequate protection against DNB and fuel centerline melt SAFDLs during normal operation and design basis anticipated operational occurrences to ensure that reactor core safety limits are satisfied.

The removal of numerical value for DNB-SAFDL in the basis to TS 2.1.1 is justified based on the fact that the DNB-SAFDL is dependent on the DNB correlation used and is different for the two DNB correlations (CE-1 and ABB-NV) that could currently be used for the DNB verification analysis for St. Lucie Unit 2.

3.4. Bases to Technical Specifications 2.2.1 (Reactor Trip Setpoints) and 3/4.2.5 DNB Parameters)

Changes proposed to the Bases to TS 2.2.1 and TS 3/4.2.5 are to remove the numerical value for DNB-SAFDL. These changes are justified based on the fact that the DNB-SAFDL is dependent on the DNB correlation used and is different for the two DNB correlations (CE-1 and ABB-NV) that could currently be used for the DNB verification analysis for St. Lucie Unit 2.

4. Analysis of Impact on Safety

A review of events described in the St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR) was performed by Westinghouse and FPL to assess the impact of the proposed reduction in design RCS flow rate from 363,000 gpm to 355,000 gpm. The margin available in the current analysis related to the use of 550°F inlet temperature as compared to the TS value of ≤ 549°F is credited, wherever necessary, to offset the adverse effects of the reduced flow. Additionally, the temperature uncertainty used in the current analyses is 4°F, which has a margin of at least 1°F based on the calculated cold/hot leg temperature uncertainty of less than 3°F.

The review of UFSAR events focused on: a) identifying the events that would be affected by the proposed changes, and b) evaluating the impact of the proposed changes on these events to ensure that available margin to the acceptance criteria continues to exist. All other event analyses were dispositioned as being not significantly affected so that they remain in compliance with the acceptance criteria. The reactor protection system (RPS) thermal margin/low pressure (TM/LP) trip function and local power density (LPD) Limiting Safety System Settings (LSSS), as described in the TS, will continue to provide adequate protection with respect to DNB and fuel centerline melt criteria. These setpoints, along with DNB and linear heat rate LCOs, are verified every reload cycle to confirm that their limits are acceptable after accounting for the thermal margin requirement of the DNB transient events.

Detailed evaluations were performed by the fuel vendor, Westinghouse Electric (W), which concluded that all the analyses would continue to meet the applicable acceptance criteria for the case of RCS flow reduction from 363,000 gpm to 355,000 gpm. The details of this work are provided in Attachment 6. The methodologies used in these supporting analyses/evaluations are the same as those previously documented.[1,2] The input changes are:

- Reduction in RCS flow from 363,000 gpm to 355,000 gpm,
- Flow basis for low flow trip setpoint reduced from 363,000 gpm to 355,000 gpm,
- Core inlet temperature reduced from 550°F to 549°F (TS value) plus or minus uncertainties, and
- RCS temperature uncertainty reduced from 4°F to 3°F.

The proposed changes with their proper implementation will have no adverse impact on the plant operation and safety.

4.1. <u>Increase in Heat Removal by the Secondary System</u>

The limiting or potentially limiting events in this category are increased main steam flow (DNBR and fuel centerline melt), inadvertent opening of a secondary safety valve (loss of shutdown margin and radiological releases), and steam system piping failures (radiological releases with fuel failure).

1 Letter L-98-308, J. A. Stall (FPL) to USNRC Document Control Desk, "Proposed License Amendment – Cycle 12 Reload Process Improvement," December 18, 1998
2 Letter K. N. Jabbour (USNRC) to T. F. Plunkett (FPL), "St. Lucie Unit 2 – Issuance of Amendment Regarding the Cycle 12 Reload Process Improvement (TAC No. MA4523)," December 21, 1999 (Amended in letter dated March 1, 2000)

Evaluation of steam system piping failures concluded that there is sufficient margin available in the current analysis to account for the minor adverse effects of the reduced flow so that the consequences and conclusions documented in the UFSAR remain unchanged. Evaluation of other events also demonstrated that the results and conclusions of the current analyses remain unchanged as these are either insensitive to the proposed reduction in the RCS flow or the effects of the flow reduction are offset by the decrease in analysis value for the core inlet temperature.

4.2. Decrease in Heat Removal by the Secondary System

The limiting or potentially limiting events in this category are loss of condenser vacuum (peak primary and secondary pressures), loss of offsite power (radiological releases), and feedwater line break (peak primary pressure and radiological releases).

Evaluation of the loss of condenser vacuum event resulted in an increase in peak RCS pressure of 1.8 psi and an increase in peak steam generator pressure of 0.033 psi. With these increases, the peak pressures continue to remain below the peak pressure limits of 2750 psia for the RCS and 1100 psia for the steam generator. Evaluation of other events confirmed that the results and conclusions documented in the UFSAR remain unchanged as the changes proposed were determined to have negligible impact on these analyses.

4.3. Decrease in Reactor Coolant Flowrate

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 degradation in the calculated DNBR. The limiting events in this category are loss of forced reactor coolant flow (DNBR), and single reactor coolant pump shaft seizure/sheared shaft (radiological releases with fuel failure).

Evaluation of these events for the proposed reduction in RCS flow demonstrated that the results and conclusions of the current analyses remain unchanged due to the initial thermal margin preserved by the LCOs. This thermal margin requirement is included in the setpoint verification analysis.

4.4. Reactivity and Power Distribution Anomalies

The limiting or potentially limiting events in this category are control element assembly (CEA) withdrawal (DNBR and fuel centerline melt), CEA drop (DNBR), chemical volume control system (CVCS) malfunction (loss of shutdown margin), and CEA ejection (radiological releases).

Although there are minor impacts of the reduced flow on the transient response of CEA withdrawal events, evaluation of these events demonstrated that the acceptance criteria with respect to the DNBR and fuel centerline melt are not violated. The conclusions of the UFSAR analyses remain unchanged.

CEA drop event is not impacted, as the DNBR limit is not violated during this event due to the thermal margin preserved by the LCOs. This thermal margin is insensitive to the flow reduction and is included as a requirement in the setpoint verification analysis.

The CVCS malfunction event is insensitive to the minimum RCS flow. For the hot full power CEA ejection event the decrease in RCS flow and the decrease in core inlet temperature offset each other with respect to the transient thermal margin, whereas the decrease in RCS flow has no impact on the hot zero power CEA ejection event. The conclusions of the UFSAR analyses therefore remain unchanged.

4.5. Increase in Reactor Coolant System Inventory

The limiting event in this category is the CVCS malfunction (pressurizer fill). The decrease in RCS flow from 363,000 gpm to 355,000 gpm has no impact on this event.

4.6. Decrease in Reactor Coolant System Inventory

The limiting events in this category are inadvertent opening of pressurizer relief valves (DNBR), steam generator tube rupture (radiological releases) and loss of coolant accidents (LOCA) (10 CFR 50.46 and radiological releases).

For the inadvertent opening of power operated relief valves (PORV), it was determined that the impact of the decrease in RCS flow is offset by the decrease in the core inlet temperature, such that the results and conclusions of the current analyses remain bounding.

For the steam generator tube rupture, the decrease in the RCS flow has a negligible impact, such that the conclusions of the current analysis remain unchanged.

The impact of the reduced flow on the large break LOCA (LBLOCA) analysis is evaluated to be an increase in the peak cladding temperature (PCT) of 15°F. The impact of the reduced flow on the small break LOCA (SBLOCA) analysis is evaluated to be an increase in the PCT of 7.6°F. The corresponding revised PCTs remain below 2200°F and all the 10 CFR 50.46 criteria are met for both the LBLOCA and SBLOCA analyses.

4.7. Miscellaneous Events/Analyses

4.7.1. Asymmetric Steam Generator Events

This event was evaluated for the impact of reduced flow. By crediting the margin available in the core inlet temperature, it was concluded that the current thermal margin requirements for this event remain bounding.

4.7.2. Peak Containment Pressure/Temperature Response

An engineering evaluation has demonstrated that the RCS flow reduction has an insignificant impact on the containment pressure and temperature response, and the component cooling water/intake cooling water system temperature response.

4.7.3. Fuel Design/Performance Analysis

Fuel mechanical design and thermal performance analyses were evaluated. It was confirmed that the St. Lucie Unit 2 operation with the reduced flow was acceptable from fuel design considerations with respect to cladding stress, strain, and fatigue. Thermal design and fuel performance analysis was also verified to be acceptable for the reduced flow condition.

5. Conclusion

The results of this evaluation demonstrate that the proposed amendment does not result in violation of any safety limit or acceptance criterion. Therefore, there is no safety concern associated with the proposed change. Additionally, it is concluded in Attachment 2 that the proposed changes do not involve a significant hazards consideration

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS AND ENVIRONMENTAL CONSIDERATION

Florida Power and Light Company (FPL) proposes 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 is 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.

The standards used to arrive at a determination that a request for amendment involves a no significant hazards consideration are included in the Commission's regulation, 10 CFR 50.92, which states that no significant hazards considerations are involved if the operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment would decrease the value of design reactor coolant system flow rate. This reduction in the reactor coolant system (RCS) flow requirement will support operation of the plant with an increased steam generator (SG) tube plugging. The changes to the Technical Specification (TS) bases either support the proposed flow reduction or are administrative in nature, consistent with the current design basis. The parameters affected by the proposed changes are not accident initiators and do not affect the frequency of occurrence of previously analyzed transients. Additionally, there are no changes to any active plant component.

This evaluation has demonstrated acceptable results for all the accidents previously analyzed. It is concluded that the radiological consequences would remain within their established acceptance criteria when including effects of the proposed reduction in the RCS flow, which would support an increased steam generator tube plugging level.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

This proposed amendment revises the RCS design flow requirement to cover plant operation with increased steam generator tube plugging. There are no physical changes to the plant systems or system interactions due to the proposed changes. The modes of operation of the plant and the design functions of all the safety systems remain unchanged.

Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The impact of the proposed changes on the design basis accident analysis was evaluated and it is concluded that the setpoint and safety analyses of all design basis accidents meet the applicable acceptance criteria with respect to the radiological consequences, specified acceptable fuel design limits (SAFDL), primary and secondary overpressurization, peak containment pressure and temperature, and 10 CFR 50.46 requirements.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the above, we have determined that the proposed amendment does not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore does not involve a significant hazards consideration.

Environmental Impact Consideration Determination

The proposed license amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released off-site, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendment involves no significant hazards consideration, and therefore, meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendment.

ATTACHMENT 3

ST. LUCIE UNIT 2 MARKED-UP TECHNICAL SPECIFICATION PAGES

TS Pages

2-5

3/4 2-15

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density – High(6) 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~~ gpm.

** 10-minute time delay after relay actuation.

355,000

TABLE 3.2-2

DNB MARGIN

LIMITS

PARAMETER	FOUR REACTOR COOLANT PUMPS OPERATING
Cold Leg Temperature (Narrow Range)	535°F* $\leq$ T $\leq$ 549°F
Pressurizer Pressure	2225 psia** $\leq$ P_{PZR} $\leq$ 2350 psia*
Reactor Coolant Flow Rate	$\geq$ ~~363,000~~ gpm 355,000
AXIAL SHAPE INDEX	COLR Figure 3.2-4

* Applicable only if power level $\geq$ 70% RATED THERMAL POWER.

** 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.

ATTACHMENT 4

ST. LUCIE UNIT 2 MARKED-UP TECHNICAL SPECIFICATION BASES PAGES

Bases Sections

Bases 2.1.1

Bases 2.2.1

Bases 3/4.2.5

BASES FOR SECTION 2.0

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

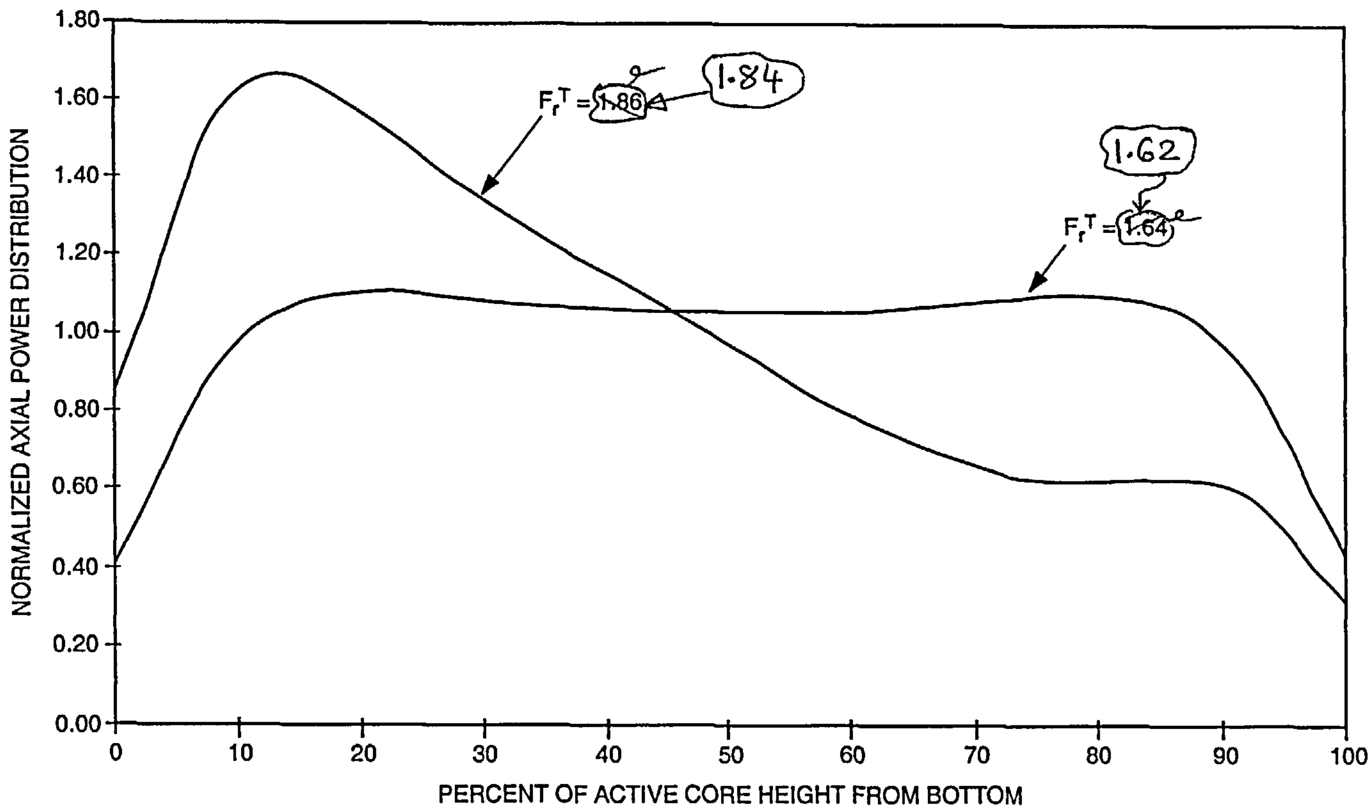
Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 or ABB-NV correlation. The CE-1 and ABB-NV DNB correlations have been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. /R1

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to the ~~DNB SAFDL of 1.28~~ in conjunction with the Extended Statistical Combination of Uncertainties (ESCU). This value is derived through a statistical combination of the system parameter probability distribution functions with the CE-1 or ABB-NV DNB correlation uncertainties. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions. /R1

appropriate correlation limit for DNB-SAFDL

SECTION NO.: 2.0	TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 1 OF ADM-25.04 SAFETY LIMITS AND LIMITING SAFETY SETTINGS ST. LUCIE UNIT 2	PAGE: 5 of 10
REVISION NO.: ~~8~~ 2		

FIGURE B 2.1-1
AXIAL POWER DISTRIBUTIONS FOR THERMAL MARGIN SAFETY LIMITS



2.2 LIMITING SAFETY SYSTEM SETTINGS (continued)

BASES (continued)

2.2.1 REACTOR TRIP SETPOINTS (continued)

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam line safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss without reactor trip. This trip's setpoint is at less than or equal to 2375 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Thermal Margin/Low Pressure

appropriate correlation limit for DNB-SAFDL

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

The trip is initiated whenever the Reactor Coolant System pressure signal drops below either 1900 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

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

3/4.2 POWER DISTRIBUTION LIMITS (continued)

BASES (continued)

3/4.2.5 DNB PARAMETERS

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

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

DNBR of greater than or equal to the appropriate correlation limit for DNB-SAFDL

ATTACHMENT 5

ST. LUCIE UNIT 2 RETYPED TECHNICAL SPECIFICATION PAGES

<u>TS Pages</u>

2-5

3/4 2-15

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
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10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	$\geq$ 636 gpm**	$\geq$ 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)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute
14. Reactor Coolant Flow – Low(1)	$\geq$ 95.4% of design Reactor Coolant flow with four pumps operating*	$\geq$ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low(5)	$\geq$ 800 psig	$\geq$ 800 psig

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

** 10-minute time delay after relay actuation.

TABLE 3.2-2

DNB MARGIN

LIMITS

PARAMETER	FOUR REACTOR COOLANT PUMPS OPERATING
Cold Leg Temperature (Narrow Range)	535°F* $\leq$ T $\leq$ 549°F
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Reactor Coolant Flow Rate	$\geq$ 355,000 gpm
AXIAL SHAPE INDEX	COLR Figure 3.2-4

* Applicable only if power level $\geq$ 70% RATED THERMAL POWER.

** 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.

ATTACHMENT 6

EVALUATION OF SAFETY ANALYSES

1.0 Purpose

St. Lucie Unit 2 is currently operating in Cycle 13 with a Technical Specifications RCS minimum flow of 363,000 gpm. This Safety Analysis Report (SAR) documents safety and setpoint evaluations performed to assess the impact of reducing the RCS minimum flow from 363,000 gpm to 355,000 gpm, and a reduction in the analysis value for inlet temperature from 550°F to 549°F, which complies with the limit specified in the current TS Table 3.2-2. There is no change to the steam generator tube plugging level used in the safety analysis. This SAR reflects only changes with respect to the current design basis accident analysis described in the St. Lucie Unit 2 UFSAR (Reference 6.1). The analysis methodology is the same as that previously used in the bounding cycle analysis approved in Reference 6.2.

2.0 Fuel Design Summary

The St. Lucie Unit 2 fuel design was evaluated as part of this work scope with consideration of a reduction in the minimum RCS flow to 355,000 gpm.

The thermal performance analysis has been verified to remain applicable with reduction in core inlet flow consistent with flow allowed by the Technical Specifications. Therefore, there is no change to the thermal design and fuel performance analysis results. The mechanical design performance evaluation was made to verify the acceptability of St. Lucie 2 fuel design under conditions of reduced coolant flow associated with increased steam generator tube plugging. The small reduction of coolant flow (associated with dropping the minimum flow limit from 363,000 to 355,000 gpm) results in small coolant and fuel rod temperature increases, and potentially small increases in internal fuel rod pressures and oxidation rates. However, the evaluation of the reduced flow effects on cladding stress, strain and fatigue versus the fuel mechanical design criteria verified that sufficient margins are still available to accommodate the reduced flow conditions. The conclusions of this evaluation remain applicable to all fuel designs currently used in the St. Lucie Unit 2 core.

3.0 Thermal Hydraulic Analysis/Setpoint Analysis

The Thermal Hydraulic Analysis was reevaluated to address the impact of the reduced flow. This is reported in Section 3.1.

In addition, an evaluation was performed which confirmed the adequacy of CEA and in core instrument (ICI) cooling under the reduced core flow condition.

3.1 DNBR Analysis

Table 3.1-1 presents a comparison of pertinent thermal-hydraulic design parameters for the bounding cycle and Cycle 13. This table also presents the thermal-hydraulic design parameters at the proposed reduced flow condition. The reduced flow condition that impacts thermal hydraulic design includes a reduction in minimum steady state reactor coolant flow to 355,000 gpm and a reduction in maximum indicated inlet temperature to 549°F. The reload cycle design is based on a core with all GUARDIAN™ fuel, which was previously introduced in Cycle 11. The Extended Statistical Combination of Uncertainties (ESCU) methodology used the calculational factors listed in Table 3.1-1 and other uncertainty factors at the 95/95 confidence/probability level to define a design limit for the minimum DNBR.

The bounding cycle and cycle specific DNBR limit includes other allowances, which do not change from those previously approved.

The combined effect of a reduction in minimum steady state reactor coolant flow and a reduction in maximum indicated inlet temperature has no impact to the bounding cycle assumptions.

Table 3.1-1
St. Lucie Unit 2 Cycle 13 and Bounding Cycle
Thermal Hydraulic Parameters at Full Power *

General Characteristics	Units	Cycle 13++		Bounding Cycle	
			Reduced Flow		Reduced Flow
Total Heat Output (Core Only)	MWt	2700	2700	2700	2700
	10^6 Btu/hr	9215	9215	9215	9215
Fraction of Heat Generated In Fuel Rod	----	0.975	0.975	0.975	0.975
Primary System Pressure (Nominal)	psia	2250	2250	2250	2250
Inlet Temperature (Maximum Indicated)	°F	550	549	550	549
Total Reactor Coolant Flow (Minimum Steady State)	gpm	363,000	355,000	363,000	355,000
	10^6 lbm/hr	136.4	133.6	136.4	133.6
Bypass Flow (Maximum for Minimum Core Flow)	10^6 lbm/hr	5.0	4.9	5.0	4.9
Coolant Flow Through Core (Minimum)	10^6 lbm/hr	131.4	128.7	131.4	128.7
Hydraulic Diameter (Nominal Channel)	ft	0.039	0.039	0.039	0.039
Average Mass Velocity	10^6 lbm/hr-ft^2	2.40	2.35	2.40	2.35
Pressure Drop Across Core (Minimum Steady State Flow Irreversible Over Entire Fuel Assembly)	psi	13.4	12.8	13.4	12.8
Total Pressure Drop Across Vessel (Based on Nominal Dimensions and Minimum	psi	35.4	34.0	35.4	34.0

General Characteristics	Units	Cycle 13++		Bounding Cycle	
			Reduced Flow		Reduced Flow
Steady State Flow)					
Core Average Heat Flux (Accounts for Fraction of Heat Generated in Fuel Rod and Axial Densification Factor)	Btu/hr-ft^2	154,723**	154,723**	154,723***	154,723***
Total Heat Transfer Area (Accounts for Axial Densification Factor)	ft^2	58,055**	58,055**	58,055***	58,055***
Film Coefficient at Average Conditions	Btu/hr-ft^2-°F	5800	5700	5800	5700
Average Film Temperature Difference	°F	26.7**	27.1**	26.7***	27.1***
Average Linear Heat Rate of Undensified Fuel Rod (Accounts for Fraction of Heat Generated in Fuel Rod)	kw/ft	4.52**	4.52**	4.52***	4.52***
Average Core Enthalpy Rise	Btu/lbm	70.1	71.6	70.1	71.6
Maximum Clad Surface Temperature	°F	656.6	656.6	656.6	656.6
Engineering Heat Flux Factor	----	1.032+	1.032+	1.032+	1.032+
Engineering Factor on Hot Channel Heat Input	----	1.030+	1.030+	1.030+	1.030+
Rod Pitch, Bowing and Clad Diameter Factor	----	1.05+	1.05+	1.05+	1.05+
Fuel Densification Factor (Axial)	----	1.003	1.003	1.003	1.003

Notes:

(*) Due to the extended statistical combination of uncertainties, the nominal inlet temperature and nominal primary system pressure were used to calculate some of these parameters.

(**) Based on a core containing a total of 100 shims and stainless steel rods.

(***) Based on a core containing a total of 100 shims and stainless steel rods, the maximum number supported in the St. Lucie Unit 2 bounding analysis.

(+) These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level and included in the design limit on the minimum DNBR when iterating on power.

(++) Cycle 13 values are provided for comparison purposes and are bounded by Bounding Cycle values.

3.2 Setpoint Analysis

The Setpoint Analysis provides or confirms the Limiting Conditions for Operation (LCO), the Limiting Safety System Settings (LSSS) and the equipment setpoint requirements for St. Lucie Unit 2. Cycle 14 Setpoint Analysis will account for the RCS flow reduction and any setpoint-related changes imposed by the transient analyses. Based on the margins available in Cycle 13, the Cycle 14 setpoint analysis is expected to accommodate the RCS flow reduction with no other DNB-related LCO or LSSS changes.

The limits will be evaluated using the same setpoint methodology and the extended statistical combination of uncertainties methodology as that currently used for the:

a. Local Power Density LSSS

b. Ex-Core Linear Heat Rate (LHR) LCO

c. Thermal Margin/Low Pressure LSSS

d. Ex-Core Departure from Nucleate Boiling (DNB) LCO

In addition, the Technical Specifications Bases (Figure B 2.1-1) for St. Lucie Unit 2 Thermal Margin Safety Limit Lines (Technical Specification Figure 2.1-1) were reviewed and revised Fr values were calculated that reflect the decreased margin associated with a reduction in the minimum guaranteed reactor coolant flow rate from 363,000 gpm to 355,000 gpm.

4.0 ECCS Performance Analysis

The results for the ECCS performance analysis presented in Reference 6.1 have been extended to support a reduction in the Technical Specifications reactor coolant system (RCS) minimum flow from 363,000 gpm to 355,000 gpm. The conclusions, however, are unchanged. The ECCS performance analysis for St. Lucie Unit 2 with reduced minimum RCS flow meets the 10CFR50.46 acceptance criteria for peak cladding temperature (PCT), maximum cladding oxidation (MCO), maximum core-wide cladding oxidation (MCWO), coolable geometry, and long term cooling.

In addition to the ECCS performance analysis, an evaluation was performed to demonstrate that the RCS flow reduction has an insignificant impact on the containment pressure and temperature response and the component cooling water/intake cooling water (CCW/ICW) system temperature response.

To quantify the impact of reduced RCS flow on the LBLOCA; the CEFLASH-4A/FII blowdown calculation was rerun with input consistent with 355,000 gpm. The results of this case were used in STRIKIN-II to determine the impact of reduced RCS flow on the PCT and MCO. Comparison of these cases to the equivalent cases for the Reference 6.1 calculation quantifies a PCT increase of 15°F and a MCO increase of 0.40%. The MCWO reported for LBLOCA in Reference 6.1 is not impacted, since sufficient conservatism is included in the MCWO result to accommodate a change of this magnitude to minimum RCS flow. These results are applicable to St. Lucie Unit 2 with reduced RCS flow from 363,000 gpm to 355,000. The LBLOCA transient behavior for RCS flow of 363,000 gpm illustrated in UFSAR Section 15.6.6.1, is representative of the LBLOCA transient with RCS flow of 355,000 gpm.

To quantify the impact of reduced RCS flow on the SBLOCA, the CEFLASH-4AS hydraulics calculation was rerun with input consistent with 355,000 gpm. The results of this case were used in PARCH to determine the impact of reduced RCS flow on the PCT and MCO. Comparison of these cases to the equivalent cases for the Reference 6.1 calculation quantifies, for SBLOCA, a PCT increase of 7.6°F, a MCO increase of 0.16%, and a MCWO increase of 0.02%. These results are applicable to St. Lucie Unit 2 with reduced RCS flow from 363,000 gpm to 355,000 gpm. The SBLOCA transient behavior for RCS flow of 363,000 gpm illustrated in UFSAR Section 15.6.6.2, is representative of the SBLOCA transient with RCS flow of 355,000 gpm.

To determine the impact of reduced RCS flow on the post-loca long term cooling (LTC), the decay heat removal part of the LTC analysis was performed with inputs, which are consistent with the 355,000 gpm RCS flow. The results indicated that there is no change in the Long Term Plan presented in Reference 6.1. The boric acid precipitation part of the LTC analysis is not affected by the changes in the operating conditions. Therefore, the result for the boric acid precipitation portion of the LTC analysis in UFSAR Section 15.6.6.3, remains applicable. In conclusion, the reduction of the RCS flow does not affect the post-LOCA LTC results.

To determine the impact of reduced RCS flow on the containment pressure/temperature and CCW/ICW system response for St. Lucie Unit 2; an engineering evaluation was made that considered a reduction in the RCS flow from 363,000 gpm to 355,000 gpm. This evaluation concluded that a reduction of about 2.3% in the RCS flow rate for St. Lucie Unit 2 would have an insignificant impact on the containment pressure and temperature response as well as the CCW/ICW system temperature response.

5.0 NON-LOCA SAFETY ANALYSIS

This section presents the results of the FPL, St. Lucie Unit 2 Non-LOCA safety analyses at 2700 MWt for a decrease in minimum reactor vessel flow from 363,000 gpm to 355,000 gpm and a decrease in maximum steady state coolant temperature from 550°F to 549°F. The Non-LOCA safety analysis was performed for Unit 2 utilizing core physics and plant parameters that are anticipated to be bounding values for future cycles as defined in Reference 6.1. Changes to initial inputs described in Reference 6.1 are defined in Section 5.0.3.

All events were reevaluated or reanalyzed except for the partial loss of forced reactor coolant flow (5.3.1), CVCS malfunction (inadvertent boron dilution) (5.4.4), startup of an inactive reactor coolant pump event (5.4.5), and the Inadvertent operation of the ECCS during power operation (5.5.2) to assure that they meet their respective criteria (see UFSAR Tables 15.0-13a, 15.0-14a, 15.0-15a, and 15.0-16a) at a reactor thermal power rating of 2700 MWt.

5.0.1 METHODS OF ANALYSIS

The method of analysis was to identify certain event specific analysis sensitivity and conservatism to trade off against the 8000 gpm decrease in minimum reactor vessel flow. To further offset the decrease in minimum reactor vessel flow, the maximum steady state coolant temperature target value 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 target value was decreased from ± 4°F to ± 3°F over the entire power range consistent with instrumentation uncertainties.

5.0.2 MATHEMATICAL MODELS

The mathematical models are the same as those used in the UFSAR.

5.0.3 INPUT PARAMETERS AND ANALYSIS ASSUMPTIONS

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 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 554°F to 552°F.

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 UFSAR Table 15.0-18c, except for the RCS flow value of 355,000 gpm used in the reactor coolant flow – low trip setpoint.

Table 5.0-1
St. Lucie Unit 2 Design Basis Events
Considered in the Safety Analysis

Section	Design Basis Event	Status
5.1	Increase in Heat Removal by the Secondary System	
5.1.1	Decrease in Feedwater Temperature	Bounded
5.1.2	Increase in Feedwater Flow	Bounded
5.1.3	Increased Main Steam Flow	Evaluated
5.1.4	Inadvertent Opening of a Steam Generator Safety Valve or Atmospheric Dump Valve	Evaluated
5.1.5*	Steam System Piping Failures	
5.1.5a*	Inside Containment Pre-Trip Power Excursions	Evaluated
5.1.5b*	Outside Containment Pre-Trip Power Excursions	Evaluated
5.1.5c*	Post-Trip Analysis	Evaluated
5.2	Decrease in Heat Removal by the Secondary System	
5.2.1	Loss of External Load	Bounded
5.2.2	Turbine Trip	Bounded
5.2.3	Loss of Condenser Vacuum	Re-analyzed
5.2.4	Loss of Offsite Power to the Station Auxiliaries (LOAC)	Bounded
5.2.5	Loss of Normal Feedwater	Bounded
5.2.6*	Feedwater Line Break Event	
5.2.6a*	Small Feedwater Line Break Event	Evaluated
5.2.6b*	Feedwater Line Break Event with a Loss of AC	Evaluated
5.3	Decrease in Reactor Coolant Flowrate	
5.3.1	Partial Loss of Forced Reactor Coolant Flow	Bounded
5.3.2	Total Loss of Forced Reactor Coolant Flow	Evaluated
5.3.3*	Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft	Evaluated
5.4	Reactivity and Power Distribution Anomalies	
5.4.1	Uncontrolled CEA Withdrawal from a Subcritical or Low Power Condition	Evaluated
5.4.2	Uncontrolled CEA Withdrawal at Power	Evaluated
5.4.3	CEA Drop Event	Evaluated

Table 5.0-1
St. Lucie Unit 2 Design Basis Events
Considered in the Safety Analysis

Section	Design Basis Event	Status
5.4.4	CVCS Malfunction (Inadvertent Boron Dilution)	Not Required
5.4.5	Startup of an Inactive Reactor Coolant System Pump Event	Not Required
5.4.6*	Control Element Assembly Ejection	Evaluated
5.5	Increase in Reactor Coolant System Inventory	
5.5.1	CVCS Malfunction	Evaluated
5.5.2	Inadvertent Operation of the ECCS During Power Operation	Evaluated
5.6	Decrease in Reactor Coolant System Inventory	
5.6.1	Pressurizer Pressure Decrease Events: Inadvertent Opening of the Pressurizer Relief Valves	Re-analyzed
5.6.2*	Small Primary Line Break Outside Containment	Evaluated
5.6.3*	Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power	Evaluated
5.7	Miscellaneous	
5.7.1	Asymmetric Steam Generator Events	Evaluated

* Postulated Accidents

Table 5.0-2
St. Lucie Unit 2
Core Parameters Input To Safety Analyses

Core Parameters	Units	Analysis Values
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2720*, 2774
Maximum Steady State Core Coolant Inlet Temperature	°F	549*, 552
Minimum Steady State Core Coolant Inlet Temperature***	°F	532, 528
Minimum Steady State RCS Pressure	psia	2225*, 2180
Minimum Reactor Vessel Flow Rate	gpm	369,500*, 355,000
Negative Axial Shape LCO Extreme Assumed at Full Power (Ex-Cores)	I_p	-0.25
Maximum CEA Insertion at Full Power	% Insertion of Lead Bank	25
Maximum Initial Linear Heat Rate	kW/ft	13.0
Steady State Linear Heat Rate for Fuel CTM Assumed in the Safety Analysis	kW/ft	22.0
CEA Drop Time from Removal of Power to Holding Coils to 90% Insertion	sec	3.1
DNBR SAFDL		
CE-1		1.28
McBeth**		1.30

* For events under category B of Table 15.0-15a, in Reference 6.1 (with the exception of events in Section 5.7.1) the effects of uncertainties on these parameters were accounted for statistically. Therefore, these values do not include uncertainties.

** Used for DNB evaluations for events with low flow/low heat flux conditions (i.e., steam system piping failures, post trip analysis).

*** A minimum temperature for criticality of 515°F was addressed for zero power events.

Table 5.0-2 (Cont'd.)
St. Lucie Unit 2
Core Parameters Input to Safety Analyses

Core Parameters	Units	Analysis Values
Integrated Radial Peaking Factor / (Unrodded)		1.7
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	-3.2 to + 0.5 (below 70% power) -3.2 to +0.3 (70% power and above)
HZP (Hot Zero Power) Shutdown Margin	$\%\Delta\rho$	-3.6

Table 5.0-3
RPS and ESFAS Trip Setpoints and Delay Times Assumed in Safety Analysis

RPS Trip Function	Technical Specification Value	Analysis Setpoint	Total Delay Time, (sec)
Variable Power Level - High (% Above Initial Power Level)	9.61	10.2	0.4[(1)]
Variable Power Level - Ceiling (% Rated Thermal Power)	107	109[(2)]	0.4[(1)]
Variable Power Level - Floor (% Rated Thermal Power)	15	23[(4)]	0.4[(1)]
Pressurizer Pressure - High (psia)	2370	2415, 2460[(5)]	1.15
Pressurizer Pressure - Low (Floor of Thermal Margin/ Low Pressure), (psia)	1900	1855, 1785[(5)]	0.9[(1)]
Steam Generator Pressure - Low (psia)	626	586, 546[(5)]	1.15
Steam Generator Pressure - High Difference, (psid)	120	175	1.15
Steam Generator Level - Low (% Narrow Range Tap Span)	20.5	5.0	1.15
Reactor Coolant Flow - Low (% of 355,000 gpm)	95.4	91.9, 70[(5)]	0.65
Containment Pressure - High (psig)	3.0	4.65	1.15

Table 5.0-3 (Cont'd.)
RPS and ESFAS Trip Setpoints and Delay Times
Assumed in Safety Analysis

ESFAS Function	Technical Specification Value	Analysis Setpoint	Total Delay Time, (sec)
Safety Injection Actuation Signal (SIAS) on Pressurizer Pressure - Low (psia)	1736	1646, 1578[(5) (6)]	30.0
Main Steam Line Isolation Signal (MSIS) on Steam Generator Pressure - Low (psia)	600	560, 520[(5)]	6.75[(7)]
Main Feedwater Isolation on Steam Generator Pressure - Low (psia)	600	560, 520[(5)]	5.15[(7)]
Auxiliary Feedwater Isolation on Steam Generator Pressure Difference - High (psid)	275	360[(5)]	120

(1) When credit is taken for ΔT power calculator portion of the trip, resistance temperature detector (RTD) maximum response time of 14.0 seconds is explicitly modeled.
(2) Credit for the ceiling of variable high power trip was not taken. The variable high power trip conservatively used a setpoint of 112.2% of rated power (10.2% above initial power of 102%).
(3) Not used.
(4) Additional uncertainties were added to this value for conservatism for the CEA ejection event initiated from hot zero power.
(5) Values include harsh environment uncertainties (for inside containment steam line break and feedline break events).
(6) This value was also used for the small primary line break outside containment event.
(7) Total closure time including delay time.

5.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

5.1.1 DECREASE IN FEEDWATER TEMPERATURE

The reduction in initial RCS vessel flow rate from 363,000 gpm to 355,000 gpm and initial core coolant inlet temperature from 554°F to 552°F does not violate the UFSAR Section 15.1.3.3.1 conclusion that the consequences of this event are bounded by the increased main steam flow event (DNB and LHR performance), Inadvertent Opening of a steam generator safety valve or atmospheric dump valve event (shutdown margin), and the feedwater line break event (radiological dose).

5.1.2 INCREASE IN FEEDWATER FLOW

The reduction in initial RCS vessel flow rate from 363,000 gpm to 355,000 gpm and initial core coolant inlet temperature from 554°F to 552°F does not violate the UFSAR Section 15.1.2.1.6 conclusion that the consequences of this event are bounded by the increased main steam flow event (DNB and LHR performance), inadvertent opening of a steam generator safety valve or atmospheric dump valve event (shutdown margin), and the feedwater line break event (radiological dose).

5.1.3 INCREASED MAIN STEAM FLOW

5.1.3.1 Changes to the Increased Main Steam Flow Event

The increased main steam flow event provided in the St. Lucie 2 UFSAR Section 15.1.3.3.2 was evaluated to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm. After accounting for uncertainties, the minimum best estimate RCS flow decreases from 377,500 gpm to 369,500 gpm. In addition, the initial core inlet temperature has been decreased from 550°F to 549°F.

All other input remained the same as that in UFSAR Section 15.1.3.3.2, except for the hot leg and cold leg minimum RTD response times, which decreased from 3 to 0 seconds. To support the reduction in minimum RTD response time, the maximum difference between the hot leg and cold leg response times is required to be no greater than 11 seconds. Table 5.1.3-1 documents the changes in input parameters due to the reduction in minimum RCS flow and the decrease in the minimum hot leg and cold leg RTD response time.

5.1.3.2 Results

The results of this reevaluation show that the impact of the decrease in RCS flow by 8000 gpm and the impact of the decrease in core inlet temperature by 1°F offset each other with respect to the transient thermal margin. The reduction in hot leg and cold leg response time from 3 to 0 seconds has a negligible impact provided that the maximum difference between the hot leg and the cold leg RTD response times is not greater than

11 seconds. Thus, there is no change in the results of the Increased Main Steam Flow event.

The UFSAR sequence of events and figures for the increased main steam flow event are unchanged and remain representative of the event.

5.1.3.3 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.1.3.3.2.

Table 5.1.3-1
Key Parameters Assumed for the Increased Main Steam Flow Event

Parameter	Units	Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2720 +
Initial Core Coolant Inlet Temperature	°F	549 +
Initial Reactor Coolant System Pressure	psia	2225 +
Initial RCS Vessel Flow Rate	gpm	369,500 +
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	-3.2 to +0.5
CEA Worth at Trip	%Δρ	-5.4
Least Negative Doppler Temperature Shadowing Factor	% power/°F	0.7
Hot Leg and Cold Leg RTD Response Time, τ	seconds	$0 \leq \tau \leq 14$ *
Pressurizer Pressure Control System		Inoperable

\+ For DNBR calculations, effects of uncertainties on these parameters were combined statistically.
* The maximum difference between the hot leg and the cold leg RTD response times is ≤11 seconds.

5.1.4 INADVERTENT OPENING OF A STEAM GENERATOR SAFETY VALVE OR ATMOSPHERIC DUMP VALVE

5.1.4.1 Identification of Causes

The inadvertent opening of a steam generator safety valve (secondary safety valve) or atmospheric dump valve event, provided in UFSAR Section 15.1.3.1.1, was evaluated to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm.

The inadvertent opening of a secondary safety valve results in the entire blowdown of one steam generator and partial blowdown of the other. The consequent cooldown exceeds those experienced in the events presented in Sections 5.1.1 and 5.1.3.

5.1.4.2 Results

Key parameters assumed for the inadvertent opening of a steam generator safety valve (secondary safety valve) or atmospheric dump valve event initiated from hot zero power are listed in Table 5.1.4-1. The transient response is insignificantly affected by the reduction in minimum RCS flow from 363,000 gpm to 355,000 gpm. This is because the reduction in minimum RCS flow does not affect the primary or secondary system inventories previously modeled in the transient simulation. The rate of primary cooldown is reduced slightly, but the total cooldown is still limited by the action of the low steam generator level trip. The inadvertent opening of a steam generator safety valve will result in the reactor core remaining in a subcritical state in post trip conditions.

The results show that reliable control of reactivity is maintained, and that the radiological dose after two hours at the site boundary is bounded by the feedwater line break event (Section 5.2.6) and are a small fraction of the 10CFR100 guidelines. A further discussion of asymmetric steam generator events, which includes evaluation of fuel performance during these events, is presented in Section 5.7.1.

5.1.4.3 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.1.3.1.1.

Table 5.1.4-1
Key Parameters Assumed for the Inadvertent Opening of a Steam Generator Safety Valve Event Initiated From Hot Zero Power

Parameter	Units	Value
Initial Core Power	MWt	1
Initial Core Coolant Inlet Temperature	°F	536
Initial Reactor Coolant System Pressure	psia	2180
Initial RCS Vessel Flow Rate	gpm	355,000
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	-3.2
CEA Worth at Trip	$\%\Delta\rho$	-3.6

5.1.5 STEAM SYSTEM PIPING FAILURES

5.1.5a Steam System Piping Failures: Inside Containment Pre-Trip Power Excursions

5.1.5a.1 Changes to the Steam System Piping Failures: Inside Containment Pre-trip Power Excursions

The Steam System Piping Failures: inside containment pre-trip power excursions event provided in the St. Lucie 2 UFSAR was reanalyzed to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm. All other input remained the same as that in UFSAR Section 15.1.4.3.5.1.

5.1.5a.2 Results

The decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm results in a slight increase in peak power for the inside containment pre-trip power excursions event. The initial DNB margin preserved by the technical specification LCO is credited to offset the more adverse transient response. As a result, the consequences of the event quoted in Reference 6.1 are unchanged and the figures are unchanged.

5.1.5a.3 Conclusion

The conclusions of the inside containment pre-trip power excursions event analysis remain unchanged from those documented in UFSAR Section 15.1.4.3.5.1.

5.1.5b Steam System Piping Failures: Outside Containment Pre-Trip Power Excursions

5.1.5b.1 Changes to the Steam System Piping Failures: Outside Containment Pre-trip Power Excursions

The Steam System Piping Failures: outside containment pre-trip power excursions event provided in the St. Lucie 2 UFSAR was reanalyzed to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm. All other input remained the same as that in UFSAR Section 15.1.5.1.5.

5.1.5b.2 Results

The decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm results in a slight increase in peak power for the outside containment pre-trip power excursions event. The initial DNB margin preserved by the technical specification LCO is credited to offset the more adverse transient response. As a result, the consequences of the event quoted in the UFSAR are unchanged and the figures are unchanged.

5.1.5b.3 Conclusion

The conclusions of the outside containment pre-trip power excursions event analysis remain unchanged from those which are documented in UFSAR Section 15.1.5.1.5.

5.1.5c Steam System Piping Failure, Post-Trip Analysis

5.1.5c.1 Changes to the Post-Trip Steam Line Break Event

The hot full power (HFP) and hot zero power (HZP) steam line break (SLB) events with and without loss of off-site power provided in the St. Lucie 2 UFSAR Section 15.1.4.3.5.2 were reanalyzed to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm. In addition, the initial core inlet temperature has been decreased from 554°F to 552°F for HFP SLB and from 536°F to 535°F for HZP SLB.

All other inputs remained the same as in UFSAR Section 15.1.4.3.5.2, Table 5.1.5c-1, documents the changes in input parameters due to the reduction in minimum RCS flow.

5.1.5c.2 Results

Re-analysis demonstrates that the decrease in RCS flow by 8000 gpm and the decrease in core inlet temperature by 2°F for HFP SLB and 1°F for HZP SLB offset each other with respect to the transient maximum return to power and maximum post-trip reactivity, hence, there is no change in the results of the post-trip steam line break event documented in UFSAR Section 15.1.4.3.5.2.

Tables 15.1.4.3-8 to 15.1.4.3-11 of Reference 6.1 remain applicable and show the sequence of events for the HFP SLB and HZP SLB cases. Figures 15.1.4.3-23 to 15.1.4.3-46 of the UFSAR remain applicable and show the typical NSSS response to power, heat flux, RCS coolant temperatures, the RCS pressure, steam generator pressures, and the reactivity changes.

5.1.5c.3 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.1.4.3.5.2.

**Table 5.1.5c-1
Key Parameters Assumed for the Post
Trip Steam Line Break Event**

Parameter & Units	Full Power	Zero Power
Total RCS Power, MWt (102% Core Rated Thermal + Pump Heat)	2774 (2754 + 20)	1.0
Initial Core Coolant Inlet Temperature,°F	552	535
Initial RCS Vessel Flow Rate, gpm	355,000	355,000
Initial Reactor Coolant System Pressure, psia	2395	2395
Moderator Temperature Coefficient, 10-4 Δρ/°F	-3.2	-3.2
CEA Worth at Trip, %Δρ	-7.3	-3.6
Inverse Boron Worth, ppm/%Δρ	115	110

5.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

5.2.1 LOSS OF EXTERNAL LOAD

The reduction in initial RCS vessel flow rate from 363,000 gpm to 355,000 gpm and initial core coolant inlet temperature from 554°F to 552°F does not violate the UFSAR Section 15.2.2.1.6 conclusion that the consequences of this event are bounded by other design basis events, namely the loss of condenser vacuum event (Section 15.2.2.2.6) for core and system performance and the feedwater line break accident (Section 15.2.5.1.1) for radiological consequences. While the feedwater line break accident is not an anticipated operational occurrence (AOO) (no prediction of fuel failure), it is analyzed to the same radiological dose criteria. Therefore, a detailed analysis was not performed.

5.2.2 TURBINE TRIP

The reduction in initial RCS vessel flow rate from 363,000 gpm to 355,000 gpm and initial core coolant inlet temperature from 554°F to 552°F does not violate the UFSAR Section 15.2.1.1.6 conclusion that the consequences of this event are bounded by other design basis events, namely the loss of condenser vacuum (Section 15.2.2.2.6) event for core and system performance and the feedwater line break accident (Section 15.2.5.1.1) for radiological consequences. While the Feedwater Line Break Accident is not an AOO (no prediction of fuel failure), it is analyzed to the same radiological dose criteria. Therefore, a detailed analysis was not performed.

5.2.3 LOSS OF CONDENSER VACUUM

5.2.3.1 Changes to the Loss of Condenser Vacuum Event

The loss of condenser vacuum event provided in the St. Lucie 2 UFSAR Section 15.2.2.2.6 was reanalyzed to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm. In addition, the low flow trip setpoint flow basis was also changed from 363,000 gpm to 355,000 gpm. The reduction in minimum RCS flow resulted in a different limiting inlet temperature input. This inlet temperature was determined via a parametric study.

All other input remained the same as that in UFSAR Section 15.2.2.2.6. Table 5.2.3-1 documents the changes in input parameters due to the reduction in minimum RCS flow.

5.2.3.2 Results

The decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm resulted in an increase in peak RCS pressure of 1.8 psi and the final value remained below the peak pressure limit of 2750 psia. The increase in peak steam generator pressure was 0.033 psi and the final value remained below the peak pressure limit of 1100 psia.

Table 5.2.3-2, documents the changes to the UFSAR Table 15.2.2.2-6, Sequence of Events. Since changes to the sequence of events are small, Figures 15.2.2.2-13 to 15.2.2.2-17 of the UFSAR remain applicable and show the typical NSSS response to power, heat flux, the RCS pressure, RCS coolant temperatures, and steam generator pressures.

5.2.3.3 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.2.2.2.6.

Table 5.2.3-1
Key Parameters Assumed for the Loss of Condenser Vacuum Event

Parameter	Units	Value
Total RCS Power (102% Core Rated Thermal + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	°F	534
Initial Reactor Coolant System Pressure	psia	2180
Initial RCS Vessel Flow Rate	gpm	355,000
Moderator Temperature Coefficient	x 10-4 $\Delta\rho$/°F	+0.3
Doppler Coefficient	---	least negative
CEA Worth at Trip	%$\Delta\rho$	-5.4
Pressurizer Safety Valve Opening Setpoint (includes +2% tolerance)	psia	2550
Number of Plugged Tubes, Symmetric	#/S.G.	1500
Number of Plugged Tubes, Asymmetric	---	400
Main Steam Safety Valve Opening Setpoint (includes +1% tolerance)	psia	4 valves at 1010 4 valves at 1050
CEDM Holding Coil Delay Time	sec	0.74

Table 5.2.3-2
Sequence of Events for the Loss of Condenser Vacuum Event

Time (sec)	Event	Setpoint or Value
0.0	Closure of Turbine Stop Valves on Turbine Trip Due to Loss of Condenser Vacuum	---
4.2	Loss of Offsite Power	---
5.2	High Pressurizer Pressure Trip/Low Flow Trip Analysis Setpoint Reached	2420 psia/91.9% of 355,000 gpm
6.4	Trip Breakers Open	---
6.4	Pressurizer Safety Valves Start to Open	2550 psia
7.1	CEAs Begin to Drop Into Core	---
7.4	Maximum Core Power	< 107.6% of 2700 MWt
9.0	Steam Generator Safety Valves Start to Open	1010 psia
9.8	Maximum RCS Pressure*	≤ 2750 psia
13.9	Pressurizer Safety Valves Close	2448 psia
17.0	Maximum Steam Generator Pressure	≤ 1100 psia

*Including pump and elevation head.

5.2.4 LOSS OF OFFSITE POWER TO THE STATION AUXILIARIES (LOAC)

The reduction in initial RCS Vessel Flow Rate from 363,000 gpm to 355,000 gpm and initial core coolant inlet temperature from 554°F to 552°F does not violate the UFSAR Section 15.10.6 conclusion that the consequences of this event are bounded by other design basis events, namely, the loss of condenser vacuum event (Section 15.2.2.2.6) for system performance and the feedwater line break accident (Section 15.2.5.1.1) for radiological consequences. While the feedwater line break accident is not an AOO (no prediction of fuel failure), it is analyzed to the same radiological dose criteria. In addition, the approach to the DNBR SAFDL is bounded by the total loss of forced reactor coolant flow event (Section 15.3.2.2.6). Therefore, a detailed analysis was not performed. The changes in key parameters are shown in Table 5.2.4-1.

Table 5.2.4-1
Key Parameters Assumed for the Loss of Offsite Power Event

Parameter	Units	Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	°F	552
Initial RCS Vessel Flow Rate	gpm	355,000
Initial Reactor Coolant System Pressure	psia	2410
Effective Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	0.3
Doppler Coefficient	---	Cycle Minimum
CEA Worth at Trip	$\%\Delta\rho$	-5.4

5.2.5 LOSS OF NORMAL FEEDWATER

The reduction in initial rcs vessel flow rate from 363,000 gpm to 355,000 gpm and initial core coolant inlet temperature from 554°F to 552°F does not violate the UFSAR Section 15.2.2.1.7 conclusion that the consequences of this event are bounded by other design basis events, namely, the loss of condenser vacuum event (Section 15.2.2.2.6) for core and system performance and the feedwater line break accident (Section 15.2.5.1.1) for radiological consequences. While the feedwater line break accident is not an AOO (no prediction of fuel failure), it is analyzed to the same radiological dose criteria. Therefore, a detailed analysis was not performed.

5.2.6 FEEDWATER LINE BREAK EVENT

5.2.6a Small Feedwater Line Break Event

5.2.6a.1 Changes to the Small Feedwater Line Break Event

The small feedwater line break provided in the St. Lucie 2 UFSAR Section 15.2.5.1.1.1 was evaluated to account for a decrease in RCS flow from 363,000 gpm to 355,000 gpm, a decrease in maximum core inlet temperature from 550°F to 549°F, and a decrease in core inlet temperature uncertainty from +4°F to + 3°F. The reduction in these flow and temperature parameter values were evaluated with respect to their impact on the event consequences and were determined to have negligible impact.

All other input remained the same as that in UFSAR Section 15.2.5.1.1.1. Table 5.2.6a-1 documents the changes in input parameters due to the reduction in minimum RCS flow.

5.2.6a.2 Results

The lower RCS flow tends to increase RCS temperatures, which is partially offset by the increased heat transfer to the intact steam generator. In addition, the primary safety valve characteristics result in increased primary safety valve flow in a smaller time interval following high pressurizer pressure trip due to the accumulation pressure setpoint being reached in a correspondingly smaller time interval. These effects combined with the offsetting RCS pressure reduction effect of a lower initial core coolant inlet temperature (552°F versus 554°F) result in UFSAR Section 15.2.5.1.1.1 continuing to remain applicable.

5.2.6a.3 Conclusion

The conclusions of this evaluation remain unchanged from those documented in UFSAR Section 15.2.5.1.1.1.

Table 5.2.6a -1
Key Parameters Assumed for the Small Feedwater Line Break Event

Parameter	Units	Value
Total RCS Power (Core Thermal + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	°F	552
Initial RCS Vessel Flow Rate	gpm	355,000
Initial Reactor Coolant System Pressure	psia	2180
Moderator Temperature Coefficient	x 10-4 Δρ/°F	+0.3
Doppler Coefficient	---	cycle minimum
CEA Worth at Trip	%Δρ	-5.4

5.2.6.b Feedwater Line Break Event with a Loss of AC

5.2.6.b.1 Changes to the Feedwater Line Break Event with a Loss of AC

The feedwater line break with a loss of ac event provided in the St. Lucie 2 UFSAR Section 15.2.5.1.1.2 was evaluated to account for a decrease in RCS flow from 363,000 gpm to 355,000 gpm, a decrease in maximum core inlet temperature from 550°F to 549°F, and a decrease in core inlet temperature uncertainty from +4°F to + 3°F. The reduction in these flow and temperature parameter values were evaluated with respect to their impact on the event consequences and were determined to have negligible impact.

All other input remained the same as that in UFSAR Section 15.2.5.1.1.2. Table 5.2.6b-1 documents the changes in input parameters due to the reduction in minimum RCS flow.

5.2.6.b.2 Results

The lower RCS flow tends to increase RCS temperatures, the impact of which is partially offset by the increased heat transfer to the intact steam generator. In addition, the primary safety valve characteristics result in increased primary safety valve flow in a smaller time interval following high pressurizer pressure trip due to the accumulation pressure setpoint being reached in a correspondingly smaller time interval. These effects combined with the offsetting RCS pressure reduction effect of a lower initial core coolant inlet temperature (552°F versus 554°F) result in UFSAR Section 15.2.5.1.1.2 continuing to remain applicable.

5.2.6.b.3 Conclusion

The conclusions of this evaluation remain unchanged from those documented in UFSAR Section 15.2.5.1.1.2.

Table 5.2.6b -1
Key Parameters Assumed for the Feedwater Line Break Event

Parameter	Units	Value
Total RCS Power (Core Thermal + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	°F	552
Initial RCS Vessel Flow Rate	gpm	355,000
Initial Reactor Coolant System Pressure	psia	2180
Moderator Temperature Coefficient	x 10-4 $\Delta\rho$/°F	+0.3
Doppler Coefficient	---	cycle minimum
CEA Worth at Trip	%$\Delta\rho$	-5.4

5.3 DECREASE IN REACTOR COOLANT FLOWRATE

5.3.1 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

The core and system performance for this event is no more adverse than those following a total loss of forced reactor coolant flow event discussed in Section 5.3.2. Therefore, no evaluation was performed.

5.3.2 LOSS OF FLOW

5.3.2.1 Changes to the Loss of Flow

The loss of flow event provided in the St. Lucie 2 UFSAR Section 15.3.2.2.6 was reevaluated to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm, a reduction in maximum core inlet temperature and core inlet temperature uncertainty. In addition, the Low Flow Trip setpoint flow basis was also changed from 363,000 gpm to 355,000 gpm. The flow uncertainty remained unchanged.

All other input remained the same as that in UFSAR Section 15.3.2.2.6. Tables 5.3.2-1 and 5.3.2-2, document the changes due to the reduction in minimum RCS flow.

5.3.2.2 Results

The decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm is judged to have no impact on this event initiated from the TS LCOs in conjunction with the low flow trip, and therefore, will not violate the DNBR limit. This is based on the fact that the DNBR limit is not violated during this event due to the initial thermal margin preserved by the LCOs. This initial margin is insensitive to the flow reduction.

5.3.2.3 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.3.2.2.6.

Table 5.3.2-1
Key Parameters Assumed for the Loss of Coolant Flow Event

Parameter	Units	Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2720+
Initial Core Coolant Inlet Temperature	°F	549+
Initial RCS Vessel Flow Rate	gpm	369,500+
Initial Reactor Coolant System Pressure	psia	2225+
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	+0.3
Fuel Temperature Coefficient	Δρ/SQRT °K	-0.00108
Low Flow Trip Response Time	sec	0.65
CEA Holding Coil Delay	sec	0.74
CEA Worth at Trip (all rods out)	%Δρ	-5.4
Total Unrodded Radial Peaking Factor (F_r^T)		1.70+
4-Pump RCS Flow Coastdown		Figure 15.3.2.2-13 of Reference 6.1
ASI Value		-0.2

\+ For DNBR calculations, effects of uncertainties on these parameters were combined statistically.

Table 5.3.2-2
Sequence of Events for Loss of Flow

Time (sec)	Event	Setpoint or Value
0.0	Loss of Power to All Four Reactor Coolant Pumps	---
1.18	Low Flow Trip Signal Generated	91.9% of 355,000 gpm
1.83	Trip Breakers Open	---
2.57	CEA's Begin to Drop Into Core	---
2.95	Minimum CE-1 DNBR	>1.28

5.3.3 SINGLE REACTOR COOLANT PUMP SHAFT SEIZURE/SHEARED SHAFT

5.3.3.1 Changes to the Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

The single reactor coolant pump shaft seizure/sheared shaft event provided in the St. Lucie 2 UFSAR Section 15.3.5.1.7, was reanalyzed to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm. As stated in UFSAR Section 15.3.5.1.7, reduction in the RCS flow to the three pump asymptotic value is balanced against the initial DNB margin reserved in the LCOs. Also, since no credit for the heat flux decay upon reactor trip was included, there is no impact due to the change in low flow trip setpoint flow basis. Therefore, the reduction in minimum RCS flow has a minimal effect on the conditions for determining the number of pins in DNB.

The reduction in minimum RCS flow resulted in different conditions for deterring the potential for DNB propagation.

All other input remained the same as that in UFSAR Section 15.3.5.1.7. Table 5.3.3-1 documents the changes in input parameters due to the reduction in minimum RCS flow.

5.3.3.2 Results

The decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm resulted in a decrease in heat flux required to define the initial thermal margin for the calculation of potential fuel pins in DNB. The conditions remain essentially the same. There is no impact on the radiological doses as the number of fuel pins that were assumed to fail were back calculated to the acceptance criteria (within 10CFR100) in Reference 6.1 Section 15.3.5.1.7, and remains unchanged

The potential for DNB propagation during the seized rotor event was reevaluated for reduced RCS flow and found to not occur. Since the fuel failure is limited and DNB propagation is not postulated to occur, a coolable geometry is maintained.

5.3.3.3 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.3.5.1.7.

Table 5.3.3-1
Key Parameters Assumed for the
Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft Event

Parameter	Units	Assumed Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	°F	554
Initial RCS Vessel Flow Rate	gpm	355,000
Initial Reactor Coolant System Pressure	psia	2180

5.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

5.4.1 UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER CONDITION

In the subsequent discussion, this event is referred to as the hot zero power (HZP) CEA withdrawal event.

5.4.1.1 Changes to the Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low Power Condition

The HZP CEA withdrawal event provided in the St. Lucie 2 UFSAR Section 15.4.2.3.7, was reanalyzed to assure that the DNBR and fuel centerline to melt (CTM) specified acceptable design limits (SAFDL's) are not exceeded after accounting for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm.

All other input remained the same as that in UFSAR Section 15.4.2.3.7. Table 5.4.1-1, documents the changes in input parameters due to the reduction in minimum RCS flow.

5.4.1.2 Results

The 8000 gpm decrease in RCS flow results in small changes in the sequence of events timing in Table 15.4.2.3-6 of the UFSAR. These changes do not impact the conclusions previously reported in the UFSAR. Namely, the results of the HZP CEA withdrawal event, with minimum RCS flow reduced to 355,000 gpm, do not result in violation of the DNBR and CTM SAFDL criteria. Revised results are presented in Table 5.4.1-2 of this SAR.

5.4.1.2 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.4.2.3.7. Note that while there are minor changes in the event timing shown in the sequence of events table, the current UFSAR Section 15.4.2.3.7, figures remain representative for the HZP CEA withdrawal analysis.

Table 5.4.1-1
Key Parameters Assumed for the
Hot Zero Power CEA Withdrawal Event

Parameter	Units	Value
Initial Core Power Level	MWt	0
Initial Core Coolant Inlet Temperature	°F	536+
Initial Reactor Coolant System Pressure	psia	2180
Initial RCS Vessel Flow Rate	gpm	355,000
Moderator Temperature Coefficient	$10^{-4}\Delta\rho$/°F	+0.5
Doppler Coefficient	----	Cycle Minimum
CEA Worth at Trip	%$\Delta\rho$	-3.6
CEA Scram Curve Axial Shape Index	ASI units	+0.8
Reactivity Insertion Rate	$10^{-4}\Delta\rho$/sec	1.95
Rod Group Withdrawal Speed	in/min	30.0
CEA Differential Worth	$10^{-4}\Delta\rho$/inch	3.9

\+ Even though the uncertainty on initial core inlet temperature has decreased from ±4°F to ±3°F, the initial core inlet temperature was not reduced for this case.

Table 5.4.1-2
Sequence of Events for CEA
Withdrawal from Hot Zero Power

0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	----
32.6	High Power Trip Signal Generated	27.5% of 2700 MWt
33.0	Reactor Trip Breakers Open	----
33.7	CEA's Begin to Drop Into Core	----
33.8	Core Power Reaches Maximum	132.6% of 2700 MWt
35.1	Minimum CE-1 DNBR	> 1.28 SAFDL
35.2	Core Heat Flux Reaches Maximum	70.5% of 2700 MWt
36.0	Pressurizer Pressure Reaches Maximum	2447 psia

5.4.2 UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL AT POWER

In the subsequent discussion, this event is referred to as the hot full power (HFP) CEA withdrawal event.

5.4.2.1 Changes to the HFP CEA Withdrawal Event

The HFP CEA withdrawal event provided in the St. Lucie 2 UFSAR was reanalyzed to assure that the DNBR specified acceptable design limit (SAFDL), the fuel centerline to melt (CTM) SAFDL and the upset pressure limit of 2750 psia are not exceeded after accounting for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm (an 8000 gpm decrease). Since the RCS flow uncertainty remains the same, this implies that the minimum best estimate RCS flow decreases from 377,500 gpm to 369,500 gpm. Additionally, for the DNBR and CTM SAFDL reanalyses, the initial core inlet temperature has been decreased from 550°F to 549°F and the uncertainty on initial core inlet temperature has been decreased from ±4°F to ±3°F. As a result of these changes, the program core inlet temperatures at lower core power levels have decreased slightly.

All other input remained the same as that in UFSAR Section 15.4.2.2.1. Table 5.4.2-1 documents the changes in input parameters due to the reduction in minimum RCS flow and core inlet temperature.

5.4.2.2 Results

The decrease in RCS flow by 8000 gpm and the decrease in core inlet temperature by 1°F do not impact the core power overshoot used to compute maximum linear heat rate (LHR) results. Hence, the LHR LCO margin results supporting the UFSAR remain unchanged. The CTM SAFDL criterion is therefore not violated.

The decrease in RCS flow by 8000 gpm adversely impacts the DNBR LCO margin results supporting the UFSAR. This adverse impact is partially compensated by the decrease in core inlet temperature as a function of power. Hence, the DNBR LCO margin results supporting the UFSAR increase slightly (less than 0.5% at HFP and less than 1.5% at and below 50% power). The DNBR SAFDL criterion is, therefore, not violated.

Based on parametric study results over a flow range of 363,000 gpm to 420,000 gpm, qualitative evaluation determined that the 8000 gpm decrease in RCS flow results in less than a 5 psi increase in the peak RCS and SG pressures observed in the HFP CEA withdrawal event reported in UFSAR Section 15.4.2.2.1. These small increases result in a maximum RCS pressure that is less than the upset pressure limit of 2750 psia and a maximum SG pressure of less than 1100 psia. Hence, the RCS and SG pressure criteria are not violated. These pressure criteria results have been incorporated into the HFP CEA withdrawal sequence of events as shown in Table 5.4.2-2. The remaining

HFP CEA withdrawal event analysis results are unchanged from those reported in UFSAR Section 15.4.2.2.1.

5.4.2.3 Conclusion

The conclusions of this analysis remain unchanged from those reported in UFSAR Section 15.4.2.2.1. Note that while there are minor changes in the event timing shown in the sequence of events table, the current UFSAR Section 15.4.2.2.1, figures remain representative for the HFP CEA withdrawal analysis.

Table 5.4.2-1
Key Parameters Assumed for the CEA Withdrawal Event at Hot Full Power

Parameter	Units	Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774*
Initial Core Coolant Inlet Temperature	°F	554**
Initial Reactor Coolant System Pressure	psia	2180*
Initial RCS Vessel Flow Rate	gpm	355,000*
Moderator Temperature Coefficient	$x10^{-4}\Delta\rho/°F$	+0.3
Doppler Coefficient	---	Cycle Minimum
CEA Worth at Trip	$\%\Delta\rho$	-5.4
Reactivity Insertion Rate	$x10^{-4}\Delta\rho/sec$	0 to 1.95
Rod Group Withdrawal Speed	in/min	30
Maximum CEA Differential Worth	$x10^{-4}\Delta\rho/inch$	3.9

* For the peak RCS pressure case, instrumentation uncertainties are included in the initial conditions. For determination of input to DNB and LHR limits, uncertainties are not included in the transient simulation and are statistically combined. See Table 5.0-2 for nominal values of key parameters.

** For the peak RCS pressure case, the initial core inlet temperature was not reduced even though the maximum HFP core inlet temperature has decreased to 549°F and the uncertainty on initial core inlet temperature has decreased from ±4°F to ±3°F. For determination of input to DNB and LHR limits, uncertainties are not included in the transient simulation and are statistically combined. See Table 5.0-2 for nominal values of key parameters.

Table 5.4.2-2
Sequence of Events for the
Hot Full Power CEA Withdrawal Event

Time (sec)	Event	Setpoint or Value
0.0	CEA Withdrawal Causes Uncontrolled Reactivity Insertion	----
49.9	High Pressurizer Pressure Trip Signal Generated (1)	2429 psia
50.7	High Power Trip Signal Generated	113.5% of 2700 MWt
51.1	Trip Breakers Open	----
51.7	Minimum CE-1 DNBR	> 1.28 SAFDL
51.8	CEAs Begin to Drop Into Core	----
51.9	Core Power Reaches Maximum	113.7% of 2700 MWt
52.1	Core Heat Flux Reaches Maximum	112.8% of 2700 MWt
53.7	RCS Pressure Reaches Maximum	≤ 2750 psia (2)
58.1	Steam Generator Pressure Reaches Maximum	≤ 1100 psia

(1) The analysis was adjusted so the HPPT signal occurred prior to the VHPT signal but was suppressed.

(2) Pressure includes pump and elevation head.

5.4.3 CEA Drop Event

5.4.3.1 Changes to the CEA DROP

the cea drop Event provided in the St. Lucie 2 UFSAR Section 15.4.2.3.8, was evaluated to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm.

All other input remained the same as that in UFSAR Section 15.4.2.3.8.

5.4.3.2 Results

The evaluation determined that the decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm has no impact on this event initiated from the S LCOs. This is because the DNBR limit is not violated during this event due to the thermal margin preserved by the LCOs. This margin is insensitive to the flow reduction.

5.4.3.3 Conclusion

The conclusions of this analysis remain unchanged from those reported in UFSAR Section 15.4.2.3.8.

5.4.4 CVCS MALFUNCTION (INADVERTENT BORON DILUTION)

This event is analyzed for operational Modes 3, 4, 5 and 6, only. This event was not impacted by a reduction of minimum RCS flow. The current analysis remains valid for this event. Therefore, no evaluation was performed.

5.4.5 STARTUP OF AN INACTIVE REACTOR COOLANT PUMP EVENT

This event was not analyzed because the TS do not permit operation at power (Modes 1 and 2) with less than four reactor coolant pumps. Therefore, no evaluation was performed.

5.4.6 CONTROL ELEMENT ASSEMBLY EJECTION

5.4.6.1 Changes to the Control Element Assembly Ejection

The control element assembly (CEA) ejection analysis provided in the St. Lucie 2 UFSAR Section 15.4.5.1.6 was re-evaluated to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm. In addition, the hot full power (HFP) initial core inlet temperature has been decreased from 554°F to 552°F.

All other input remained the same as that in UFSAR Section 15.4.5.1.6.

5.4.6.2 Results

The decrease in RCS flow by 8000 gpm and the decrease in core inlet temperature by 2? F for HFP event offset each other with respect to the transient thermal margin. There is no change in the results of the HFP CEA ejection. The decrease in RCS flow by 8000 gpm has no impact on the HZP CEA ejection results.

5.4.6.3 Conclusion

The conclusions of this analysis remain unchanged from those reported in UFSAR Section 15.4.5.1.6.

5.5 INCREASE IN REACTOR COOLANT SYSTEM INVENTORY

5.5.1 CVCS MALFUNCTION - PRESSURIZER LEVEL CONTROL SYSTEM (PLCS) MALFUNCTION WITH A SIMULTANEOUS CLOSURE OF THE LETDOWN CONTROL VALVE TO THE ZERO FLOW POSITION

5.5.1.1 Changes to the CVCS Malfunction - Pressurizer Level Control System (PLCS) Malfunction with a simultaneous closure of the letdown control

The CVCS malfunction provided in the St. Lucie 2 UFSAR Section 15.5.3.2.2 was evaluated to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm.

All other input remained the same as that in UFSAR Section 15.5.3.2.2.

5.5.1.2 Results

The evaluation determined that the decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm has negligible impact on this event initiated from within TS LCOs. This analysis is insensitive to RCS vessel flow rate because it is driven by volume changes and initial flow has negligible impact on volume changes caused by letdown or charging. This analysis uses the maximum core coolant temperature as input. Since this value is decreasing, the current value is bounding. No credit for this decrease is assumed.

5.5.1.3 Conclusion

The conclusions of this analysis remain unchanged from those reported in UFSAR Section 15.5.3.2.2.

5.5.2 INADVERTENT OPERATION OF THE ECCS DURING POWER OPERATION

This event during power operation (Mode 1) is caused by a malfunction which results in startup of the safety injection (SI) pumps due to an inadvertent safety injection actuation signal (SIAS). The shutoff head of the SI pumps is much less than the RCS pressure in Mode 1. The impact on initiating of charging pump flow upon SIAS is no more adverse that the CVCS malfunction – PLCS malfunction with simultaneous closure of letdown control valve to the zero flow position event is discussed in Section 5.5.1. Therefore, no evaluation was performed.

5.6 DECREASE IN REACTOR COOLANT SYSTEM INVENTORY

5.6.1 PRESSURIZER PRESSURE DECREASE EVENTS: INADVERTENT OPENING OF THE PRESSURIZER RELIEF VALVES

5.6.1.1 Changes to the Pressurizer Pressure Decrease Events: Inadvertent Opening of the Pressurizer Relief Valves

The pressurizer pressure decrease events: inadvertent opening of the pressurizer relief valves provided in the St. Lucie 2 UFSAR Section 15.6.3.1.6 was reanalyzed to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm. In addition, the initial core inlet temperature has been decreased from 554°F to 552°F. The analysis setpoint for the TM/LP trip was increased from 1799 psia to the value of 1855 psia documented in Table 5.0-3, to remove unnecessary conservatism.

All other input remained the same as that in UFSAR Section 15.6.3.1.6. Table 5.6.1-1 documents the changes in input parameters due to the reduction in minimum RCS flow.

5.6.1.2 Results

The decrease in RCS flow by 8000 gpm to 355,000 gpm has a small effect on the rate of primary system pressure decrease with respect to the transient thermal margin. The decrease in RCS temperature in combination with the higher TM/LP analysis setpoint more than offsets any increase in thermal margin due to the lower initial RCS flow. There is no change in the results of the inadvertent opening of the pressurizer relief valves event.

The results remain unchanged from those reported in the UFSAR. The UFSAR figures for inadvertent opening of the pressurizer relief valves event are unchanged and remain representative of this event.

5.6.1.3 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.6.3.1.6.

Table 5.6.1-1
key parameters assumed For The Inadvertent Opening of The Pressurizer Power Operated Relief Valves Event

Parameter	Units	Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	°F	552
Initial RCS Pressure	psia	2300
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/°F$	-3.2
CEA Worth at Trip	$\%\Delta\rho$	-5.4

5.6.2 SMALL PRIMARY LINE BREAK OUTSIDE CONTAINMENT

5.6.2.1 Changes to the Small Primary Line Break Outside Containment

The small primary line break outside containment provided in the St. Lucie 2 UFSAR Section 15.6.3.1.7 was evaluated to account for a decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm.

All other input remained the same as that in UFSAR Section 15.6.3.1.7.

5.6.2.2 Results

The evaluation determined that the decrease in the minimum RCS flow from 363,000 gpm to 355,000 gpm has negligible impact on this event initiated from within TS LCOs. This analysis is insensitive to RCS vessel flow rate. The analysis uses the maximum core coolant temperature as input. Since this value is decreasing, the current value is bounding. No credit for this decrease is assumed.

5.6.2.3 Conclusion

The conclusions of this analysis remain unchanged from those reported in UFSAR Section 15.6.3.1.7.

5.6.3 STEAM GENERATOR TUBE RUPTURE WITH A CONCURRENT LOSS OF OFFSITE POWER

5.6.3.1 Changes to the Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power

The steam generator tube rupture with a concurrent loss of offsite power event provided in the St. Lucie 2 UFSAR Section 15.6.2.1.7 was evaluated to account for a decrease in the minimum RCS flow from 360,000 gpm to 355,000 gpm.

All other input remained the same as that in UFSAR Section 15.6.2.1.7. Table 5.6.3-1 documents the changes in input parameters due to the reduction in minimum RCS flow.

5.6.3.2 Results

The decrease in the minimum RCS flow to 355,000 gpm is demonstrated by evaluation to have a negligible impact on the calculated radiological doses. In addition, TS leakage of both steam generators was applied to the intact steam generator after operator action at 1800 seconds for the duration of the time period. This conservatism plus the negligible impact of the RCS flow reduction are sufficient to conclude that the results of the analysis described in the UFSAR remain unchanged.

5.6.3.3 Conclusion

The conclusions of this analysis remain unchanged from those documented in UFSAR Section 15.6.2.1.7.

Table 5.6.3-1
Key Parameters Assumed for the Steam Generator Tube Rupture Event With a Loss of Offsite Power

Parameter	Units	Value
Total RCS Power (Core Thermal Power + Pump Heat)	MWt	2774
Initial Core Coolant Inlet Temperature	°F	553
Initial RCS Vessel Flow Rate	gpm	355,000
Initial Steam Generator Pressure	psia	770
Steam Generator U-Tube Break Size	in^2	0.336
CEA Worth at Trip	$\%\Delta\rho$	-5.4
Initial Pressurizer Pressure	psia	2400

5.7 MISCELLANEOUS

5.7.1 ASYMMETRIC STEAM GENERATOR EVENTS

5.7.1.1 Changes to the Asymmetric Steam Generator Transient

the asymmetric steam generator transient event provided in the St. Lucie 2 UFSAR Section 15.2.2.2.7 was evaluated to account for a decrease in RCS flow from 363,000 gpm to 355,000 gpm, a decrease in maximum core inlet temperature from 550°F to 549°F, and a decrease in core inlet temperature uncertainty from ±4°F to ± 3°F. The reduction in the above mentioned temperature parameter values lowers the thermal margin requirements. This effect combined with the unchanged thermal margin requirements resulting from the flow reduction ensures that the current requirements are bounding.

All other input remained the same as that in UFSAR Section 15.2.2.2.7. Table 5.7.1-1 documents the changes in input parameters due to the reduction in minimum RCS flow.

5.7.1.2 Results

The results of this evaluation remain unchanged from those reported in UFSAR Section 15.2.2.2.7.

5.7.1.3 Conclusion

The conclusions of this evaluation remain unchanged from those documented in UFSAR Section 15.2.2.2.7.

Table 5.7.1-1
Key Parameters Assumed for the Analysis of the Loss of Load to One Steam Generator Event

Parameter	Units	Value
Total RCS Power (Core Thermal + Pump Heat)	MWt	2774
Initial Core Inlet Temperature	°F	552
Initial Reactor Coolant System Pressure	psia	2180
Moderator Temperature Coefficient	x 10-4 $\Delta\rho$/°F	-3.2
Doppler Coefficient	---	most negative

6.0 References

6.1 St. Lucie Unit 2 Updated Final Safety Analysis Report, Amendment 14, Docket No. 50-389.

6.2 Letter K. N. Jabbour (USNRC) to T. F. Plunkett (FPL), "St. Lucie Unit 2 – Issuance of Amendment Regarding the Cycle 12 Reload Process Improvement (TAC No. MA4523)," December 21, 1999 (Amended in letter dated March 1, 2000)