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

April 21, 2011

LICENSEE: Florida Power & Light Company

FACILITIES: Turkey Point, Units 3 and 4

SUBJECT: SUMMARY OF MARCH 31, 2011, PUBLIC MEETING FLORIDA POWER & LIGHT COMPANY, ON TURKEY POINT, UNITS 3 AND 4 EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST CURRENTLY UNDER REVIEW (TAC NOS. ME4907 AND ME4908)

On March 31, 2011, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) and representatives of Florida Power & Light Company (FPL, the licensee) at NRC Headquarters, One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The purpose of the meeting was to discuss the extended power uprate (EPU) license amendment request currently under review by the NRC staff. More specifically, the NRC and licensee discussed draft requests for additional information (RAIs) generated by the NRC technical staff to gain a common understanding of the questions. The draft RAIs discussed were generated by the Quality and Vendor (EQVB), Environmental Review (RERB), Health Physics and Human Performance (IHPB), Containment and Ventilation (SCVB), Mechanical and Civil Engineering (EMCB), and Reactor Systems (SRXB) branches. A list of attendees is provided as Enclosure 1.

The licensee provided a PowerPoint presentation prior to the meeting to help facilitate the discussion (Agencywide Document and Management System Accession No. ML110890726). The licensee presented on the amount of final and draft RAIs they have received from the NRC staff and the amount of responses the licensee has issued. Also, the NRC provided a copy of the draft RAIs to the licensee to help facilitate the meeting (see Enclosure 2).

The first RAIs discussed were generated by the EQVB and RERB branches. The NRC reviewers summarized their questions and provided clarification from the licensee asking questions. At the conclusion of the discussions, there was a common understating of the question, and it was agreed upon that the questions could be formally issued by the NRC and the licensee would provide its responses within 30 days of issuance. The next RAIs discussed were generated by the IHPB NRC reviewer. At the conclusion of the discussion, there was a common understanding of the questions but RAI 2IHPB-1.2 needed to be revised to be more specific and reflect the discussions during the meeting. Once the question is revised, it was agreed upon that the questions could be formally issued by the NRC and the licensee would provide its responses within 30 days of issuance.

The next set of questions discussed were from the SRXB NRC staff. Since the NRC provided the draft RAIs to the licensee a couple of days before the meeting, FPL did not have time to ask any specific questions regarding the RAIs. So, to get the rationale for the NRC reviewers generating the draft RAIs, the NRC summarized each of the draft RAIs and provided clarification when asked by the licensee. At the end of the discussion, it was concluded that the licensee would review the draft RAIs and determine if a follow-up draft RAI telephone call will be needed at a later date. Similar to the SRXB questions, the EMCB questions were summarized by the

NRC EMCB reviewer, and it was concluded that the licensee would review the questions in more detail to determine the need for a follow-up draft RAI call with the NRC staff at a later date.

The last group of questions discussed were generated by the NRC SCVB staff. These draft RAIs were originally sent to the licensee March 8, 2011, via email. The NRC and FPL held a teleconference on March 17, 2011, to discuss the draft RAIs. At the conclusion of the teleconference, it was determined that a followup call was needed to discuss draft RAI questions SCVB-1.10, SCVB-1.11, and SCVB-1.12. These questions relates to: 1) the error Westinghouse discovered in the computer code EPITOME used to generate the mass and energy release analysis for postulated loss-of-coolant accident; and 2) the resolution of net positive suction head (NPSH), as it relates to generic safety issue (GSI) 191, while the NRC staff is reviewing a EPU application. It was stated that FPL believes GSI-191 is a separate licensing action than the EPU, and is currently being resolved through Generic Letter 2004-02. During the meeting, it was concluded that draft questions, SCVB-1.11 and SCVB-1.12 regarding GSI 191 and NPSH should be withdrawn until the NRC staff finalizes guidance on how to resolve this issue while EPUs are under NRC staff review. Also, regarding question SCVB-1.10, FPL plans on providing a commitment in its response to provide a new analysis at a later date. Regarding the rest of the questions, including question SCVB-1.10, it was determined that there was a common understanding of the questions, and it was agreed that the questions could be formally issued by the NRC and the licensee would provide its responses within 30 days of issuance.

Members of the public were in attendance. After the meeting, a member of the public asked questions via email ranging from the number of participants from FPL and the NRC to what was meant by shine dose. Public Meeting Feedback forms were not received.

Please direct any inquiries to me at 301-415-5888, or Jason.Paige@nrc.gov.

Sincerely,

Jason C. Paige, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. List of Attendees
2. Draft Requests for Additional Information

cc w/encls: Distribution via ListServ

<u>LIST OF ATTENDEES</u>

<u>MARCH 31, 2011</u>

<u>MEETING WITH FLORIDA POWER & LIGHT COMPANY</u>

<u>EXTENDED POWER UPRATE LICENSE AMENDMENT REQUEST</u>

<u>U.S. Nuclear Regulatory Commission</u>

C. Clemons-Webb
N. Karipineni
B. Lee
D. Palmrose
S. Klementowicz
R. Pettis
D. Broaddus
C. Basavaraju
A. Sallman
R. Lobel
W. Lyon
B. Parks
U. Shoop
M. Khanna

<u>Florida Power & Light Co.</u>

J. Hoffman
L. Abbott
C. Wasik
P. Tiemann
C. O'Farrill
K. Rydman
S. Brain

<u>Westinghouse</u>

M. Watson

<u>Public</u>

M. Jesse
L. Gunderson
S. Salisbury

DRAFT Requests for Additional Information RE March 31, 2011, Public Meeting

Turkey Point, Units 3 and 4 Extended Power Uprate License Amendment Request

The below draft requests for additional information (RAIs) were generated while the Nuclear Regulatory Commission (NRC) staff was reviewing the Turkey Point Units 3 and 4 extended power uprate (EPU) license amendment request (LAR) dated October 21, 2010. The below draft RAIs will be used to help facilitate the discussions during the March 31, 2011, public meeting between the NRC and Florida Power & Light Co (FPL, the licensee). These RAIs are in draft form and at the conclusion of the public meeting, the RAIs could either be revised, deleted, remain the same, or added to. In addition to the below RAIs, the NRC and FPL will discuss the draft SCVB RAIs sent to the licensee March 8, 2011, via email.

Agenda

9:00 - 9:10am	Introductions
9:10 – 9:25am	EQVB Draft RAIs
9:25 – 9:40am	IHPB Draft RAIs
9:40 – 9:55am	RERB Draft RAIs
9:55 – 10:00am	BREAK
10:00 – 10:30am	SCVB follow-up discussion
10:30 – 11:30am	SRXB Draft RAIs
11:30am – 12:30pm	EMCB Draft RAIs
12:30 – 1pm	Questions

Quality and Vendor (EQVB) Draft RAI

EQVB-1.1 The licensee stated in the LAR that satisfactory post EPU industry operating experience has been demonstrated at greater than original power levels at two other pressurized-water reactors (PWRs) of similar design to Turkey Point (PTN). Section 2.12.1.2.2, "Background," of Attachment 4 to the LAR states, in part, that "In addition to Beaver Valley, Units 1 and 2, and the R.E. Ginna Nuclear Power Plant, PTN has benefited from industry operating experience in power uprate implementation from several industry sources, including the Institute of Nuclear Power Operations.

However, in Section 2.12.1.2.6.2, "Justification for Exception to Transient Testing," of Attachment 4, a discussion of such industry operating experience was not provided. Additionally, no discussion of any PTN plant-specific transient operating experience relative to operating events, planned and unplanned reactor trips, and overall plant transient performance was presented. Such information may be considered by the NRC staff, as discussed in Section III.C.2 of NRC Standard Review Plan 14.2.1, to support the basis for the licensee's request not to perform certain initial startup tests as part of the proposed EPU PATP. The licensee's primary basis for not performing certain transient testing as part of the proposed EPU LAR appears to rely solely on an analytical justification using LOFTRAN.

Health Physics (IHPB) Draft RAIs

2IHPB-1.1 Provide an estimate of the current shine dose that is references in Section 8.2.2 of Attachment 7 titled, "Offsite Doses at Power Uprate Conditions."

2IHPB-1.2 Provide an estimate of the impact on dose regarding the gas and liquid effluent levels.

2IHPB-1.3 In section 2.10.1, "Occupational and Public Radiation Doses," it states that the NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR 10. Provide additional information on the specific portions of 10 CFR 10 referenced in Section of the application.

Environmental (RERB) Draft RAI

RERB-1.1 The Supplemental Environmental Report (ER), Section 8.2.2, Offsite Doses at EPU conditions, discusses the projected offsite doses from the proposed power uprate. The ER states that the evaluation of the post-EPU total dose to an offsite member of the public includes the dose contribution from low-level radioactive waste (LLRW). Specifically, the ER's dose evaluation considered the following: radioactive decay of stored waste, stored waste being routinely moved offsite for disposal, and waste generated post-EPU entering storage.

The State of South Carolina's licensed low-level radioactive waste disposal facility, located in Barnwell, South Carolina, has limited access to radioactive waste generators in States that are not part of the Atlantic Low-Level Waste Compact. Since the State of Florida is not a member of the Atlantic Low-Level Waste Compact, the disposal of LLRW from Turkey Point Units 3 & 4 may be impacted. The ER's assumption that stored waste will be routinely moved offsite for disposal may need to be reevaluated in light of the Barnwell facility's limited access. However, the ER does not contain sufficient information about LLRW management, storage, and disposal capabilities at the Turkey Point Units 3& 4 for the NRC staff to determine whether the dose impact from the long-term storage of LLRW needs to be considered in the post-EPU dose evaluation.

Provide additional information on the management, storage, and disposal of LLRW as it relates to the potential impact on offsite doses to members of the public from the proposed EPU. The evaluation of the post-EPU total dose to an offsite member of the public should be revised as appropriate.

Reactor Systems (SRXB) Draft RAIs

SRXB-2.1 Provide plant piping configuration drawings that include scale three dimensional (3D) drawings or isometrics with dimensional information and piping and instrumentation diagrams (P&IDs) that describe the feedwater piping associated with the CheckPlus installation. Scaled 3D drawings comparable to Figures 1 and 2 in ER-748 Revision 1[1] that illustrate the Alden Laboratory test configurations are preferred in place of isometrics that cover the plant piping

[1] "Meter Factor Calculation and Accuracy Assessment for Turkey Point 3 Nuclear Power Plant," Engineering Report No. 748, Rev 1, Caldon Ultrasonics, June, 2010.

configuration comparable to the Alden Laboratory figures but the overall response to this RAI is for configuration information that describes piping, valves, flow straighteners (if any), feedwater flow meters, and any other components from the feedwater pumps to at least 10 pipe diameters downstream of each feedwater flow meter.

SRXB-2.2 Section 2.8.4.3 discusses reactor vessel neutron fluence calculations performed to support the pressure and temperature limits and cold over-pressure mitigation system setpoint confirmation. Describe how the fluence calculations account for uprated core operation.

SRXB-2.3 Assumption 2 in Section 2.8.4.4.2.2 states that the reactor coolant system (RCS) is assumed to be at a uniform temperature during residual heat removal (RHR) system operation and that there are no "hot spots" that could cause an unexpected increase in the bulk RCS temperature. Discuss why the uniform temperature assumption was made. Justify this assumption with respect to the core, upper plenum, upper head, and the pressurizer for both RCPs running and not running. Include a discussion of the implications where RCS temperature is not uniform and equal to average RCS temperature in the hot legs. Include drain down of RCS inventory and emptying the pressurizer in this discussion.

SRXB-2.4 Section 2.8.4.4.2.3 identifies a higher heat load that often extends cooldown time, yet, in the case of normal cooldown, time is 28 hours versus the lower heat load cooldown time of 30 hours. Explain since the expectation is that an increased heat generation rate would result in a longer cooldown time. The potential concern is that the analysis methodology used for normal cooldown may be applicable to other conditions and an understanding is necessary to assess predicted plant behavior for other conditions.

SRXB-2.5 Sections 2.8.4.4.2.3 and 2.8.4.4.2.5 state that there are no safety-related design criteria for normal plant cooldown times and, therefore, the calculated cooldown times are acceptable. Yet Section 2.8.4.4.2.5 also states a Technical Specification (TS) time limit for achieving cold shutdown of 6 + 30 = 36 hours after reactor shutdown and states that the 28 hours applies. Explain the rationale for concluding that calculated cooldown times are acceptable since there are stated to be no safety-related design criteria in contrast to using these results for meeting TS and other cooldown criteria.

SRXB-2.6 There appears to be no information that addresses the effect of the EPU on heat exchanger fouling factors. Address the behavior of heat exchanger fouling factors due to the higher heat load, longer cooldown times, and greater differential temperatures.

SRXB-2.7 The end of Section 2.8.4.4.2.3 states "The EPU has no effect on the ability of the RHR system to remove residual heat at reduced reactor coolant system inventory, and therefore the PTN (Turkey Point Nuclear Power Plant) will continue to meet the current licensing basis requirements with respect to NRC Generic Letter 88-17." Justify this conclusion in light of the increased decay heat generation rate that must be removed after shutdown. Include the effect on temperature, RHR flow rate including any limitations on flow rate as a function of RCS water level, and potential hot leg vortexing in your justification.

SRXB-2.8 Section 2.8.7.1 shows "deleted." Reduced inventory operation is one of the most risk-significant per unit time conditions where the plant is operated. Consequently, the NRC staff expects in-depth information similar to that provided in the Point Beach Units 1 and 2 EPU Licensing Report that addresses "Loss of Residual Heat Removal at Reduced Inventory."[2] Provide this information or provide readily available references that provide the information.

SRXB-2.9 Section 2.8.7.2 states that a minimum subcooling margin of 50˚F is maintained during natural circulation cooldown until RCS temperature is below 350˚F. What temperature sensors are used to determine RCS temperature and where are they located within the RCS? If hot leg temperatures are used, address the distribution of temperature that is expected in the hot legs. What is used to determine maximum upper head temperature during natural circulation cooldown? Compare the upper head temperatures predicted to exist during natural circulation cooldown for the existing power level and the proposed power level. Include saturation temperature at the uppermost upper head elevation in the comparison.

SRXB-2.10 Pages 2.8.7.2-3 and -4 state that "As indicated in UFSAR Section 9.11, the Condensate Storage Tanks were originally designed to have sufficient storage capacity after loss of offsite power to support AFW System operation required to maintain the plant at Hot Standby conditions for a 15-hour period followed by a 4-hour cooldown to Residual Heat Removal (RHR) System entry conditions (350°F). However, based on the current sizing criteria and administrative procedure limitations for cooldown rate (25°F/hr), the maximum time that the plant can remain at Hot Standby is currently 12 hours if a natural circulation cooldown for RHR System operation is required.." The wording implies that these values do not apply to EPU conditions. Is this correct? If so, what are the EPU values?

SRXB-2.11 The 4 hour cooldown to 350˚F implies a start of cooldown from 450˚F. What are the Turkey Point temperatures associated with hot standby and how do these compare to the 450˚F temperature? For example, Section 2.8.7.2.2.5 states that "After reactor trip Tavg decreases to approximately 560°F and stays relatively constant during the hot standby period." Is this representative and, if so, how is it consistent with 450˚F? What reasonably ensures that the condensate tank inventory is sufficient to address cooldown given the response to the above questions?

SRXB-2.12 Section 2.8.7.2.2.1 states "The design and licensing basis in UFSAR Section 9.11 and TS 3/4.7.1.3 Bases will be revised to reflect CST (condensate storage tank) storage capacity capable of supporting a 4 hour hold at hot standby conditions followed by a 9-hour natural circulation cooldown with at least one CRDM fan in operation to the RHR cut-in conditions consistent with NRC recommendations contained in the Standard Review Plan (SRP) Branch Technical Position (BTP) 5-4. This evaluation will demonstrate the ability to cool

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[2] Contained in material submitted with "License Amendment Request 261, Extended Power Uprate," FPL Energy Point Beach, LLC, ML091250564, April 7, 2009.

down the plant on natural circulation to RHR cut-in conditions (PRCS < 450 psig and Tavg< 350°F)." Address the various times identified in RAIs 9, 10, and 11 with respect to expected cooldown behavior should a reactor trip occur due to loss of offsite power and provide information to confirm that the times can be achieved. Include the Section 2.8.7.2.3 statement that "RHR cut-in conditions can be achieved in ~13 hours" in the discussion.

SRXB-2.13 Provide representative RCS natural circulation temperatures that have been observed during operation of the Turkey Point plant and compare these to selected natural circulation temperatures provided in Section 2.8.7.2.2.3.

SRXB-2.14 Is the Table 2.8.7.2-1 620.8°F core outlet temperature an average value or the peak value located immediately above the hottest region of the core?

Mechanical and Civil (EMCB) Draft RAIs

EMCB-1.1 In section 1.0.4.5 of the licensing report (LR), it is mentioned that a supplemental heat exchanger will be added to maintain the design limits for spent fuel pool (SFP) cooling system for EPU conditions. The licensee is requested to address if any new piping additions are modifications to existing piping and provide a summary of the piping stresses and support qualification results along with margins as applicable.

EMCB-1.2 Section 1.0.4.6: The licensee is requested to clarify if there are any increases in loadings to emergency diesel generator (EDG) fuel supply and return lines, and jacket water supply and cooling lines due to increases in EDG loading from EPU.

EMCB-1.3 In section 2.6.5.2.1 it was mentioned that EPU increases the heat available for release to the containment which increases the subsequent loads on the containment heat removal. Systems. The licensee is requested to clarify if the normal containment coolers (NCC) will be replaced with new larger capacity cooling units and address if these units are seismically qualified and the supporting system for the cooling units redesign. It is requested to provide a summary of the results, if applicable.

EMCB-1.4 In section 2.2.1, the licensee is requested to clarify if the criteria for pipe rupture locations and associated dynamic effects for EPU considerations are the same as those of the current licensing basis (CLB). The licensee is also requested to clarify if the piping stresses utilized in break postulations include any local stresses from integral pipe attachments and occasional or upset loadings such as steam hammer or water hammer when applicable.

EMCB-1.5 In section 2.2.2.1.2.3 it was mentioned that the maximum potential earthquake (MPE) seismic analyses performed for the reactor coolant loop (RCL) piping considered various primary equipment support activity. The licensee is requested to describe what is meant by support activity.

EMCB-1.6 Table 2.2.2.1.1: The licensee is requested to add a note to describe how the MPE and loss of coolant accident (LOCA) results are combined for RCL (hot leg, cold leg, cross over leg) piping stresses.

EMCB-1.7 In section 2.2.2.1 under the current licensing basis it was described that class I, class III, and non-safety piping were designed to ASA B31.1-1955 with the exception of the pressurizer surge lines. It was also mentioned that the associated piping supports were designed to the requirements of AISC Manual of steel construction 1963 edition. However in section 2.2.2.2.2.1 for EPU, the balance of plant (BOP) piping was evaluated based on ANSI B31.1 - 1973 edition through winter 1976 addenda, and the supports were evaluated based on AISC Manual of Steel Construction 8th edition. The licensee is requested to address if a code reconciliation was performed for the differences between B31.1-1955 and B31.1-1973-1976, and AISC Manual 1963 edition and the 8th edition.

EMCB-1.8 In section 2.2.2.2.2.3: (a) It is mentioned that plant walkdowns were performed on portions of the BOP piping systems to review the piping layouts and support configurations to assess the adequacy of the dead weight spans and thermal flexibility. The licensee is requested to clarify if the above mentioned walkdowns are for non-analyzed or cook book supported or field routed small bore piping; (b) Five different computer programs, namely NUPIPE-SWPC, PC-PREPS, STEHAM-PC, WATHAM-PC, and ANSYS/Mechanical not currently described in the UFSAR were used to perform the EPU piping stress evaluation, piping welded attachment stress evaluation, and for the generation of fluid transient forcing functions. The licensee is requested to describe whether these programs are widely used in the nuclear industry for similar applications. The licensee is requested to address which one of the above computer programs was utilized for piping welded attachment local stress evaluation.

EMCB-1.9 Section 2.2.2.2.2.5 does not address the impact of EPU on containment penetration anchor qualification for any increase in loads due to thermal expansion, water hammer, steam hammer from temperature and flow rate increases as applicable in the affected systems such as main steam, feed water, and component cooling water. The licensee is requested to address the qualification of containment penetration anchors as applicable.

EMCB-1.10 Table 2.2.2.2-1: (a) The licensee is requested to clarify if the stresses due to steam hammer fluid transient (FT) loading for the main steam line are based on a time history analysis; (b) In the second column of the table under stress combination or as a note to the table, the licensee is requested to provide the applicable allowable stress criteria (for example: similar to those allowable limits shown in column 1 of Table 2.2.2.1-1); (c) In the second column of the table there are several rows showing stress combination G+E (for stresses due to gravity G and thermal expansion E) and not P+G+E, and it is not clear why the pressure stress (P) is not in this pipe stress combination per the applicable code. The licensee is requested to explain this discrepancy; (d) The licensee is requested to clarify the allowable stresses and its reference for the pipe stress combinations G+P+ SRSS (OBE,FT) , G+P+ SRSS (SSEE,FT), E; G+E. The licensee is requested to verify whether the allowable stress values listed in the 5th column of the table are in accordance with the criteria and the code specified limits. (e) The licensee is requested to clarify if there is no fluid transient (FT) due to water hammer for feedwater system; and (f) In column 3 of the table for component cooling problems SP-016 & SP-017, note #3 was mentioned for G+E combination. The licensee is requested to address if this note is really applicable here for the stress combination.

EMCB-1.11 Table 2.2.2.2-2: (a). The licensee is requested to clarify if the stresses due to steam hammer fluid transient (FT) loading for the main steam line are based on a time history analysis; (b) In the second column of the table under stress combination or as a note to the table, the licensee is requested to provide the applicable allowable stress criteria (for example: similar to those allowable limits shown in column 1 of Table 2.2.2.2-2); (c) In the second column of the table there are several rows showing stress combination G+E (for stresses due to gravity G and thermal expansion E) and not P+G+E, and it is not clear why the pressure stress (P) is not in this pipe stress combination per the applicable code. The licensee is requested to explain this discrepancy; (d) The licensee is requested to clarify the allowable stresses and its reference for the pipe stress combinations G+P+ SRSS (OBE,FT), G+P+ SRSS (SSEE,FT), E; G+E, & P+G+E. The licensee is requested to verify whether the allowable stress values listed in the 5th column of the table are in accordance with the criteria and the code specified limits; (e) The licensee is requested to clarify if there is no fluid transient (FT) due to water hammer for feedwater system; and (f) In column 3 of the table for component cooling problems CCW-18, note #3 was mentioned for G+E combination. The licensee is requested to address if this note is really applicable here for the stress combination.

EMCB-1.12 Section 2.2.2.3.2.1, Reactor Vessel (RV) and Supports: It is mentioned that four of the RV supports have one cantilever beam skewed to clear ex-core detectors in the reactor cavity concrete. The licensee is requested to clarify the total number of RV supports present.

EMCB-1.13 The LR mentions various editions of the following Codes:

ASME Boiler and Pressure Vessel (B&PV) Code , Section III 2004 Edition Appendix-F,
ASME B&PV Code, Section III 1965 Edition – summer 1965 addenda,
ASME B&PV Code , Section III 1971 Edition
ASME B&PV Code , Section III 1974 Edition through Summer 1976 Addenda,
ASME B&PV Code, Section III 1986 Edition.
ASME B&PV Code, Section III 1989 Edition.
ASME B&PV Code, Section III 1998 Edition through 2000 Addenda.
ASME B&PV Code, Section III 1998 Edition, Subsection NF
American Standards Association (ANSI) B31.1-1973 Edition through Winter 1976 Addenda
American National Standards (ASA) B31.1-1955 Edition
AISC Manual of Steel Construction -1963 Edition
AISC Manual of Steel Construction -8th Edition

The licensee is requested to provide a summary table listing the various editions of B31.1 Code, ASME B&PV Code Section III, AISC Manual of Steel Construction used for the evaluation of piping systems and pipe supports in the design basis code of record and for specific portion of the EPU application. The licensee is also requested to clarify whether the various editions of the codes used for EPU evaluations are reconciled with the codes of record as applicable.

EMCB-1.14 Section 2.2.2.3.2.2, Reactor Vessel (RV) and Supports: The licensee is requesting NRC approval for the use of Appendix-F of the ASME B&PV Code 2004 Edition in evaluating Normal plus LOCA load combination. The licensee is requested to provide information to clarify what Code was used by the licensee previously for this Normal plus LOCA load combination along with the corresponding computed stresses and the allowable stresses.

EMCB-1.15 Table 2.2.2.3-1: The licensee is requested to provide a clarification note for those cases with EPU values lower than the current values explaining the reason; (b) The licensee is also requested to provide a note whether the cumulative fatigue usage factor for the inlet and outlet nozzles for EPU is based on 40 years or 60 years life; (c) The licensee is also requested to provide a note on the value of the Ke factor used in the computation of the cumulative fatigue usage factor for the bottom mounted instrumentation nozzles for EPU; and (d) The licensee is also requested to clarify the cumulative fatigue usage factor of 0.478 for EPU, for shell at core support pads is conservatively used instead of the values shown under note 3.

EMCB-1.16 Table 2.2.2.3-2, Bottom mounted instrumentation nozzle simplified elastic plastic results: The licensee is requested to provide a note showing the magnitudes of the thermal bending stress and the total bending stress.

EMCB-1.17 Table 2.2.2.3-3, Existing Design basis RV Support Loads: For EPU conditions (Table 2.2.2.-3-4), RV Support loads are shown separately for Inlet and Outlet, while for existing design basis the inlet and outlet are not shown. The licensee is requested to clarify whether the existing design basis loads represent bounding loads for inlet and outlet.

EMCB-1.18 Table 2.2.2.3-4, Revised Design basis RV Support Loads at EPU: Note 1 to this table is not clear. (a) The licensee is requested to clarify whether the revised loads E (Design Earthquake), E' (Maximum hypothetical Earthquake), and R (LOCA) include loads from D (dead weight), T (Normal thermal expansion), and L (Live load), and (b) The licensee is also requested to clarify why the E' load for EPU is smaller than E' for existing design basis.

EMCB-1.19 Table 2.2.2.3-6: The licensee is requested to clarify how the allowable loads in this table are obtained.

EMCB-1.20 Table 2.2.2.4-1: This table provides only cumulative fatigue usage factors. The licensee is requested to provide a summary of primary, primary plus secondary stresses for the control rod drive mechanism (CRDM).

EMCB-1.21 Section 2.2.2.5.2.3: The licensee is requested to discuss whether thermal stratification is significant for this steam generator's feedwater nozzle

EMCB-1.22 Section 2.2.2.5.5.2: The licensee is requested to provide a simple summary of the basis for the assumptions regarding the fluid elastic instability, turbulence, and wear.

EMCB-1.23 Section 2.2.2.5.5.5: (a) The licensee is requested to provide the maximum cross flow velocity for the steam generator tube bend region and the calculated critical velocity of the fluid to initiate fluid elastic instability; (b) The licensee is requested to briefly explain how FIV stress of 0.45 ksi was obtained; and (c) The licensee is also requested to clarify if the drilled holes in stainless steel support plates for steam generator tubes broached or not.

EMCB-1.24 Section 2.2.2.5.5.5: The LR did not address flow induced vibration (FIV), and the potential for acoustic resonance due to standing waves in any stagnant side branches in main steam and feed water lines, and the potential to generate loose parts impacting any safety related components. The licensee is requested to address the above items.

EMCB-1.25 Table 2.2.2.5-3: Case 2 in this table addresses pin connection at tube sheet to channel head joint. This connection is more like weld connection or a moment connection rather than a pinned connection. The licensee is requested to provide a justification for the validity or applicability of Case 2 treating the connection as pinned connection.

EMCB-1.26 Table 2.2.2.5-4: This table provides maximum/minimum in-plane tube bending stresses. The licensee is requested to also provide the corresponding allowable stress limits or margins.

EMCB-1.27 Table 2.2.2.5-6: This table provides integrated tube support plate loads. The licensee is requested to briefly explain how these loads are related to those provided in table 2.2.2.5-5

EMCB-1.28 Table 2.2.2.5-7: The title of this table is steam generator support member stresses, however, the table contains faulted actual and allowable loads. (a) Clarify if these are loads or stresses, and if there is a separate table for stresses and (b) The licensee is also requested to briefly explain how the allowable loads are obtained.

EMCB-1.29 Section 2.2.2.6: The end of this section does not contain the list of references. The licensee is requested to include any applicable references.

EMCB-1.30 Table 2.2.2.6-1: (a) For comparison purposes, the licensee is requested to show the analysis of record (AOR) values of the stresses and usage factors; and (b) The third column of this table for casing points to note 1 do not seem to be applicable. The licensee is also requested to provide the usage factor value for the casing for EPU.

EMCB-1.31 Table 2.2.2.6-2: The title of this table is reactor coolant pump support member stresses, however, the table contains faulted actual and allowable loads. (a) Clarify if these are loads or stresses, and if there is a separate table for stresses; and (b) The licensee is also requested to briefly explain how the allowable loads are obtained.

EMCB-1.32 Table 2.2.2.7-1: The licensee is requested to clarify with a note to explain why the fatigue usage values for spray nozzle, upper head, surge nozzle, and safety-relief nozzle for EPU are smaller than current or pre-EPU values.

EMCB-1.33 Table 2.2.2.7-2: The licensee is requested to clarify with a note providing the primary plus secondary stress intensity range including thermal bending, numerical value of the Ke factor for the spray nozzle. The licensee is also requested to clarify whether a check for thermal ratcheting was performed for the spray nozzle.

EMCB-1.34 Table 2.2.3-1: (a) The licensee is requested to clarify why the EPU values decreased significantly compared to the current AOR values for the upper support columns; and (b) The licensee is also requested to provide the primary plus secondary stress intensity range including thermal bending, thermal bending stress, numerical value of the Ke factor, ratio pf yield to ultimate strength, and thermal ratcheting evaluation as applicable for upper core plate alignment pins, lower support plate & weld, and outlet nozzle.

EMCB-1.35 Section 2.2.7.2.2: The licensee is requested to provide the following information regarding the bottom mounted instrumentation (BMI): (a) Is the 2 inches interface for the change of temperature from 547°F to 120°F based on calculation of thermal attenuation distance or actual measurement; (b) Is the BMI guide tubing insulated or not; and (c) Are the RV connections qualified for the loads from BMI guide tubing?

EMCB-1.36 Table 2.2.7-1: (a) In the second column of the table under loading combination or as a note to the table, the licensee is requested to provide the applicable allowable stress criteria (for example: similar to those allowable limits shown in column 1 of Table 2.2.2.1-1); (b) The licensee is requested to clarify if this table is for EPU conditions; (c) The licensee is requested to supplement this table with the current AOR results; and (d) The licensee is requested to explain why the allowable stress of 23550 psi is the same for primary stress case and thermal expansion case.

EMCB-1.37 The licensee is requested to provide a summary of BMI guide tubing support qualification.

EMCB-1.38 Table 2.5.5.1-2: (a) The licensee is requested to clarify why the EPU values decreased significantly compared to the current AOR values for the upper support columns; and (b) The licensee is also requested to provide the primary plus secondary stress intensity range including thermal bending, thermal bending stress, numerical value of the Ke factor, ratio pf yield to ultimate strength, and thermal ratcheting evaluation as applicable for upper core plate alignment pins, lower support plate & weld, and outlet nozzle.

EMCB-1.39 Table 2.5.5.1-2: This table summarized and showed that the EPU steam flow velocities are exceeding the industry guidance. The licensee is requested to briefly explain the significance of these EPU flow velocities and the impact of these values on flow induced vibration, acoustic resonance due to any side branch stand pipes, and any loose parts that could potentially affect safety related components.

NRC EMCB reviewer, and it was concluded that the licensee would review the questions in more detail to determine the need for a follow-up draft RAI call with the NRC staff at a later date.

The last group of questions discussed were generated by the NRC SCVB staff. These draft RAIs were originally sent to the licensee March 8, 2011, via email. The NRC and FPL held a teleconference on March 17, 2011, to discuss the draft RAIs. At the conclusion of the teleconference, it was determined that a followup call was needed to discuss draft RAI questions SCVB-1.10, SCVB-1.11, and SCVB-1.12. These questions relates to: 1) the error Westinghouse discovered in the computer code EPITOME used to generate the mass and energy release analysis for postulated loss-of-coolant accident; and 2) the resolution of net positive suction head (NPSH), as it relates to generic safety issue (GSI) 191, while the NRC staff is reviewing a EPU application. It was stated that FPL believes GSI-191 is a separate licensing action than the EPU, and is currently being resolved through Generic Letter 2004-02. During the meeting, it was concluded that draft questions, SCVB-1.11 and SCVB-1.12 regarding GSI 191 and NPSH should be withdrawn until the NRC staff finalizes guidance on how to resolve this issue while EPUs are under NRC staff review. Also, regarding question SCVB-1.10, FPL plans on providing a commitment in its response to provide a new analysis at a later date. Regarding the rest of the questions, including question SCVB-1.10, it was determined that there was a common understanding of the questions, and it was agreed that the questions could be formally issued by the NRC and the licensee would provide its responses within 30 days of issuance.

Members of the public were in attendance. After the meeting, a member of the public asked questions via email ranging from the number of participants from FPL and the NRC to what was meant by shine dose. Public Meeting Feedback forms were not received.

Please direct any inquiries to me at 301-415-5888, or Jason.Paige@nrc.gov.

Sincerely,

/RA/

Jason C. Paige, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. List of Attendees
2. Draft Requests for Additional Information

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