Florida Power & Light, 9760 S.W. 344 St. Homestead, FL 33035

FPL

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L-2011-414
10 CFR 50.90
10 CFR 2.390

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Supplemental Response to NRC Mechanical and Civil Engineering Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205

References:

(1) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request No. 205: Extended Power Uprate (EPU)," (TAC Nos. ME4907 and ME4908), Accession No. ML103560169, October 21, 2010.

(2) Email from J. Paige (NRC) to T. Abbatiello (FPL), "Turkey Point EPU – Mechanical and Civil (EMCB) Request for Additional Information – Round 1", Accession No. ML111090285, April 19, 2011.

(3) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-140), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Mechanical and Civil Engineering Issues, May 19, 2011.

(4) Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU – Mechanical and Civil (EMCB) Request for Additional Information – Round 2", Accession No. ML11193A111, July 12, 2011.

(5) M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-248), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205, August 25, 2011.

By letter L-2010-113 dated October 21, 2010 [Reference 1], Florida Power and Light Company (FPL) requested to amend Renewed Facility Operating Licenses DPR-31 and DPR-41 and revise the Turkey Point Units 3 and 4 Technical Specifications (TS). The proposed amendment will increase each unit's licensed core power level from 2300 megawatts thermal (MWt) to 2644 MWt and revise the Renewed Facility Operating Licenses and TS to support operation at this increased core thermal power level. This represents an approximate increase of 15% and is therefore considered an extended power uprate (EPU).

By email from the U.S. Nuclear Regulatory Commission (NRC) Project Manager (PM) on April 19, 2011 [Reference 2], additional information regarding mechanical and civil engineering issues was requested by the NRC staff in the Mechanical and Civil Engineering Branch (EMCB) to support the review of the EPU License Amendment Request (LAR) [Reference 1]. The Request for Additional Information (RAI) consisted of thirty-eight (38) questions previously discussed with the NRC staff at a public meeting in Rockville, MD on March 31, 2011. FPL provided its response to these RAI questions on May 19, 2011 via letter L-2011-140 [Reference 3].

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By email from the NRC PM on July 12, 2011 [Reference 4], additional information regarding mechanical and civil engineering issues was again requested by the NRC EMCB staff to support their review of the EPU LAR. The RAI consisted of six (6) follow-up questions regarding EPU stress values and margins for several piping systems and components. FPL provided its response to these RAI questions on August 25, 2011 via letter L-2011-248 [Reference 5]. The submittal contained two new commitments, one in the form of a proposed license condition on each unit associated with the spent fuel pool supplemental heat exchanger modifications [RAI question EMCB-2.1] and the other associated with completion of additional reactor vessel support structural analyses by October 31, 2011 [RAI question EMCB-2.4]. The results of the analyses addressing the reactor vessel (RV) support structure as well as the evaluations of other affected components including the RV, RV head, RV internals, reactor coolant piping and components, and nuclear fuel are documented in Attachments 1 (non-proprietary) and 2 (proprietary) to this letter.

Attachment 3 contains the application for withholding the proprietary information contained in Attachment 2 from public disclosure. As Attachment 2 contains information proprietary to Westinghouse Electric Company, LLC (Westinghouse) and AREVA NP, Inc (AREVA), it is supported by affidavits signed by Westinghouse and AREVA, the owners of the information. The affidavits set 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 AREVA be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items provided in Attachment 2 of this letter or the supporting Westinghouse affidavit should reference CAW-11-3268 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, PA 16066.

Correspondence with respect to the copyright or proprietary aspects of the items provided in Attachment 2 of this letter or the supporting AREVA affidavit should be addressed to Gayle F. Elliot, Product Licensing, AREVA NP, Inc, P.O. Box 10935, Lynchburg, VA 24506-0935.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-113 [Reference 1]. This submittal contains no new commitments and no revisions to existing commitments.

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

Should you have any questions regarding this submittal, please contact Mr. Robert J. Tomonto, Licensing Manager, at (305) 246-7327.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on October 12, 2011.

Very truly yours,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Attachments (3)

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Resident Inspector, Turkey Point Nuclear Plant
Mr. W. A. Passetti, Florida Department of Health (w/o Attachment 2)

Turkey Point Units 3 and 4

SUPPLEMENTAL RESPONSE TO
NRC MECHANICAL AND CIVIL ENGINEERING BRANCH (EMCB)
REQUEST FOR ADDITIONAL INFORMATION REGARDING
EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST NO. 205

ATTACHMENT 1
(Non-Proprietary)

Response to Request for Additional Information

The following information is provided by Florida Power and Light Company (FPL) in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support License Amendment Request (LAR) 205, Extended Power Uprate (EPU), for Turkey Point Nuclear Plant (PTN) Units 3 and 4 that was submitted to the NRC by FPL via letter L-2010-113 dated October 21, 2010 [Reference 1].

By email from the NRC Project Manager (PM) on April 19, 2011 [Reference 2], additional information regarding mechanical and civil engineering issues was requested by the NRC staff in the Mechanical and Civil Engineering Branch (EMCB) to support the review of the EPU LAR. The RAI consisted of thirty-eight (38) questions previously discussed with the NRC staff at a public meeting in Rockville, MD on March 31, 2011. FPL provided its response to these RAI questions on May 19, 2011 via letter L-2011-140 [Reference 3].

By email from the NRC PM on July 12, 2011 [Reference 4], additional information regarding mechanical and civil engineering issues was again requested by the NRC EMCB staff to support their review of the EPU LAR. The RAI consisted of six (6) follow-up questions regarding EPU stress values and margins for several piping systems and components. FPL provided its response to these RAI questions on August 25, 2011 via letter L-2011-248 [Reference 5]. The submittal contained two new commitments, one in the form of a proposed license condition on each unit associated with the spent fuel pool supplemental heat exchanger modifications [RAI question EMCB-2.1] and the other associated with completion of additional reactor vessel support structural analyses by October 31, 2011 [RAI question EMCB-2.4]. The results of the analyses addressing the reactor vessel (RV) support structure as well as evaluations of other affected components including the RV, RV head, RV internals, reactor coolant piping and components, and nuclear fuel are documented below.

NRC's RAI question EMCB-2.4 in Reference 4 stated:

"In response to EMCB-1.19 on the allowable load for the reactor vessel support structure, the licensee utilized yield strength of the material based on certified material test reports (CMTRs). Typically, design basis calculations are based on minimum yield strength of the material from the applicable codes rather than actual strength values from CMTRs. The licensee is requested to address if the current licensing basis (CLB) permits the use of the material CMTR values in the design basis evaluations."

FPL's response to EMCB-2.4 in Reference 5 stated the following:

"Based on discussions with the Mechanical and Civil Branch personnel during the public meeting held on June 23, 2011, FPL reviewed the Turkey Point current licensing basis, original design codes, and design specifications. FPL agrees with the NRC staff input that code specified material allowable values in design basis evaluations should be used. Accordingly, FPL is performing additional analyses to demonstrate that stresses for the reactor vessel (RV) support structure will be less than the minimum yield stress for support materials as allowed by the ASTM and ASME codes of record for the faulted load conditions. Evaluations will also be performed to confirm that changes in loads as a result of the revised RV support analyses to other affected components including the RV, RV head, RV internals, and nuclear fuel are acceptable. A detailed schedule for this work is currently under development and will be provided to the NRC project manager once finalized. At this time, FPL plans to complete these additional analyses for the reactor vessel supports by October 31, 2011 and provide a follow up response to this RAI at that time. Note that for the Normal, Upset and Emergency load conditions, stresses for support materials in the reactor vessel support structure are less than applicable code allowables without credit for CMTRs. This response will address the faulted load conditions."

Revised Reactor Vessel Support Stress Analysis, RAI EMCB 2.4

The Turkey Point Units 3 and 4 current licensing bases for the primary equipment supports does not use CMTRs in the structural evaluations. An alternate approach to evaluating the system for EPU conditions and the subsequent qualification of the RV supports is presented below.

The original RV structural analyses for EPU conditions were performed using plant specific RV support stiffnesses. The plant specific support stiffnesses were developed using typical "free body" beam models of the structure and the GT STRUDL structural analysis computer code. The "free body" beam model is widely accepted approach to account for the global axial, bending, torsion, and shear stiffnesses of an assembly of typical structural members.

In order to show that the design basis loads for the EPU conditions are less than the allowable design basis loads computed using "nominal" material properties the loads had to be reduced. The "nominal" material properties are defined as the minimum material yield and ultimate strength values based on the applicable ASTM material specification and adjusted for the operating metal temperature. A number of assessments of the loads was performed and it was determined that if the RV support stiffnesses were reduced the corresponding design basis loads will be reduced.

The RV support design was reviewed with respect to the primary load path within the steel support structure as it transfers loads from the RV nozzle to the surrounding concrete. The design of the support is such that local bending of flanges of the support beams as well as shear deflection of connecting bolts are a significant contributor to the overall stiffness of the support. These local effects are not accounted for in the "free-body" beam model. The local effects were investigated in greater detail with a detailed 3-dimensional (3D) solid model of the support structure with the ANSYS general purpose finite element analysis computer code. The detailed 3D model includes the geometric non-linearities of the structure by including contact elements between contact surfaces of bolts and bolt holes and beams mounted to other beams.

The detailed 3D model was used to produce RV support stiffnesses reflecting the local effects of the structure. The revised RV support stiffnesses were incorporated and a reanalysis of the RV internals structural model for EPU seismic and loss-of-coolant accident (LOCA) cases performed.

This RV internals structural analysis produced revised RV support loads. The revised support loads were reconciled using allowable loads computed with "nominal" material properties for the materials used to construct the RV support assembly. The results are summarized in the revised Licensing Report (LR) Tables 2.2.2.3-5 and 2.2.2.3-6 presented below and demonstrate that the stresses for the RV support structure will be less than the minimum yield stress for support materials as allowed by the ASTM and ASME codes of record for the faulted load conditions.

In addition to the changes required to the RV supports analysis, there were impacts to other NSSS analyses due to the change in the calculated RV support stiffnesses. These are summarized below.

Other Potentially Affected LAR Sections

2.2.2.1 NSSS Piping, Components and Supports

The Reactor Coolant Loop (RCL) piping analyses documented in LR Section 2.2.2.1 were evaluated for the change in reactor vessel support stiffness for resolution of RAI EMCB-2.4. The deadweight, thermal, and seismic analyses of the RCL are not impacted because the reactor vessel is modeled as a fixed anchor, and the reduction in the reactor vessel support stiffness is not sufficient to invalidate this modeling technique for these analyses. The LOCA analysis is impacted because the reactor vessel LOCA displacements provided in LR Section 2.2.3 were revised due to the change in reactor vessel support stiffness. The RCL LOCA cases were reanalyzed with the revised reactor vessel displacements. The impact of the revised RV support stiffness on steam generator and reactor coolant pump support loads, and reactor vessel and reactor coolant pump inlet and outlet nozzle loads are discussed further below. Steam generator inlet and outlet nozzle loads were compared to the allowable loads and found to be bounded by the original EPU results. Piping stresses were recalculated and found to be bounded by the stresses provided in LR Table 2.2.2.1-1.

2.2.2.3 Reactor Vessel and Supports

The RV internals system model was reanalyzed using revised stiffness values for the RV supports. As a result of this reanalysis, some of the seismic and LOCA interface loads on the RV nozzles, core support pads, and bottom mounted instrumentation (BMI) tubes have changed from the previously-analyzed values. Since the previous interface loads produced the primary plus secondary (P+Q) stress intensity ranges and fatigue usage values currently in the Turkey Point EPU LR, the revised interface loads were assessed for their impact on those current LR values using a simplified, conservative method of calculation. The revised support stiffness for the RV results in increases in the P+Q stress intensity range and fatigue usage factor for the core support pads and increases in the fatigue usage factors for the inlet and outlet nozzles of the RV. Table A below presents the maximum possible values for the P+Q stress intensity range and fatigue usage values for the core support pads and for the inlet or outlet nozzle as listed in LR Table 2.2.2.3-1. Table A also presents the maximum possible stress intensity range and fatigue usage values for those components due to the revised support stiffness. The fatigue usage presented for the inlet and outlet nozzles is the maximum value for either nozzle. All revised values continue to meet ASME Code allowables. All other values from LR Table 2.2.2.3-1 remain bounding.

Table A
Comparison of Current EPU Values with Revised EPU Values

Component	Category	Current EPU Maximum from LR Table 2.2.2.3-1	EPU with Revised Stiffness – Maximum Value Compared to ASME Code Allowable
Shell at Core Support Pads	P+Q	35,637 psi	35,950 psi < 80,100 psi
	Fatigue Usage	0.478	0.585 < 1.0
Inlet or Outlet Nozzle	Fatigue Usage	0.0732	0.0869 < 1.0

The revised RV support stiffnesses, considering the "local" effects, result in reduced RV support loadings that have been shown to be within design basis allowable load limits established with "nominal" material properties. The new results are presented below.

Table 2.2.2.3-5
Revised Final RV Support Load Combinations (per Support)

Combination	Vertical	Horizontal
Normal = (D + L + T)	491 kips	0 kips
Upset = (D + L + T + E)	462 kips	57 kips
Emergency = (D + L + T + E')	500 kips	128 kips
Faulted = (D + L + T + R)	2,073 kips	1,180 kips

Per LR Section 2.2.2.3, D is Dead Load Stress, L is Live Load Stress, T is Thermal Stress, E is Design Earthquake Stress, E' is Maximum Hypothetical Earthquake Stress, and R is Stress due to Pipe Rupture (LOCA) reactions.

Table 2.2.2.3-6
Summary of RV Support Critical Component Stress Interaction Ratios

Component	Comparison	Normal	Upset	Emergency	Faulted
14WF342 Beam (Web Shear)	Actual Load	491 kips	462 kips	500 kips	2073 kips
	Allowable Load(2)	1587 kips	2111 kips	2182 kips	2182 kips
	IR (<100%)(1)	30.94%	21.89%	22.91%	95.00%
Tangential Bracket (Web Shear)	Actual Load	0 kips	57 kips	128 kips	1180 kips
	Allowable Load(2)	892 kips	1186 kips	1226 kips	1226 kips
	IR (<100%)(1)	0%	4.81%	10.44%	96.25%
Cross Beam Bolt (Shear)	Actual Load	0 kips	57 kips	128 kips	1180 kips
	Allowable Load(2)	1145 kips	1523 kips	1603 kips	1603 kips
	IR (<100%)(1)	0%	3.74%	7.99%	73.61%

1. Interaction Ratio IR = (Actual Load / Allowable Load) X 100%
2. Component "Nominal" material properties were used to compute the design basis allowable loads.

2.2.2.4 Control Rod Drive Mechanism

AREVA performed the necessary calculations to determine the impact of the revised RV support stiffness on the AREVA supplied Integrated Head (IHA) package. The results show that for all the components provided by AREVA, the design remains within code allowables and meets the current licensing basis.

In addition, in support of FPL's submittal to the NRC, AREVA has reviewed the EPU LR Sections 2.2.2.3, 2.2.2.4 and 2.8.4 and confirms that, with respect to the components provided by AREVA (RV Head, CRDMs, IHA), the conclusions presented in these sections regarding these components remain valid. The analyses adequately address the effects of the EPU and meet all aspects of the current licensing bases. A revised LR Table 2.2.2.4-1 is provided below.

Table 2.2.2.4-1
Primary Membrane Stress Intensity, Primary Membrane + Bending Stress Intensity, and Cumulative Usage Factor

CRDM Location	Primary Membrane Stress Intensity: Pm (Allowable) Ksi	Primary Membrane + Bending Stress Intensity: Pl + Pb (Allowable) Ksi	Cumulative Usage Factor U Maximum (U Allowable)
Latch Housing	[ ] (Note 1) (38.9)	[ ] (Note 1) (58.3)	[ ] (1)
Rod Travel Housing	[ ] (Note 1) (38.9)	[ ] (Note 1) (58.3)	[ ] (1)
Cap	[ ] (Note 1) (38.9)	[ ] (Note 1) (58.3)	[ ] (1)
Lower Joint	[ ] (Note 1) (38.9)	Note 2	[ ] (1)
Middle Joint	[ ] (Note 1) (38.9)	Note 2	[ ] (1)
Upper Joint	[ ] (Note 1) (38.9)	Note 2	[ ] (1)

Note 1: Pm, Pl + Pb values listed are for faulted conditions only.
Note 2: [ ]

2.2.2.5 Steam Generators and Supports

In response to RAI EMCB-2.4 for the Turkey Point Units 3 and 4 RV support analysis, new RCL piping LOCA loads on the steam generators (SG) and SG supports were evaluated. Based on the assessment, the SG supports are acceptable for the revised loads.

Some changes to LR Table 2.2.2.5-7 for SG supports were required. The revised table is presented below.

Table 2.2.2.5-7
RCL Primary Equipment Steam Generator Support Member Stresses

Member	Faulted		
	Actual	Allowable	Stress Ratio, %***
Steam Generator Columns*	[ ] b,c	806.00 kips	[ ] b,c
Steam Generator Lower Lateral Support Bumpers*	[ ] b,c	892.00 kips	[ ] b,c
Steam Generator Upper Support Bumpers**	[ ] b,c	1470.00 kips	[ ] b,c

*The values remain bounding

**Governing bumper is now different than the one previously identified.

***Percent of Allowable = (Actual/Allowable) x 100%

2.2.2.6 Reactor Coolant Pumps and Supports

In response to RAI EMCB-2.4 for the Turkey Point Units 3 and 4 RV support analysis, new RCL piping LOCA loads on the reactor coolant pump (RCP) and RCP supports were evaluated for impact. It was concluded that the revised loads do not affect the loading conditions previously evaluated for the component. Therefore, the results presented in LR Section 2.2.2.6 for RCPs and RCP supports remain bounding.

2.2.2.7 Pressurizer and Supports

As a result of the changes identified from the RV support stiffness change, it was determined that there was no effect on the loading conditions previously evaluated for the component. Therefore, results presented in LR Section 2.2.2.7 for the pressurizer and pressurizer supports remain bounding.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

The nonlinear time history LOCA and seismic analyses of the reactor pressure vessel (RPV) system has been revised to incorporate the change in RV support horizontal stiffness. As a result, the core plate motions and the RV/internals interface loads have been revised. The core plate motions were used in fuel grid analysis and to confirm the structural integrity of the fuel. The loads calculated at vessel/internals interfaces are used to evaluate the structural integrity of the RV and its internals. The qualification of the fuel structural integrity and of the RV is addressed in other sections of this response.

For the RV internals, the results presented in the LR Table 2.2.3-1, "Reactor Internal Components Stresses and Fatigue Usage Factors," have been updated below to address the change in the RV support stiffness input. Only two stress intensity values for EPU increased slightly: the Deep Beam and Upper Support Columns but they remain within the allowable values. All other results remain bounding. It was determined that no other changes to LR Section 2.2.3 were required due to this change in the EPU analysis.

Table 2.2.3-1
Reactor Internal Components Stresses and Fatigue Usage Factors

Component	**Stress Intensity (ksi) S.I. = (P_m + P_b + Q)**		**Allowable(6) S.I. (3 Sm) ksi**	**Fatigue Usage**	
	Current AOR	**After EPU**		**Current AOR(8)**	**After EPU**
Upper Support Plate	[ ]a,c	[ ]a,c	48.60	[ ]a,c	[ ]a,c
Deep Beam	[ ]a,c (3)	[ ]a,c	48.60	[ ]a,c(3)	[ ]a,c
Upper Core Plate	[ ]a,c	[ ]a,c	48.60	[ ]a,c	[ ]a,c
Upper Core Plate Alignment Pins	[ ]a,c(1)	[ ]a,c(1)	49.20	[ ]a,c	[ ]a,c
Upper Support Columns	[ ]a,c	[ ]a,c	48.60	[ ]a,c	[ ]a,c
Lower Support Plate	[ ]a,c (3)(1)	[ ]a,c	48.60(7)	[ ]a,c(3)	[ ]a,c
LSP to Core Barrel Weld	[ ]a,c(1)	[ ]a,c	48.60(7)	[ ]a,c	[ ]a,c
Lower Core Plate	[ ]a,c	[ ]a,c	48.60	[ ]a,c	[ ]a,c
Lower Support Columns	[ ]a,c	[ ]a,c	48.60	[ ]a,c	[ ]a,c
Core Barrel:					
Flange	[ ]a,c	[ ]a,c	49.20	[ ]a,c	[ ]a,c
Outlet Nozzle	[ ]a,c (1)	[ ]a,c(1)	21.90(2,4)	[ ]a,c	[ ]a,c
Radial Keys and Clevis Insert Assembly	[ ]a,c	[ ]a,c	170.10(5)	[ ]a,c	[ ]a,c

Table 2.2.3-1 (con't'd)
Reactor Internal Components Stresses and Fatigue Usage Factors

Notes:

1. Exceeded $3S_m$ limit, simplified elastic-plastic analysis performed to calculate fatigue strength.
2. Allowable based on weld quality factor.
3. This component is bounded by limiting location. Stresses are reported and evaluated only at limiting location.
4. For current AOR, 34.40 ksi is allowable value.
5. For current AOR, 96.75 ksi is allowable value.
6. The PTN reactor internals were designed and built prior to the implementation of Subsection NG of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, and therefore, a plant-specific stress report on the reactor internals was not required. The structural integrity of the PTN reactor internals design has been ensured by analyses performed on both generic and plant-specific bases to meet the intent of the ASME Code. These analyses were used as the basis for evaluating critical PTN reactor internal components for EPU RCS conditions and revised NSSS design transients. The original NG criteria used the allowable stress levels of the 1965 Edition of the ASME B&PV Code, Section III, Article 4, through Summer 1966 Addenda. A reactor internals structural and fatigue evaluation for a three-loop plant similar to PTN was performed using the rules and structural limits in the 1969 and 1971 Editions of the ASME Code, Section III, Division 1, and the criteria of the ASME Code for Design by Analysis in Section VIII, Division 2. Recent plant specific evaluations for PTN use the NRC approved version of ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NG, which is the 1998 Edition up to and including 2000 Addenda.
7. For current AOR, 48.30 ksi is allowable value.
8. In some cases, the fatigue usage for the current AORs is greater than the after EPU conditions due to the conservatisms in the current AORs consisting of generic analyses and comparisons to similar plants at bounding conditions.

Table 1 has previously been provided in response to NRC RAI EMCB-1.34 and is shown below for information. The primary plus secondary stress intensity results from Table 1 have been incorporated into LR Table 2.2.3-1.

Table 1
Turkey Point EPU Primary Plus Secondary Stress Intensity Summary for UCP Alignment Pins, LSP and Weld, and Outlet Nozzle

Component	Primary Plus Secondary Stress Intensity P_m+P_b+Q (psi) NG-3222.2	Secondary Stress Q (psi)	Ke NG-3228.3(b)	S_y/S_u NG-3228.3(f)	Thermal Ratcheting NG-3228.3(d) (NG-3222.5)
Upper Core Plate Alignment Pins	[ ][a,c(1)]	[ ][a,c(3)]	1.0	[ ][a,c]	N/A
Lower Support Plate	[ ][a,c(1)]	[ ][a,c(3)]	1.0	[ ][a,c]	N/A
Lower Support Plate Weld	[ ][a,c(1)]	[ ][a,c(3)]	1.0	[ ][a,c]	N/A
Outlet Nozzle	[ ][a,c(2)]	[ ][a,c(4)]	1.0	[ ][a,c]	Met Criteria[5]

Notes:

1. Primary plus secondary stress intensity is less than 3Sm = [49,200 psi UCPAP and 48,600 psi LSP and Weld][a,c], and simplified elastic-plastic analysis is not required.
2. Primary plus secondary stress intensity is greater than 3Sm = [21,879 psi][a,c], and simplified elastic-plastic analysis is required.
3. Primary plus secondary stress intensity including thermal bending is less than 3Sm (Pm+Pb+Qm+Qb<3Sm). Therefore, primary plus secondary stress intensity excluding thermal bending is less than 3Sm (Pm+Pb+Qm<3Sm).
4. Excluding secondary thermal bending stress (Qb = [12,500 psi=30,300-17,800][a,c], the primary plus secondary stress intensity is less than 3Sm (Pm+Pb+Qm<3Sm).
5. Thermal ratcheting requirement of the ASME Code Subsection NG-3222.5 for core barrel nozzle is met since the maximum allowable range of thermal stress (y') calculated on the elastic basis [73.7 ksi][a,c] is greater than the calculated value of Q [26.3 ksi][a,c].

2.2.7 Bottom-Mounted Instrumentation Guide Tubes

The Bottom Mounted Instrumentation (BMI) tubing analyses documented in the Turkey Point EPU LR Section 2.2.7 were evaluated for the change in RV support stiffness for resolution of RAI EMCB-2.4. The deadweight and thermal analyses of the BMI tubing are not impacted. The seismic and LOCA analyses are impacted because the RV seismic and LOCA displacements calculated as part of the RV internals dynamic analysis were revised due to the change in RV support stiffness. The BMI seismic and LOCA cases are reanalyzed with the revised RV displacements. Loads at the seal table and BMI supports are bounded by those previously provided for the EPU project. BMI tubing upset and faulted stresses provided in LR Table 2.2.7-1 were recalculated. There is a slight increase in upset stress; however all stresses remain below the allowable values. A revised LR Table 2.2.7-1 is provided below.

Table 2.2.7-1
Maximum BMI Thimble Tubing Stress Qualification

Loading Condition	**Loading Combination (Allowable)**	**Actual Stress (psi)**	**Reference 1 Code Allowable Stress (psi)**
Primary Stress	Pressure + Deadweight ($< 1.5S_h$)	[ ]^{a,c,e}	23,550
Thermal Expansion Stress	Thermal ($< S_A$**)	[ ]^{a,c,e}	23,550
Upset Condition Stress	Pressure + Deadweight +OBE ($< 1.8S_h$)	[ ]^{a,c,e}	28,260
Faulted Condition Stress	Pressure + Deadweight + $(SSE^2 + LOCA^2)^{1/2}$ ($< 2.4S_h$)	[ ]^{a,c,e}	37,680

**$S_A = 1.25S_c + 0.25\ S_h$ per Reference 6. Note also that $S_h = S_c$ at 120°F, resulting in $S_A = 1.5S_h$

2.8.1 Fuel System Design

The fuel Seismic/LOCA analysis was regenerated with revised core plate motions to account for the change in RV stiffness values. The Seismic/LOCA LR Section 2.8.1.2.3 was reviewed for any necessary changes. This review concluded that the description of crushed grid locations requires updating in LR Sections 2.8.1.2.3.1 and 2.8.1.2.3.5. The revised description of grid crush locations is presented below.

New core plate motions were developed for Turkey Point Units 3 and 4 and, subsequently, an updated Fuel Seismic/LOCA Analysis was performed. The analysis shows that all seismic/LOCA fuel criteria are met, for the EPU conditions, for a homogeneous core of Westinghouse 15x15 Upgrade fuel and for mixed cores with Westinghouse 15x15 Debris Resistant Fuel Assembly (DRFA) design fuel. However, some analysis details are different than those documented in the LR Section 2.8.1; specifically the locations of fuel assemblies containing predicted crushed grids. The analysis results show that the maximum impact force of 15x15 Upgrade fuel in the homogenous core and the mixed cores are below the allowable crush limit except for a few assemblies in the three-fuel-assembly and seven-fuel-assembly rows. The bounding configuration of fuel assemblies containing crushed grids occurs in the mixed core with the Upgrade design fuel at the ends of the rows and DRFA design fuel elsewhere. For this configuration, fuel assemblies at both ends of the 3-FA row and two (2) assemblies at one end of the 7-FA row experience 1 or 2 crushed grids per assembly. The two crushed FAs from the 3-FA row plus the two FAs from the 7-FA row and their half-core symmetric assemblies yield eight total assemblies that are predicted to contain crushed grids. Two of these eight locations are RCCA locations. The stress analysis results indicated that adequate margin for both fuel rod and thimble tube exists. Fragmentation of the fuel rods and thimble tubes will not occur. Since there was no thimble tube damage observed during the grid crush test and the stress analysis shows that no fractures occur for the thimble tube under the combined seismic and LOCA loads, both the RCCA insertability and a coolable geometry are maintained, as confirmed by LR Sections 2.8.5.6.3.2 and 2.8.5.6.3.3.

The results also show that the maximum impact force of the 15x15 DRFA fuel assemblies is below the allowable crush limit except a few assemblies in the three fuel assembly row for one of the two mixed core loading conditions. Considering the grid elevation mismatch between the 15x15 Upgrade fuel and 15x15 DRFA fuel, a stepped surface impact test and analysis were performed. The grid impact forces between the elevation mismatch grids were checked and found that no additional crushed grids were predicted during the seismic and LOCA events.

2.8.4.1 Functional Design of Control Rod Drive System

AREVA performed the necessary calculations to determine the impact of the revised RV support stiffness on the AREVA supplied IHA package. The results show that for all the components provided by AREVA, the design remains within code allowables and meets the current licensing basis.

In addition, AREVA reviewed EPU LR Section 2.8.4.1 and confirms that, with respect to the components provided by AREVA (RV Head, CRDMs, IHA), the conclusions presented in this section regarding these components remain valid. The analyses adequately address the effects of the EPU and meet all aspects of the current licensing bases.

2.8.5.6.3 Emergency Core Cooling System

2.8.5.6.3.2 Large Break LOCA

The grid crush results of the LOCA /seismic loads were reviewed for impact on the LBLOCA EPU analysis. The fuel assemblies for which grid crush is calculated to occur are located in the peripheral area of the core. These assemblies are confirmed to be at low power locations; therefore, core coolable geometry is maintained and the conclusions in LR Section 2.8.5.6.3.2.5 remain valid.

2.8.5.6.3.3 Small Break LOCA

The grid crush results of the LOCA /seismic loads were reviewed for impact on the SBLOCA EPU analysis. The fuel assemblies for which grid crush is calculated are located in the peripheral area of the core. These assemblies are confirmed to be at low power locations; therefore, any flow blockage induced penalty would be more than offset by the low power credit. Given this, and that all RCCAs were confirmed to be able to insert, the conclusions in LR Section 2.8.5.6.3.3.2 remain valid.

2.8.5.6.3.5 LOCA Forces

The LOCA Vessel Forces analyses were regenerated with revised core-barrel beam data to account for the change in RV stiffness values. LR Section 2.8.5.6.3.5 was reviewed for any necessary changes. This review concluded that no changes are needed and the text contained in LR Section 2.8.5.6.3.5 is still valid.

Conclusion

FPL has revised the RV support stress analysis and re-evaluated the effects of the proposed EPU on the structural integrity of the RV and its support structures, the RV head, CRDM, and IHA, the RV internals and core support structures, the RCL piping, the SGs and supports, the RCPs and supports, the PZR and supports, BMI guide tubes, and the nuclear fuel. FPL has adequately addressed the effects of the proposed EPU on all of these components and support structures. Further, FPL has demonstrated that these components and support structures will continue to meet the requirements of PTN's current licensing basis following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to the design of these components and support structures.

References

1. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2010-113), "License Amendment Request for Extended Power Uprate (LAR 205)," Accession No. ML103560169, October 21, 2010.
2. Email from J. Paige (NRC) to T. Abbatiello (FPL), "Turkey Point EPU – Mechanical and Civil (EMCB) Request for Additional Information – Round 1", Accession No. ML111090285, April 19, 2011.
3. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-140), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Mechanical and Civil Engineering Issues, May 19, 2011.
4. Email from J. Paige (NRC) to S. Hale (FPL), "Turkey Point EPU – Mechanical and Civil (EMCB) Request for Additional Information – Round 2", Accession No. ML11193A111, July 12, 2011.
5. M. Kiley (FPL) to U.S. Nuclear Regulatory Commission (L-2011-248), "Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205, August 25, 2011.
6. American Society of Mechanical Engineers (ASME), New York, Boiler and Pressure Vessel Code, Section III, NB and NC Subsections, 1989 Edition.

Turkey Point Units 3 and 4

SUPPLEMENTAL RESPONSE TO
NRC MECHANICAL AND CIVIL ENGINEERING BRANCH (EMCB)
REQUEST FOR ADDITIONAL INFORMATION REGARDING
EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST NO. 205

ATTACHMENT 3

Westinghouse & AREVA Affidavits
for Attachment 2

This coversheet plus 11 pages

Westinghouse

Westinghouse Electric Company
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USA

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Rockville, MD 20852

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Proj letter: FPL-11-261

CAW-11-3268

October 12, 2011

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: FPL-11-261 P-Attachment, "Turkey Point Units 3 and 4 – Response to NRC Request for Additional Information (RAI) from the Mechanical and Civil Branch (EMCB) Related to Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 (TAC Nos. ME 4907 and ME 4908)" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-11-3268 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

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

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-11-3268, 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,

BFMann for

J. A. Gresham, Manager
Regulatory Compliance

Enclosures

CAW-11-3268

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared B. F. Maurer, 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:

BFMaurer

B. F. Maurer, Manager
ABWR Licensing

Sworn to and subscribed before me
this 12th day of October 2011

Cynthia Olesky
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, ABWR Licensing, 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-11-261 P-Attachment, "Turkey Point Units 3 and 4 – Response to NRC Request for Additional Information (RAI) from the Mechanical and Civil Branch (EMCB) Related to Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 (TAC Nos. ME 4907 and ME 4908)" (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 proprietary information as submitted by Westinghouse for use by Turkey Point Units 3 and 4 is expected to be applicable for other licensee submittals in response to certain NRC requirements for Extended Power Uprate submittals and may be used only for that purpose.

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

(a) Provide input to the U.S. Nuclear Regulatory Commission for review of the Turkey Point EPU submittals.

(b) Provide revised stress values for equipment supports.

(c) Provide licensing support for customer submittals.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of the information to its customers for the purpose of meeting NRC requirements for licensing documentation associated with EPU submittals.

(b) Westinghouse can sell support and defense of the technology to its customer in licensing process.

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

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in Table 2.2.2.4-1 of Turkey Point Units 3 and 4 Licensing Report entitled "Primary Membrane Stress Intensity, Primary Membrane + Bending Stress Intensity, and Cumulative Usage Factor," Docket Nos. 50-250 and 50-251, referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secret and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 7th

day of October 2011.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/14
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2014