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

FPL

~~Proprietary Information – Withhold From Public disclosure Under 10 CFR 2.390~~

September 23, 2011

L-2011-361
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 1
Docket No. 50-335
Renewed Facility Operating License No. DPR-67

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

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2010-259), "License Amendment Request (LAR) for Extended Power Uprate," November 22, 2010, Accession No. ML103560419.

(2) Email from T. Orf (NRC) to C. Wasik (FPL), "St. Lucie Unit 1 EPU draft Mechanical and Civil RAIs (EMCB)," July 27, 2011.

By letter L-2010-259 dated November 22, 2010 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. DPR-67 and revise the St. Lucie Unit 1 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 July 27, 2011 [Reference 2], additional information related to mechanical and civil engineering topics was requested by the NRC staff in the Mechanical and Civil Engineering Branch (EMCB) to support their review of the EPU LAR. The request for additional information (RAI) identified forty-five (45) questions. Response to these RAIs is provided in Attachment 1 to this letter with the exception of a response to EMCB RAI-23. FPL's response to EMCB RAI-23 will be provided by October 10, 2011. Attachment 1 contains Westinghouse and Babcock and Wilcox (B&W) proprietary information and Attachment 2 is the fully non-proprietary version of Attachment 1.

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Attachments 3 and 4 contain the Westinghouse and B&W Proprietary Information Affidavits, respectively. The purpose of these attachments is to withhold the proprietary information contained in Attachment 1 from public disclosure. The Affidavits, signed by Westinghouse and B&W as the owners 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 and B&W be withheld from public disclosure in accordance with 10 CFR 2.390.

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

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-259 [Reference 1].

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

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 September 23, 2011.

Very truly yours,

Robert J. Hughes FOR RLA

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachments (4)

cc: Mr. William Passetti, Florida Department of Health

ATTACHMENT 2

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

NON-PROPRIETARY VERSION

(Cover page plus 43 pages)

Response to 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 1 that was submitted to the NRC by FPL via letter (L-2010-259) dated November 22, 2010, Accession Number ML103560419.

In an email dated July 27, 2011 from NRC (Tracy Orf) to FPL (Chris Wasik), Subject: St. Lucie 1 EPU draft Mechanical and Civil RAIs (EMCB), the NRC requested additional information regarding FPL's request to implement the EPU. The RAI consisted of forty-five (45) questions from the NRC's Mechanical and Civil Branch (EMCB). These forty-five RAI questions and the FPL responses are documented below.

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:

With the exception of the replacement steam generator (RSG) nozzle analyses discussed in EMCB-23 and the final design of the hot leg injection modification, applicable safety related and 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 response to EMCB-23 regarding the RSG nozzle analyses and updated information regarding the hot leg injection modification will be provided to NRC by October 28, 2011.

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:

The piping systems which experienced an increase in temperature, pressure flow and/or mechanical loads due to EPU include reactor coolant, main steam, feedwater, condensate, extraction steam, heater drains, component cooling water, intake cooling water, chemical and volume control, pressurizer spray, safety injection, shutdown cooling, steam generator blowdown, and containment hydrogen purge.

The CLTP and EPU stress values for the specific piping systems impacted by EPU were provided in LAR Attachment 5, LR Sections 2.2.2.1 and 2.2.2.2 (Tables 2.2.2.1-1 and 2.2.2.1-2 for NSSS Piping Systems and Table 2.2.2.2-1 for BOP Piping Systems).

The piping systems that contain high energy piping include reactor coolant, main steam, feedwater, condensate, extraction steam, heater drains, chemical and volume control, pressurizer spray, safety injection, shutdown cooling and steam generator blowdown. 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 summary data for the containment hydrogen purge system, which was not included in the St. Lucie Unit 1 EPU LAR submittal, is as follows:

Stress Summary at EPU Conditions					
Piping Analysis Description	**Loading Condition (Note 2)**	**Existing Stress (psi)**	**EPU Stress (psi)**	**Allowable Stress (psi)**	**Design Margin (Note 1)**
Containment Hydrogen Purge System – Outside Containment	Equation 8	Not Available	3,407	15,000	0.227
	Equation 9U	Not Available	4,770	18,000	0.265
	Equation 9E	Not Available	5,415	27,000	0.201
	Equation 10	Not Available	9,342	22,500	0.415

Notes:
1. Stress Interaction Ratio (also called "Design Margin") is based on the ratio of EPU Stress divided by the Allowable Stress.
2. The pipe stress analysis equation numbers listed in this table correspond to ASME Section III, NC / ND – 3650 equation numbers.

The stress summaries for the remaining piping systems identified in "item a" above that were affected by EPU were included in the following tables of the EPU LAR submittal.

- Class 1 piping system stress summaries are included in LAR Attachment 5, LR Section 2.2.2.1-1, Table 2.2.2.1-1 for NSSS piping, components, and supports.
- BOP piping system stress summaries are included in LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-1.

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:

BOP piping evaluations were performed using both computer analyses and scaling factors. In instances where scaling factors were used 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 the piping running from drain coolers 1A/B to heater drain pumps 1A/B was 310°F. The corresponding EPU temperature for this piping is 319°F. Hence, the thermal change factor equals (319-70)/(310-70) = 1.04.

Using this change factor, the existing thermal stress of 16,917 psi (based on 310°F) was increased by the 1.04 change factor (16,917 X 1.04) to determine the corresponding EPU thermal stress of 17,594 psi. The stress values shown in this example were included in LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-1.

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 LAR Attachment 5, LR Table 2.2.2.1-1. The stresses calculated using scaling factors followed the methodology described herein. The scaling factors are initially applied in a conservative manner (i.e., the resultant stress is multiplied by the largest component of load increase). An example analysis for the safety injection nozzles is included here:

Table 1 and Table 2 show the loads on the safety injection nozzles due to thermal expansion and deadweight (DWt) from the tributary piping at current and EPU conditions in global and local coordinate systems, respectively. The maximum Mb loads (bending moment) at EPU conditions are divided by the maximum Mb loads at current conditions to determine the Mb ratio. The maximum Mt (torsion moment) loads at EPU conditions are divided by the maximum Mt loads at current conditions to determine the Mt ratio. Based on the loads from Table 2 the Mb ratio is [ ]$^{(a,c)}$ and the Mt ratio is [ ]$^{(a,c)}$. The total current primary plus secondary stress is [ ]$^{(a,c)}$. The maximum as-calculated current stress due to tributary piping thermal expansion and DWt is [ ]$^{(a,c)}$ of the

[ ]$^{(a,c)}$. The Mb factor is applied to this stress, resulting in a total primary plus secondary EPU stress of [ ]$^{(a,c)}$, which is less than the allowable stress of [ ]$^{(a,c)}$ from LAR Attachment 5, LR Table 2.2.2.1-1. The safety injection nozzle cumulative fatigue usage factor is [ ]$^{(a,c)}$.

Table 1: Safety Injection Nozzle Loads - Global Coordinate System

Condition	Nozzle	Moments (in-kips)		
		Mx	My	Mz
Current	P9			
	P5			
	P14			
	P18			
EPU	P9			
	P5			
	P14			
	P18			

[a,c]

Table 2: Safety Injection Nozzle Loads - Local Coordinate System

Condition	Nozzle	Moments (in-kips)(4)			Applied Loads (in-kips)	
		Mx	My	Mz	Mb (in-kips)(2)	Mt (in-kips)(3)
Current(1)	P9					
	P5					
	P14					
	P18					
EPU(1)	P9					
	P5					
	P14					
	P18					

[a,c]

Notes:

(1) Global loads are transformed to the local coordinate system. Mx and Mz are bending moments; My is the torsion moment.

(2) $Mb = \sqrt{Mx^2 + Mz^2}$

(3) $Mt = My$

(4) The local coordinate system is defined as: the x-axis is parallel to the centerline of the safety injection nozzle; the y-axis and z-axis are orthogonal and perpendicular to the centerline of the safety injection nozzle. The horizontal plane projection of the local x-axis is [ ]$^{(a,c)}$ off of the global X-axis for nozzles P5 and P14, and parallel to the global X-axis for nozzles P9 and P18. For all nozzles, the local x-axis is [ ]$^{(a,c)}$ off of the horizontal plane towards the positive global Y-axis, and points away from the center of the RCS.

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 evaluations performed for 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 primary loop piping for St. Lucie Unit 1 meets all of 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. The criteria for pipe rupture postulation for other piping inside containment is based on the guidance provided in Regulatory Guide 1.46 (May 1973), which is part of the St. Lucie Unit 1 CLB, except for RCL branch piping which considered a break anywhere approach consistent with UFSAR Section 3.6.4.1.

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 1 CLB.

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 EPU, the code used for developing stress data for postulating pipe failures inside and outside containment for Class II and III systems was the ASME Section III, 1971 edition through Summer 1973 Addenda. Although the piping code for Class II and III piping systems is identified as ANSI B31.7 in the CLB, reconciliation for the use of this ASME III 1971 edition through Summer 1973 Addenda code was performed in accordance with ASME Section XI.

For EPU, there was no stress criteria used in postulating pipe failures inside containment for the primary RCL Class 1 piping and associated RCL branch piping. The primary RCL piping used LBB criteria in accordance with NUREG-1061 and RCL branch piping considered a break everywhere consistent with the CLB (UFSAR Section 3.6.4.1).

Response 4b:

The stress equations used for EPU 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 (Primary RCL)
Leak Before Break (LBB) criteria in accordance with NUREG-1061 is applicable to the RCS main loop piping.

Inside Containment (RCL Branch Piping)
A break anywhere was considered consistent with UFSAR Section 3.6.4.1.

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 1 CLB.

The basis for the inside containment stress equations is the guidance provided in Regulatory Guide 1.46 (May 1973), UFSAR Section 3.6.4.1, and NUREG-1061 which are part of the St. Lucie Unit 1 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:

As stated in UFSAR Section 3.6, Generic Letter 87-11 was adopted as an alternative means to provide pipe break protection for Class 2, Class 3, and Non-ASME Class systems and as a means to minimize the addition of or facilitate the removal of excess arbitrary intermediate pipe whip restraints.

Generic Letter 87-11 eliminated the requirement for all dynamic effects (missile generation, pipe whipping, pipe break reaction forces, jet 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 (SRP), 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, which is presented in Appendix 3.J of the St. Lucie Unit 1 UFSAR, in lieu of the original criteria.

The regulatory process that allowed the insertion of Appendix 3.J into the UFSAR is 10 CFR 50.59. A modification to steam generator blowdown system isolation valves performed in 1993 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 1 was justified and the UFSAR change package that added Appendix 3.J was provided. Application of the criteria presented in Appendix 3.J 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 1 CLB (UFSAR Section 3.6) for postulating piping failures inside containment allows the use of either RG 1.46, "Protection Against Pipe Whip Inside Containment (May 1973)," or MEB 3-1, Rev 2, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment."

For EPU, the guidance provided in Regulatory Guide 1.46 (May 1973) was used for postulating piping failures inside containment, except for RCL branch piping which considered a break anywhere approach consistent with UFSAR Section 3.6.4.1.

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 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 RCS branch piping breaks include pressurizer surge and spray 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 EPU piping evaluations for HE piping systems located inside containment was 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 SG sliding base support function is to provide vertical support of the SG dead weight and LOCA restraint in the North-South direction as the RCS (hot and cold legs) expands and contracts with varied thermal modes of operation. It 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 bent shim plate attached to the steam generator (SG) sliding base support was to maintain the 1/16" (+/- 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 leg and cold leg LOCAs.

The basis for the removal of the bent 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 a North-South restraint capacity of the support was no longer required, the SG bent 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.

<u>Response:</u>

Removal or modification of existing 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 components:

- 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 design for main loop LOCA accidents is no longer required, changes involving removal of attributes specifically designed to resist only LOCA loadings have been reviewed to ensure that the RCS pipe whip restraint configuration adequately restrains the RCS under a branch line LOCA event. If it was determined that removal of a restraint or other attribute potentially compromised the validity of the original pipe rupture analysis, additional evaluations were conducted that ensured the operability of the Unit.

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 LR 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. Thus, 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 LR 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. LR 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 RAB utilized for equipment qualification remains unchanged by the EPU. In addition, LR 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. Thus, additional structural analyses are not required for SSCs inside or outside containment.

- As indicated in LR Section 2.6.1, the containment pressure response analyses performed to evaluate the consequences of the PSL1 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 LR 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 applicable to PSL1 following implementation of LBB. Thus, 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 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, and peak stresses are in accordance with ANSI B31.7 as shown in the table below. All components analyzed with simplified elastic-plastic methodology used ASME Section III, as shown in the table below.

Class I Piping Code Editions

<table>
<tr><td>Component</td><td>Code Edition</td></tr>
<tr><td>Reactor Coolant System Piping and Nozzles</td><td rowspan="2">ANSI B31.7, Code for Nuclear Power Piping, Class 1, Feb. 1, 1968 Draft Edition for Trial Use and Comment & ASME Section III, Nuclear Vessels, Summer 1969 Addenda</td></tr>
<tr><td>Pressurizer Surge Line</td></tr>
</table>

The evaluation of Class I branch lines connected to RCL primary loop 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 Code, using the 1971, 1986 or 1989 editions. A code reconciliation documenting the use of these later ASME III codes for the subject RCL branch piping was performed in accordance with ASME Section XI.

The code used for Class II and III piping system evaluations for EPU was ASME Section III, 1971 edition through Summer 1973 Addenda. Although the piping code for Class II and III piping systems is identified as ANSI B31.7 in the CLB, reconciliation for the use of this ASME III 1971 edition through Summer 1973 Addenda code was performed in accordance with ASME Section XI.

The code used for non safety-related/non seismic piping system evaluations for EPU was the ANSI B31.1 1967 edition. This code is consistent with the CLB for non safety-related/ non seismic 1 piping evaluations.

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 EPU have used allowable stress values contained in existing design basis analyses which utilized original code of construction allowable values.

The ANSI and ASME allowable values used for the EPU are in accordance with the correct Code editions for each component. As discussed in the EMCB-16 response, the allowable values for the hot leg elbow, pressurizer spray and shutdown cooling nozzles at the bi-metallic weld, and the hot leg surge nozzle were revised to include corrected 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 GL 96-06 evaluation all containment penetrations and pipe sections inside containment that were vulnerable to thermal overpressurization during 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-DBA 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.

Two penetrations and five isolated piping segments within three non-CCW systems were identified during FPL's initial GL 96-06 evaluation that either did not have self relieving capabilities or are not drained / partial filled. These non-CCW systems are Containment Spray, Reactor Coolant and Chemical Volume and Control. FPL has installed thermal relief valves on the affected penetrations. FPL has updated an administrative procedure to drain pipe segments of two of the systems, installed a thermal relief valve on a pipe segment of one system, changed a valve configuration to the open position on a pipe segment of one system, and updated a procedure to provide air in a pipe segment of the remaining affected system inside the Unit 1 containment. NRC letter dated March 27, 2000 documented NRC acceptance of these changes as adequate resolution of GL 96-06 issues for non-CCW piping.

EPU will not require any 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 LAR Attachment 5 pages 2.5.4.3-10, and 11, refer to MSLB conditions while LAR Attachment 5, LR Table 2.5.5.1-1, provides normal plant operation parameters. The MSLB containment analysis that is discussed in LAR Attachment 5, LR 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 LAR Attachment 5, LR 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 operating conditions at EPU as presented in LAR Attachment 5, LR 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.

In regard to whether MSLB is the limiting condition for thermally induced overpressurization of piping, it is determined that LOCA is the limiting condition.

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 RCS loop piping. These changes affect fluid weight, thermal expansion mechanical loads, thermal transient definitions, and thermal displacements.

Thermal expansion analyses were performed to evaluate changes in RCS temperatures due to the EPU using the ANSYS code. The differences between the EPU and CLB normal operation analyses are an increase of [ ]$^{(a,c)}$ in the cold leg temperature and a decrease of [ ]$^{(a,c)}$ in the hot leg temperature. The temperature changes 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, 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 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 used the Integrated Civil Engineering Systems Structural Design Language computer code (ICES STRUDL II).

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:

St. Lucie Unit 1, LAR Attachment 5, LR Table 2.2.2.1-1 summarized the maximum primary plus secondary reactor coolant system (RCS) piping stresses for extended power uprate (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 Paragraphs I-705.1 through I-705.3 of ANSI B31.7, February 1968. Upon further research it was determined that four of the allowable stresses listed in LAR Attachment 5, LR Table 2.2.2.1-1 were inconsistent with current S_m determination methodology. All critical allowable stresses, as detailed in LAR Attachment 5, LR Table 2.2.2.1-1, have been reviewed. The revised LR Table 2.2.2.1-1 is supplied here. Additionally, applicable ASME Code versions are referenced throughout LAR Attachment 5, LR Section 2.2.2.1.2.4, and are in accordance with the current licensing basis.

Table 2.2.2.1-1
Maximum RCS Piping Stress and Usage Factor Results

RCS Piping	Criterion		Pre-EPU Stress (ksi)	Additional Stress due to EPU (ksi)	Total Stress with EPU (ksi)	Allowable (ksi)	Margin
Hot Leg Straight Pipe[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
Hot Leg Elbow[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
All Cold Leg Straight Pipes and Elbows (RCP Suction and Discharge Legs) [1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
Safety Injection Nozzles[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
Spray Nozzles[1]	Prim + Bending Fatigue Prim. + Sec. at Bi-metallic Weld Fatigue at Bi-metallic Weld	$P_m + P_b$ CUF $P_L + P_b + Q$ CUF					
Charging Nozzles[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
Letdown & Drain and Drain Nozzles, Cold Leg[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
Letdown & Drain and Drain Nozzles, Cold Leg, at Bi-metallic Boundary[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
Drain Nozzle, Hot Leg[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
Drain Nozzle, Hot Leg, at Bi-metallic Boundary[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					
Shutdown Cooling Outlet Nozzle[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					

[a,c]

Table 2.2.2.1-1 (continued)
Maximum RCS Piping Stress and Usage Factor Results

RCS Piping	Criterion		Pre-EPU Stress (ksi)	Additional Stress due to EPU (ksi)	Total Stress with EPU (ksi)	Allowable (ksi)	Margin
Shutdown Cooling Outlet Nozzle at Bi-metallic Boundary[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + Q$ CUF					%
Hot Leg Surge Nozzle[1]	Primary Prim. + Sec. Fatigue	P_m $P_L + P_b + P_e + Q$ CUF					
RCP Suction Nozzles[1]	Prim. + Sec. Fatigue	$P_L + P_b + P_e + Q$ CUF					
RCP Discharge Nozzles[1]	Prim. + Sec. Fatigue	$P_L + P_b + P_e + Q$ CUF					

[a,c]

Notes:

1) Stresses are in accordance with USA Standard Code for Pressure Piping, Nuclear Power Piping, ANSI B31.7, February 1968 and ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Summer 1969 Addendum.
2) Fatigue was determined by a simplified elastic-plastic discontinuity analysis. The [ ][(a,c)] value is the $3mS_m$ upper limit associated with the determination of K_e

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 years for the ASME stress levels provided are ASME Section III, 1971, 1986 and 1989 editions.

Updated stress data for the feedwater and main steam piping systems is provided in the following table. The stress data for all remaining safety related piping systems provided in LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-1, remain unchanged. The allowable stresses shown are consistent with the original code of record.

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)**
Feedwater Inside containment: From containment penetration P3 to Steam Gen. 1A.	Equation 8	5,564	6,384	15,000	0.426
	Equation 9U	5,976	7,053	18,000	0.392
	Equation 9E	6,311	7,282	27,000	0.27
	Equation 9F	(Note 4)	7,282	45,000	0.162
	Equation 10	13,509	11,706	22,500	0.52
Feedwater Inside containment: From containment penetration P4 to Steam Gen. 1B	Equation 8	5,271	6,158	15,000	0.411
	Equation 9U	7,105	8,096	18,000	0.45
	Equation 9E	7,546	8,364	27,000	0.31
	Equation 9F	(Note 4)	8,364	45,000	0.186
	Equation 10	12,602	9,714	22,500	0.432
Feedwater Outside containment: Feedwater Pumps 1A & 1B	Equation 8	7,930	8,834	15,000	0.589
	Equation 9U	8,100	11,760	18,000	0.653
	Equation 9E	(Note 4)	11,770	27,000	0.436
	Equation 9F	(Note 4)	11,770	45,000	0.262
	Equation 10	12,602	17,752	22,500	0.789
Main Steam Inside Containment: Steam Gen. 1A to Cont. Penetration P1	Equation 8	6,742	8,694	15,000	0.58
	Equation 9U	(Note 4)	9,576	18,000	0.532
	Equation 9E	(Note 4)	9,678	27,000	0.358
	Equation 9F	(Note 4)	9,678	45,000	0.215
	Equation 10	(Note 4)	7,238	22,500	0.322
Main Steam Inside Containment: Steam Gen. 1B to Cont. Penetration P2	Equation 8	7,485	8,182	15,000	0.545
	Equation 9U	(Note 4)	10,496	18,000	0.583
	Equation 9E	(Note 4)	10,655	27,000	0.395
	Equation 9F	(Note 4)	10,655	45,000	0.237
	Equation 10	(Note 4)	7,204	22,500	0.32
Main Steam Outside Containment: Penetrations P1 & P2 to Turbine Inlet.	Equation 8	(Note 4)	10,040	15,000	0.669
	Equation 9U	(Note 4)	13,188	18,000	0.733
	Equation 9E	(Note 4)	14,092	27,000	0.522
	Equation 9F	(Note 4)	14,092	45,000	0.313
	Equation 10	(Note 4)	19,665	22,500	0.874

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, NC / ND – 3650 equation numbers.
3. Description is based on pipe stress analysis calculation number or piping segment of a given system included in the analysis.
4. When information is not provided, the information was not available.

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:

For EPU piping analyses that were performed in accordance with B31.7 or 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:

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 these piping systems, the deadweight, pressure and seismic stresses are unchanged due to EPU (i.e., no change to Equations 8 or 9). Hence, for these piping systems, only the thermal expansion loading condition was included in LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-1, since it was the only loading condition that was affected by EPU.

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 LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-1 since the deadweight, seismic and fluid transient stress levels were all affected by EPU. In summary, LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-1, 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:

Piping modifications for safety-related piping and/or seismic II/I piping systems are as follows:

Piping Modifications		
Main Line	Branch Description	Modification Description
8"-BF-16	Vent with valves V09228 and V09229	Replacement of branch pipe to coupling socket weld
8"-BF-16	PX connection with valves V09233 and V09234	Replacement of branch pipe to coupling socket weld
8"-BF-16	Vent with valves V09236 and V09237	Replacement of branch pipe to coupling socket weld
8"-BF-16	PX connection with valves V09231 and V09232	Replacement of branch pipe to coupling socket weld
I-4"-MS-11	Vent with valves V08450 and V08533	Replacement of branch pipe to coupling socket weld Replacement of spool piece between valves Replacement of branch pipe to valve welds
I-4"-BF-58	PX connection with valves V09115 and V09116	Replacement of socket weld at tee connection to allow for rotation of V09291

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:

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 main steam piping, the stress summaries contained in LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-1 include stresses due to fluid transient loads associated with turbine stop valve and main steam isolation valve closure events.

For feedwater piping, the stress summaries contained in Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-1, include stresses due to fluid transient loads associated with feedwater regulating valve and feedwater isolation valve closure 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 steam hammer analyses (for turbine stop valve and main steam isolation valve closure events) and the feedwater water hammer analyses (for feedwater regulating valve and feedwater isolation valve closure events).

The load combinations which include the fluid transient stresses are as follows:
Equation 9U (Pressure + Deadweight + OBE + Fluid Transient) ≤ 1.2 Sh
Equation 9E (Pressure + Deadweight + SSE + Fluid Transient) ≤ 1.8 Sh
Equation 9F (Pressure + Deadweight + SSE + Fluid Transient) ≤ 3.0 Sh

Response 22c:

The EPU structural evaluations described within Section 2.2 reflect the revised piping and support configurations related to the replacement feedwater pumps and associated piping.

The stress and load summaries presented in LAR Attachment 5, LR Section 2.2.2.2, Tables 2.2.2.2-1, 2.2.2.2-3, 2.2.2.2-4, 2.2.2.2-5, 2.2.2.2-6 and 2.2.2.2-7 include results from the fluid transients associated with the replacement feedwater pumps and the 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:

The steam generator nozzle analyses are being revised to reflect updates to the main steam and feedwater piping analyses. Accordingly FPL will provide a response to this RAI by October 28, 2011.

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 loads provided in LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-5 (Main Steam) and Table 2.2.2.2-6 (Feedwater) are 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.

Response 24b:

The load combination for the calculated loads presented in LAR Attachment 5, LR Section 2.2.2.2, Table 2.2.2.2-5 (Main Steam) and Table 2.2.2.2-6 (Feedwater) is as follows:

Deadweight + Thermal + SSE + Fluid Transient

Response 24c:

The allowable stress intensity values were obtained from the allowable stress values summarized in UFSAR Appendix 3G5.

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 feedwater pump is 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 assessments.

Response 25b and 25c:

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

New/replacement snubbers were 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.

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:

LAR Attachment 5, LR Table 2.2.2.3-4 has been revised to correct the original design thermal plus deadweight loads per UFSAR Table 3H-1. Deadweight and thermal plus deadweight loads have increased due to EPU. The maximum vertical load under EPU increased by 6.7 % and the friction force increased by 5.7%. The maximum vertical EPU load is 1232 kips. The allowable vertical compression load is 5587 kips for normal operating conditions. A friction force of 350 kips was used in the original analysis, providing 18% margin over allowables. The increase in friction force due to EPU reduces that margin to 13%. Therefore the reactor vessel supports are adequate for the increased loads due to EPU.

Table 2.2.2.3-4
Vessel Support Load Comparison

Point	EPU(kips)			Original Design (kips)		
	Thermal (TH)	Deadweight (DWt)	TH + DWt	Thermal (TH)	Deadweight (DWt)	TH +DWt
H1	0	0	0	28	0	28
V1	510	722	1232	489	666	1155
H2	0	0	0	91	0	91
V2	94	709	803	92	634	726
H3	0	0	0	79	0	79
V3	96	709	805	107	634	741
F	±370			±350		

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:

Reanalysis of the CEDMs 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 CEDMs.

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:

A reanalysis of the SBP for the EPU was required due to changes in the deadweight loads at EPU conditions. The changes made to the RCS component weights that affected the SBP included:

- Replacing the original SG weight with the RSG weight.
- Replacing the original RCP motor weight with the replacement RCP motor weight for all four RCPs.

An analysis of the SBP was performed to ensure that the maximum loads on the SBP and building supports resulting from these changes were considered.

Response 29b:

The PSL1 SBP is considered a support structure; therefore, it was analyzed for EPU conditions to Subsection NF and Appendix F of the ASME Code. The editions used were the 1972 Addendum for Appendix F and the 1973 Addendum for Subsection NF. The ASME Code of record for the SBP is the 1971 Edition. Because the 1971 Edition does not contain Subsection NF and Appendix F, the first editions of the ASME Code that contain these sections were used. Accordingly, no Code reconciliation is required. The RSG was analyzed to the appropriate edition(s) of Section III of the ASME Code because it is a component, not a support.

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:

The SBP was analyzed using the finite element method and hand calculations when appropriate. The model is composed of a full representation of the SBP using solid brick and tetrahedral elements. The flange that connects the SBP to the RSG was fixed, and the interface loads were applied to the building interface locations where the vertical pads interface with the building's support structure. For the normal operation analysis, six load cases were defined to combine deadweight and thermal loads with the operational basis earthquake (OBE) loads. Vertical loads were applied as distributed loads at the sliding base socket interfaces. Lateral loads were applied to the shear keys and x-direction stop. The frictional load was applied as distributed forces at the sliding pad locations.

The condition that shows that only one out of four corners of the SG1B base support is bearing weight is common in similar plant designs. This condition is a result of the orientation of the supports relative to RCS geometry and the input load path. As the system expands and the SG moves slightly outward from the center of the RCS, more of its weight is distributed to the outermost support pad than the other three vertical supports. This steady-state SG support configuration is normal for CE-designed plants.

Response 30b:

The St. Lucie Unit 1 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. The replacement of the SGs and RCP motors is not part of the EPU project.

Response 30c:

The limiting vertical hot gap size is [ ]$^{(a,c)}$ or [ ]$^{(a,c)}$. The difference in the SG liftoff 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 load due to friction listed for current conditions in column one of LAR Attachment 5, LR Table 2.2.2.5-1, 375 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:

LAR Attachment 5, LR Section 2.2.2.5.2.1.3 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 (NOP) 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 dead weight loads are shown in LAR Attachment 5, LR Table 2.2.2.5-1 as "EPU Load". The original design basis dead weight loads on the steam generator lower supports (as shown in LAR Attachment 5, LR Table 2.2.2.5-1 as "Pre-Uprate Load") did not reflect the flooded steam generators.

Both the (flooded) dead weight load and the combination of thermal plus dead weight load, as listed under "EPU Load" in LAR Attachment 5, LR Table 2.2.2.5-1, 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 steam generator sliding base support and foundation are designed for thermal expansion loads in the N-S direction transmitted via friction forces from the sliding base. The term "lateral forces" in the above quoted excerpt from the St. Lucie Unit 1, EPU LAR was intended to mean forces in the E-W direction. Because of the symmetry of the RCS cold and hot legs, there is negligible thermal movement in the E-W 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 E-W 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 B & PV Code, Sect. III, 1971 Edition
- SGSB support – AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, dated February 12, 1969
- Steam generator foundation – ACI 318-63, 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:

The steam generator secondary side pressure boundary components that are identified as exempt from fatigue in LAR Attachment 5, LR Table 2.2.2.5-4 were analytically demonstrated to have a combination of materials and service loadings at EPU conditions that satisfy the rules of the 1986 Edition (no addenda) of the ASME B&PV Code Section III, Subsection NB subsubparagraph 3222.4(d) - *Components Not Requiring Analysis for Cyclic Service.* The service loadings for the feedwater nozzle, which is dominated by large fluctuations in feedwater temperatures and flow rates, could not be demonstrated to satisfy these rules; hence, a fatigue analysis was performed for this component only.

The rules in NB-3222.4(d) place limitations on the number of full range pressure cycles, partial range pressure cycles, magnitude of thermal gradients within startup and shutdown cycles, magnitude of partial range thermal gradients during service and the magnitude of cyclic mechanical loads such as piping loads. The exemption provisions are based on performing conservative evaluations using allowable membrane stresses and the ASME B&PV Code fatigue design curves for the materials being considered.

[ ]a,c.

EMCB-34

Please provide the basis for the SG two-phase stability ratio of greater than [ ]a,c In addition, discuss the terms, the purpose and significance of the two-phase stability ratio.

Response:

The basis for the acceptance criterion that the SG two-phase stability ratio be greater than [ ]a,c is industry operating experience and [ ]a,c.

The two modes of instability of concern in a recirculating steam generator are Ledinegg (flow excursion) instability and density wave instabilities. Both instabilities can be avoided by increasing the pressure loss in the single-phase (liquid) flow region of the recirculating loop relative to the two-phase loss.

The SG two-phase stability ratio is calculated by dividing the flow loss in the downcomer (liquid flow) region by the flow losses in the riser (two-phase flow) region. The purpose of designing for stable two-phase flow is to prevent flow oscillations which might result in excessive water level fluctuations in the steam drum or an increased rate of tube fretting wear at the tube supports.

[ ]a,b,c

EMCB-35

Show how the maximum fluid elastic stability ratio of [ ]a,c was derived and whether it is in the U-bend region. Provide the calculated critical gap cross-flow velocity (Ucr) and the calculated maximum effective gap velocity (Ueff) equations and values for CLTP and EPU conditions and discuss the methodology and basis used to derive these values. If a benchmark case of another plant was used, show its applicability to St. Lucie U1.

Response:

The flow-induced vibration analysis is performed specifically for the St. Lucie Unit 1 steam generator geometry and thermal-hydraulic conditions and not from benchmarking using data from another plant. The analysis is performed using the computer code [ ]a,c

(Ref. 1). [ ]^{a,c} is a PC based flow-induced vibration computer code that has been developed by B&W for predicting the vibration response of tube bundles (or single tube) subjected to cross-flow. [ ]^{a,c}. This program has been validated in accordance with the B&W Canada Quality Assurance program for nuclear products.

The cross-flow velocity distributions used for the FIV analysis are extracted from a thermal-hydraulic analysis of the steam generator tube bundle carried out using [ ]^{a,c}. [ ]^{a,c} is an EPRI computer program developed for thermal hydraulic analysis of steam generators.

The critical and effective cross-flow velocities are calculated within the [ ]^{a,c} code using the equations described below.

The following empirical expression has been established for the critical gap velocity (i.e. the velocity within the space between the tubes), which marks the onset of fluidelastic instability. It is based on uniform flow over a single span length of a simply supported tube array.

$$U_{cr} = \beta f \sqrt{m\delta / \rho_o} \qquad \text{(Ref. 1)}$$

Where

U_{cr}	=	critical gap velocity
f	=	natural frequency of the tube including all hydrodynamic (added) mass effects
m	=	mass per unit length including all hydrodynamic (added) mass effects
δ	=	logarithmic decrement of damping = $2\pi\zeta$
ζ	=	critical damping ratio
ρ_0	=	density of the flowing fluid
β	=	instability coefficient

The actual gap velocity (effective gap velocity) for a tube array that weighs the effect of the velocity distribution over the mode shape of vibration of the tube is as follows:

$$U_{eff} = \sqrt{\frac{\int_0^L U^2(x)\phi^2(x)\,dx}{\int_0^L \phi^2(x)\,dx}} \qquad \text{(Ref. 1)}$$

where U(x) is the actual velocity distribution and $\phi(x)$ is the mode shape of vibration that is being excited.

The fluidelastic instability results for the RSG tube bundle are presented in terms of fluidelastic instability ratio (FEI ratio) U_{eff}/U_{cr}, which is the ratio of the effective fluid gap velocity (average in the gap between tubes) to the critical gap velocity at the onset of fluidelastic instability. This ratio is defined as follows:

$$\frac{U_{eff}}{U_{cr}}=\frac{\sqrt{\dfrac{\int_0^L U^2(x)\phi^2(x)\,dx}{\int_0^L \phi^2(x)\,dx}}}{\beta f\sqrt{\dfrac{m\delta}{\rho_O}}} \qquad \text{(Ref. 1)}$$

A maximum FEI ratio of [ ]a,c was calculated and it occurs at the U-bend Region (See Figure 2). [ ]a,c.

[ ]a,c.
[ ]a,c

Figure 1: The tube mode shape corresponding to the maximum FEI ratio of [ ]a,c

[ ]a,c

Figure 2: Gap velocity profile for U-tube shown in Figure 1 (Taken from [ ]a,c output file)

References

[1] [ ]a,c.

EMCB-36

For the EPU estimated maximum of [ ]a,c and [ ]a,c tube wear thickness depths, show how these values were calculated and state the basis for the acceptance criterion of 40% through wall tube wear.

Response:

The maximum tube wear thickness depths of [ ]a,c and [ ]a,c reported in LAR Attachment 5, LR Section 2.2.2.5.2.5 were calculated using the methodology described below:

Wear Analysis

[ ]a,c

The basis for the acceptance criterion of 40% through wall tube wear is founded upon the requirements presented in ASME Code Section XI, 2001 Edition, "Requirements for Class 1 Components"; specifically IWB-3521 "Standards for Examination Category B-Q, Steam Generator Tubes." Furthermore FPL incorporates the 40% through wall wear acceptance criterion in St. Lucie Unit 1 Technical Specification (TS) 6.8.4.l.c, "Steam Generator Program." This TS requires that tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness be plugged.

[ ]a,c

EMCB-37

The EPU LR states that SGs have ample tube support to ensure that flow induced vibration amplitudes are very small and tube cyclic stresses are negligible.

a) Please discuss the analysis performed and the methodology and criteria employed to determine that the vibratory tube stresses are below the material endurance limit and show the maximum calculated alternating stress intensity compared to the endurance limit.

b) Provide the SG tube spacing distance and the EPU maximum tube vibration amplitude due to turbulence excitation. The LR shows [ ]^{a,c} mils for maximum vortex shedding resonance amplitude. Address whether these values are in the U-bend region area and show how they were derived. If a benchmark case of another plant was used, discuss its applicability to St. Lucie U1. In addition, please show how the acceptance limit of [ ]^{a,c} mils vibratory amplitude was derived and the basis of its derivation (i.e., whether the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code Part 3, Salt equation for steady state vibration was used).

Response:

a) Maximum random turbulence amplitudes ([ ]^{a,c} mils from a model of the U-Bend region ([ ]^{a,c}), and [ ]^{a,c} mils from a model of the bundle entrance region ([ ]^{a,c}), are used in conjunction with calculated tube fretting-wear predictions as explained in the response to EMCB-RAI-36. The methodology used in calculating the tube vibration amplitude due to random turbulence excitation, and in assessing through-wall fret depths for tubes is also explained in the response to EMCB-RAI-36. The amplitudes compare with a tube-to-tube spacing of [ ]^{a,c}.

The vibration amplitude due to random turbulence excitation is small and, based on previous B&W experience, the alternating stress associated with amplitudes of this magnitude are well below the endurance limit. Nevertheless, the alternating stress due to an induced displacement from random turbulence vibration is calculated below for both the U-bend region and bundle entrance regions of tubes with the highest amplitudes. Using a closed form solution for stress analysis of a straight beam (Ref. 4), the bending stress occurring in the tube is determined by considering a single span beam model (with tube cross sectional area) with an imposed displacement (listed in Table 1) acting at its mid-span corresponding to the maximum amplitude of vibration. The tube is modelled using both simply supported and clamped boundary conditions at its ends. The results are listed in Table 1.

[ ]^{a,c}.

Table 1
Alternating stress intensity due to random turbulence vibration amplitude

Tube Number	Random Turbulence Vibration Amplitude (zero to peak)	Maximum Stress Intensity	
		Simply Supported Beam Model	Fixed-Fixed Beam Model
[ ]a,c Tube model for U-bend region	[ ]a,c	[ ]a,c	[ ]a,c
[ ]a,c Tube model for Bundle Entrance region	[ ]a,c	[ ]a,c	[ ]a,c

1 mils = 10^{-3} inch; 1 ksi =1000 psi
£ See the third paragraph of item (a)

[ ]a,c.

b) The existence of a vortex shedding resonance is determined specifically for the St. Lucie Unit 1 RSG geometry and thermal-hydraulic conditions by comparing the vortex shedding frequency with the tube natural frequency. No benchmarking from another plant was performed. The vortex shedding frequency is given by the following equation [Ref. 1]:

$$f_{sh} = SU / D_o \qquad (1)$$

where

f_{sh} : vortex shedding frequency

S : Strouhal number ([ ] a,c used in calculating the vortex shedding frequency)

D_o : Outer tube diameter

If the shedding frequency in the operating range is equal to any of the natural frequencies of the tube, a resonance may occur. If a resonance occurs, a forced response analysis can be performed and the resulting vibration amplitude is:

$$y(x) = \frac{C_L \rho_o D_o}{16 \zeta_\alpha \pi^2 f_\alpha^2 M_\alpha} \phi_\alpha(x) \int_0^L U^2(x) \, | \phi_\alpha(x) | \, dx \qquad (2)$$ [Ref. 1]

where C_L = vortex shedding lift coefficient (see Table 1)
ρ_o = density of flowing fluid
D_o = outside diameter of tube
$\zeta_\propto$ = critical damping ratio of $\propto^{th}$ mode
$f_\propto$ = natural frequency of $\propto^{th}$ mode
$M_\propto$ = modal mass for $\propto^{th}$ mode
$\phi_\propto(x)$ = mode shape of the $\propto^{th}$ mode of vibration
$U(x)$ = velocity distribution

The modal mass is given by:

$$M_\alpha = \int_0^L m(x)\phi_\alpha^2(x)\,dx \qquad (3) \qquad \text{[Ref. 1]}$$

where $m(x)$=mass per unit length of the tube at location "x" including all hydrodynamic mass effects

[ ]a,c.

The alternating stress intensity occurring in the tube due to vortex-shedding amplitude is also determined here and listed in Table 2.

Table 2
Alternating stress intensity due to vortex-shedding vibration amplitude

Tube Number	Vortex-Shedding Vibration Amplitude (zero to peak)	Maximum Stress Intensity	
		Simply Supported Beam Model	Fixed-Fixed Beam Model
[ ]a,c Bundle Entrance Tube Model	[ ]a,c	[ ]a,c	[ ]a,c

1 mils = 10^{-3} inch; 1 ksi =1000 psi
£ See the third paragraph of item (a)

[ ]a,c.

References:

[1] [ ]a,c.

[2] [ ]a,c.

[3] [ ]a,c.

[4] W. C. Young, "Roark's Formulas for Stress and Strain", 6th edition, McGraw-Hill, 2005.

[5] ASME Boiler and Pressure Vessel Code, Section II and Section III, Division 1, Subsection NB, including Appendices, 1986 Edition.

EMCB-38

Discuss whether any acoustic resonance could be generated at EPU flow or during power ascension to EPU power in the feedwater and main steam lines (due to standing waves in stagnant side branches) and describe how the acoustics driven dynamic pressure loading acting on the components inside the steam generator under EPU conditions will be estimated.

Response:

A piping vibration program for EPU, as described in EPU LAR Attachment 5 Section 2.12.1.2.3.4, Vibration Monitoring, has been established to ensure that any steady state flow induced piping vibrations are not detrimental to the plant piping systems, including the main steam and feedwater piping systems. Piping systems that will experience an increase in process flow rates as a result of EPU have been monitored at current power conditions (i.e., baseline plant walkdowns), and will be monitored during power ascension to ensure piping system acceptability with respect to piping vibration. Branch piping and vents/drains directly connected to main piping systems experiencing flow rate increases due to EPU have also been monitored at current plant conditions, and will be monitored during power ascension, to ensure that potential acoustic resonance affecting these lines are acceptable with respect to piping vibration. The piping vibration program and related monitoring will be performed in accordance with ASME OM-S/G-2007 Part 3.

The St. Lucie steam generators are Replacement Steam Generators (RSG) fabricated by Babcock & Wilcox Canada (B&W). [ ][a,c].

The design details described above make the steam separation equipment a compact and rigid structure which has a very high natural frequency; hence, they are not susceptible to flow-induced vibration and acoustic resonance.

Operating experience at steam flows greater than the steam flow at EPU conditions confirms that the steam separators used in the St. Lucie Unit 1 RSGs are not likely to suffer any degradation due to acoustic or flow induced vibration following the EPU power uprate. To date, no station with B&W RSGs has ever observed any degradation of the steam separators that could be attributed to flow induced vibration or acoustic pressure fluctuations in the steam space.

EMCB-39

Please discuss procedures in place for preparation, response and preventive actions designed to detect and remove loose parts that could potentially occur due to component degradation as a result of the EPU increased steam feed flows. Also please discuss the potential for damage that these loose parts could have to safety related SSCs.

Response:

FPL employs procedures which address examination, monitoring and maintenance activities associated with ensuring the integrity of the steam generator secondary side components.

With regards to inspection of the tube bundle, two types of inspection techniques are routinely performed: 1) primary side eddy current (ECT) examinations, and 2) secondary side foreign object search and retrieval (FOSAR). FPL procedures provide the schedule for SG primary and secondary side inspections.

St. Lucie Unit 1 employs a Loose Parts Monitoring System (LPMS), which consists of transducers, preamplifiers, a computer and a flat panel display to automatically detect and record metal to metal contact within the RCS. A description of the LPMS as well as detection capabilities and type of data collected is provided in St. Lucie Unit 1 UFSAR Section 5.2.5.2. The LPMS that is permanently installed in St Lucie Unit 1 provides the in-service monitoring function during plant operation. In addition refer to LAR Attachment 5, LR Section 2.2.2.5.2.7 "Loose Parts and Foreign Objects" for a discussion pertaining to secondary side loose parts analysis.

The potential for damage to safety related SSCs in the SGs due to loose parts is limited to tube wear, which has the potential of causing a primary to secondary side leak if the wear is not detected. Damage from loose parts is a function of object material type and size, applied drag force, object vibration and tube displacement.

Most loose parts will first enter the tube region at the top of the tubesheet. Typically, there will be insufficient drag force in the vertical direction to transport the parts to the upper bundle or the U-Bend region and the parts will remain on the top of the tubesheet. Before a part can reach the U-Bend, it needs to first pass through multiple lattice grid tube supports. Therefore, any parts reaching the U-Bend must be small and light, and therefore the potential for tube wear that could result in a tube leak is very low.

The risk of a primary to secondary tube leak due to loose part damage is managed through regularly scheduled inspection and maintenance activities, including eddy current (ECT) inspections, tubesheet flushing, and foreign object search and retrieval (FOSAR). ECT inspections will detect tube wear due to loose parts so that the affected tubes can be plugged and/or the objects can be removed. Tubesheet flushing and FOSAR will identify parts on the top of tubesheet region so the parts can be removed and/or the affected tubes plugged if required.

[ ]^{a,c}.

[ ]^{a,c}

Figure 1: Typical Dynamic pressure in U-Bend region

[ ]^{a,c}

Figure 2: Typical [ ]^{a,c} in U-Bend region

[ ]^{a,c}

Figure 3: Typical Dynamic pressure at Bundle Entrance region

[ ]^{a,c}

Figure 4: Typical [ ]^{a,c} at Bundle Entrance Region

EMCB-40

Please discuss planned inspections to identify degradation, due to the EPU increased feed and steam flow rates, of the steam drum and SG upper internals and of the feedwater ring with J-nozzles and supports.

Response:

Consistent with the requirements of the Steam Generator Secondary Side Integrity Plan, FPL plans to perform a baseline visual inspection of the steam separators during secondary side inspection of the steam generators prior to the implementation of the EPU. FPL also plans to perform a follow-up post-EPU visual inspection of the steam separators during the refueling outage following the first operating cycle under EPU conditions. Visual inspections are performed in accordance with FPL procedures. Inspections are expanded, if necessary, based on the results of the inspection. Subsequent visual inspections of the steam separators are based upon the results of the refueling outage inspection.

The St. Lucie steam generators are replacement steam generators (RSG) fabricated by Babcock & Wilcox Canada (B&W). [ ][a,c]; hence, the steam separation equipment forms a compact, rigid structure which has a very high natural frequency and is immune to flow-induced vibration.

Prior to the EPU, the steam flow per separator was approximately [ ][a,c] lb/hr. Following EPU, the steam flow per separator will increase to approximately [ ][a,c] lb/hr. Exelon Energy has implemented a 5% power uprate for both Braidwood Unit 1 and Byron Unit 1 and both units have B&W RSGs that use CAP-3 steam separators which are similar to the CAP-2 steam separators in the St. Lucie Unit 1 RSGs. After the power uprate, the steam flow per steam separator at Braidwood Unit 1 was approximately [ ][a,b] lb/hr. Exelon Energy has not reported any SG degradation due to the power uprate. Secondary side inspections have also been conducted at four other PWRs having B&W RSGs. Those inspections showed no signs of damage. The only unusual observation was at one plant, where some deposit flow patterning and possible FAC was observed on the bottom plate of several of the secondary separators. To date, no customer with B&W RSGs has ever observed any degradation to RSG components other than the tubes, that could be attributed to flow induced vibration or acoustic pressure fluctuations in the steam space.

Therefore, operating experience at steam flows similar to the expected steam flow at EPU conditions confirms that the steam separators used in the St. Lucie Unit 1 RSGs are not likely to suffer any degradation due to acoustic or flow induced vibration following the EPU power uprate.

[ ][a,c]

To date there have been no indications of perforations in the any steam separator components in B&W RSGs in operation for many years at 11 PWR plants, including St. Lucie Unit 1. [ ].[a,c]

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:

The comparison of the controlling normal operation (deadweight plus thermal) discharge and suction nozzle loads are shown in the table below. LAR Attachment 5, LR Figure 2.2.2.6-1 shows the orientation of each RCP in the RCS. The A2 and B1 pumps are grouped together as shown in the table below because these two pumps have the same orientation relative to the global coordinate system of the RCS. Since the RCS support systems for all the RCS major components are symmetrical, each of these pumps experiences normal operation condition loads of the same magnitude. The A1 and B2 pumps are grouped following the same reasoning as the A2 and B1 pumps.

Seismic and pipe break loads were not affected by the EPU. The table below shows that, for each nozzle type (suction or discharge), the shear forces and bending moments for EPU conditions are less than the largest current values. Therefore, the design basis stresses in the AOR remain bounding for EPU conditions.

RCP Nozzle SRSS Loads

Condition	**Pump**	**Nozzle**	**Current(1),(2)**		**EPU(1)**	
			F (kips)	**M (in-kips)**	**F (kips)**	**M (in-kips)**
Thermal + DWt	A2 & B1	D				
	A1 & B2	D				
	A2 & B1	S				
	A1 & B2	S				

(a,c)

Notes:

(1) The SRSS values are of the local Y and Z values for each discharge nozzle, and of the X and Z values for each suction nozzle. This captures the entire shear force and bending moment for each nozzle (by excluding loads in the axial direction).

(2) Maximum current loads are shown in **bold.**

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 RCPs and RCP motors are only supported on snubbers and springs.

Response 42b:

The increase of 24,000 lbs 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 EPU LAR changed the AOR results for the RVI. LAR Attachment 5, LR Table 2.2.3-1 values are for EPU conditions.

Response 43b:

The methodology used to derive EPU results is:

1. Primary stresses in the St. Lucie Unit 1 RVI were obtained from the AOR.
2. RVI design loads associated with the EPU were compared with those used in the AOR. Any design load changes that could potentially increase stresses, as calculated in the AOR, were identified.
3. The impact of the design load changes on the AOR stresses was determined.
4. Additional primary stresses and non-thermal secondary stresses (not addressed in the AOR) were calculated using design loads that reflect the EPU. In some cases, stresses that were addressed in the AOR were re-calculated to remove excess conservatism.
5. 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.
6. Stresses were combined and evaluated with respect to the acceptance criteria.
7. 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 3Sm, the value of 3Sm; show that when removing the thermal bending stresses, the primary plus secondary stress intensity value is less than 3Sm; 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 double asterisk is shown adjacent to the text for the core shroud (fatigue usage**).

Only the core shroud did not meet the $3S_m$ limit. The specific values shown below reflect the worst case stress results at the core shroud baffle.

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

Without thermal bending:
$P_m + P_b + Q = [\quad]^{(a,c)} < 3S_m = [\quad]^{(a,c)}$

As shown in the table, CUF = $[\quad]^{(a,c)}$.

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 SSC.

The various evaluations related to the impact of non-safety related SSCs upon safety-related SSCs are included in the St. Lucie Unit 1 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 LAR and identified below.

Evaluation	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 Unit 1 Design Change Package procedure contains 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. As per the procedure, 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 Package procedure with respect to performing interaction evaluations ensures that compliance with the requirements of 10 CFR 54.4(1)(a) 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 relative to St. Lucie Unit 1 EPU is discussed in Attachment 5 of the LAR in the sections identified below.

Regulation	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

ATTACHMENT 3

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

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-11-224

CAW-11-3230

September 2, 2011

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: St. Lucie Unit 1 – Responses to NRC Requests for Additional Information (RAIs) EMCB RAI-2, EMCB RAI-15, EMCB RAI-16, EMCB RAI-30, EMCB RAI-41 and EMCB RAI-44 on the Extended Power Uprate License Amendment Request (Proprietary)

Reference:

1. NRC E-Mail, T. Orf (NRC) to C. Wasik (FPL), "St. Lucie Draft Mechanical and Civil RAIs (EMCB)," July 27, 2011, 10:45 AM.

The proprietary information for which withholding is being requested is that included in the response to NRC Requests for Additional Information (RAIs) EMCB RAI-2, EMCB-15, EMCB-16, EMCB-30, EMCB RAI-41 and EMCB RAI-44 transmitted by Reference 1 and further identified in Affidavit CAW-11-3230 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 Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-11-3230, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared J. A. Gresham, 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:

J. A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 2nd day of September 2011

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Cynthia Olesky, Notary Public
Manor Boro, Westmoreland County
My Commission Expires July 16, 2014
Member. Pennsylvania Association of Notaries

(1) I am Manager, 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 FPL's Responses to NRC Requests for Additional Information (RAIs) EMCB RAI-2, EMCB RAI-15, EMCB RAI-16, EMCB RAI-30, EMCB RAI-41 and EMCB RAI-44 on the Extended Power Uprate License Amendment Request (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 are included in NRC E-Mail, T. Orf (NRC) to C. Wasik (FPL), "St. Lucie Draft Mechanical and Civil RAIs (EMCB)," July 27, 2011, 10:45 AM. The proprietary information provided by Westinghouse supports the St. Lucie Unit 1 Extended Power Uprate (EPU) License Amendment Request (LAR), and may be used only for that purpose.

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

(a) Support the St. Lucie Unit 1 EPU LAR by demonstrating acceptability of various components under EPU conditions.

Further this information has substantial commercial value as follows:

(a) The information can provide baseline data for future engineering and analytical services.

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 engineering, analytical and licensing defense services 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.

In order for competitors of Westinghouse to duplicate this information, 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).

Copyright Notice

ATTACHMENT 4

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

**Babcock and Wilcox
Application for Withholding
Proprietary Information
from Public Disclosure**

(Cover page plus 4 pages)

babcock & wilcox canada ltd.
581 coronation boulevard cambridge, on n1r 5v3 canada
phone 519.621.2130 fax 519.621.2310 www.babcock.com

August 29, 2011

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
U.S.A.

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Florida Power & Light Letter L-2011-361 (Responses to EMCB RAI-33 through 40 for the St. Lucie Unit 1 EPU Submittal)

Dear Sir/Madam:

The proprietary information for which withholding is being requested in the above-referenced document is identified in the attached affidavit signed by the owner of the proprietary information, Babcock & Wilcox Canada Ltd. 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 Babcock & Wilcox Affidavit should reference this letter, and should be addressed to the undersigned.

Yours truly,

BABCOCK & WILCOX CANADA LTD.

Jeff Millman

Jeffrey Millman,
Manager, Nuclear Engineering

Attach./

Cc: K. McHugh
J. MacQuarrie
J. Albert

PROVINCE OF ONTARIO

REGIONAL MUNICIPALITY OF WATERLOO

AFFIDAVIT OF JEFFREY MILLMAN

I, Jeffrey Millman, of the Village of Ayr, in the Township of North Dumfries, Regional Municipality of Waterloo, in the Province of Ontario, being sworn, make oath and say as follows:

1. I am the Manager, Nuclear Engineering of Babcock & Wilcox Canada Ltd. ("B&W"), 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 rulemaking proceedings, and am authorized to apply for its withholding on behalf of B&W.

2. I am making this Affidavit in conformance with the provisions of 10CFR Section 2.390 of the Commission's regulations and in conjunction with the Babcock & Wilcox Canada Ltd. Application for Withholding accompanying this Affidavit.

3. I have personal knowledge of the criteria and procedures utilized by B&W in designating information as a trade secret, proprietary 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 B&W.

   (ii) The information is of a type customarily held in confidence by B&W and not customarily disclosed to the public. B&W 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 B&W 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 follow:

   (a) The information reveals the distinguishing aspects of a process, component, structure, tool, method, etc., where prevention of its use by any of B&W's competitors without license from B&W 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 its expenditure of resources or improve its competitive position in the design, manufacture, shipment, installation, quality assurance, or licensing of a similar product.

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

There are sound policy reasons behind the B&W system which include the following:

- The use of such information by B&W gives B&W a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect B&W's competitive advantage.
- It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the B&W ability to sell products and services involving the use of such information.
- Use by a competitor of B&W would put B&W at a competitive disadvantage by reducing the competitor's expenditure of resources at B&W's expense.
- B&W's 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 10CFR 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 identified in Florida Power & Light Letter L-2011-361 (Responses to EMCB RAI-33 through 40 for the St. Lucie Unit 1 EPU Submittal) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk.

The information which is proprietary in the proprietary version 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 (d) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary. These lower case letters refer to the types of information B&W customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(d) of this affidavit pursuant to 10 CFR 2.390(b)(l).

SWORN BEFORE ME in the)
City of Cambridge in the)
Province of Ontario, this)
29th day of August, 2011.)

JEFFREY MILLMAN

A Commissioner, etc.
Jon R. Johnson