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

March 1, 2000

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

SUBJECT: ST. LUCIE, UNIT 2 - CORRECTION TO WITHDRAWAL OF AMENDMENT REQUEST AND ISSUANCE OF REVISED BASES (TAC NO. MA5619) AND CORRECTION TO SAFETY EVALUATION ISSUED AS PART OF A LICENSE AMENDMENT REGARDING THE CYCLE 12 RELOAD PROCESS IMPROVEMENT (TAC NO. MA4523)

Dear Mr. Plunkett:

The U.S. Nuclear Regulatory Commission (NRC) issued a letter documenting your withdrawal of a license amendment request and issuance of revised technical specification (TS) bases for the St. Lucie Plant, Unit No. 2, on February 17, 2000. The second paragraph of this letter incorrectly stated the date of your request for the TS bases change. The correct date is December 13, 1999.

NRC issued Amendment No. 105 to Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2, on December 21, 1999. This amendment consisted of changes to the TS in response to your application dated December 18, 1998, as supplemented September 13, 1999.

The amendment revised the St. Lucie, Unit 2, TS Index Page III; TS 1.10, Dose Equivalent I-131; TS 2.1.1.2, Linear Heat Rate; TS 3.1.1.1/4.1.1.1.1, Shutdown Margin - T_{avg} Greater than 200°F; TS 3/4.1.1.2, Shutdown Margin - T_{avg} Less Than or Equal to 200°F; TS 3.1.2.2, Boration Systems Flow Paths - Operating; TS 3.1.2.4, Charging Pumps - Operating; TS 3.1.2.6, Boric Acid Makeup Pumps - Operating; TS 3.1.2.8, Borated Water Sources - Operating; and TS 6.9.1.11, Core Operating Limits Report (COLR). The amendment also relocated the core operating limits for shutdown margin to the St. Lucie, Unit 2, COLR. In addition, several TS bases sections were changed.

In the safety evaluation related to this amendment, the first paragraph of page 9 (the final paragraph of Section 2.3.3.2, "Steam System Piping Failures") reads:

> The results of analyses show that the calculated DNBRs for all cases are above the safety limit DNBRs and, thus ensure that no fuel failure will occur. Since assumptions used in the analyses are conservative and the results of the analyses show that the minimum calculated DNBRs are greater than the acceptable safety limits, satisfying SRP 15.1.5 with respect to the fuel failure criteria, the staff concludes that the SLB analyses are acceptable.

DF01

This paragraph states that no fuel failure will occur, which is inconsistent with the licensee's submittals and the staff's evaluation. This paragraph has been changed as follows:

> The results of analyses show that the calculated DNBRs for all post-trip cases are above the safety limit DNBRs and thus ensure that no fuel failure will occur. For pre-trip cases, the calculated percentage of failed fuel rods is small. Since assumptions used in the analyses are conservative and the results of the analyses show that the ~~minimum DNBRs are greater than the acceptable safety limits, satisfying SRP 15.1.5. with respect to the fuel failure criteria~~ maximum calculated percentage of failed fuel rods is small and within the limit used to calculate the radiological consequences that satisfy the dose release limits applicable to the steam line break, the staff concludes that the SLB analyses are acceptable.

The last paragraph in Section 2.3.3.6, "Reactor Coolant Pump Shaft Seizure/Sheared Event," on page 11 of the safety evaluation reads:

> Since the approved method is used to assess the fuel performance, and the results show that no DNB occurs during the transient, satisfying SRP 15.3.3 with respect to the fuel failure criteria, the staff concludes that the reanalysis is acceptable.

The statement regarding fuel failure criteria is inconsistent with the licensee's submittals and the staff's evaluation. This paragraph has been changed as follows:

> Since the approved method is used to assess the fuel performance and the results show that ~~no DNBR occur during the transient, satisfying SRP 15.3.3 with respect to the fuel failure criteria~~ the calculated percentage of failed fuel rods is small and within the limit used to calculate the radiological consequences that satisfy the dose release limits applicable to the RCP shaft seizure event, the staff concludes that the reanalysis is acceptable.

The title of Table 2.4 (page 17 of the safety evaluation) is incorrectly stated as "Licensee Calculated Offsite Dose Due To A Feedwater Line Break With Loss of AC." This title has been changed to read "Licensee Calculated Offsite Dose Due To A SGTR With Concurrent Loss of Offsite Power."

A copy of the revised safety evaluation is enclosed. These changes do not affect the conclusions reached by the staff regarding the acceptability of the TS changes as proposed in your submittals.

Please contact me if you have any questions regarding this matter.

Sincerely,

Allen G. Hansen for

Kahtan N. Jabbour, Sr. Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosure: Revised Safety Evaluation

cc w/enclosure: See next page

Template NRR-056

Please contact me if you have any questions regarding this matter.

Sincerely,

/RA by A. Hansen for/

Kahtan N. Jabbour, Sr. Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosure: Safety Evaluation

cc w/enclosure: See next page

DISTRIBUTION:

File Center	**KJabbour**	**LKopp**	**OGC**
PUBLIC	**OGC**	**MHart**	**ACRS**
ST. Lucie Reading	**GHill (2)**	**KKavanagh**	**LWert, RII**
HBerkow	**WBeckner**	**JWermiel**	**AHansen**
RCorreia	**FOrr**	**JHannon**	
BClayton	**SSun**	**RBarrett**	

DOCUMENT NAME: G:\PDII-2\St. Lucie\ma4523unit2amdcorrection.wpd

OFFICE	PDII-2/PM	PDII-2/LA	SRXB/BC	PDII-2/SC
NAME	KJabbour AH for	BClayton	JWermiel	RCorreia
DATE	2/23/00	2/28/00	2/23/00	3/ /00

OFFICIAL RECORD COPY

Mr. T. F. Plunkett
Florida Power and Light Company

ST. LUCIE PLANT

cc:
Senior Resident Inspector
St. Lucie Plant
U.S. Nuclear Regulatory Commission
P.O. Box 6090
Jensen Beach, Florida 34957

Joe Myers, Director
Division of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

M. S. Ross, Attorney
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. Douglas Anderson
County Administrator
St. Lucie County
2300 Virginia Avenue
Fort Pierce, Florida 34982

Mr. William A. Passetti, Chief
Department of Health
Bureau of Radiation Control
2020 Capital Circle, SE, Bin #C21
Tallahassee, Florida 32399-1741

Mr. Rajiv S. Kundalkar
Vice President
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Mr. R. G. West
Plant General Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

E. J. Weinkam
Licensing Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Mr. John Gianfrancesco
Manager, Administrative Support
and Special Projects
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. J. A. Stall
Vice President - Nuclear Engineering
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. J. Kammel
Radiological Emergency
Planning Administrator
Department of Public Safety
6000 SE. Tower Drive
Stuart, Florida 34997

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letters dated December 18, 1998 (Reference 1), as supplemented September 13, 1999, Florida Power & Light Company (FPL) requested changes to the technical specifications (TS) and the reload evaluation process to be implemented for St. Lucie, Unit 2 (SL2), as part of the nuclear fuel fabrication and related services contract between FPL and ABB-Combustion Engineering, Incorporated (ABB-CE). This process is referred to as the reload process improvement (RPI). The safety analyses affected by the RPI process have been reanalyzed, taking into account the bounding physics parameters corresponding to various fuel management schemes, such as 24-month cycles and the use of gadolinia and erbia burnable absorber in the fuel rods (Reference 2). The proposed license amendment would also incorporate the proposed TS revisions and the new analyses of record for certain transients in connection with FPL's Reload Process Improvement program. According to the amendment request, the new analyses are based on a bounding fuel cycle core design which will provide flexibility in the fuel management scheme for any specific reload cycle and will provide enhanced margins for a future reanalysis and plant operations. The amendment request also proposed new dose analyses for certain design basis accidents and a proposed revision to TS 1.10 to change the reference for thyroid dose conversion factors used in Dose Equivalent Iodine-131 calculations from those listed in Table III of U.S. Atomic Energy Commission (AEC) document TID-14844 (Reference 3), to those listed in the International Commission on Radiological Protection (ICRP) Publication 30.

The supplemental September 13, 1999 letter provided additional information that did not change the initial proposed no significant hazards consideration determination, or extend the scope of the amendment request beyond that described in the Federal Register Notice. The staff's evaluation of the proposed TS changes and the RPI follows.

2.0 EVALUATION

2.1 TS Changes

The following TS changes have been proposed.

TS 2.1.1.2: Peak Linear Heat Rate

This TS, specifying the peak linear heat rate (LHR) limit corresponding to the fuel centerline melt (CTM), will be deleted.

The fuel CTM limit is the basis for the peak LHR. Although the proposed change will eliminate the peak LHR from the TS, the UO_2 CTM limit remains applicable as a specified acceptable fuel design limit (SAFDL) and the safety analysis will continue to comply with the fuel CTM acceptance criterion. Therefore, the proposed elimination is acceptable.

TS 3/4.1.1.1: Shutdown Margin - T_{avg} Greater Than 200°F

The shutdown margin limit in this TS will be relocated to the core operating limit report (COLR).

The shutdown margin requirement for T_{avg} greater than 200°F (Modes 1 through 4) during times late in cycle is determined by the results of the steamline break event. Because the scram worth and power distribution may vary substantially from cycle-to-cycle, the shutdown margin requirement may also be cycle-dependent. Therefore, the proposed relocation to the COLR would obviate the need for license amendments and is in accordance with Generic Letter (GL) 88-16, "Guidance for Technical Specification Changes for Cycle-Specific Parameter Limits," October 4, 1988 (Reference 4). The proposed relocation is, therefore, acceptable.

TS3/4.1.1.2: Shutdown Margin - T_{avg} Less Than Or Equal To 200°F

The shutdown margin limit in this TS will be relocated to the COLR.

The shutdown margin requirement for T_{avg} less than or equal to 200°F (Mode 5) may also vary from cycle-to-cycle based on cycle-specific fuel management and design considerations. The proposed change would, therefore, allow for accommodating cycle-to-cycle variations in shutdown margin requirements without the need for a license amendment. The proposed change is in accordance with GL 88-16 and is acceptable.

TS 3.1.2.2, 3.1.2.4, 3.1.2.6, 3.1.2.8: Reactivity Control Systems

The reference to a shutdown margin of at least 3000 pcm in the Action statements of these TS will be changed to refer to the COLR limit.

Since the shutdown margin in these Action statements is the value corresponding to TS 3/4.1.1.2 at 200°F that has been previously approved for relocation to the COLR, deletion of the numerical value and referring to the value in the COLR is an administrative change and is acceptable.

TS 6.9.1.11: Core Operating Limits Report (COLR)

The shutdown margin specifications (TS 3.1.1.1 and 3.1.1.2) that have been approved for relocation to the COLR will be added to the list of COLR specification limits listed in

TS 6.9.1.11.a. In addition, the list of U.S. Nuclear Regulatory Commission (NRC)-approved analytical methods that can be used to determine the COLR parameters will be expanded to include recently NRC-approved reports that are used in the SL2 RPI safety analyses report. The proposed changes are acceptable.

2.2 Analytical Model Changes

In addition to the above TS changes, FPL has also proposed the following changes to the existing safety analysis of record.

2.2.1 Rod Bow Penalty

The rod bow penalty effects will be calculated as described in CEN-289(A)-P (Reference 5). In particular, the rod bowing effects will be extrapolated from the 14x14 channel closure data using the L^2/I dependence, where L is the span length between two adjacent grids and I is the moment of inertia of the fuel rod cladding.

The rod bow penalty used for fuel assembly burnups up to 31,700 megawatt days per metric ton uranium (MWD/MTU) is 1.2 percent on minimum departure from nucleate boiling ratio (DNBR). This penalty is included in the 1.28 DNBR limit. For burnups greater than 31,700 MWD/MTU, sufficient margin is available due to the reduction in radial power peaking to offset potential rod bow penalties. Therefore, the rod bow penalty for 31,700 MWD/MTU is bounding for all assembly burnups expected for the bounding cycle. The staff found the use of the L^2/I dependence acceptable for the Arkansas Nuclear One Unit 2 (ANO2) fuel design. The SL2 fuel design represents less of an extrapolation from the 14x14 design than the ANO2 design because relative to the ANO2 design, it has a longer span length, the same moment of inertia of the cladding, fewer spacer grids, and less differential growth between the fuel rods and the guide tubes. Therefore, since this methodology was previously approved for ANO2 and the SL2 design is conservatively represented by the ANO2 design, the L^2/I dependence in extrapolating from the 14x14 channel closure data is acceptable for determining the rod bow penalty for SL2 fuel.

2.2.2 Use of HERMITE Code in Loss-Of-Flow (LOF) Analysis

The HERMITE computer code (Reference 6) will be used in the one-dimensional mode to simulate the four-pump LOF transient.

In the current LOF analysis, it is assumed that the hot channel normalized heat flux decay is equivalent to the core average normalized heat flux decay, and the axial heat flux distribution is constant in time. The minimum DNBR assumes no decay of the hot channel heat flux. The use of HERMITE would allow the core power in one dimension (axially) to be calculated directly from control element assembly (CEA) position versus time. The hot channel normalized power decay is assumed to be equivalent to the core average normalized power decay prior to the insertion of the CEAs. As the CEAs are inserted into the core, the planar radial peaking factors are increased so that the hot channel power decreases less rapidly than core average power for the rodded planes. The hot bundle and core average axial heat flux distributions are each time-dependent. The calculated minimum DNBR is based on the decay heat flux calculated by HERMITE at the time of minimum DNBR. In each application of the

HERMITE LOF methodology, plant-specific data (e.g., trip setpoints, reactor protection system delay times, holding coil delay times and CEA motion characteristics) will be used to determine the appropriate time dependent core response. The use of HERMITE in one-dimension has been previously approved for use on other plants (Palo Verde, San Onofre, Waterford, and Calvert Cliffs) to simulate the four-pump LOF transient when additional spatial detail is needed to model the core behavior. We find the use of HERMITE in the one-dimensional mode acceptable for analysis of the LOF transient.

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

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

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

2.3 <u>Safety Analyses</u>

2.3.1 Analytical Methods

The analytical methods used for the transient analyses to support the reload application and TS changes are normally reviewed on a generic basis. The methods for the transient analyses include the following computer codes:

CESEC: The CESEC code (Reference 7) provides a simulation of the system response and calculates system parameters such as core power, RCS flow, primary and secondary temperatures and pressures during a transient. The code was previously approved by the NRC for licensing applications.

TORC and CETOP-D: TORC (References 8 and 9) is used to simulate three-dimension fluid conditions within the reactor core. Results from TORC, including the core radial distribution of the relative channel axial flow, are used to calibrate CETOP-D (Reference 9). With transient core heat flux and thermal-hydraulic conditions from CESEC as input, CETOP-D calculates the DNBR using approved critical heat flux correlations. Both TORC and CETOP-D have been reviewed and approved by the NRC for use in the design basis analysis for licensing applications.

STRIKIN-II: The STRIKIN-II code is used in the analyses of the CEA ejection event (Reference 11). This code calculates the clad and fuel temperatures for an average or hot fuel rod. STRIKIN-II was previously approved by the NRC for the licensing calculations.

HERMITE: The HERMITE code (Reference 6) is used to determine the reactor core response during the total loss of reactor coolant pump (RCP) flow event. The application of HERMITE to simulate the total loss of RCP flow event was previously approved by the NRC for similar CE plants. HERMITE/TORC is used to calculate the negative reactivity feedback during the post-trip steamline break (SLB) conditions. The application of HERMITE/TORC for the negative reactivity calculations is consistent with the method used in the original SLB analysis for the Cycle 1 core. Both HERMITE and TORC were previously approved by the NRC for the licensing applications.

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

2.3.2 Safety DNBR Limits

The safety limit of the DNBR has been imposed to assure that there is at least a 95 percent probability at a 95 percent confidence level that the hottest fuel rod in the core does not experience a departure from nucleate boiling (DNB) during either normal operation or an AOO initiated within the limiting conditions for operation. For transient analyses other than the SLB post-trip analysis, the licensee applied the extended statistical combination of uncertainty (ESCU) methodology (Reference 12) and the CE-1 critical heat flux (CHF) correlation (References 13 and 14) to calculate DNBRs with the safety limit DNBR of 1.28. The 1.28 value incorporates all applicable penalties, including rod bow. As previously described in Section 2.2.1, the rod bow penalty of 1.2 percent on minimum DNBR was calculated for burnups up to 31,700 MWD/MTU. Because of lower radial power peaks in fuel assemblies and rods at burnups above 31,700 MWD/MTU, sufficient margin exists to offset the rod bow penalty at these higher burnups. For the post-trip return-to-power portion of the SLB analysis, the licensee calculated DNBRs using the modified McBeth CHF correlation with the safety limit DNBR of 1.30. Since the ESCU methodology was previously approved by the NRC for the SL2 licensing calculations, and the use of the modified McBeth correlation in the SLB post trip analysis is consistent with that used in the existing SLB analysis, the staff concludes that the use of the approved ESCU methodology and CHF correlations with the associated safety DNBR limits to assess the fuel failure in the SL2 transient analyses are acceptable.

2.3.3 Transient Analyses

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

The AOOs were either reanalyzed or evaluated to assure that SAFDLs for DNB and fuel CTM are not exceeded. The AOOs require either RPS trips or RPS trips and/or sufficient initial steady state margin to prevent exceeding the SAFDLs. The events reanalyzed are:

1. Increased Main Steam Flow
2. Steam System Piping Failures
3. Loss of Condenser Vacuum
4. Feedwater Line Break Events
5. Total Loss Forced Reactor Coolant Flow
6. Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft
7. Chemical and Volume Control System (CVCS) Malfunction (Increase in Reactor Coolant System Inventory)
8. Pressurizer Pressure Decrease Events
9. Steam Generator Tube Rupture
10. Asymmetric Steam Generator Events
11. Uncontrolled CEA Withdrawal from a Subcritical Condition or Low Power
12. Uncontrolled CEA Withdrawal at Power
13. CEA Drop Event
14. CVCS Malfunction (Inadvertent Boron Dilution)
15. Control Element Assembly Ejection

Events were reanalyzed to account for the effects of changes in core physics and plant parameter values intended to bound anticipated changes in future cycles. Major analysis input changes include higher core inlet temperature, increase in primary safety valve tolerance, more negative moderator temperature coefficient (MTC) and lower analysis value for low RCS flow trip setpoint than that used in the existing analysis of record. The values of the core physics and plant parameter assumed in the analyses are summarized in Table 8.0-6 in Reference 2. For the cases analyzed with respect to the RCS pressure and shutdown margin criteria, a combination of initial core and plant parameters with uncertainties is selected to reduce the margin to the safety RCS pressure limits and the shutdown margin requirements. For the cases analyzed with respect to fuel performance where the ESCU methodology is applied, the effects of uncertainties on the initial plant parameters are accounted for statistically. Thus, the values without uncertainties for initial plant parameters (such as the total power, maximum RCS inlet temperature, minimum RCS pressure and minimum reactor vessel flow rate) are used. The following is the staff's evaluation of the results of transient reanalyses.

2.3.3.1 Increase in Steam Flow Event

An increased main steam flow (IMSF) event is a rapid increase in a steam generator (SG) steam flow other than an SLB or an inadvertent opening of a secondary safety valve. Rapid increases in a steam flow result in a decrease in reactor coolant temperature and pressure. In the presence of a negative MTC, the event results in an increase in core power and a reduction in DNBR. During the transient, the RPS (e.g., the high power level and thermal margin/low pressure trips, low SG water level and low SG pressure trips) will trip the reactor to avoid violation of safety DNBR limits. In the analysis credit was taken only for the variable high-power level trip in the determination of the minimum DNBR. The licensee previously analyzed the following IMSF cases: (1) opening of the turbine control valves at the hot full power conditions, (2) opening of the turbine control valves at hot standby conditions and (3) opening of a single valve within either the steam dump and bypass system, or the atmospheric dump system. The results showed that case 1, resulting in the minimum DNBR, is the limiting case. Therefore, the licensee reanalyzed case 1 for reload applications and presented the results of the reanalysis in Section 8.1.3 of Reference 2.

In the analysis, the licensee used the minimum fuel temperature coefficient (FTC). This FTC results in the least amount of negative reactivity addition to mitigate the transient increase in the core heat flux, resulting a minimum DNBR, and is, therefore, a conservative assumption. The pressurizer control system was assumed to be inoperable. This assumption reduces the RCS pressure during the event and, therefore, reduces the calculated DNBR. At time of the trip, the minimum CEA worth corresponding to the full-power conditions was used. The range of the MTCs used in the analysis is shown in Table 8.1.3-1 of Reference 1. The values of the MTCs are within the operating range and are acceptable for use in the analysis. The results of the analysis showed that the reactor trip occurred at 42.3 seconds on a high power level trip signal, and the calculated minimum DNBR was greater than the safety limit DNBR of 1.28.

The analysis uses acceptable values for the core design parameters, and the results of the analysis show that the safety limit DNBR is not violated; therefore, the staff concludes that the analysis has met the acceptance criterion of Standard Review Plan (SRP) 15.1.1 through 15.1.4 (Reference 15) and therefore, the analysis for the IMSF event is acceptable.

2.3.3.2 Steam System Piping Failures

The licensee presented in Section 8.1.5 of Reference 2, the results of reanalyses for six cases of SLB events: two cases analyzed for pre-trip core conditions and four cases analyzed for post-trip conditions. The following are pre-trip SLB cases:

1. an SLB inside containment at full power in combination with a loss of AC power (LOAC) (Section 8.1.5a of Reference 2)
2. an SLB outside containment at full power in combination with an LOAC (Section 8.1.5b of Reference 2)

In the analysis of SLB inside containment, environmental degradation of the sensor input to the delta T power calculator and the pressure measurement sensors was considered. For the pre-trip SLBs, sensitivity studies of the break size and MTC were done to identify the limiting SLB case that resulted in lowest minimum DNBRs. The results showed that the limiting cases were: for case (1), an SLB with maximum power occurring simultaneously with the high containment pressure trip signal and for case (2), an SLB with the maximum heat flux occurring before the high power trip. For both cases, the LOAC was assumed to occur concurrently with the reactor trip breakers opening. The LOAC caused a coastdown of the reactor coolant pump. The analysis showed that no single failure for pre-trip SLB events would increase the amount of calculated fuel failure.

The following list is the balance of the post trip SLB cases studied (the results are presented in Section 8.1.5c of Reference 2):

3. an SLB inside containment at full power with concurrent LOAC in combination with a single failure and a stuck CEA
4. case 3 with AC available
5. an SLB inside containment at zero power with concurrent LOAC in combination with a single failure and a stuck CEA
6. case 5 with AC available

An SLB will result in an increase in steam flow that causes a decrease in the RCS temperature. In the presence of a negative MTC, the decreased RCS temperature results in an addition of positive reactivity. With conservative assumptions regarding MTC and a stuck CEA, the plant system responses after the reactor trip may lead to an increase in core power, or return-to-power (r-t-p) from a previously subcritical state, and may result in highly skewed core power distributions. A combination of a high power, low flow and low RCS pressure may result in fuel failures. According to SRP Section 15.1.5, the input parameters and initial conditions applied for the SLB analysis should be selected such that they maximize the degradation of fuel performance.

The assumptions used in the analysis of cases 3 through 6 included the following:

The CEA with the maximum worth was assumed stuck in the fully withdrawn position after a reactor trip. This assumption is consistent with the General Design Criterion (GDC) 26, which requires that the system be designed with a redundant reactivity control system that is capable for providing appropriate margin for malfunction such as a stuck CEA. Therefore, the assumption is conservative.

The most negative value of MTC allowed by the COLR was used. This assumption provides the greatest potential for r-t-p. Since this assumption maximizes the potential for r-t-p, it is conservative.

Steam was assumed to blow down through the break. This assumption maximizes the severity of the cooldown and is therefore, conservative.

The break was postulated to be a double-ended rupture upstream of the main steam isolation valve. Postulating the break at the location that maximizes the break area is conservative.

For single failure consideration, the failure of one of two available high pressure safety injection pumps was identified as the most limiting single failure. The selection of the most limiting single failure reduces the boron flow injection rate and increases the potential for r-t-p and is conservative.

The LOAC was assumed to occur simultaneously with the main steam isolation signal. This assumption results in the coastdown of the main feedwater pumps concurrent with the main feedwater isolation valve closure. The combination of an early reactor trip time and delayed feedwater isolation results in the most severe cooldown of the RCS and is conservative.

The results of analyses show that the calculated DNBRs for all post-trip cases are above the safety limit DNBRs and thus ensure that no fuel failure will occur. For pre-trip cases, the calculated percentage of failed fuel rods is small. Since assumptions used in the analyses are conservative and the results of the analyses show that the maximum calculated percentage of failed fuel rods is small and within the limit used to calculate the radiological consequences that satisfy the dose release limits applicable to the steam line break, the staff concludes that the SLB analyses are acceptable.

2.3.3.3 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum Events

This category of the events results in a reduction of steam flow to the turbine generator. The loss of steam flow results in a rapid rise in secondary system pressure and temperature and a reduction of the heat transfer rate in the SGs, which in turn causes the RCS pressure and temperature to rise. The events are ended by a reactor trip on high pressurizer pressure. The licensee reanalyzed a loss of condenser vacuum (LOCV), previously identified as the limiting case, resulting in a highest peak RCS pressure and lowest minimum DNBR, for the category of the events.

The results of the LOCV analysis (in Section 8.2.3 of Reference 2) show that decreasing the initial core inlet temperature reduces the initial SG pressure, by delaying the opening of the main steam safety valves and the associated heat removal effects, and results in a higher peak RCS pressure. Thus, the lowest allowable initial core inlet temperature was assumed in the analysis. This assumption will result in a higher RCS pressure and is therefore, conservative.

The initial core average axial power distribution was chosen to be a bottom-peaked shape. This distribution reduces the negative reactivity inserted during the initial portion of the control rod insertion following a reactor trip. This assumption is conservative because it delays a reactor shutdown and thus maximizes RCS pressure.

An LOAC was assumed in the analysis. This assumption is consistent with GDC 17 that requires an LOAC be considered in the analysis of AOOs (such as the LOCV event). The time of an LOAC was selected such that the high pressurizer pressure trip and low coolant flow trip occurred simultaneously. This assumption maximizes the peak RCS pressure and is therefore, conservative. Additionally, 1500 tubes from each SG were assumed to be plugged. This assumption decreases the primary-to-secondary heat transfer and results in higher RCS pressure, and therefore, is conservative.

Based on its review, the staff concludes that the reanalysis of the LOCV is acceptable because conservative assumptions are made and the results show that the peak RCS pressure is less than 110 percent of the design pressure, satisfying the acceptance criteria of SRP Sections 15.2.1 through 15.2.5.

2.3.3.4 Feedwater Line Break

The feedwater line break (FLB) reduces the ability to remove heat generated by the core from the RCS because fluid in the SG is discharged from the break, and the break may be large enough to prevent the addition of main feedwater after the trip. During the event the auxiliary feedwater system (AFS) and pressurizer safety valves will function to prevent substantial overpressurization of the RCS. A reactor trip may be actuated on reactor signals such as the high pressurizer pressure signal.

In Section 8.2.6 of Reference 2, the licensee presented the results of the analyses for FLB events with and without an LOAC. To maximize the calculated peak RCS pressure, the licensee made the following assumptions: (1) the least negative fuel reactivity feedback coefficient for each fuel cycle was used to maximize the power increase, (2) the initial power was assumed at the full power level with measurement uncertainty, (3) at initiation of the FLB, the feedwater flow to both SGs was instantaneously reduced to zero, and (4) AFS was available after sufficient delay time following an auxiliary feedwater actuation signal on low SG water level.

For the FLB with an LOAC, the time of LOAC was selected to coincide with a turbine trip, resulting from a reactor trip. Because of LOAC, the reactor coolant pumps began to coastdown. The emergency diesel generators were assumed to start automatically after the loss of all non-emergency AC power.

The FLB analyses identified that the worst breaks, resulting in a highest peak RCS pressure, were 0.19 and 0.2 ft^2 for cases with and without AC power available, respectively. For the FLB with AC power available, the analysis shows that the peak pressure is within 110 percent of the design pressure in compliance with SRP Section 15.2.8, item II.D.1, which allows RCS pressure up to 110 percent of the design pressure for low probability events. The staff considers an FLB with available AC power is a low probability event. For the FLB with LOAC, the analysis shows that the peak RCS pressure is within 120 percent of the design pressure and shows the compliance of SRP Section 15.2.8, item II.D.1, which limits the RCS pressurization to 120 percent of the design pressure for very low probability events. The staff considers the FLB with LOAC to be a very low probability event. The results of the reanalysis show that the calculated peak pressures are within the acceptable ranges of SRP Section 15.2.8. Therefore, the staff concludes that the analysis is acceptable.

2.3.3.5 Loss of Forced Reactor Coolant Flow Event

This event is initiated by a loss of power supplied to, or a mechanical failure in, the RCS pumps. As a result, the core flow rate will decrease and core temperature will increase. Before the reactor trip, the combination of the decreased RCS flow and increased temperature may result in core conditions that violate the safety DNBR limits. The licensee reanalyzed events involving total loss of RCS flow that leads to a decrease in RCS flow. The partial loss of forced reactor coolant flow, resulting in similar loss in the DNBR margin, is bounded by the total loss of forced reactor coolant flow. The analysis assumed nominal values for the core at full power level, RCS flow, temperature and pressure. For DNBR calculations, effects of uncertainties on the initial plant parameters such as core power, core inlet temperature, RCS pressure, flow and radial peaking factor were combined statistically. The low RCP speed trip was credited to trip the reactor.

The results of the reanalysis (Section 8.3.2 of Reference 2) show that the minimum DNBR is greater than the safety DNBR limit of 1.28. The use of nominal values for initial plant parameters and statistical treatment of the uncertainties associated with initial plant parameters used in DNBR calculations are consistent with the approved ESCU methodology (discussed in Section 2.3.2 of this evaluation). The calculated minimum DNBR is greater than the safety DNBR limit and, therefore, satisfies the acceptance criterion of SRP Sections 15.3.1 and 15.3.2. Therefore, the staff concludes that the reanalysis is acceptable.

2.3.3.6 Reactor Coolant Pump Shaft Seizure/Sheared Event

RCP shaft seizure is caused by an instantaneous seizure of an RCP rotor, and the RCP shaft break is caused by an instantaneous failure of RCP shaft. For both cases, flow through the affected reactor loop drops rapidly, leading to a reactor trip on a low-flow signal. After the reactor trip, energy stored in the fuel rods continues to transfer to the coolant, causing the coolant temperature to increase. The combination of decreased RCS flow and increased RCS temperature may result in low DNBRs and thus, fuel failures.

The case discussed in Section 8.3.3 of Reference 2 is the RCP shaft seizure event. This case bounds the RCP shaft break event because the RCP flow coastdown for the shaft seizure event is faster, resulting in a lower minimum DNBR. An LOAC was assumed to occur following a turbine trip. An LOAC caused a simultaneous loss of feedwater flow, condenser inoperability and coastdown of all RCPs.

In the analysis, an atmospheric dump valve (ADV) was assumed to be stuck-open at initiation of the event and remained open for the first 30 minutes. The stuck-open ADV causes excessive steam to be released to the environment from the SGs. Thus, the failure of an ADV to close in combination with the LOAC maximizes the radiological consequences of the event.

The statistical convolution method (Reference 16) was used to calculate the number of failed fuel rods. This method assigns a probability of DNB occurrence as a function of DNBR. Since the statistical convolution method was previously approved by the NRC for CE 16x16 fuel designs and the CE-1 CHF correlation (Reference 17), the staff concludes that the licensee's use of the convolution method is acceptable.

Since the approved method is used to assess the fuel performance and the results show that the calculated percentage of failed fuel rods is small and within the limit used to calculate the radiological consequences that satisfy the dose release limits applicable to the RCP shaft seizure event, the staff concludes that the reanalysis is acceptable.

2.3.3.7 CVCS Malfunction (Increase in Reactor Coolant System Inventory)

In Section 8.5 of Reference 2, the licensee included the discussion for the following cases that resulted in an increase in RCS inventory:

1. inadvertent operation of the emergency core cooling system (ECCS) during power operation
2. CVCS malfunction

The licensee identified that case 2 is more limiting than case 1. For case 1, the shutoff head of the safety injection pump is less than the RCS pressure during full power conditions, resulting in no injection of fluid into the RCS. In addition, the impact of initiation of charging flow upon the safety injection actuation signal is the same as case 2 (CVCS malfunction). Therefore, the licensee analyzed case 2 for the reload application.

The CVCS malfunction can be caused by operator action, an electrical actuation signal, or valve failure. The analysis assumed that the event was initiated by a CVCS malfunction that resulted in injection from two CVCS pumps. For the single failure consideration, the licensee identified that the limiting failure was the complete closure of the letdown control valve that occurred concurrently with initiation of the event. The effect of this single failure increased the RCS inventory at a faster rate. As the CVCS injection flow increased RCS inventory, pressurizer water volume began increasing. Eventually, the pressurizer water level reached the high-level alarm setpoint. The results of the analysis show that the rate of filling is slow enough that operator action to end the charging-letdown flow imbalance in 20 minutes from the time the high level alarm occurs is sufficient to prevent filling of the pressurizer. Since the analysis for the limiting case shows that sufficient time is available for the operator to end the CVCS imbalance flow to prevent the pressurizer from filling with the water, the staff concludes that the analysis is acceptable.

2.3.3.8 Inadvertent Opening of the Pressurizer Relief Valves

An accidental depressurization of the RCS can occur because of an inadvertent opening of the power operated relief valves (PORV). During the transient, the RCS pressure rapidly decreases and, the pressurizer water also decreases rapidly. The depressurization of the RCS can cause the fuel to approach the safety limit DNBR.

In Section 8.6.1 of Reference 2, the licensee discussed the analytical results of the inadvertent opening of the PORV event. In the analysis, a combination of initial plant conditions of maximum core power, maximum RCS inlet temperature and pressure, and minimum core flow was selected to reduce the calculated DNBRs. A low value of the low pressurizer pressure trip setpoint was chosen to delay the reactor trip. An LOAC was assumed to occur following a turbine trip. The LOAC caused the RCPs to coastdown. A low RCS flow with low RCS pressure resulted in a low DNBR.

The results of the analysis show that the calculated minimum DNBR is greater than the safety limit DNBR and thus, ensure fuel integrity during the transient. Since conservative assumptions are used and the results of the analysis show that the no failure of fuel pins will occur during an inadvertent opening of PORVs, the staff concludes that the analysis meets the acceptance criterion of SRP Section 15.6.1 and is acceptable.

2.3.3.9 Steam Generator Tube Rupture (SGTR)

The SGTR event is defined by a penetration of the barrier between the RCS and the main steam system. This event can be caused by the failure of a SG U-tube.

The licensee analyzed the SGTR event for a case with complete severance of a single SG tube. At initiation of an SGTR, a combination of maximum values of core power, core inlet temperature and RCS pressure was assumed to maximize a flow from the primary to the secondary side through the SG tube break, resulting in high radioactivity in the affected SG. An LOAC was assumed following a turbine trip. Consistent with the assumption of an LOAC, RCS and main feedwater pump coastdown occurred after the reactor trip following a turbine trip. With the availability of stand-by emergency power, auxiliary feedwater (AFW) was automatically initiated on a low SG water level signal. To maximize the offsite dose release, the analysis assumed that the AFW was not available until 30 minutes after the initiation of the event. The low pressurizer pressure trip was credited for a reactor trip, which initiated a turbine trip. Two high pressure safety injection pumps were assumed to be available to maximize the RCS pressure, and consequently, primary to secondary leakage.

Following a turbine trip, the main steam system pressure increased until the main steam safety valves (MSSVs) opened. A maximum SG pressure occurred shortly after the trip. After the SG peak pressure, the pressure decreased and the MSSVs cycled to maintain the secondary pressure. Operator actions were delayed until 30 minuted after the event initiation. The operator isolated the affected SG by closing the main steam isolation valves (MSIVs). The operator then cools down the system by using the atmospheric dump valves and AFW flow to the unaffected SG. After the pressure and temperature were reduced to the entry point of the shutdown cooling system (SDC), the operator started the SDC and isolated the unaffected SG.

Since the SGTR analysis (in Section 8.6.3 of Reference 2) uses conservative assumptions, maximizing the primary-to-secondary leakage, and the results show that the maximum RCS and secondary pressures are within 110 percent of the design pressure, and the minimum calculated DNBR is greater than the safety limit DNBR, satisfying SRP 15.6.3 with respect to the fuel failure criteria, the staff concludes that the SGTR analysis is acceptable.

2.3.3.10 Asymmetric Steam Generator Events

The licensee considered the following asymmetric SG events: (1) loss of load to one SG, (2) an excess load to one SG, (3) loss of feedwater to one SG and (4) excess feedwater to one SG. For the reload application, the licensee reanalyzed case 1 because this case was previously identified as limiting, resulting in a lowest minimum DNBR, and presented the results of the reanalysis in Section 8.7.1 of Reference 2.

The event analyzed was initiated by the inadvertent closure of a single MSIV, which resulted in a loss of load to the affected SG. As a result, the primary loop associated with the closed MSIV experienced a heatup due to loss of heat sinks and the primary loop associated with the open MSIV experienced a cooldown due to the load increase. In the presence of a negative MTC, the radial peaking increased, resulting in a low minimum DNBR.

In the analysis, a combination of maximum initial core power, maximum core inlet temperature and minimum RCS pressure was assumed to reduce the initial DNBR. Only the asymmetric SG pressure trip signal was credited for the reactor trip. In addition, the most negative value of the MTC was used to maximize the calculated severity of the asymmetry.

Since the assumptions used for the input plant parameters result in low calculated DNBRs and are conservative, and the results of the analysis show that the asymmetric SG pressure trip ensures that the safety limit DNBR is not exceeded during a limiting asymmetric SG transient, the staff concludes that the analysis of the closure of a single MSIV event is acceptable.

2.3.3.11 Reactivity and Power Distribution Anomalies

The uncontrolled CEA withdrawal events from subcritical or low power and from power conditions were analyzed to assure that the DNBR and fuel CTM limits are not exceeded. The results indicate that the DNBR and CTM limits are not violated, thereby meeting the acceptance criteria of SRP Sections 15.4.1 and 15.4.2.

A reanalysis of the CEA drop event assured that sufficient initial thermal margin exists such that violation of either the DNBR or fuel CTM limits does not occur, thereby meeting the acceptance criteria of SRP Section 15.4.3.

The CVCS malfunction (inadvertent boron dilution event) was reanalyzed and showed that the setpoints of the startup channel alarms ensure that the criteria for the time between alarms and the loss of shutdown margin (15 minutes for Modes 2 through 5, 30 minutes for Mode 6) are met. Therefore, the acceptance criteria of SRP Section 15.4.6 are met.

The CEA ejection accident was evaluated to assess the impact of fuel design changes on previously calculated results. The limiting full power and zero power cases show that the maximum total energy deposited during the event is less than the criterion for incipient centerline melting and, consequently, no fuel pin failures occur. The boundary doses remain well within 10 CFR Part 100 guidelines meeting the acceptance criteria of SRP Section 15.4.8.

2.3.4 Loss of Coolant Accident (LOCA) Analysis

2.3.4.1 Large Break LOCA (LBLOCA) and Small Break LOCA (SBLOCA) Analyses

In the RPI of December 1998 for SL2, as supplemented by a letter dated September 13, 1999 (Reference 18), FPL proposed to include reference to the ABB-CE SBLOCA evaluation model described in the topical report CENPD-137, Supplement 2-P-A (Reference 19). This topical report has been generically approved as applicable to CE reactor designs. ABB-CE identifies to its licensees those analysis input parameters to which LOCA analysis results are sensitive. The supplemental letter showed that the CENPD-137, Supplement 2-P-A methodology applies to SL2 by describing FPL processes for determining values for analysis input parameters which assure that the input values bound the actual as-operated plant values for peak cladding temperature (PCT)-sensitive parameters.

RPI Section 7 also proposes to include the generically approved June 1985 ABB-CE LBLOCA methodology in the SL2 RPI references. Item B.1 of the supplemental letter discusses the RPI processes for generating LOCA analysis input parameters, and how parameter variances, or instrument drift, uncertainties, inaccuracies, and other variances are accounted for. The supplemental letter identifies that the FPL input value determination processes are also used to specify LBLOCA input values to assure that the input values bound the actual as-operated plant values for PCT-sensitive parameters. Although the licensee states in its letter dated September 13, 1999, "In general, the LOCA analysis for St. Lucie Unit 2 presented in FPL letter L-98-308 has been performed to achieve results that conservatively bound expected consequences for this event," and ". . . the ABB-CE evaluation models used in this analysis provide substantial margins over realistic conditions that are sufficient to bound variances in input parameters including instrument drift, uncertainties and inaccuracies," it is our position that the values of PCT-sensitive parameters input to the LOCA analyses must bound the as-operated plant values for these parameters. Notwithstanding the licensee's contentions, we find the parameter LOCA analysis input value designation processes of the RPI, as described in the supplemental letter, acceptable based on compliance with the staff position.

Based on the generic acceptability and applicability of the SBLOCA and LBLOCA models, and on the licensee's description of plant-specific processes it will use to assure that PCT-sensitive parameter values input to the SBLOCA and LBLOCA models bound the actual as-operated plant values for SL2, we conclude that the SBLOCA and LBLOCA methodologies discussed above apply to SL2. Therefore, these methodologies are acceptable for use in performing SL2 SBLOCA and LBLOCA analyses, and are acceptable for reference in the SL2 RPI, TS and COLR.

2.3.4.2 Post-LOCA Boron Precipitation and Long Term Cooling

Post-LOCA boron precipitation and long-term cooling are currently under generic consideration. The RPI addresses these issues by means which had been approved for previous operation of SL2. The ongoing generic consideration has not provided new plant-specific guidance for SL2 to date. Therefore, the means described in the RPI continue to be acceptable, until such time, if at all, that generic guidance applicable to SL2 is prescribed.

2.4 <u>Review of Bounding Dose Analyses and Thyroid Dose Conversion Factors</u>

The licensee performed several new analyses based on a "bounding fuel cycle" core design. To allow flexibility in future reload core designs, FPL used conservative physics inputs that are expected to bound a wide range of fuel management strategies. This process resulted in the calculated dose consequences of several events being closer to the acceptance criteria than current analyses. For events that are predicted to have fuel failure, the licensee derived acceptable values for fuel failure from the dose acceptance criteria.

The licensee performed new bounding dose consequences analyses for the following accidents:

1. Steam System Piping Failures; Inside and Outside Containment
2. Feedwater Line Break Event: Small Break and Break with Loss of AC
3. Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft
4. Small Primary Line Break Outside Containment
5. Steam Generator Tube Rupture with a Concurrent Loss of Offsite Power

2.4.1 Steam System Piping Failures; Inside and Outside Containment and Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

The licensee performed these analyses to demonstrate the level of acceptable fuel failure associated with maximum allowed offsite doses. The licensee back-calculated the fuel failure from the 10 CFR Part 100 dose acceptance criteria of 25 rem to the whole body or 300 rem to the thyroid from iodine exposure. The staff finds this approach acceptable.

2.4.2 Small Primary Line Break Outside Containment

The licensee evaluated the small primary line break occurring outside containment to ensure the resulting doses are within the SRP acceptance criteria of a small fraction (10 percent) of 10 CFR Part 100. The licensee determined that the current analysis dose results remain valid, since no changes to the sequence of events for the accident are required for the requested amendment. The staff finds this acceptable.

2.4.3 Feedwater Line Break Event: Small Break and Break with Loss of AC

The licensee analyzed feedwater system pipe breaks to ensure that the site boundary doses would not exceed a small fraction (10 percent) of 10 CFR Part 100 acceptance criteria, as given in NUREG-0800, Standard Review Plan. The licensee determined that the feedwater break with loss of AC is the bounding feedwater break event. The staff reviewed the licensee's assumptions used in the dose analyses and determined them to be acceptable. The licensee's calculated offsite doses at the exclusion area boundary (EAB) and Low Population Zone (LPZ) are within the acceptance criteria of 2.5 rem to the whole body or 30 rem to the thyroid from iodine exposure, as shown in Table 2.3.

Table 2.3
Licensee Calculated Offsite Dose Due To
A Feedwater Line Break With Loss of AC

	Dose (rem)		SRP Acceptance
	EAB	LPZ	Criteria (rem)
Thyroid	1.56	0.71	30
Whole Body	<0.002	<0.003	2.5

2.4.4 SGTR with a Concurrent Loss of Offsite Power

The licensee analyzed the SGTR to assure that the offsite dose consequences are within the SRP acceptance criteria. The staff reviewed the licensee's assumptions and determined they are acceptable. The licensee's calculated offsite dose results are within the SRP acceptance criteria as shown in Table 2.4, below. NRC staff performed a confirmatory dose calculation using licensee assumptions and confirmed licensee results. The staff has determined that the licensee's SGTR analysis is acceptable.

Table 2.4
Licensee Calculated Offsite Dose Due To
A SGTR With Concurrent Loss Of Offsite Power

	Dose (rem) EAB	LPZ	SRP Acceptance Criteria (rem)
Thyroid			
PIS*	12.0	6.0	300
GIS*	2.0	8.0	30
Whole Body	0.2	0.1	2.5

* PIS = pre-accident iodine spike, GIS = accident initiated iodine spike

2.4.5 ICRP-30 Dose Conversion Factors

The licensee's justification for making use of ICRP-30 thyroid dose conversion factors in place of those from TID-14844 is that the values in ICRP-30 are more recent and incorporate the considerable advances in the state of knowledge of limits for intakes of radionuclides. The staff has generally accepted the use of ICRP-30 dose conversion factors, and such use is consistent with current industry standards. The licensee stated that the analyses of record for dose consequences use thyroid dose conversion factors from TID-14844. Since TID-14844 thyroid dose conversion factors give conservative dose results as compared to those in ICRP-30, the current dose analyses of record remain bounding and no reanalysis is necessary, assuming the only change is the use of ICRP-30. The licensee may and should use ICRP-30 thyroid dose conversion factors for future reanalyses. The staff finds the licensee's proposed change to TS 1.10 and the use of the ICRP-30 thyroid dose conversion factors to be acceptable.

2.5 Review of Containment Analysis Methodologies

As part of the license amendment request, the licensee proposed to change TS 6.9.1.11, Core Operating Limits Report (COLR). The change incorporates additional topical reports, which would be consistent with the methodology used in the RPI safety analysis report. Two of the fifteen additional reports for inclusion into the TS are related to the analysis of the pressure and temperature of the containment. The two reports are CENPD-140-A (Reference 20), "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," and DP-456 (Reference 21), F.M. Stern (CE) to E. Case (NRC), dated August 19, 1974, Chapter 6, Appendix 6B to CESSAR System 80 PSAR [preliminary safety analysis report].

The licensee stated that the methodology reported in CENPD-140-A will be used to calculate the containment high pressure trip time for SLB analysis. CENPD140-A was found acceptable by the staff, as stated in a letter dated April 11, 1976 (Reference 22). In that letter, the staff concluded the following:

> CENPD-140 provides an acceptable analytical procedure for predicting the containment temperature and pressure transients following a loss-of-coolant accident and main steamline break accident. CENPD-140 may be referenced by application with dry or subatmospheric containment structures. For each plant analyzed, the specific plant dependent input information such as containment initial conditions, active heat removal system operation, and passive heat sink structures should be provided and justified.

Based on this information, the NRC staff finds it acceptable to reference CENPD-140-A in the St. Lucie, Unit 2, TS 6.9.1.11.

The licensee stated that DP-456 is a reference which, "supports the use of the SGN-111 computer code to recalculate the time of Containment High Pressure Trip for Steamline Break. This is an acceptable code for calculating mass and energy releases for Steamline Breaks per SRP 6.2.1.4." The staff notes that SRP 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," provides the acceptance criteria for meeting the requirements of GDC 50 with respect to providing sufficient conservation in the mass and energy release analysis. According to the staff safety evaluation report (SER) NUREG-75/112 (Reference 23), DP-456 was a response to a request for additional information during the review of the CESSAR System 80 PSAR. In the SER, the staff concluded that the SGN-111 code as used in the CESSAR was a conservative method of analysis of secondary system ruptures for the purposes of containment pressure analysis. Based on this information, the staff believes the reference to CENPD-140-A in the St. Lucie, Unit 2, TS 6.9.1.11, is acceptable.

3.0 STAFF CONCLUSIONS

The staff has reviewed the licensee's safety analyses to support the proposed TS changes for operation of fuel cycle 12 and future cycles at the SL2 plant. Based on this review, the staff concludes that the proposed TS changes and supporting safety analyses are acceptable.

NRC staff reviewed the bounding dose analyses done by the licensee to support FPL's Reload Process Improvement Plan, and found them acceptable. The staff also finds the licensee's proposed change to TS 1.10 and the use of the ICRP-30 thyroid dose conversion factors to be acceptable.

The staff has reviewed the FPL's submittal with respect to the inclusion of two references to containment analysis methodologies into the St. Lucie, Unit 2, TS. The staff finds that both reports are acceptable references for the St. Lucie, Unit 2, TS. The staff believes that there is reasonable assurance that plant operation in this manner poses no undue risk to the health and safety of the public.

The staff also found that the supplemental, September 13, 1999, letter provided additional information that did not change the original proposed no significant hazards consideration determination.

4.0 STATE CONSULTATION

By Letter dated March 8, 1991, Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, informed Deborah A. Miller, Licensing Assistant, U.S. NRC, that the State of Florida does not desire notification of issuance of license amendments. Thus, the State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (64 FR 6697). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: F. Orr, S. Sun, L. Kopp, M. Hart, K. Kavanagh, W. Gleaves

Date: December 21, 1999

Revised March 1, 2000

7.0 REFERENCES

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

2. Attachment 2 to Reference 1, "St. Lucie Unit 2 Safety Analysis Report for Initial Application of PAC and NPAC."

3. AEC, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," March 1962.

4. NRC Generic Letter 88-16, "Guidance for Technical Specification Changes for Cycle-Specific Parameter Limits," October 4, 1988.

5. CENPD-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984.

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

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

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

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

10. CEN-191(B)-P, "CETOP Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981.

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

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

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

14. CEN-207-P-A, "Critical Heat Flux Correlations for CE Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Power Distribution," June 1976.

15. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

16. CENPD-183-A, "Loss of Flow: CE Methods for Loss of Flow Analysis," July 1975.

17. NUREG-1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," August 1994.

18. Letter from J.A. Stall (FPL) to NRC, Proposed License Amendment, Cycle 12 Reload Process Improvement, Response to Request for Additional Information, September 13, 1999.

19. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

20. CENPD-1 40-A, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," June 1976.

21. Letter from F.M. Stern, (CE), to E. Case (NRC), "DP-456, Chapter 6, Appendix 6B to CESSAR System 80 PSAR," August 19, 1974.

22. Letter from O.D. Parr, (NRC) to A.E. Scherer, "Staff Evaluation of CENPD-140," April 11, 1976.

23. NUREG-75/112, "Safety Evaluation Report Related to the Preliminary Design of the Standard Reference System CESSAR System 80, Docket No. STN 50-470," December 1975.