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

November 28, 2000

L-2000-224
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 - Unreviewed Safety Question
Revised Post-Trip Steam Line Break Analysis

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating License NFP-16 for St. Lucie Unit 2. FPL proposes to amend the St. Lucie Unit 2 post-trip steam line break (SLB) analysis. The design basis for the current analysis of record ensures that no fuel failure will occur for all post-trip SLB cases. The revised analysis supports a change to the fuel failure criterion, to limit fuel failure to less than or equal to 2%, as compared to the current analysis limit of no fuel failure. The analysis continues to meet the 10 CFR 100 dose criteria previously addressed in the FPL submittal supporting License Amendment 105. The change in allowed fuel failure fraction provides additional flexibility in the core design. There are no changes to the St. Lucie Unit 2 Technical Specifications due to the proposed change to the post-trip SLB analysis.

Attachment 1 is a Safety Analysis in support of the proposed amendment. Attachment 2 is the Determination of No Significant Hazards Consideration.

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.

A001

The proposed license amendment, if approved, is requested to be issued by May 15, 2001, to support planning and analysis for the Unit 2 fall 2001 refueling outage (SL2-13).

Please contact us if there are any questions about this submittal.

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/GRM

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. William A. Passetti, Florida Department of Health and Rehabilitative Services

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

Rajiv S. Kundalkar 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.

Rajiv S. Kundalkar

STATE OF FLORIDA

COUNTY OF ST. LUCIE

Sworn to and subscribed before me

this 28 day of November, 2000
by Rajiv S. Kundalkar, who is personally known to me.

Name of Notary Public - State of Florida

Leslie J. Whitwell
MY COMMISSION # CC646183 EXPIRES
May 12, 2001
BONDED THRU TROY FAIN INSURANCE, INC.

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

ATTACHMENT 1

SAFETY ANALYSIS

Introduction

Florida Power & Light Company (FPL) proposes to amend the St. Lucie Unit 2 post-trip steam line break (SLB) analysis. The design basis for the current analysis of record ensures that no fuel failure will occur for all post-trip SLB cases. The revised analysis supports a change to the fuel failure criterion, to limit fuel failure to less than or equal to 2%, as compared to the current analysis limit of no fuel failure. The analysis continues to meet the 10 CFR 100 dose criteria previously addressed in the FPL submittal supporting License Amendment 105. The change in allowed fuel failure fraction results in a shutdown margin benefit and provides additional flexibility in the core design. There are no changes to the St. Lucie Unit 2 Technical Specifications due to the proposed change to the post-trip SLB analysis. The proposed change continues to support the current cycle (Cycle 12) operation of St. Lucie Unit 2.

Description of Proposed Change

The proposed change is a revision to the post-trip SLB analysis. The revised analysis limits the fuel failure to less than or equal to 2%, which ensures a coolable geometry and meets the 10 CFR 100 dose acceptance criteria.

Basis for Proposed Change/Analysis of Impact on Safety

The proposed change, resulting in a fuel failure limit of less than or equal to 2% for the post-trip SLB event, is supported by the analysis described in this section.

Identification of Causes

The hot full power (HFP) and hot zero power (HZP) SLB events with and without loss of off-site power (LOOP) were analyzed to determine the maximum post-trip return-to-power and reactivity for each case. The worst single failure considered is the failure of one high pressure safety injection (HPSI) pump to start.

A break in the main steam system piping increases the rate of heat extraction by the steam generators and causes cooldown of the reactor coolant system (RCS). With a negative moderator temperature coefficient of reactivity, the cooldown will produce a positive reactivity addition. If the break occurs between the steam generators and the main steam isolation valve (MSIV), blowdown of the affected steam generator continues after main steam line isolation. Flow from the intact steam generator is terminated with closure of the MSIVs, either of which is capable of terminating flow from the intact steam generator.

Analysis Methodology

Plant response was simulated using the CESEC computer code (Reference 3). Consistent with the Reference 1 analysis, the post-trip SLB analysis assumed negative reactivity credit due to the local heat up of the inlet fluid in the hot channel, which occurs near the location of the stuck control element assembly (CEA). This credit is based on three-dimensional coupled neutronic-thermal-hydraulic calculations performed with the HERMITE/TORC code (References 4 and 5).

Simulation of the fluid conditions within the hot channel of the reactor core and prediction of departure from nucleate boiling (DNB) was performed using the CETOP-D and TORC computer codes (References 4, 6 and 7). Determination of DNB ratio (DNBR) for the post-trip return to power portion of the steam piping failure events is based on the correlation developed by R. V. McBeth (Reference 8) with corrections developed by Lee, consistent with the methodology employed in the Reference 1 and the Cycle 1 analysis.

Analysis of Effects and Consequences

The transient response of the post-trip SLB Events, described below, is also documented in Reference 1. The analysis includes HFP and HZP events with and without LOOP.

The HFP case was initiated at the conditions listed in Table 1. The following considerations were also included (applies to both HFP and HZP analysis):

a) A reactor trip is initiated by low steam generator pressure.

b) A safety injection actuation signal (SIAS) is actuated when the pressurizer pressure drops below the setpoint. Time delays associated with the safety injection pump acceleration, valve opening, and flushing of the unborated safety injection lines are accounted for.

c) The cooldown of the RCS is terminated when the affected steam generator blows dry. As the coolant temperatures begin increasing, positive reactivity insertion from moderator reactivity feedback decreases. The decrease in moderator reactivity combined with the negative reactivity inserted via boron injection cause the total reactivity to become more negative.

d) The time of LOOP, when it occurred, was assumed to be simultaneous with the main steam isolation signal (MSIS). This assumption results in the coastdown of the main feedwater pumps concurrent with the main feedwater isolation valve closure. The combination of reactor trip and delayed feedwater isolation results in a severe cooldown of the RCS. For the cases analyzed, if the LOOP is assumed at the time of the reactor trip, there are no significant changes in the results since the MSIS and the reactor trip are within 1 second of each other.

e) The moderator temperature coefficient (MTC) of reactivity assumed in the analysis corresponds to the most negative value allowed by the *Core Operating Limits Report* (COLR). This negative MTC results in the greatest positive reactivity addition during the RCS cooldown caused by the SLB. Since the reactivity change associated with moderator feedback varies significantly over the moderator density covered in the analysis, a curve of reactivity insertion versus density rather than a single value of MTC is assumed in the analysis. The moderator cooldown reactivity used in the analysis was conservatively calculated assuming that on reactor trip, the highest worth control element assembly is stuck in the fully withdrawn position.

The reactivity defect associated with fuel temperature decrease is also based on the most negative fuel temperature coefficient (FTC). This FTC, in conjunction with the decreasing fuel temperatures, causes the greatest positive reactivity insertion during the SLB event. The β fraction assumed is the maximum value including uncertainties for end-of-cycle conditions. This too, is conservative since it maximizes subcritical multiplication and thus, enhances the potential for post-trip return-to-power (RTP).

The minimum HFP CEA worth assumed to be available for shutdown at the time of reactor trip at the maximum allowed power level is -7.3%$\Delta\rho$. This available SCRAM worth corresponds with the moderator cooldown curve and stuck rod worth used in the analysis.

The analysis assumed that, on a safety injection actuation signal, one high pressure safety injection pump fails to start. Maximum inverse boron worth of 115 ppm/%$\Delta\rho$ was conservatively assumed for safety injection.

For larger break sizes, increased mass flow rate results in more heat removal from the RCS (all mass leaving the break is conservatively assumed to be in the pure steam phase). Results show that a break size of 6.358 ft^2 is the limiting HFP SLB event. Since the post-trip analysis does not credit the full array of RPS trips (e.g., high containment pressure) there is an initial power increase. The impact of potential SLBs on the pre-trip power increase are explicitly analyzed as documented in Sections 8.1.5a and 8.1.5b of Attachment 2 of Enclosure to L-98-308 (Reference 1).

The reactor trips on low steam generator pressure at 540 psia. Auxiliary feedwater flow is assumed to enter each steam generator 120 seconds after AFAS on each generator. The affected steam generator will be identified as ruptured, (by the auxiliary feedwater actuation system - AFAS) and isolated 120 seconds after the steam generator differential pressure setpoint for auxiliary feedwater isolation is reached. In the events analyzed, no auxiliary feedwater enters the affected steam generator, as the AFW isolation signal occurs early in the transient and before the initiation of AFW. No AFW enters the intact steam generator prior to the time of maximum return to power.

The HZP case was initiated at the conditions given in Table 1. The cooldown curve corresponds to the most negative MTC allowed by the COLR. The most negative FTC was also used for the reasons previously discussed.

The minimum CEA shutdown worth available is conservatively assumed a minimum value of -3.6%$\Delta\rho$. The analysis also assumed that, on safety injection actuation signal, one high pressure safety injection pump fails to start. Maximum inverse boron worth of 110 ppm/%$\Delta\rho$ was conservatively assumed for the safety injection during the HZP case. In the events analyzed, no auxiliary feedwater enters the affected steam generator as the AFW isolation signal occurs early in the transient and before the initiation of AFW. There is no actuation of AFW for the intact steam generator prior to the time of maximum return to power.

Radiological consequences for the RTP SLB event for St. Lucie Unit 2 have been calculated to infer the allowed fuel failure fraction from the 2-hour and 8-hour 10 CFR 100 dose limits and are consistent with the results presented in Reference 1. Releases were calculated based on fuel that violates Centerline-Melt (CTM) criteria and produced fuel failure limits of 13.5% for inside containment SLB and 3.4% fuel failure for an outside containment SLB.

These fuel failure values represent an upper bound limit corresponding to the 10 CFR 100 dose criteria. A conservative value of 2% fuel failures (from violation of CTM and/or DNBR specified acceptable fuel design limits - SAFDL) will be utilized as a cycle specific limit for post-trip SLB. The peak power density during the post-trip SLB also will be limited to less than or equal to 30 kW/ft. For each fuel cycle core design, these limits will be verified based on the calculated physics data for that cycle.

The limit of 2% fuel failures, in conjunction with the 30 kW/ft on peak power density, ensures a coolable geometry during and subsequent to the post-trip SLB return-to-power. Although the peak power fuel rods during the post-trip SLB return-to-power are actually expected to be relatively low burnup/high reactivity fuel, it is assumed that any failed rods are high burnup rods, and that these fuel rods operate with an internal pressure exceeding system pressure. Since fuel failure is assumed when either the CTM criterion is violated or when the McBeth DNBR is below its SAFDL, additional fuel failures due to DNB propagation may be postulated. For the purpose of the propagation evaluation, it was assumed that a fuel rod in DNB would expand asymmetrically towards an adjacent fuel rod. The adjacent fuel rod would, because of the non-uniform heating, balloon into the subchannel of the original ballooned rod and fail due to DNB. However, propagation would stop with only these rods being affected. High power densities during the post-trip SLB return-to-power are limited to the region beneath the stuck rod. Propagation is consequently limited to fuel rods adjacent to this high power density region. The number of such adjacent fuel rods was estimated to be less that 0.5% of the fuel rods in the core. The total number of fuel failures (2.5% when propagation is included) remains well below the limits established for acceptable radiological consequences (13.5% and 3.4% for inside and outside containment breaks, respectively).

Results

Table 2 presents the sequence of events for the HFP without LOOP case initiated at the conditions given in Table 1. The results of the transient show that a reactor trip signal on low steam generator pressure is generated at 3.5 seconds. Low steam generator pressure will result in generation of MSIS at 3.8 seconds. Main steam isolation and main feedwater isolation valves begin to close as a result of MSIS generation and are completely closed following the maximum allowed Technical Specification delay times.

The results of the analysis show that SIAS is actuated at 20.0 seconds. The affected steam generator blows dry at 48.9 seconds and terminates the cooldown of the RCS. The peak reactivity attained is -0.617%$\Delta\rho$ at 53.9 seconds. A peak post-trip total power (fission and decay) of 10.0% is produced at 51.8 seconds. The continued production of decay heat from the fuel after termination of blowdown, causes the reactor coolant temperatures to increase. This in turn reduces the magnitude of the positive moderator reactivity inserted and thus the total reactivity becomes more negative.

Table 3 presents the sequence of events for the HFP with LOOP case initiated at the conditions given in Table 1. The results of the transient show that a reactor trip signal on low steam generator pressure is generated at 3.5 seconds. Low steam generator pressure will result in generation of MSIS at 3.8 seconds. Main steam isolation and main feedwater isolation valves begin to close as a result of MSIS generation and are completely closed following the maximum allowed Technical Specification delay times. Occurrence of LOOP at the time of MSIS leads to the coastdown of the main feedwater pumps.

The results of the analysis show that SIAS is actuated at 22.1 seconds. The affected steam generator blows dry at 70.9 seconds and terminates the cooldown of the RCS. The peak reactivity attained is -0.190%$\Delta\rho$ at 123.6 seconds. A peak post-trip total power of 7.2% is produced at 102.6 seconds. The continued production of decay heat from the fuel after termination of blowdown causes the reactor coolant temperatures to increase. This in turn reduces the magnitude of the positive moderator reactivity inserted and thus the total reactivity becomes more negative.

Table 4 presents the sequence of events for the HZP without LOOP case initiated from the conditions given in Table 1. Reactor trip occurs on low steam generator pressure at 3.6 seconds and is followed by MSIS at 3.9 seconds. The closure of the MSIVs is completed following the maximum allowed Technical Specification delay time.

The results of the analysis show that SIAS is actuated at 28.1 seconds. The peak reactivity attained is +0.441%$\Delta\rho$ at 39.9 seconds and leads to a peak post-trip total power of 15.8% at 169.9 seconds. The affected steam generator blows dry at 167.4 seconds.

Table 5 presents the sequence of events for the HZP with LOOP case initiated from the conditions given in Table 1. Reactor trip occurs on low steam generator pressure at 3.6 seconds and is followed by MSIS at 3.9 seconds. The closure of the MSIVs is completed following the maximum allowed Technical Specification delay time.

The results of the analysis show that SIAS is actuated at 35.3 seconds. The peak reactivity attained is +0.415%$\Delta\rho$ at 54.5 seconds and leads to a peak post-trip total power of 8.4% at 368.5 seconds. The affected steam generator blows dry at 334.9 seconds.

The typical response of the plant for power, heat flux, RCS temperature, RCS pressure, steam generator pressure, and reactivity for each of the above cases remains unchanged from that of the analysis documented in Reference 1.

Conclusions

The representative analysis of the steam line rupture event from HFP and HZP conditions with and without LOOP demonstrates that a significant return to power can occur when bounding physics inputs are utilized. The MTC reactivity defect and scram reactivity are the physics parameters determined by the cycle core design that most affect the results of the post-trip SLB event. With the placement of the MTC and scram reactivity requirements into the COLR, FPL limits these parameters on a cycle specific basis such that less than or equal to 2% of the fuel is predicted to exceed either the CTM or the MacBeth DNB criteria during the post-trip SLB event and that 30 kW/ft peak power density is not exceeded. The limit on fuel failure, in conjunction with the limit on peak power density ensures that coolable geometry is maintained. This fuel failure limit, along with a conservative allowance for DNB propagation failures, remains well below the upper bound limits of 13.5% and 3.4% fuel failure for the inside and the outside containment breaks, respectively, corresponding to the 10 CFR 100 dose criteria.

References

1. Letter L-98-308, J. A. Stall (FPL) to Document Control Desk (USNRC), *St. Lucie Unit 2 Docket No. 50-389: 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, as corrected in Letter dated March 1, 2000.
3. Letter, A. E. Scherer, Enclosure 1-P to LD-82-001, *CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System*, December 1981.
4. CENPD-161-P-A, *TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core*, April 1986.
5. CENPD-188-A, *HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients*, July 1976.

6. CENPD-206-P-A, *TORC Code, Verification and Simplified Modeling Methods*, June 1981.
7. CEN-191(B)-P, *CETOP-D Code, Structure and Modeling Methods for Calvert Cliffs Units 1 and 2*, December 1981.

Table 1
Key Parameters Assumed For the Post-Trip Steam Line Break Event

Parameter & Units	Full Power	Zero Power
Total RCS Power, MWt (102% Core Thermal Power + Pump Heat)	2774 (2754 + 20)	1.0
Initial Core Coolant Inlet Temperature, °F	554	536
Initial RCS Vessel Flow Rate, GPM	363,000	363,000
Initial Reactor Coolant System Pressure, psia	2395	2395
Moderator Temperature Coefficient, $10^{-4}\Delta\rho/°F$	-3.2	-3.2
CEA Worth at Trip, $\%\Delta\rho$	-7.3	-3.6
Inverse Boron Worth, $ppm/\%\Delta\rho$	115	110

Table 2
Sequence of Events for the Post-Trip Steam Line Break Event, Inside Containment, at Hot Full Power, without Loss of Off-Site Power, and with HPSI Pump Failure

Time (s)	Event	Setpoint or Value
0.0	Guillotine break of main steam line occurs inside containment	6.358 ft^2
3.50	Low Steam Generator Pressure Trip Signal setpoint is reached	540 psia
3.80	Main Steam Isolation Signal is generated	520 psia
4.65	Reactor trips	
4.95	Main Steam and Feedwater Isolation Valves begin to close	
5.45	CEAs drop into core	
8.95	Main Feedwater Isolation Valves are fully closed	
10.55	Main Steam Isolation Valves are fully closed	
11.47	Steam Generator Differential Pressure setpoint is reached for AFW isolation of the affected steam generator	
19.62	Pressurizer empties	
19.97	Safety Injection Actuation Signal generated on low pressurizer pressure	1578 psia
48.91	Ruptured steam generator empties	< 5000 lbm
49.97	High Pressure Safety Injection Pump reaches full speed	
51.80	Maximum Return to Power	10.0% of 2700 MWt
53.85	Maximum post-trip reactivity	-0.6171%$\Delta\rho$
104.30	Safety injection, boron enters the core	

Table 3
Sequence of Events for the Post-Trip Steam Line Break Event, Inside Containment, at Hot Full Power, with Loss of Off-Site Power and HPSI Pump Failure

Time (s)	Event	Setpoint or Value
0.0	Guillotine break of main steam line occurs inside containment	6.358 ft^2
3.50	Low Steam Generator Pressure Trip Signal setpoint is reached	540 psia
3.80	Main Steam Isolation Signal is generated Loss of Off-Site Power occurs	520 psia
4.65	Reactor trips	
4.95	Main Steam and Feedwater Isolation Valves begin to close	
5.45	CEAs drop into core	
8.95	Main Feedwater Isolation Valves are fully closed	
10.55	Main Steam Isolation Valves are fully closed	
11.34	Steam Generator Differential Pressure setpoint is reached for AFW isolation of the affected steam generator	
21.99	Pressurizer empties	
22.12	Safety Injection Actuation Signal generated on low pressurizer pressure	1578 psia
52.22	High Pressure Safety Injection Pump reaches full speed	
70.90	Ruptured steam generator empties	< 5000 lbm
102.60	Maximum Return to Power	7.2% of 2700 MWt
123.60	Maximum post-trip reactivity	-0.1902%$\Delta\rho$
213.50	Safety injection, boron enters the core	

Table 4
Sequence of Events for the Post-Trip Steam Line Break Event, Inside Containment, at Hot Zero Power, without Loss of Off-Site Power, and with HPSI Pump Failure

Time (s)	Event	Setpoint or Value
0.0	Guillotine break of main steam line occurs inside containment	6.358 ft^2
3.60	Low Steam Generator Pressure Trip Signal setpoint is reached	540 psia
3.90	Main Steam Isolation Signal is generated	520 psia
4.75	Reactor trips	
5.05	Main Steam and Feedwater Isolation Valves begin to close	
5.55	CEAs drop into core	
9.05	Main Feedwater Isolation Valves are fully closed	
10.65	Main Steam Isolation Valves are fully closed	
28.12	Safety Injection Actuation Signal generated on low pressurizer pressure	1578 psia
28.82	Pressurizer empties	
39.91	Maximum post-trip reactivity	0.4407%$\Delta\rho$
58.12	High Pressure Safety Injection Pump reaches full speed	
82.65	Safety injection, boron enters the core	
167.40	Ruptured steam generator empties	< 5000 lbm
169.90	Maximum Return to Power	15.8% of 2700 MWt
174.20	Reactor subcriticality reestablished	< 0%$\Delta\rho$

Table 5
Sequence of Events for the Post-Trip Steam Line Break Event, Inside Containment, at Hot Zero Power, with Loss of Off-Site Power and HPSI Pump Failure

Time (s)	Event	Setpoint or Value
0.0	Guillotine break of main steam line occurs inside containment	6.358 ft^2
3.60	Low Steam Generator Pressure Trip Signal setpoint is reached	540 psia
3.90	Main Steam Isolation Signal is generated, Loss of Off-Site Power occurs	520 psia
4.75	Reactor trips	
5.05	Main Steam and Feedwater Isolation Valves begin to close	
5.55	CEAs drop into core	
9.05	Main Feedwater Isolation Valves are fully closed	
10.65	Main Steam Isolation Valves are fully closed	
35.28	Safety Injection Actuation Signal generated on low pressurizer pressure	1578 psia
37.08	Pressurizer empties	
54.50	Maximum post-trip reactivity	0.4151%$\Delta\rho$
65.28	High Pressure Safety Injection Pump reaches full speed	
93.25	Safety injection, boron enters the core	
334.90	Ruptured steam generator empties	< 5000 lbm
368.50	Maximum Return to Power	8.4% of 2700 MWt
376.10	Reactor subcriticality reestablished	< 0%$\Delta\rho$

ATTACHMENT 2
DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Florida Power & Light Company (FPL) proposes to amend the St. Lucie Unit 2 post-trip steam line break (SLB) analysis. The design basis for the current analysis of record ensures that no fuel failure will occur for all post-trip SLB cases. The revised analysis supports a change to the fuel failure criterion, to limit fuel failure to less than or equal to 2%, as compared to the current analysis limit of no fuel failure. The analysis continues to meet the 10 CFR 100 dose criteria previously addressed in the FPL submittal supporting License Amendment 105. The change in fuel failure fraction provides additional flexibility in the core design. There are no changes to the St. Lucie Unit 2 Technical Specifications due to the proposed change to the post-trip SLB analysis. The proposed change continues to support the current cycle (Cycle 12) operation of St. Lucie Unit 2.

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 revises the post-trip SLB analysis of record to support a fuel failure limit of 2% as compared to the current criterion of no fuel failure. Post-trip SLB is a current design basis event for St. Lucie Unit 2 and is defined in the St. Lucie Unit 2 Updated Final Safety Analysis Report (UFSAR). The revision to the analysis does not impact the event initiator and requires no change to any plant component or system. The plant configuration remains unchanged, and thus the probability of occurrence of previously analyzed accidents is not affected by the proposed change.

Radiological consequences for the return to power (RTP) SLB event for St. Lucie Unit 2 have been calculated to infer the allowed fuel failure fraction from the 2-hour and 8-hour 10 CFR 100 dose limits and are consistent with the results presented in License Amendment 105. Releases were calculated based on fuel that violates Centerline-Melt (CTM) criteria and produced fuel failure limits of 13.5% for inside containment SLB and 3.4% fuel failure for an outside containment SLB.

These fuel failure values represent an upper bound limit corresponding to the 10 CFR 100 dose criteria. A conservative value of 2% fuel failures (from violation of CTM and/or departure nucleate boiling ratio (DNBR) specified acceptable fuel design limits

(SAFDL)) will be utilized as a cycle specific limit for post-trip SLB. The peak power density during the post-trip SLB also will be limited to less than or equal to 30 kW/ft. For each fuel cycle core design, these limits will be verified based on the calculated physics data for that cycle.

The limit of 2% fuel failures, in conjunction with the 30 kW/ft on peak power density, ensures a coolable geometry during and subsequent to the post-trip SLB RTP. This fuel failure limit, along with a conservative allowance for DNB propagation failures, remains well below the upper bound limits of 13.5% and 3.4% fuel failure for the inside and the outside containment breaks, respectively, corresponding to the 10 CFR 100 dose criteria.

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) Use of the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed amendment is merely a revision to the post-trip SLB event analysis, which continues to meet the applicable limits of 10 CFR 100 dose criteria. There is no change to the plant configuration, systems, or components that would create new failure modes. The modes of operation of the plant 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) Use of the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendment revises the post-trip SLB analysis and supports a change to the fuel failure acceptance criterion. The revised analysis, with the limit of 2% fuel failure, would continue to provide margin to the applicable limits of 10 CFR 100 dose criteria. The proposed change, including any core design variations, will have no adverse impact on other plant safety analysis. The plant operation would continue to remain within all design basis requirements, which would ensure that a safety margin to the acceptance criteria would continue to remain available during plant operation at all power levels.

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 probability 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.