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

FPL

This letter forwards proprietary information in accordance with 10 CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment 1.

February 29, 2012

L-2012-059
10 CFR 50.90
10 CFR 2.390

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

Re: St. Lucie Plant Unit 2
Docket No. 50-389
Renewed Facility Operating License No. NPF-16

Response to NRC Mechanical and Civil Engineering Branch (EMCB) Request for Additional Information Regarding Extended Power Uprate License Amendment Request

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2011-021), "License Amendment Request for Extended Power Uprate," February 25, 2011, Accession No. ML110730116.

(2) Email from T. Orf (NRC) to C. Wasik (FPL), St. Lucie 2 EPU draft RAIs – Mechanical & Civil Engineering Branch (EMCB), dated January 13, 2012.

By letter L-2011-021 dated February 25, 2011 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. NPF-16 and revise the St. Lucie Unit 2 Technical Specifications (TS). The proposed amendment will increase the unit's licensed core thermal power level from 2700 megawatts thermal (MWt) to 3020 MWt and revise the Renewed Facility Operating License and TS to support operation at this increased core thermal power level. This represents an approximate increase of 11.85% and is therefore considered an Extended Power Uprate (EPU).

By email from the NRC Project Manager dated January 13, 2012 [Reference 2], additional information was requested by the NRC staff in the Mechanical and Civil Engineering Branch (EMCB) to support their review of the EPU License Amendment Request (LAR). The request for additional information (RAI) identified 47 questions.

A001
NRR

Attachment 1 to this letter contains FPL's response to RAIs EMCB-1 through EMCB-33 and EMCB-41 through EMCB-47. Attachment 1 contains Westinghouse proprietary information and Attachment 2 is the fully non-proprietary version of Attachment 1. Attachment 3 contains the Westinghouse Proprietary Information Affidavit. The purpose of this attachment is to withhold the proprietary information contained in Attachment 1 from public disclosure. The Affidavit, signed by Westinghouse as the owner of the information, sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of § 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

Note that the responses to RAIs EMCB-1 and EMCB-23 are not complete at this time. Supplemental responses to these two RAIs will be provided to the NRC in a separate submittal. The responses to RAIs EMCB-34 through EMCB-40 will also be provided to the NRC in a separate submittal.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2011-021 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the designated State of Florida official.

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate LAR Project Manager, at 772-467-7138.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed on 29-February-2012

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachment (3)

cc: Mr. William Passetti, Florida Department of Health

ATTACHMENT 2

Response to NRC Mechanical and Civil Engineering Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

NON-PROPRIETARY INFORMATION

(Cover page plus 40 pages)

Response to NRC Mechanical and Civil Engineering Branch (EMCB)
Request for Additional Information

The following information is provided by Florida Power & Light (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support Extended Power Uprate (EPU) License Amendment Request (LAR) for St. Lucie Nuclear Plant Unit 2 that was submitted to the NRC by FPL via letter (L-2011-021) dated February 25, 2011, Accession Number ML110730116.

In an email dated January 13, 2012 from NRC (Tracy Orf) to FPL (Chris Wasik), Subject: St. Lucie 2 EPU draft RAIs - Mechanical & Civil Engineering Branch (EMCB), the NRC requested additional information regarding FPL's request to implement the EPU. The RAI consisted of forty-seven (47) questions from the NRC's Mechanical and Civil Engineering Branch (EMCB). The responses to RAIs EMCB-1 through EMCB-33 and EMCB-41 through EMCB-47 are documented below. The responses to RAIs EMCB-34 through 40 will be transmitted to NRC under a separate submittal.

EMCB-1

The staff requests that the licensee provide assurance that all structural modifications and/or additions have been identified and designed and that all structural evaluations and required design calculations to demonstrate that all systems, structures and components (SSCs) credited to and/or affected by the proposed extended power uprate (EPU) have been completed and controlled documentation exists which finds said SSCs structurally adequate to perform their intended design functions under EPU conditions.

Response:

For St. Lucie Unit 2, with the exception of the EPU modifications listed below, applicable safety related and/or seismic II/I piping and associated structural evaluations and design calculations for affected systems, structures and components (SSCs) credited to and/or affected by the proposed EPU have been completed. These calculations document that affected SSCs are structurally adequate to perform their intended design functions under EPU conditions.

The following EPU modifications contain safety related and/or seismic II/I piping and are not complete at this time:

- Control Room Air Conditioning Margin Improvement,
- Main Steam, Feedwater, and Condensate Pipe Support Modification,
- Chemical and Volume Control System (CVCS) Vent Modification.

Design information regarding these specific EPU modifications will be provided to NRC in a supplemental response.

EMCB-2

The EPU licensing report (LR) states that "The method used to evaluate piping systems that experienced an increase in temperature, pressure, and/or flow rate is the preparation of detailed pipe stress computer analyses."

a) **Provide a list of systems inside and outside containment for which temperature, pressure, flow or mechanical loads has increased due to EPU. Please also provide the associated original licensed thermal power (OLTP), current licensed thermal power (CLTP) and EPU values. Also, in this list, identify the high energy (HE) lines for which breaks/cracks need to be postulated.**
b) **If stress summaries of these systems identified above are not included in EPU LR Section 2.2, please provide such stress summaries for these systems similar to the ones presented in the EPU LR tables. If the stresses did not change for EPU provide a justification.**
c) **If scaling factors have been utilized to calculate pipe stresses, please describe the method and provide an example of the scaling factor derivation and how the scaling factors have been used to determine the code equation stresses.**

Response 2a:

For St. Lucie Unit 2, the piping systems which experience an increase in temperature, pressure, flow or mechanical loads due to EPU are the reactor coolant, main steam, feedwater, condensate, heater vents and drains, component cooling water, chemical and volume control and safety injection systems.

The reactor coolant system (RCS) is impacted by changes in temperature and mechanical loads due to the EPU. As indicated in EPU LAR Attachment 5, Table 2.2.2.1-1, the hot leg straight pipe and the cold leg spray nozzles are the only affected components. Other RCS components were analyzed, but only these two components were affected by the EPU.

Current and EPU primary plus secondary stresses and fatigue usage factors for the main coolant loop and pressurizer surge line are reported in EPU LAR Attachment 5, Tables 2.2.2.1-1 and 2.2.2.1-2. The effects of breaks in high energy lines (i.e., the main loop piping) are no longer considered a part of the design basis following the application of leak-before-break, as described in EPU LAR Attachment 5, Section 2.1.6.

A summary of the CLTP and EPU stress values for the specific BOP piping systems impacted by EPU are provided in the response to RAI EMCB-2b.

The piping systems that contain high energy piping are the reactor coolant, main steam, feedwater, condensate, extraction steam, heater vents and drains, chemical and volume control, safety injection, shutdown cooling and steam generator blowdown systems. The EPU piping evaluations performed to reconcile changes in operating temperatures, pressures and flow rates due to EPU did not result in any new postulated pipe break/crack locations.

Response 2b:

Stress summaries for the two affected reactor coolant system components listed in the response to RAI EMCB-2a are included in EPU LAR Attachment 5, Table 2.2.2.1-1.

Stress summary data for BOP piping systems impacted by EPU are provided in the following table.

Stress Summary at EPU Conditions

Piping Analysis Description (Note 3)	Loading Condition (Note 2)	Existing Stress (psi) (Note 4)	EPU Stress (psi)	Allowable Stress (psi)	Design Margin (Note 1)
Main Steam Piping from SG-2A to Containment Penetration P-1	Equation 8	6,624	9,062	15,000	0.60
	Equation 9U	4,169	10,975	18,000	0.61
	Equation 9F	4,582	10,600	36,000	0.29
	Equation 10	8,208	7,986	22,500	0.35
Main Steam Piping from SG-2B to Containment Penetration P-2	Equation 8	6,613	8,711	15,000	0.58
	Equation 9U	4,755	11,025	18,000	0.61
	Equation 9F	5,033	11,037	36,000	0.31
	Equation 10	8,427	7,943	22,500	0.35
Main Steam Piping from Containment Penetrations P-1 and P-2 to HP Turbine	Equation 8	7,828	7,926	15,000	0.53
	Equation 9U	9,567	13,170	18,000	0.73
	Equation 9F	5,693	13,186	36,000	0.37
	Equation 10	19,149	20,535	22,500	0.91
Feedwater Piping from Containment Penetration P-3 to SG-2A	Equation 8	5,222	6,061	15,000	0.40
	Equation 9U	8,840	10,013	18,000	0.56
	Equation 9F	8,276	9,477	36,000	0.26
	Equation 10	14,089	12,096	22,500	0.54
Feedwater Piping from Containment Penetration P-4 to SG-2B	Equation 8	5,664	6,325	15,000	0.42
	Equation 9U	7,567	9,010	18,000	0.50
	Equation 9F	9,144	9,339	36,000	0.26
	Equation 10	9,330	11,304	22,500	0.50
Feedwater Pumps to Containment Penetrations P-3 and P-4	Equation 8	8,321	8,730	15,000	0.58
	Equation 9U	7,371	13,911	18,000	0.77
	Equation 9F	7,371	13,924	36,000	0.39
	Equation 10	16,119	17,950	22,500	0.80
Component Cooling Water Suction Piping to Pumps 2A, 2B and 2C	Equation 8	7,441	5,533	15,000	0.37
	Equation 9U	17,005	15,933	18,000	0.89
	Equation 9F	26,632	25,336	36,000	0.70
	Equation 11	36,572	32,273	37,500	0.86

Stress Summary at EPU Conditions					
Piping Analysis Description (Note 3)	**Loading Condition (Note 2)**	**Existing Stress (psi) (Note 4)**	**EPU Stress (psi)**	**Allowable Stress (psi)**	**Design Margin (Note 1)**
Component Cooling Water SDHX Return Piping	Equation 8	10,025	6,838	15,000	0.46
	Equation 9U	19,468	15,875	18,000	0.88
	Equation 9F	31,598	20,204	36,000	0.56
	Equation 11	36,448	23,791	37,500	0.63
Component Cooling Water Piping to Surge Tank A	Equation 8	3,776	3,256	15,000	0.22
	Equation 9U	9,464	3,744	18,000	0.21
	Equation 9F	10,466	4,057	36,000	0.11
	Equation 11	28,368	26,958	37,500	0.72
Component Cooling Water Piping to Surge Tank B	Equation 8	2,510	2,535	15,000	0.17
	Equation 9U	9,623	10,974	18,000	0.61
	Equation 9F	10,603	11,979	36,000	0.33
	Equation 11	7,553	11,166	37,500	0.30
Component Cooling Water CCW Cooler Return Piping from Control Room ACU	Equation 8	7,429	3,788	15,000	0.25
	Equation 9U	12,985	4,988	18,000	0.28
	Equation 9F	16,336	5,533	36,000	0.15
	Equation 11	25,514	36,793	37,500	0.98
Condensate Pump Discharge Piping to Feedwater Pump Suction	ANSI B31.1 Equation 11	N/A	13,880	15,000	0.93
	ANSI B31.1 Equation 13	N/A	18,670	22,500	0.83
Heater Vents and Drains MSR-2C/2D to Heater 5A	ANSI B31.1 Equation 13	19,749	22,119	22,500	0.98
Heater Vents and Drains MSR-2A/2B to Heater 5A	ANSI B31.1 Equation 14	28,752	32,202	37,500	0.86
Chemical Volume and Control Charging Loop 2B1 Cold Leg	Equation 12	13,353	13,620	51,900	0.26
Safety Injection RC Loop to Safety Injection Tank 2A1	Equation 12	40,581	46,546	50,700	0.92
Safety Injection RC Loop to Safety Injection Tank 2A2	Equation 12	23,429	26,662	50,700	0.53

Stress Summary at EPU Conditions					
Piping Analysis Description (Note 3)	**Loading Condition (Note 2)**	**Existing Stress (psi) (Note 4)**	**EPU Stress (psi)**	**Allowable Stress (psi)**	**Design Margin (Note 1)**
Safety Injection RC Loop to Safety Injection Tank 2B1	Equation 12	8,226	9,641	50,700	0.19
Safety Injection RC Loop to Safety Injection Tank 2B2	Equation 12	1,519	1,780	50,700	0.04

Notes:

1. Stress Interaction Ratio (also called "Design Margin") is based on the ratio of EPU stress divided by the Allowable stress.
2. Unless otherwise indicated, the pipe stress analysis equation numbers listed in this table correspond to ASME Section III, NB-3650, NC-3650, and ND-3650 equation numbers.
3. Description is based on pipe stress analysis segment of a given system included in the analysis.
4. When stress data is not provided, the information is not available.

Piping systems that were not affected by EPU (i.e., existing and/or currently analyzed temperatures, pressures and flow rates bound the corresponding EPU values) do not require re-evaluation (i.e., stress values remain acceptable/unchanged).

Response 2c:

For reactor coolant system (RCS) components, comparisons of current and EPU loads on a component are used to develop EPU results. The stress summaries for RCS piping are reported in EPU LAR Attachment 5, Table 2.2.2.1-1. In the case where scaling factors are used to calculate Code equation stresses, stresses are multiplied by the ratio of the EPU load divided by the current load. For example, the cold leg spray nozzle resultant moment of the branch line pipe for EPU exceeds the current moment. The current maximum stress intensity is scaled to an EPU stress intensity of 17.53 ksi, as follows:

(a,c)

The St. Lucie Unit 2 methodology scales the total stress intensity, which is conservative. In the example provided, the resultant branch moment ratio is multiplied by the total stress intensity, thereby scaling thermal transient stress contributions as well.

BOP piping evaluations were performed using both computer analyses and scaling factors. In instances where scaling factors were used in BOP piping evaluations, the scale factors (i.e., change factors) were used to determine applicable thermal expansion pipe stresses for EPU conditions. The thermal change factors were based on the ratio of the power uprate to pre-uprate operating temperatures. That is, the thermal change factor equals (T_{uprate} - 70°F) / ($T_{pre\text{-}uprate}$ - 70°F).

For example, the existing/analyzed temperature for heater vents and drain piping running from MSR-2C/2D to feedwater heater 5A was 520°F. The corresponding EPU temperature for this piping is 572°F. Hence, the thermal change factor equals (572-70)/(520-70) = 1.12.

Using this change factor, the existing thermal stress of 19,749 psi (based on 520°F) was increased by the 1.12 change factor (19,749 x 1.12) to determine the corresponding EPU thermal stress of 22,119 psi. The stress values shown in this example are included in table provided in the response to RAI EMCB-2b above.

EMCB-3

Please confirm that the proposed EPU does not introduce any changes to the current licensing basis (CLB) in determining pipe break or crack locations and dynamic effects associated with the postulation of pipe failures.

Response:

The primary loop piping for St. Lucie Unit 2 meets the criteria for the application of leak-before-break (LBB) presented in NUREG-1061, Volume 3. The changes in mechanical loads on the primary loop piping due to the EPU would have a negligible effect on CLB pipe breaks, and internal pressure does not change for the EPU. As a result, the EPU does not introduce any changes to the postulated pipe breaks.

The evaluations performed for the St. Lucie Unit 2 EPU did not introduce any changes to the current licensing basis (CLB) in determining pipe break or crack locations and dynamic effects associated with the postulation of pipe failures.

The criteria for pipe rupture postulation for inside containment piping is based on the guidance provided in Regulatory Guide 1.46 (May 1973) which is part of the St. Lucie Unit 2 CLB. The criteria for pipe rupture postulation for outside containment piping is based on the guidance provided in the A. Giambusso Letter (December 1972) which is also part of the St. Lucie Unit 2 CLB.

As noted in the response to RAI EMCB-5, NRC Generic Letter 87-11 may be invoked for Class 2 and 3 piping which allows for the elimination of arbitrary intermediate breaks.

EMCB-4

According to the St. Lucie U1 CLB (Final Safety Analysis Report Section 3), Class I piping systems have been designed in accordance with the 1969 ANSI B31.7 and Class II and III piping systems have been designed in accordance with the 1967 ANSI B31.1. (Please note that there is a separate RAI, EMCB RAI-13, that addresses codes and code editions for pipe stress evaluations other than postulating pipe failures.)

a) **Please provide the code and code year edition used for postulating pipe failures inside and outside containment using the stress criteria. If different than the CLB code, provide the basis for justifying use of codes other than CLB codes (whether a documented code reconciliation exists) and discuss the regulatory process utilized that allowed the use of codes that are different than those stated in the CLB for postulating pipe failures.**

b) **Please provide the stress equations used for postulating pipe breaks and pipe cracks including stress equations for calculating local stresses due to pipe welded attachments and discuss the basis which allows the use of these equations.**

Response 4a:

For the St. Lucie Unit 2 EPU, there was no stress criteria used for postulating pipe failures inside containment for the primary reactor coolant loop (RCL) Class 1 piping. The primary RCL piping used leak-before-break (LBB) criteria in accordance with NUREG-1061.

The stress evaluations used for postulating pipe failures of Class I branch lines connected to the RCL primary loop piping were performed using simplified change factor methodology to reconcile minor changes in thermal expansion displacements. All other loading conditions such as deadweight, seismic, etc., were unchanged due to EPU for these branch lines. The pre-EPU design basis stress levels that were increased were generated in accordance with the ASME Section III, 1971 edition through Summer 1973 (i.e., CLB code) or in some cases in accordance with the ASME III, 1980 edition. For those instances where the ASME III 1980 edition was used, a code reconciliation documenting the use of this later ASME III code was performed in accordance with ASME Section XI.

The current licensing basis (CLB) code was used for developing stress data for postulating pipe failures inside and outside containment for Class II and III systems. This CLB code is ASME Section III, 1971 edition through Summer 1973 Addenda.

Response 4b:

The EPU stress equations used for postulating pipe breaks are as follows:

Outside Containment (For Class 2 and 3 Piping)

Pressure + Deadweight + OBE + Fluid Transient (if applicable) +Thermal ≤ 0.8(Sh + Sa)
Thermal ≤ 0.8 Sa

Inside Containment (For Class 2 and 3 Piping)

Pressure + Deadweight + OBE + Fluid Transient (if applicable) +Thermal ≤ 0.8(Sh + Sa)

Inside Containment (Reactor Coolant Loop (RCL))

Leak-before-break (LBB) criteria in accordance with NUREG-1061 is applicable to the reactor coolant system main loop piping.

Inside Containment (RCL Class 1 Branch Piping)

Primary + Secondary Stress Intensities ≤ 2.0 Sm for ferritic steel and 2.4 Sm for austenitic steel
Cumulative Usage Factor ≤ 0.10

The basis for the outside containment stress equations is the guidance provided in the A. Giambusso Letter (December 1972) which is part of the St. Lucie Unit 2 CLB.

The basis for the inside containment stress equations is the guidance provided in Regulatory Guide 1.46 (May 1973) and NUREG-1061 which are part of the St. Lucie Unit 2 CLB.

With respect to local stresses from integral pipe attachments, these stresses were not included in the determination of pipe break locations. The guidance within the CLB used for EPU (i.e., Giambusso Letter and Regulatory Guide 1.46) does not require that local pipe stresses from integral welded attachments be included in determining pipe break locations.

Also, there are no pipe crack stress equations contained in the Giambusso Letter or Regulatory Guide 1.46.

EMCB-5

Please provide a copy of the regulatory process which allowed the insertion of FSAR Appendix 3.J.

Response:

Appendix 3.J of the St. Lucie Unit 1 UFSAR provides the portions of the revised Branch Technical Position MEB 3-1 that are applicable to St. Lucie Unit 1. The St. Lucie Unit 2 UFSAR does not contain an equivalent to St. Lucie Unit 1 UFSAR Appendix 3.J. The design and licensing basis information regarding the protection against the dynamic effects associated with the rupture of piping is included in Section 3.6 of the St. Lucie Unit 2 UFSAR.

As stated in St. Lucie Unit 2 UFSAR Section 3.6, NRC Generic Letter 87-11 was adopted as an alternative means to provide pipe break protection for Class 2 and Class 3 piping systems. NRC Generic Letter 87-11 eliminated the requirement for all dynamic effects (missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, compartment, subcompartment and cavity pressurizations and decompression waves within the ruptured pipe) and all environmental effects (pressure, temperature, humidity and flooding) resulting from arbitrary intermediate pipe ruptures. It also allows the elimination of pipe whip restraints and jet impingement shields placed to mitigate the effects of arbitrary intermediate pipe ruptures.

Generic Letter 87-11 revised Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," as contained in the Standard Review Plan, Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." Modifications to Class 2 and 3 piping systems may invoke the criteria set forth in Generic Letter 87-11 in lieu of the original criteria.

The regulatory process that added the criteria included in NRC Generic Letter 87-11 to Section 3.6 of the St. Lucie Unit 2 UFSAR is 10 CFR 50.59. The modification to replace the original steam generators in 1996 applied the criteria associated with MEB 3-1 relative to high energy line break analysis. As part of the modification process, use of MEB 3-1 for St. Lucie Unit 2 was justified and the UFSAR change package that updated UFSAR Section 3.6 was provided. Application of this criteria presented in UFSAR Section 3.6 was not exclusive to this modification, but was intended for use in future modifications.

EMCB-6

Please discuss whether the St. Lucie U1 current licensing thermal power (CLTP) criteria for postulating piping failures inside containment are in accordance with RG 1.46, "Protection Against Pipe Whip Inside Containment," or MEB 3-1, Rev 2, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment."

Response:

The St. Lucie Unit 2 current licensing basis (CLB) (UFSAR Section 3.6) for postulating piping failures inside containment is in accordance with RG 1.46, "Protection Against Pipe Whip Inside Containment (May 1973)."

EMCB-7

The EPU LR states the following:

> **FPL conducted a review of pipe break postulation and associated pipe rupture analyses to ensure that SSCs [systems, structures, and components] are adequately protected from the dynamic effects of pipe ruptures such as pipe whip and jet impingement.**

For HE piping systems that will experience an increase in loads due to EPU (see EMCB RAI-2(a), above), provide a quantitative summary which shows that the dynamic effects of pipe whip and jet impingement have been evaluated and provide a comparison of results to the acceptable limits. If the loads resulting from pipe whip and jet impingement at EPU conditions are enveloped by the CLTP loads, provide a discussion that justifies this condition.

Response:

The St. Lucie Unit 2 EPU piping evaluations performed to reconcile changes in operating temperatures, pressures and flow rates due to EPU did not result in any new postulated pipe break locations. Also, operating parameters associated with EPU did not result in any load increases which would adversely impact existing pipe whip and jet impingement assessments. As such, no quantitative evaluations were required to be performed for EPU.

EMCB-8

Identify the reactor coolant system (RCS) branch piping breaks and discuss the analyses performed due to these breaks at EPU conditions.

Response:

The St. Lucie Unit 2 RCS branch piping breaks are the pressurizer surge, spray and relief line breaks, high pressure safety injection line breaks, shutdown cooling line breaks, and chemical and volume control line breaks for letdown and charging piping. For EPU, applicable RCS branch line piping evaluations performed to reconcile changes in operating temperatures, pressures and flow rates due to EPU did not result in any new postulated pipe break locations. As such, no additional analyses were required due to EPU.

EMCB-9

The EPU LR Section 2.2.1.2.4 provides a discussion of the evaluation results for the postulation of pipe failures and their associated dynamic effects at EPU conditions for the balance of plant (BOP) systems. Please provide a discussion that addresses the evaluation results for the postulation of pipe failures and their associated dynamic effects at EPU conditions for the remainder of the HE systems inside containment.

Response:

The St. Lucie Unit 2 EPU piping evaluations for high energy (HE) piping systems located inside containment were performed to reconcile changes in operating temperatures, pressures and flow rates. These evaluations did not result in any new postulated pipe break locations. Also, operating parameters associated with EPU did not result in any load increases which would adversely impact existing pipe whip and jet impingement assessments. As such, no quantitative evaluations were required to be performed for EPU.

EMCB-10

Explain the purpose of the shim plate attached to the steam generator (SG) sliding base support and discuss the basis which allowed its removal and thus deletion of the north south restraint part of this support.

Response:

The St. Lucie Unit 2 SG sliding base support function is to provide vertical support of the SG dead weight and loss of coolant accident (LOCA) restraint in the North-South direction as the reactor coolant system (RCS) hot and cold legs expand and contract with the varied thermal modes of operation. The SG sliding base also provides seismic and LOCA restraint for the East-West direction by transferring the loads into the key embedded in the concrete SG pedestal.

The purpose of the shim plate attached to the steam generator (SG) sliding base support was to maintain the 3/64" (+/- 1/64") "hot gap" between the SG sliding base casting and the embedded key in the concrete SG pedestal to provide a North-South restraint for pipe rupture loads as a result of RCS hot and cold leg break LOCAs.

The basis for the removal of the shim plate follows the application of leak-before-break (LBB) criteria, which allowed for the removal of LOCA loads from the design basis of the SG sliding base support while postulated pipe breaks of the RCS branch lines (small break LOCA) remain part of the design basis. Only guillotine ruptures of the RCS hot leg and cold leg piping load the SG sliding base support in the North-South direction. In addition, none of the postulated slot ruptures in the RCS hot and cold leg piping nor the seismic loads credit the North–South direction support of the sliding base plate, eliminating the need to maintain a maximum gap between the SG sliding base and the embedded key. Since the North-South restraint capacity of the support was no longer required, the SG shim plate was permanently removed.

EMCB-11

In accordance with the EPU LR, leak before break (LBB) is credited for the proposed EPU. Approval of LBB eliminates pipe breaks and permits removal of protective barriers and redesign of piping and equipment supports. Approval of LBB methodology in the current St. Lucie U1 licensing basis has eliminated pipe breaks and their dynamic effects from the RCS main loop piping and may have eliminated or redesigned SSCs required to mitigate the RCS main loop loss of coolant accident (LOCA) dynamic effects. As indicated in the EPU LR (page 2.2.2-47), as a result of the LBB methodology application on the RCS main loop, the limiting pipe breaks considered in the EPU design basis with respect to the RCS mechanical/dynamic response, are branch line pipe breaks (BLPBs). The EPU LR makes the statement that, "The response of the RCS loop to BLPBs is bounded by the response of the RCS loop to the originally postulated LOCAs." Please discuss what action(s) have been taken to safeguard the validity of this statement, given that approval of LBB can result in the removal and modifications of SSCs and assure that documented consideration has been given, where required, for existing changes allowed due to LBB so that the structural integrity of any SSCs due to this or other statements that use acceptance by bounding conditions has not been affected.

Response:

For St. Lucie Unit 2, removal or modification of existing systems, structures, or components (SSCs) would fall under the control of the plant modification process. Plant modifications are prepared, reviewed and approved by competent personnel trained in the preparation of design change packages in accordance with plant approved procedures. The process requires examination of the UFSAR, Design Basis Documents and all applicable calculations. If a particular expertise is required, the original equipment manufacturers (OEM) and owners groups are engaged and consultants and companies cognizant in Nuclear Engineering design are employed.

Every design change package contains the following required attributes:

- A summary of the current design basis and functions of the SSCs affected by or involved in the modification. This requires consulting, among other things, the existing licensing basis documentation.
- A summary of the purpose and design objective of the design change. This would require an examination of any applicable licensing commitments underlying the change.
- A description of the design change, in sufficient detail to identify the affected SSCs.
- A justification of the design change. If the design change modifies (or in this case, removes) the basic function of the SSC, the modified function will be examined from the perspective of any system interactions that are affected by the change. The critical attributes and the potential effects on design basis system and component functions must be identified.

Accordingly, potential impact of a modification on existing structures provided for the mitigation of the effects of pipe rupture, even those pipe ruptures no longer postulated, would necessarily be identified and evaluated in the normal course of the development of the design change package.

Following the adoption of leak-before-break, since the design for main reactor coolant loop LOCAs is no longer required, changes involving removal of attributes specifically designed to resist LOCA loadings would be reviewed to ensure that the RCS pipe whip restraint configuration adequately restrains the RCS under a branch line LOCA event.

EMCB-12

Provide a discussion with a summary of the structural evaluations performed which demonstrate that plant compartments, the containment with its sub-compartments and plant SSCs important to safety, including containment penetrations, are structurally capable to withstand the EPU mass and energy (M&E) releases from postulated pipe failures.

Response:

New EPU structural evaluations were not required since the EPU assessments demonstrated that plant compartments, the containment with its sub-compartments, and plant SSCs important to safety (including containment penetrations), are currently designed to withstand the consequences of the mass and energy (M&E) releases from postulated licensing basis pipe failures at EPU conditions.

Summarized below are the key elements of the relevant EPU assessments that demonstrate that the existing structural design of plant SSCs bound EPU conditions following postulated pipe failures:

- As indicated in EPU LAR Attachment 5, Section 2.2.1, the EPU did not result in any new or revised break locations. Existing pipe whip dynamic analyses and results including jet thrust and impingement forcing functions and pipe whip dynamic effects remain valid for EPU. Therefore, it is concluded that existing design of SSCs both inside and outside containment remain acceptable to protect safety related SSCs from the effects of pipe whip and jet impingement loading following postulated pipe breaks at EPU conditions, and that new structural analyses are not required.
- As indicated in EPU LAR Attachment 5, Section 2.3.1, Figure 2.3.1-2, the current in-containment EQ accident pressure profile bounds the EPU LOCA and MSLB accident pressure profiles. The peak-tested conditions of the in-containment SSCs (includes the containment penetrations) envelope the current EQ accident pressure profile, which in turn bounds the EPU LOCA and MSLB accident pressure profiles. EPU LAR Attachment 5, Section 2.3.1 also discusses the EPU pressure transients outside containment due to postulated pipe failures and notes that the post-accident pressure in the Reactor Auxiliary Building (RAB) utilized for equipment qualification remains unchanged by the EPU. In addition, EPU LAR Attachment 5, Section 2.3.1 notes that the pressure in the steam trestle area remains unchanged by the EPU since it is open to the atmosphere. Therefore, additional structural analyses are not required for SSCs inside or outside containment.

- As indicated in EPU LAR Attachment 5, Section 2.6.1, the containment pressure response analyses performed to evaluate the consequences of the St. Lucie Unit 2 licensing basis spectrum of breaks inside containment, demonstrate that the containment peak pressure at EPU conditions remains within containment design pressure. Therefore, no additional structural analyses are required to support containment integrity following postulated pipe failures.

- As indicated in EPU LAR Attachment 5, Section 2.6.2, the containment subcompartment walls are designed to withstand differential pressures in excess of that expected at EPU conditions as a result of licensing basis pipe breaks following implementation of the leak-before-break (LBB) methodology. Therefore, no additional analyses are required to support the structural integrity of the in-containment subcompartments following the EPU.

EMCB-13

According to the St. Lucie U1 CLB (FSAR Section 3), Class I piping systems have been designed in accordance with the 1969 ANSI B31.7 and Class II and III piping systems have been designed in accordance with the 1967 ANSI B31.1. For the design of structural steel, the FSAR makes reference to the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," dated February 12, 1969. The EPU LR (Page 2.2.2-21) indicates that AISC, B31.1, B31.7 and/or ASME Section III codes of various year editions have been used for the EPU structural evaluations.

a) **Please provide a discussion, which addresses for SSCs important to safety, the code edition used for EPU (ASME Section III, B31.7, B31.1, AISC Manual, etc.) and the code used in CLB. Where codes other than the CLB codes have been utilized, please provide the basis for justifying use of those codes (and include in the discussion whether a documented code reconciliation exists) and discuss the regulatory process utilized that allowed use of those codes that are different than the FSAR listed codes. This information needs to include only those specific items (SSCs) that use different code or code edition/addenda for EPU evaluations other than those mentioned in the FSAR.**

b) **Please provide assurance that structural calculations for the SSCs credited or affected by the EPU utilized original code of construction allowable values.**

Response 13a:

The Code editions used for the St. Lucie Unit 2 EPU analysis of Class 1 reactor coolant system (RCS) piping correspond with the original Code of construction and the Code editions used in the analysis of record (AOR). Therefore, no reconciliation is required. The AOR Class I piping primary, primary plus secondary, peak stresses, and simplified elastic-plastic methodology are in accordance with ASME Section III, 1971 Edition with Addenda up to and including Winter 1972. The exception is the pressurizer surge line analysis, which utilizes ASME Section III, 1986 Edition.

The Code editions for the St. Lucie Unit 2 analysis of nuclear steam supply system (NSSS) supports are ASME Section III, 1971 through Summer 1971, 1977 through Summer 1977, and 1977 through Winter 1978 for the reactor vessel supports, steam generator (SG) sliding baseplate (SBP), and SG SBP bearing, respectively. Reconciliation of the NSSS supports for EPU was not required.

The evaluation of Class I branch lines connected to the reactor coolant loop (RCL) primary loop piping was performed using simplified change factor methodology to reconcile minor changes in thermal expansion displacements. All other loading conditions such as deadweight, seismic, etc., were unchanged due to EPU for these branch lines. The pre-EPU design basis stress levels that were increased were generated in accordance with the ASME Section III, 1971 edition through Summer 1973 (i.e., current licensing basis (CLB) code) or in some cases in accordance with the ASME Section III, 1980 edition. For those instances where the ASME Section III, 1980 edition was used, a code reconciliation documenting the use of this later ASME code was performed in accordance with ASME Section XI.

The St. Lucie Unit 2 CLB piping code of record for Class 2 and 3 piping is ASME Section III, 1971 edition through Summer 1973 Addenda. The code used for the evaluation of Class 2 and 3 piping systems for EPU was ASME Section III, 1971 edition through Summer 1973 addenda, consistent with the CLB.

The St. Lucie Unit 2 CLB piping code of record for non safety-related/non seismic piping systems is ANSI B31.1, 1967 edition through Summer 1973 addenda. The code used for the evaluations of non safety-related/non seismic piping systems for EPU was ANSI B31.1, 1967 edition through Summer 1973 addenda. This code is consistent with the CLB code of record for non safety-related/non seismic piping.

For EPU pipe support evaluations, the AISC Manual, 7th edition was used to perform these assessments. This 7th edition of the AISC Manual is consistent with the CLB.

Structural steel that is credited for support of piping systems or is otherwise affected by the EPU has been analyzed for EPU conditions using the allowable stresses specified by the original code of construction (i.e., the AISC Manual, 7th edition).

Response 13b:

Structural evaluations of SSCs performed for the St. Lucie Unit 2 EPU have used allowable stress values contained in existing design basis analyses which utilized original code of construction allowable values.

EMCB-14

a) **EPU LR page 2.5.4.3-10 discusses the impact of the proposed EPU on the resolution of the GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions," issues of component cooling water (CCW) waterhammer, two-phase flow and CCW piping sections subject to thermally induced overpressurization. Please discuss the impact that the proposed EPU has on piping sections subject to thermally induced overpressurization, other than the CCW.**

b) **Please discuss the decrease in containment temperature following a main steam line break (MSLB) at EPU conditions that is mentioned in the EPU LR (pgs 2.5.4.3-10,11), while EPU LR Table 2.5.5.1-1 shows no significant change (less than 0.5%) in the main steam temperature. Also, please discuss whether the MSLB is the limiting condition for thermally induced overpressurization of piping.**

Response 14a:

During FPL's Generic Letter (GL) 96-06 evaluation, containment penetrations and pipe sections inside containment that were vulnerable to thermal overpressurization during loss of coolant accidents (LOCA) and MSLB events were evaluated. The screening process excluded systems within containment not handling liquids; sections of fluid filled piping inside containment normally operating at higher than post-accident containment temperatures; systems with thermal relief provided by relief devices, check valves, or solenoid / air operated valves (AOV) with pressure under the seat; and sections of piping open to vessels containing compressible fluids or provided with pressure relief devices. The majority of the penetrations and isolated pipe sections were determined to be not susceptible to thermal overpressurization based on the aforementioned criteria.

Six penetrations and isolated piping segments within three systems inside the Unit 2 containment were identified during FPL's GL 96-06 evaluation as being vulnerable to a water solid volume that may be subjected to an increase in pressure due to heating of trapped fluid. FPL has implemented modifications as part of the resolution to GL 96-06 to address these concerns.

EPU will not require any additional modifications to these pipe sections. Further, EPU will not create any new configurations, nor change existing procedural controls that will result in overpressurization of piping during accident conditions.

Response 14b:

The discussion on EPU LAR Attachment 5, pages 2.5.4.3-10, and 11, refer to MSLB conditions while EPU LAR Attachment 5, Table 2.5.5.1-1, provides normal plant operation parameters for the main steam system. The MSLB containment analysis that is discussed in EPU LAR Attachment 5, Section 2.6.1.2.2.1 is performed using the SGNIII code, which conservatively neglects the effect of safety injection (which would act to reduce the primary side temperatures and therefore reduce the MSLB mass and energy release rates into the containment atmosphere). Neglecting the safety injection in the MSLB containment analysis has the additional effect of preventing boron addition to the RCS. With no boron addition modeled, the reactivity addition due to the cooldown induced by the rapid blowdown of the secondary inventory (in the presence of a negative temperature feedback), may result in unnecessarily conservative predicted return to power. The MSLB containment analysis therefore limits the return to power to a value which conservatively bounds the return to power predicted in the analysis of the core and fuel response to a steam line break (see EPU LAR Attachment 5, Section 2.8.5.1.2). The bounding reactivity effect selected for the EPU MSLB containment analysis results in a lower peak containment temperature than that seen in the pre-EPU analysis. In addition, the maximum temperature reached inside containment for the MSLB event is a function of the total mass and energy released, as well as other mitigating systems and analytical techniques applied. A direct correlation between initial system operating temperature and post accident containment temperature does not necessarily exist.

The main steam temperature under normal EPU operating conditions as presented in EPU LAR Attachment 5, Table 2.5.5.1-1 was determined in a separate analysis using a plant thermal performance model tuned to pre-EPU conditions which was then revised for EPU conditions by increasing thermal power, applying the EPU steam generator pressure and modeling components replaced for EPU.

The containment response to a LOCA is the limiting condition for thermally induced overpressurization of piping.

EMCB-15

Please discuss why it was required to reanalyze the RCS loop piping for deadweight and thermal expansion loading conditions for EPU using the ANSYS program (stated in LR page 2.2.2-11). Show the difference in these loading cases for EPU and CLTP. The stress summaries of LR Tables 2.2.2.1-1 and 2.2.2.1-2 show that there are no additional stresses due to EPU and that the analyses on record (AOR) stresses are still applicable. In addition, please provide information on the original RCS loop piping stress analysis computer program code. Also, please discuss the math model validation and verification performed.

Response:

The EPU causes changes in thermal conditions imposed on the reactor coolant system (RCS) main loop piping. These changes affect fluid weight, thermal expansion mechanical loads, thermal transient definitions, and thermal displacements.

Thermal expansion analyses were performed using the ANSYS code to evaluate changes in RCS temperatures due to the EPU. The difference between the EPU and current licensed thermal power (CLTP) normal operation analyses is an increase of 1°F in the cold leg temperature. This temperature change also affected vertical support preloads, so deadweight analyses were also repeated.

Changes in thermal transient temperature versus time curves, if significant enough, will cause changes in through-wall thermal gradients. The EPU transients were reviewed for any changes relative to the current transient definitions, and it was determined that the changes were negligible.

Therefore, a reanalysis of the RCS for the effects of the EPU was required to determine if the EPU affected the deadweight and thermal expansion loads on the RCS piping, components, and supports. The results indicated that there was negligible change in the loads acting on the main piping loop and pressurizer surge line piping. Therefore, the AOR Class 1 stresses were unchanged.

Benchmarking was conducted to validate the ANSYS model by comparing frequencies and normal operation displacements to the corresponding AOR information. The AOR for St. Lucie Unit 2 used the Mare Island Naval Shipyard Computer Program, MEC-21.

EMCB-16

Please state the code and code edition utilized for the stress summaries shown in EPU LR Table 2.2.2.1-1. Please verify that the allowable stress values shown are correct and have been derived from the original code of construction for the RCS piping.

Response:

Table 2.2.2.1-1 of Attachment 5 to the St. Lucie Unit 2 EPU License Amendment Report summarizes the maximum primary plus secondary reactor coolant system (RCS) piping stresses for EPU and current conditions and compared them to the Code allowable for each piping component.

The EPU stresses were determined and compared to the Code allowable stresses as found in the analysis of record, which is based on ASME Section III, 1971 Edition with Addenda up to and including Winter 1972.

The allowable stress values included in EPU LAR Attachment 5, Table 2.2.2.1-1 were reviewed and found to be correct.

EMCB-17

From the notes on Table 2.2.2.2-1, it appears that for all listed pipe stresses, with the exception of the condensate and heater vent system, the code utilized is ASME Section III. Please list the year for the ASME code. Please verify that the shown values are correct and that the allowable stresses have been derived from the original code of construction.

Response:

The code year for the ASME stress levels provided in the response to RAI EMCB-2b is ASME Section III, 1971 edition through Summer 1973 Addenda. The allowable stresses shown are correct and are consistent with the original code of record.

EMCB-18

The EPU LR states the following:

> **Operating pressure increases due to EPU mostly affect systems related to the main power cycle (main steam, condensate, feedwater, extraction steam, heater drains). Since the pipe stress evaluations for these piping systems have been determined in accordance with the B31.1 Code or ASME Code Section III, increases in operating pressures are acceptable, as long as the EPU operating pressure remains within the current design pressure of the system. If the EPU operating pressure exceeds the design pressure, the impact is evaluated relative to the applicable pipe stress analysis calculations.**

It is noted that although B31.1 utilizes design pressure for calculating pipe stresses, B31.7 and ASME Section III utilize operating and maximum operating pressures. Therefore, the above statement is not valid for B31.7 and ASME Section III pipe stress calculations. Please provide a solution which will resolve this issue.

Response:

The applicable piping codes for St. Lucie Unit 2 are ASME Section III and ANSI B31.1. For St. Lucie Unit 2 EPU piping analyses that were performed in accordance with ASME Section III, the evaluation of Equations 9 and/or Equation 11 used the larger of the applicable design pressure or maximum operating pressure in determining the longitudinal pressure stress.

EMCB-19

The EPU LR indicates that Table 2.2.2.2-1 shows summaries of pipe stresses for EPU affected piping. Please explain why for some systems or sections of piping, stresses for only a limited number of code equations have been included.

Response:

Summaries of pipe stresses for St. Lucie Unit 2 EPU affected piping are provided in the response to RAI EMCB-2b.

For the main steam and feedwater piping systems that experienced changes in fluid transient loads/stresses (i.e., revised steam hammer and water hammer loads due to EPU), as well as changes in pipe support configurations, the deadweight, thermal expansion and seismic loading conditions are affected by EPU. Hence, for these piping systems, revised Equation 8, 9 and 10 stress data were included in stress summary table provided in the response to RAI EMCB-2b since the deadweight, seismic, thermal expansion and fluid transient stress levels were all affected by EPU.

For piping systems that only experience a temperature increase due to EPU, the only load/loading condition that is impacted is the "thermal expansion" loading condition (i.e., Equation 10 for AMSE III and Equation 13 for ANSI B31.1 piping). For these piping systems, the deadweight, pressure and seismic stresses are unchanged due to EPU (i.e., no change to ASME III Equations 8 or 9 or ANSI B31.1 Equations 11 or 12). Hence, for these piping systems, only the thermal expansion loading condition was included in the stress table provided in the response to RAI EMCB-2b since it was the only loading condition that was affected by EPU.

The stress table provided in the response to RAI EMCB-2b provides a summary of the specific loading conditions for those systems, and/or portions of systems, that were affected by EPU.

EMCB-20

EPU LR Table 2.2.2.2-2 lists EPU required pipe support modifications. Please discuss and list EPU required piping modifications or additions.

Response:

Safety related and/or seismic II/I piping modifications resulting from EPU are summarized in the tables below.

The first table is an updated version of EPU LAR Attachment 5, Table 2.2.2.2-5. The second table identifies modifications to address piping vibration.

Piping Reinforcement Locations Component Cooling Water Piping			
Pipe Stress Analysis Description	**Number of Modifications**	**Piping Stress Isometric Node Number**	**Reinforcement Description**
SDHX and CFC Cooler Return from CCW Bldg to CCW Pumps	4	7	0.625” pad on 24” x 16” Tee
		29	
		102	0.625” pad on 24” x 24” Tee
		104	
SDHX and CFC Cooler Return to CCW Bldg	1	7427	0.625” pad on 20” x 12” Tee
I-6-CC-153 From CCW Surge Tank to Return Header B (I-20-CC-26)	1	20	0.625” pad on 20” x 6” Tee
CCW Cooler Return from Control Room Air Conditioning	1	6421	0.625” pad on 20” x 4” Tee

Piping Modifications to Address Piping Vibration	
Line Number	**Modification Description**
Branch off of line 8"-MS-21 containing valve V8298	Modify weld at existing branch connection
Branch off line 8" BF-16 containing valve V9236	Modify weld at existing branch connection and V9236 location
Branch off line 8" BF-16 containing valve V9231	Modify weld at existing branch connection and V9231 location
Branch off line 8" BF-16 containing valve V9228	Modify weld at existing branch connection and V9228 location
Branch off line 8" BF-21 containing valve V9270	Modify weld at existing branch connection and V9270 location
Branch off line 8" BF-21 containing valve V9267	Modify weld at existing branch connection and V9267 location
Branch off line 8" BF-21 containing valve V9265	Modify weld at existing branch connection and V9265 location
Branch off line 8" BF-21 containing valve V9262	Modify weld at existing branch connection and V9262 location

Safety related and/or seismic II/I piping modifications for EPU that are not complete at this time are identified in the response to RAI EMCB-1.

EMCB-21

Please provide a discussion which addresses the methodology and criteria used for the detailed analyses that were performed to determine piping vibration stresses at locations of vibration concern that is mentioned in the EPU LR Section 2.2.2.2.2.4.

Response:

For St. Lucie Unit 2, the methodology used was to perform a PIPESTRESS computer analysis of the piping configurations to evaluate piping vibration responses at specified locations due to imposed vibration response spectra. The analyses were performed to generate piping displacements that correlated to the field observed piping displacement magnitudes at specified locations. The resulting pipe stress values from these analyses were verified to be within the acceptance criteria (i.e., permitted endurance limit) as provided in ASME OM-S/G-2007.

EMCB-22

a) **Confirm whether stress summaries of Table 2.2.2.2-1 include stresses due to fluid transient loads associated with the EPU such as main feedwater pump trips and valve closure transients due to turbine stop valve, main steam isolation valve and main feedwater regulating and isolation valves.**

b) **Please discuss and explain whether a force time history dynamic analysis was performed utilizing the PIPESTRESS or another pipe stress program code for the steam hammer or water hammer loads and provide the load combinations which include the transient loads for the Table 2.2.2.2-1 stresses.**

c) **The EPU LR indicates that the feedwater pumps will be replaced and the feedwater control system will be modified for EPU. Please discuss whether the structural evaluations in Section 2.2 reflect the configuration of the replacement pumps and piping. In addition, discuss whether the stress and load summaries presented in the tables of Section 2.2.2.2 include results from the actual transients of the replacement feedwater pumps and the modified feedwater control system.**

Response 22a:

For St. Lucie Unit 2 main steam piping, the stress summaries provided in the response to RAI EMCB-2b include stresses due to fluid transient loads associated with turbine stop valve and main steam isolation valve closure events.

For St. Lucie Unit 2 feedwater piping, the stress summaries provided in the response to RAI EMCB-2b include stresses due to fluid transient loads associated with feedwater regulating valve closure, feedwater isolation valve closure, and feedwater pump trip events.

Response 22b:

A force time history dynamic analysis was performed using the PIPESTRESS computer program to generate piping stresses/loads related to the main steam hammer analyses (for turbine stop valve and main steam isolation valve closure events) and the feedwater water hammer analyses (for feedwater regulating valve closure, feedwater isolation valve closure, and feedwater pump trip events).

The load combinations which include fluid transient stresses provided in the response to RAI EMCB-2b for main steam and feedwater piping are as follows:

Equation 9U (Pressure + Deadweight + OBE + Fluid Transient) $\leq$ 1.2 Sh

Equation 9F (Pressure + Deadweight + SSE + Fluid Transient) $\leq$ 2.4 Sh

Response 22c:

The St. Lucie Unit 2 EPU structural evaluations contained in EPU LAR Attachment 5, Section 2.2.2.2 reflect the revised piping and support configurations related to the replacement feedwater pumps and associated piping.

The stress summaries included in the stress table provided in the response to RAI EMCB-2b include results from the fluid transients associated with the replacement feedwater pumps, the modified feedwater control system, and revised feedwater piping and support configurations.

EMCB-23

To prove acceptability of the shown calculated loads for the steam generator nozzles shown on Tables 2.2.2.2-3 and 2.2.2.2-4, please provide the allowable loads and allowable load derivation.

Response:

Allowable nozzle loads were not provided by the supplier of the St. Lucie Unit 2 replacement steam generators (RSGs) during their original design. Instead, the RSG nozzles were analyzed based on loading conditions provided by FPL as part of their original design and procurement.

For the St. Lucie Unit 2 EPU, the RSG nozzle stresses and fatigue usage factors have been calculated for EPU based on EPU loads that bound those provided in EPU LAR Attachment 5, Tables 2.2.2.2-1 and 2.2.2.2-2. The results of the RSG nozzle stress and fatigue analyses using bounding EPU loads are provided in EPU LAR Attachment 5, Table 2.2.2.5-9.

As part of the EPU modification process, reanalysis of the main steam and feedwater piping has been performed. These revised EPU piping analyses establish new loads for the RSG main steam and feedwater nozzles. The RSG main steam and feedwater nozzles are being reanalyzed for these new EPU loads. Accordingly, an update to EPU LAR Attachment 5, Tables 2.2.2.2-1, 2.2.2.2-2, and 2.2.2.5-9 will be provided in a supplemental response.

EMCB-24

For the containment penetration qualification summaries on EPU LR Tables 2.2.2.2-5 and 2.2.2.2-6, please provide the following:

a) **Show whether the calculated EPU loads include reactions from piping from both sides of the penetration and discuss how the shown EPU calculated stress intensities were derived.**

b) **Provide the load combinations for the calculated loads.**

c) **Show how the allowable stress intensity values were derived.**

Response 24a:

The containment penetration load summaries for St. Lucie Unit 2 were provided in EPU LAR Attachment 5, Table 2.2.2.2-3 (Main Steam) and Table 2.2.2.2-4 (Feedwater).

The tables that follow provide an updated version of EPU LAR Attachment 5, Tables 2.2.2.2-3 and 2.2.2.2-4 and contain the final containment penetration load summaries resulting from EPU. The containment penetration loads in these tables are the total loads developed from the combined loads from the inside containment and outside containment piping.

The EPU calculated stress intensities were determined by combining individual calculated stresses due to the axial load, shear load, bending moment and torsional moment.

Containment Penetration Load Summary

Main Steam Penetration P-1				
Description	Forces (Kips)		Moments (in-Kips)	
	Axial (F_a)	Shear (F_V)	Bending (M_b)	Torsion (M_t)
Total Loads (EPU)	114	49	6,444	4,966
Calculated Stress intensity (EPU)	34,671			
Allowable Stress Intensity	61,141			
Design Margin	0.57			

Main Steam Penetration P-2				
Description	Forces (Kips)		Moments (in-Kips)	
	Axial (F_a)	Shear (F_V)	Bending (M_b)	Torsion (M_t)
Total Loads (EPU)	94	47	5,582	4,968
Calculated Stress Intensity (EPU)	30,935			
Allowable Stress Intensity	61,141			
Design Margin	0.51			

Feedwater Penetration P-3				
Description	Forces (Kips)		Moments (in-Kips)	
	Axial (F_a)	Shear (F_V)	Bending (M_b)	Torsion (M_t)
Total Loads (EPU)	29	18	1,731	878
Calculated Stress Intensity (EPU)	38,683 psi			
Allowable Stress Intensity	60,520 psi			
Design Margin	0.64			

Feedwater Penetration P-4				
Description	Forces (Kips)		Moments (in-Kips)	
	Axial (F_a)	Shear (F_V)	Bending (M_b)	Torsion (M_t)
Total Loads (EPU)	29	22	2,280	710
Calculated Stress Intensity (EPU)	38,623 psi			
Allowable Stress Intensity	60,520 psi			
Design Margin	0.64			

Response 24b:

The load combination for the calculated loads presented in the response to RAI EMCB-24a and in EPU LAR Attachment 5, Tables 2.2.2.2-3 (Main Steam) and Table 2.2.2.2-4 (Feedwater) is as follows:

Deadweight + Thermal + SSE + Fluid Transient

Response 24c:

The allowable stress intensity values were obtained from the containment penetration allowable stress values summarized in St. Lucie Unit 1 UFSAR Appendix 3G5. The justification for using the St. Lucie Unit 1 allowable stress intensities is based on a comparison of applicable St. Lucie Unit 1 and Unit 2 containment penetration attributes including component materials, physical geometric details, design temperature and pressure.

EMCB-25

For the replacement feedwater pump nozzle load summary shown on EPU LR Table 2.2.2.2-2; (a) please discuss the basis for not including seismic loads; (b) discuss the basis for comparing the calculated loads to twice the American Petroleum Institute (API) allowable value and (c) show whether the calculated loads meet the pump specification/vendor nozzle allowable loads.

Response 25a:

The St. Lucie Unit 2 feedwater pump nozzle load summaries are provided in EPU LAR Attachment 5, Table 2.2.2.2-6. The tables that follow provide an updated version of EPU LAR Attachment 5, Table 2.2.2.2-6 and contain the final feedwater pump nozzle load summaries resulting from EPU.

The feedwater pumps are located upstream of the seismic/non-seismic piping system boundary (i.e., located in a non safety-related/non seismic portion of the piping system, with no seismic II/I concerns). As such, there are no seismic loads that need to be considered in the feedwater pump nozzle analyses.

Feedwater Pump 2A						
Case Number	Forces (Lbs)			Moments (Ft-Lbs)		
	Fx (Shear)	Fy (Shear)	Fz (Axial)	Mx (Bending)	My (Bending)	Mz (Torsion)
Deadweight (DW)	-46	-455	108	-111	-111	-1455
ABSMAX (TH-I, TH-2, TH-3, TH-4)	1574	1814	642	1718	3593	1955
MAX (FRV, FIV)	130	227	537	342	322	616
Static Total	1,620	1,359	534	1,607	3,704	3,410
Dynamic Total	1,750	1,586	1,071	1,949	4,026	4,026
2 x API Allowable	3,200	2,600	4,000	9,400	7,000	4,600
Static Interaction Ratio	0.51	0.52	0.13	0.17	0.53	0.74
Dynamic Interaction Ratio	0.55	0.61	0.27	0.21	0.58	0.88

Notes:

ABSMAX = Absolute Maximum Value
MAX = Maximum Value
TH-1 = Thermal Mode 1 – Both Pumps On
TH-2 = Thermal Mode 2 – Feed Pump 2B Off
TH-3 = Thermal Mode 3 – Feed Pump 2A Off
TH-4 = Thermal Mode 4 – Both Pumps Off, Lines @ 40°F
FRV = Feedwater Regulating Valve
FIV = Feedwater Isolation Valve

Static Total = Max [DW, (DW + TH_1), (DW + TH_2), (DW + TH_3), (DW+ TH_4)]
Dynamic Total = {MAX(FRV, FIV) + Max [DW, (DW + TH_1), (DW + TH_2), (DW + TH_3), (DW+ TH_4)]}

Static Interaction Ratio = Static Total / 2xAPI Allowable
Dynamic Interaction Ratio = Dynamic Total / 2xAPI Allowable

Feedwater Pump 2B						
Loadings	Forces (Lbs)			Moments (Ft-Lbs)		
	Fx (Shear)	Fy (Shear)	Fz (Axial)	Mx (Bending)	My (Bending)	Mz (Torsion)
Deadweight (DW)	-226	-897	-12	484	109	-884
ABSMAX (TH-l, TH-2, TH-3, TH-4)	1628	976	1198	4051	1134	647
MAX (FRV, FIV)	102	157	363	77	97	238
Static Total	1,854	1,873	1,210	4,535	1,243	1,531
Dynamic Total	1,956	2,030	1,573	4,612	1,340	1,769
2 x API Allowable	3,200	2,600	4,000	9,400	7,000	4,600
Static Interaction Ratio	0.58	0.72	0.30	0.48	0.18	0.33
Dynamic Interaction Ratio	0.61	0.78	0.39	0.49	0.19	0.38

Notes:

ABSMAX = Absolute Maximum Value
MAX = Maximum Value
TH-1 = Thermal Mode 1 – Both Pumps On
TH-2 = Thermal Mode 2 – Feed Pump 2B Off
TH-3 = Thermal Mode 3 – Feed Pump 2A Off
TH-4 = Thermal Mode 4 – Both Pumps Off, Lines @ 40°F
FRV = Feedwater Regulating Valve
FIV = Feedwater Isolation Valve

Static Total = Max [DW, (DW + TH_1), (DW + TH_2), (DW + TH_3), (DW+ TH_4)]
Dynamic Total = {MAX(FRV, FIV) + Max [DW, (DW + TH_1), (DW + TH_2), (DW + TH_3), (DW+ TH_4)]}

Static Interaction Ratio = Static Total / 2xAPI Allowable
Dynamic Interaction Ratio = Dynamic Total / 2xAPI Allowable

Response 25b and 25c:

The 2 times API allowable values used correspond to the allowable nozzle load limits that are summarized on the replacement feedwater pump vendor drawing. These allowable nozzle values are also contained in the feedwater pump specification. The calculated feedwater pump nozzle loads for EPU are within the allowable nozzle loads contained in the feedwater pump specification.

EMCB-26

Table 2.2.2.2-2 shows that a number of new supports are required to be added to the existing piping. Given that the EPU does not change the deadweight and seismic loads, please explain why new snubber and spring supports are required to be added.

Response:

The St. Lucie Unit 2 EPU LAR submittal did not contain a specific list of pipe support modifications. The table that follows provides a list of the St. Lucie Unit 2 pipe support modifications that were required due to EPU.

With respect to the new/replacement snubbers that are shown, these supports were mainly required to accommodate revised fluid transient and vibration loads on the main steam, feedwater and condensate piping systems. Spring hanger modifications were mainly required to accommodate revised thermal expansion piping displacements resulting from EPU.

Pipe Support Modifications for EPU Conditions					
Item	**Support Mark Number**	**System**	**Pipe Size (in.)**	**Building Location**	**Modification Description**
1	CC-2063-6461	CC	6	Reactor Auxiliary Building	Add tube steel above pipe to make vertical restraint from vertical support
2	CC-2063-6464A	CC	6	Reactor Auxiliary Building	Replace vertical strut with variable spring
3	CC-2064-86	CC	10	Reactor Auxiliary Building	Lower horizontal strut to reduce cantilever length, add weld at baseplate
4	CC-2074-59	CC	24	CC Building	Replace axial strut with snubber, shim pipe clamp lugs
5	CC-2074-7388	CC	24	CC Building	Shim pipe clamp lugs for axial strut
6	CC-2074-7810	CC	20	CC Building	Add weld between baseplate and embedded plate
7	CC-2074-8088	CC	14	CC Building	Add weld between anchor's trunnion and tube steel
8	CC-2074-8200	CC	20	CC Building	Add weld at bottom TS to TS connection and at brace to frame connection
9	CC-2163-8050	CC	4	Reactor Auxiliary Building	Replace existing trunnion with larger dia. trunnion, add TS
10	CC-2063-7427	CC	20	Reactor Auxiliary Building	Replace portions of W10 with TS to accommodate pipe branch reinforcing pad
11	MS-4100-6080	MS	34	Reactor Containment	New support
12	MS-4101-315A	MS	34	Reactor Containment	New support
13	MS-4102-274	MS	38	Turbine Bldg	Add weld
14	MS-4102-32B	MS	38	Turbine Bldg	Add cover plate
15	MS-4102-48	MS	38	Turbine Bldg	Add TS brace
16	MS-4102-3910	MS	38	Turbine Bldg	Add weld
17	BF-4004-1811	FW	20	Turbine Bldg	Replace snubber
18	BF-4004-7106	FW	20	Turbine Bldg	Add weld
19	BF-4004-7060	FW	16	Turbine Bldg	Add weld
20	BF-4004-6029A	FW	20	Turbine Bldg	Add weld
21	BF-4004-258	FW	20	Turbine Bldg	Add weld
22	BF-4004-7185B	FW	20	Turbine Bldg	Add weld
23	BF-4004-7168A	FW	20	Turbine Bldg	Replace frame members
24	BF-4004-41	FW	20	Turbine Bldg	Add weld
25	BF-4004-176A	FW	20	Turbine Bldg	New support
26	BF-4004-188	FW	20	Turbine Bldg	Replace snubber
27	BF-4004-192A	FW	20	Turbine Bldg	New support
28	BF-4004-1671	FW	20	Turbine Bldg	New support
29	BF-4004-1770	FW	20	Turbine Bldg	New support
30	BF-4004-1851	FW	20	Turbine Bldg	New support

Pipe Support Modifications for EPU Conditions					
Item	**Support Mark Number**	**System**	**Pipe Size (in.)**	**Building Location**	**Modification Description**
31	BF-4004-1861	FW	20	Turbine Bldg	New support
32	BF-4004-1925	FW	20	Turbine Bldg	New support
33	BF-4004-6054	FW	20	Turbine Bldg	Replace springs
34	BF-4004-6057A	FW	20	Turbine Bldg	Replace springs
35	BF-4004-7185A	FW	20	Turbine Bldg	Replace snubber
36	BF-4004-38	FW	20	Turbine Bldg	Replace snubbers
37	BF-4004-7026	FW	20	Turbine Bldg	Add weld
38	BF-4004-7084	FW	20	Turbine Bldg	Replace strut
39	BF-4004-162	FW	20	Turbine Bldg	Add weld
40	CH-7	C	24	Turbine Bldg	Add IWA stiffener
41	CH-13	C	24	Turbine Bldg	Replace rod & beam
42	CH-36	C	20	Turbine Bldg	Add IWA stiffener & replace rod
43	C-E-13-H1	C	24	Turbine Bldg	New support
44	C-E-13-H2	C	24	Turbine Bldg	New support
45	C-E-14-H1	C	24	Turbine Bldg	New support
46	CH-38	C	20	Turbine Bldg	Replace rod & pipe clamp
47	CVR-15	C	8	Turbine Bldg	Replace strut with snubber
48	CVR-38	C	14	Turbine Bldg	Reset spring
49	CH-56	C	24	Turbine Bldg	Replace spring
50	CH-57	C	24	Turbine Bldg	Add structural members
51	CH-61	C	24	Turbine Bldg	Add IWA stiffener & structural members
52	CH-175	C	20	Turbine Bldg	Replace spring
53	CH-176	C	20	Turbine Bldg	Replace spring, rod & pipe clamp
54	CH-817	C	8	Turbine Bldg	Replace spring and add structural members
55	C-E-4-H1	C	20	Turbine Bldg	New support
56	C-E-4-H2	C	20	Turbine Bldg	New support
57	CH-816	C	20	Turbine Bldg	Add structural members
58	CH-28	C	24	Turbine Bldg	Replace rod
59	CH-631	C	20	Turbine Bldg	Replace rod & pipe clamp
60	CH-42	C	24	Turbine Bldg	Replace trapeze & rod
61	CH-60A	C	24	Turbine Bldg	Replace strut with snubber
62	C-E-34-H4	C	24	Turbine Bldg	Replace rod with strut
63	CH-60	C	24	Turbine Bldg	Replace struts with frame
64	CH-656	C	24	Turbine Bldg	Replace rod with strut
65	CH-54	C	24	Turbine Bldg	Reset spring
66	CH-713	C	8	Turbine Bldg	Add cover plates
67	C-E-35-H2	C	24	Turbine Bldg	Add brace
68	C-E-35-H1	C	24	Turbine Bldg	Add brace
69	CH-55	C	24	Turbine Bldg	Replace strut
70	C-E-36-H1	C	24	Turbine Bldg	Add structural members
71	CVR-10	C	8	Turbine Bldg	Replace struts with snubbers
72	CVR-14	C	8	Turbine Bldg	Replace struts with spring and snubber
73	HDH-300	HD	20	Turbine Bldg	Reset springs
74	C-E-36-H2	C	24	Turbine Bldg	Add brace
75	HDH-188	HD	14	Turbine Bldg	Replace spring
76	CH-41	C	24	Turbine Bldg	Replace saddle
77	CH-25	C	24	Turbine Bldg	Replace saddle
78	HDH-66	HD	14	Turbine Bldg	Replace springs, trapeze and frame

Pipe Support Modifications for EPU Conditions					
Item	**Support Mark Number**	**System**	**Pipe Size (in.)**	**Building Location**	**Modification Description**
79	HDH-184	HD	14	Turbine Bldg	Add weld
80	HDH-301	HD	14	Turbine Bldg	Replace spring, rod & pipe clamp
81	HDVR-8	HD	14	Turbine Bldg	Replace strut with snubber
82	CH-48	C	24	Turbine Bldg	Replace pipe clamp
83	HDH-183	HD	14	Turbine Bldg	Replace springs, trapeze and frame
84	CVR-17	C	8	Turbine Bldg	New support
85	CVR-12	C	8	Turbine Bldg	Replace frame with snubber
86	CH-633	C	20	Turbine Bldg	Replace rods, trapeze and frame
87	CH-617	C	24	Turbine Bldg	Replace rod with strut
88	CH-30	C	20	Turbine Bldg	Replace springs, trapeze and frame
89	CH-714	C	8	Turbine Bldg	Remove U-Bolt
90	CH-715	C	8	Turbine Bldg	Remove U-Bolt
91	C-E-35-H4	C	24	Turbine Bldg	Remove rod
92	MS-12-7	MS	1	Turbine Bldg	New snubber
93	MS-3023-13	MS	4	Steam Trestle	Modify support member
94	HDVR-5	HD	10	Turbine Bldg	Replace snubber
95	HDVR-6	HD	10	Turbine Bldg	Replace snubber
Notes: C = Condensate CC = Component Cooling Water FW = Feedwater HD = Heater Drains MS = Main Steam IWA = Integral Welded Attachment					

EMCB-27

In qualifying the reactor pressure vessel (RPV) structural steel supports for EPU, the EPU LR makes the statement, on pages 2.2.2-50 and 2.2.2-51, that the deadweight (DWT) plus thermal load combination is bounded by the original design and provides LR Table 2.2.2.3-4. This table includes the original design loads shown in the FSAR Table 3H-1. In the LR table, though, the "Thermal" column under the "Original Design" section mistakenly shows the loads of the FSAR table under the "Thermal + D. WT" column. Because of this error, the sum of the DWT plus thermal loads shown in the LR table exceed the original design load combination case of thermal plus DWT for all points listed. In addition, the maximum frictional load ("F") of (+/-)540 (kips) shown on the LR table should have been (+/-)346.5 (kips), which is also less than the EPU frictional load of (+/-)370 (kips). Please explain how this issue will be resolved for the EPU qualification of the RPV supports.

Response:

This specific question is not applicable to the St. Lucie Unit 2 EPU LAR. There is no corresponding table to the St. Lucie Unit 1 UFSAR Table 3H-1 in the St. Lucie Unit 2 UFSAR. St. Lucie Unit 2 EPU LAR Attachment 5, Table 2.2.2.3-4 compares EPU loads to the pre-EPU loads specified in the reactor pressure vessel specification. There are no errors in St. Lucie Unit 2 EPU LAR Attachment 5, Table 2.2.2.3-4.

EMCB-28

Please discuss whether the control element drive mechanisms (CEDMS) were reanalyzed for EPU or were the AOR utilized and scaling factors employed to produce the stresses shown in tables 2.2.2.4-2 through 2.2.2.4-7. Please discuss how the scaling factors were derived and employed.

Response:

Control element drive mechanism (CEDM) stresses for St. Lucie Unit 2 are provided in EPU LAR Attachment 5, Tables 2.2.2.4-2 through 2.2.2.4-6. A reanalysis of the St. Lucie Unit 2 CEDM stresses for EPU conditions was not required because design conditions for the current CEDM analysis bound the EPU conditions. Scaling factors were not required for the EPU analysis of the CEDM stresses.

EMCB-29

a) **Please discuss why it was necessary to reanalyze the SG support sliding base plate (SBP) for EPU (whether the SG flooded DWT load case was the only difference for the EPU reanalysis). Discuss the differences between the current analysis load cases and the EPU load cases.**

b) **Please discuss the basis that justifies the use of different than CLB ASME code sections for the SBP EPU reanalysis and whether these sections were utilized for the replacement SG (RSG).**

Response 29a:

As described in St. Lucie Unit 2 EPU LAR Attachment 5, Section 2.2.2.5.2.10.3, it was not necessary to reanalyze the St. Lucie Unit 2 SG SBP for EPU conditions since the SBP loads under EPU normal operating conditions are less than the current licensing basis loads. The SG SBP load cases for the EPU are the same as the current analysis load cases.

Response 29b:

As discussed above in the response to RAI EMCB-29a, a reanalysis of the SBP for the EPU was not required. As such, use of a different ASME Code section is not applicable.

EMCB-30

Discussion in the EPU LR indicates that the reanalysis of the SG SBP has shown some SG uplift at normal operating (NOP) conditions, documented on LR Table 2.2.2.5-2. It also states that "the vertical uplift is bounded by the pre-EPU design since the EPU displacements were obtained from a model using a heavier SG, heavier reactor coolant pump (RCP) motors and a negligible rise in EPU temperature of 1°F."

a) **Describe the analysis model (including its components, boundary conditions and loading cases) and discuss whether this condition, which shows that only one out of four corners of the SG1B base support is bearing weight, is common in similar plant designs.**

b) **Please explain why the EPU modeled SGs and RCP motors are heavier than the existing design and whether these components are been replaced for the EPU.**

c) **Please list the built-in gaps at the SG base supports and show the bounding pre-EPU lift-off movements.**

d) **From EPU LR Table 2.2.2.5-1, it can be shown that the lower SG sliding base support has a coefficient of friction of 0.3 for all table columns with the exception of the first column. Please review the frictional loads and coefficient of friction for possible errors.**

Response 30a:

As discussed in the response to RAI EMCB-29, the St. Lucie Unit 2 SG SBP was not reanalyzed for EPU conditions. Similar to the St. Lucie Unit 1 response to RAI EMCB-30, the condition of only one of the four corners of the SG 2B base support bearing weight is common in similar plant designs. NOP gaps between the St. Lucie Unit 2 SG pads and the sliding base at EPU conditions are shown in the table below.

NOP Steam Generator Sliding Base Plate (SBP) Gaps

SG 2A	**Lift-off (in)**
Y1	0.026
SG 2B	**Lift-off (in)**
Y1	0.051
Y2	0.007
Y4	0.007

Response 30b:

The St. Lucie Unit 2 reactor coolant system (RCS) model was updated for EPU conditions. The EPU analysis model used SGs and RCP motors that are heavier than those used in the original analysis because the replacement SGs and replacement RCP motors are heavier than the original components. The appropriate analyses were performed for the replacement SGs and replacement RCP motors as part of the design modification process. Replacement of the SGs and RCP motors is not part of the EPU project.

Response 30c:

The limiting vertical hot gap size is 1/64 or 0.0156 inches. The difference in the SG lift-off movement between current and EPU conditions is insignificant. The resulting EPU hot gap satisfies the gap size requirement while maintaining a clearance that prevents the SG support system from binding up during operation.

Response 30d:

The comparison of the SG SBP pre-uprate and EPU loads is provided in St. Lucie 2 EPU LAR Attachment 5, Table 2.2.2.5-14. The load due to friction listed for current conditions in EPU LAR Attachment 5, Table 2.2.2.5-14, 483 kips, was conservatively calculated and has been reviewed; this table contains no errors.

EMCB-31

LR Page 2.2.2-74 states that, "The increase in total DWt load was evaluated and found to be bounded by the pre-EPU design basis loads." Please confirm whether this DWT increase is referring to the flooded SG DWT loading case.

Response:

The St. Lucie Unit 2 EPU LAR Attachment 5, Section 2.2.2.5.2.10.3 states, in part, "In addition [to the full power DWt and NOP analyses], a DWt only analysis for ambient temperature conditions with flooded SGs was performed to maximize the loads on the SG supports."

This case was run to capture a condition that might possibly govern over the normal operating pressure thermal plus dead weight case in order to ensure that the maximum loads on the sliding base and the foundation were considered in the EPU analysis. The resulting deadweight loads are shown in EPU LAR Attachment 5, Table 2.2.2.5-14 as "EPU Load". The original design basis deadweight loads on the steam generator lower supports (as shown in EPU LAR Attachment 5, Table 2.2.2.5-14 as "Pre-Uprate Load") did not reflect the flooded steam generator condition.

The (flooded) deadweight load and the combination of thermal plus deadweight load, acting on the steam generator foundation are listed under "EPU Load" in EPU LAR Attachment 5, Table 2.2.2.5-14. The total combined foundation loads are bounded by the original design basis accident condition loads that governed the design of the steam generator foundation.

EMCB-32

LR Page 2.2.2-74 states that, "SGSB supports were not evaluated for lateral forces, since SSE and rupture loads remain unchanged for EPU. In all cases, the stresses for all SG support components satisfy applicable acceptance criteria." Please explain how the lateral frictional forces have been accounted for in the EPU evaluation of the SG supports and provide the basis of the "applicable design criteria."

Response:

The St. Lucie Unit 2 steam generator sliding base (SGSB) support and foundation are designed for thermal expansion loads in the North-South direction transmitted via friction forces from the sliding base. The term "lateral forces" in the above quoted excerpt from the St. Lucie Unit 2 EPU LAR was intended to mean forces in the East-West direction. Because of the symmetry of the reactor coolant system (RCS) hot and cold legs, there is negligible thermal movement in the East-West direction. The results of the RCS analysis indicate that there are no thermal loads (and hence, no frictional forces) imposed on the steam generator sliding base support in the East-West direction. There are seismic and LOCA pipe rupture loads in this direction and these are transferred to the steam generator foundation by means of the shear keys in the SGSB support.

The acceptance criteria for the design of the steam generator supports is as follows:

- Steam generator sliding base (SGSB) – ASME Code, Section III, 1977 Edition through Summer 1977 Addenda, Subsection NF
- SGSB support – AISC Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, dated February 12, 1969
- Steam generator foundation – ACI 318-71, Building Code Requirements for Reinforced Concrete

EMCB-33

LR Table 2.2.2.5-4 shows that fatigue evaluation was performed for the feedwater nozzle, while other secondary pressure boundary components, including the main steam nozzle, are exempt from fatigue. Please provide a technical justification which shows that these components are not required to be evaluated for fatigue.

Response:

Table 2.2.2.5-9 of Attachment 5 to the St. Lucie Unit 2 EPU LAR provides the results of the stress and fatigue analyses performed for the replacement steam generators (RSG) at EPU conditions. This table is equivalent to Table 2.2.2.5-4 of Attachment 5 of the St. Lucie Unit 1 EPU LAR.

As indicated in St. Lucie Unit 2 EPU LAR Attachment 5, Table 2.2.2.5-9, fatigue analyses have been performed for the RSG feedwater nozzles, main steam nozzles, and other secondary pressure boundary components. None of the Unit 2 RSG secondary pressure boundary components were exempt from fatigue at EPU conditions.

EMCB-41

Please provide a comparison of the RCP nozzle loads used in the current design basis analysis to the EPU-derived RCP nozzle loads to support the statement addressed in the LR that the increased EPU RCP nozzle forces will not affect the existing analysis on record stresses and fatigue cumulative usage factors (CUFs).

Response:

All RCP nozzle loads for the EPU are less than the current RCP nozzle loads, except for the shear force at pumps A1 and B2 discharge nozzles (see table below). The shear force at pumps A1 and B2 discharge nozzles exceeds the current force by [ ](a,c) kip. This results in an insignificant increase in stress, as shown below:

(a,c)

This increase in stress is negligible. Seismic and pipe break loads were not affected by the EPU.

RCP Nozzle Loads

Condition	Pump	Nozzle	Current			EPU		
			Axial (kips)	Shear (kips)	Resultant Moment (in-kips)(1)	Axial (kips)	Shear (kips)	Resultant Moment (in-kips)(1)
Thermal + Deadweight	A2 & B1	D						
	A1 & B2	D						
	A2 & B1	S						
	A1 & B2	S						

(a,c)

Note: (1) Resultant moments are calculated by the square root of the sum of squares of all three directions.

EMCB-42

EPU LR Page 2.2.2-117 states that:

> **Since the hydraulic snubber located at the top of the each RCP motor is inactive (i.e., offers no resistance) under normal operation conditions, there is no load path above the RCP nozzles. Since the seismic and pipe break effects are also not changed by the EPU (as discussed in LR Section 2.2.2.6.2.2), the stress AOR for the RCP casing, motor, motor mount flange and flange studs is not changed by the EPU.**

a) **Please verify whether the RCPs and RCP motors are only supported on snubbers and springs.**

b) **Table 2.2.2.6-2 shows an approximate increase of 24,000 lbs DWT load going through the RCP and reacted by the RCP springs (assuming that (a) above is applicable). Please provide a technical justification which shows that the increase of 24,000 lbs does not impact the structural integrity of the RCP and its pressure retaining components as calculated in the AOR design calculations.**

Response 42a:

It has been verified that the reactor coolant pumps (RCPs) and RCP motors are only supported on snubbers and springs.

Response 42b:

The increase in load does not impact the structural integrity of the RCP or its pressure-retaining components because maintaining the pressure boundary is controlled by changes in the RCP nozzle loads, not by changes in the pump hanger loads. As long as increases in pump hanger loads do not exceed the capacity of the variable spring hangers, which is the case for the EPU, the pump vertical support systems will remain intact and provide the required support. The EPU analysis demonstrated that the structural integrity of the RCP nozzles is not compromised; therefore, the RCP pressure boundary remains intact for the EPU changes in RCP loadings.

EMCB-43

The EPU LR indicates that evaluations for critical reactor vessel internals (RVI), listed on pages 2.2.3-11 and 2.2.3-12, were performed to assess the EPU effects on the RVI.

a) **Please discuss whether these evaluations changed the AOR results for the RVI and whether the Table 2.2.3-1 values are for EPU conditions.**

b) **If the EPU evaluations were performed using as basis the AOR design calculations, explain the methodology used to derive EPU results from the existing calculations on record.**

Response 43a:

Evaluations performed for the St. Lucie 2 EPU LAR changed the analysis of record (AOR) results for thermal stresses in the RVI components. Primary stresses from the AOR were not changed for EPU. EPU LAR Attachment 5, Table 2.2.3-1 values reflect EPU conditions.

Response 43b:

The methodology used to derive the EPU results is:

1. Primary stresses in the St. Lucie Unit 2 RVI components were obtained from the AOR. These primary stresses were not affected by EPU.
2. Temperatures and thermal stresses in RVI components were obtained from analyses performed for EPU conditions. No analyses of record were used to calculate thermal stresses.
3. Stresses were combined and evaluated with respect to the acceptance criteria.
4. A fatigue evaluation of the RVI components was performed.

EMCB-44

The EPU LR Page 2.2.3-13 states that:

> **LR Table 2.2.3-1 indicates that the core shroud primary plus secondary stress intensity exceeded the 3Sm limit imposed by the ASME Code. Acceptability of the core shroud was shown by applying the simplified elastic-plastic analysis identified by ASME Code Paragraph NG-3228.3.**

In relevance to the above statement, Table 2.2.3-1 only contains a double asterisk (**) note which does not correspond to any values or areas in the table. Please review this table and (in addition to the justification statement shown on LR Page 2.2.3-13) quantitatively show, as a minimum, the calculated values which exceed the primary plus secondary stress intensity limit of $3S_m$, the value of $3S_m$; show that when removing the thermal bending stresses, the primary plus secondary stress intensity value is less than $3S_m$; and show that for these components the fatigue CUF is less than 1.0, when calculated in accordance with the provisions of NG-3228.3.

Response

The subject RAI is applicable to St. Lucie Unit 2. However, instead of a double asterisk, the same condition is indicated in St. Lucie Unit 2 EPU LAR Attachment 5, Table 2.2.3-1 with a superscript 1 (^{1}). The superscript 1 (^{1}) is shown adjacent to the text for the core shroud girth rib (fatigue usage[1]).

Only the core shroud girth rib did not meet the $3S_m$ limit. The specific values shown below reflect the worst case stress results, which occur at girth rib 6. The primary plus secondary stress intensity without thermal bending is calculated below:

$P_m + P_b + Q = [\qquad]^{(a,c)} > 3S_m = [\qquad]^{(a,c)}$

Without Thermal Bending:
Primary Membrane plus Bending Stress Intensity: $SI_{m+b} = [\qquad]^{(a,c)}$
Thermal Membrane Stress Intensity: $SI_{tm} = [\qquad]^{(a,c)}$
$P_m + P_b + Q_m = SI_{m+b} + SI_{tm} = [\qquad]^{(a,c)} < 3S_m = [\qquad]^{(a,c)}$

As shown in EPU LAR Attachment 5, Table 2.2.3-1, CUF = $[\qquad]^{(a,c)}$.

EMCB-45

Please provide assurance that all existing or added non-safety related SSCs have been evaluated to preclude failure that could prevent the satisfactory accomplishment of a function required by 10 CFR 54.4(a)(1) and (a)(3).

Response:

Non-safety related systems, structures, and components (SSCs) whose failure could prevent satisfactory accomplishment of any of the functions identified for safety-related SSCs as required by 10 CFR 54.4(a)(1) and (a)(3) have been evaluated in terms of interactions with safety-related SSCs.

The various evaluations related to the impact of non-safety related SSCs upon safety-related SSCs are included in the St. Lucie Unit 2 EPU LAR by virtue of the requirements of RS-001 "Review Standard for Extended Power Uprates." These evaluations are presented in Attachment 5 of the EPU LAR and identified below.

Evaluation	EPU LAR Section
Pipe Rupture Locations and Associated Dynamic Effects	Attachment 5, Section 2.2.1
Balance of Plant Piping, Components, and Supports	Attachment 5, Section 2.2.2.2
Flooding	Attachment 5, Section 2.5.1.1
Missile Protection	Attachment 5, Section 2.5.1.2
Pipe Failures	Attachment 5, Section 2.5.1.3

In addition to the above evaluations, as a matter of practice, the St. Lucie design change process includes requirements to evaluate the interaction of non-safety related SSCs with safety-related SSCs under all modes of operation, including plant start-up, shutdown, normal power operation, off-normal operation, abnormal operation, and emergency operation. In accordance with design change procedures, the interactions reviewed include high-energy pipe breaks and interaction of seismically supported non-safety related systems with safety-related SSCs, i.e. "seismic II over I." The requirements inherent in the design change process with respect to performing interaction evaluations ensures that compliance with the requirements of 10 CFR 54.4(a)(1) is afforded for EPU.

10 CFR 54.4(a)(3) states that SSCs relied on in safety analyses or plant evaluations need to demonstrate compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63). FPL compliance with the aforementioned regulations is discussed in Attachment 5 of the EPU LAR in the sections identified below.

Regulation	EPU LAR Section
Fire Protection (10 CFR 50.48)	Attachment 5, Section 2.5.1.4
Environmental Qualification (10 CFR 50.49)	Attachment 5, Section 2.3.1
Pressurized Thermal Shock (10 CFR 50.61)	Attachment 5, Section 2.1.3
Anticipated Transients Without Scram (10 CFR 50.62)	Attachment 5, Section 2.8.5.7
Station Blackout (10 CFR 50.63)	Attachment 5, Section 2.3.5

EMCB-46

a) **Confirm that the cumulative fatigue usage factors contained in the EPU licensing report, Section 2.2, Mechanical and Civil Engineering, have been derived to include the 60 year plant life extension (60yr-PLEX).**

b) **Confirm that, when required to perform stress-based fatigue monitoring, the St. Lucie 1 current and plant licensing renewal to 60yr-PLEX fatigue monitoring program(s) is performed in accordance with the ASME Section III, NB-3200, and includes all six stress components.**

Response:

a) St. Lucie Unit 2 EPU LAR Attachment 5, Section 2.2.6 "NSSS Design Transients" provides a discussion of the design transients utilized as inputs to the EPU component fatigue analyses contained in EPU LAR Attachment 5, Section 2.2 "Mechanical and Civil Engineering." EPU LAR Attachment 5 Tables 2.2.6-1, 2.2.6-2, and 2.2.6-3 list the Nuclear Steam Supply System (NSSS) design transients and frequencies of occurrence assumed in the EPU component fatigue analyses. The number of transient occurrences assumed for EPU are the same as the existing design basis transient occurrences. EPU LAR Attachment 5, Section 2.2.6 further states that the selected transients were considered for the 60-year operating license period of the plant. EPU LAR Attachment 5, Section 2.14, "Impact of EPU on the Renewed Plant Operating License" also states that for ASME Section III, Class 1 components, the number of design cycles that systems or components will be subjected to during the 60-year license renewal period of operation will not be exceeded.

b) As discussed in St. Lucie Unit 2 UFSAR Section 18.2.6, the Fatigue Monitoring Program is based on the manual logging of design cycles throughout the life of the plant. The program does not rely on an online fatigue monitoring system. Stress-based fatigue analyses performed for St. Lucie do not rely on the simplified single stress methodology described in NRC Regulatory Issue Summary (RIS) 2008-30, "Fatigue Analysis of Nuclear Power Plant Components." Fatigue analyses for St. Lucie Unit 2 license renewal were perfomed in accordance with the rules of ASME Section III, Subsection NB-3200 which considers the six stress components.

EMCB-47

The EPU LAR proposes to add Metamic neutron-absorbing inserts to the fuel assemblies in the rack storage cells of the spent fuel pool. Provide a technical justification for the acceptability of the structural integrity of the spent fuel pool and its contents due to the addition of the Metamic inserts.

Response:

The structural analysis of the St. Lucie Unit 2 spent fuel storage racks and spent fuel pool for EPU was performed by Holtec International. The structural response of the spent fuel storage racks is influenced by their self-weight and the weight of the stored fuel assemblies. The Holtec analysis treats the Metamic insert as an integral part of the fuel assembly; its weight is considered as contributing to the mass of the fuel assembly but its structural properties are not credited in the analysis. The weight of the individual Metamic insert is assumed in the analysis as 25 lbs. Approximately half of the fuel storage locations will be fitted with the inserts. This would amount to an increase in the inertial mass of the loaded rack due to the inserts of less than 1%. The Holtec analysis demonstrated that the most stressed location of the spent fuel storage rack for the governing loading condition is bounded by an allowable stress that represents a margin of safety of 7.9%. Inasmuch as the dynamic response of the structure changes in an approximately proportional manner under slight perturbations in its mass, the increases in stresses and displacements in the spent fuel storage racks due to the presence of the Metamic inserts is accordingly expected to be no greater than 1%, and therefore well within the available margin.

The analysis of the spent fuel pool floor structure performed for the interface loads imposed by the spent fuel racks considered maximum horizontal and vertical loads of 66 k and 158 k, respectively vs. allowable bearing loads of 119 k and 238 k, respectively. This results in a minimum safety factor of 1.51 for the bearing stress in the concrete pool floor slab. The small (< 1%) increase in the weight of the loaded spent fuel storage rack attributed to the addition of Metamic inserts in half the cells will have no appreciable effect on this margin. Thus, it is concluded that the concrete pool floor slab design is capable of accommodating the increase in dead weight and seismic loading imparted to the pool floor slab as a result of the addition of the Metamic inserts.

ATTACHMENT 3

Response to NRC Mechanical and Civil Engineering Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

NON-PROPRIETARY INFORMATION

Westinghouse Application for Withholding Proprietary Information from Public Disclosure

(Cover page plus 7 pages)

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 720-0754
e-mail: greshaja@westinghouse.com
Proj letter: FPL-12-61

CAW-12-3418

February 23, 2012

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: St. Lucie Unit 2 Responses to NRC Requests for Additional Information (RAIs) EMCB 2, 41 and 44, (Proprietary)

References:

1. NRC E-Mail, T. ORF (NRC) to C. Wasik (FPL), "St. Lucie 2 EPU Draft RAIs - Mechanical & Civil Engineering Branch (EMCB)," January 13, 2012, 12:48 PM

Reference 1 requested additional information on aspects of the St. Lucie Unit 2 Extended Power Uprate (EPU) License Amendment Request. The proprietary information for which withholding is being requested is contained in the responses to RAIs EMCB 2, 41 and 44, and is further identified in Affidavit CAW-12-3418 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Florida Power and Light.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-12-3418, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

for J. A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

STATE OF CONNECTICUT:

ss Windsor Locks

COUNTY OF HARTFORD:

Before me, the undersigned authority, personally appeared C. M. Molnar, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

C. M. Molnar, Senior Engineer
Regulatory Compliance

Sworn to and subscribed before me
this 23rd day of February 2012

Notary Public

Subscribed and Sworn to before me, a Notary Public, in and for County of Hartford and State of Connecticut, this 23rd day of February 2012

JOAN GRAY
Notary Public
My Commission Expires January 31, 2017

(1) I am Senior Engineer, Regulatory Compliance, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in responses to EMBC RAIs 2, 41 and 44 (Proprietary), for submittal to the Commission, being transmitted by Florida Power and Light letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The RAIs identified above were transmitted by NRC E-Mail, T. ORF (NRC) to C. Wasik (FPL), "St. Lucie 2 EPU Draft RAIs - Mechanical & Civil Engineering Branch (EMCB)," January 13, 2012, 12:48 PM. The proprietary information as submitted by Westinghouse is provided in support of the St. Lucie Unit 2 EPU License Amendment Request, and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Support FPL's efforts to obtain approval of the St. Lucie Unit 2 EPU License Amendment Request.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of this information to its customers for purposes of justifying piping analyses and certain component nozzle loads and stresses under EPU conditions.

(b) Westinghouse can sell support and defense of the use of various plant components under EPU conditions.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).