NEXTera ENERGY
POINT BEACH

December 15, 2010

NRC 2010-0186
10 CFR 50.90

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

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Amendment Request 261
Extended Power Uprate
Response to Request for Clarification

References:
(1) FPL Energy Point Beach, LLC letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)
(2) NRC electronic mail to NextEra Energy Point Beach, LLC, dated August 26, 2010, Point Beach Nuclear Plant, Units 1 and 2–Requests for Additional Information Associated with Extended Power Uprate (TAC Nos. ME1044 and ME1045) (ML102440095)
(3) NextEra Energy Point Beach, LLC, letter to NRC, dated September 8, 2010, License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information (ML102520325)
(4) NRC electronic mail to NextEra Energy Point Beach, LLC, dated November 18, 2010, Point Beach Nuclear Plant, Units 1 and 2 - Comments to NextEra's Request for Additional Information Responses (ML103330408)

NextEra Energy Point Beach, LLC (NextEra) submitted License Amendment Request (LAR) 261 (Reference 1) to the NRC pursuant to 10 CFR 50.90. The proposed amendment would increase each unit's licensed thermal power level from 1540 megawatts thermal (MWt) to 1800 MWt, and revise the Technical Specifications to support operation at the increased thermal power level.

Via Reference (2), the NRC staff determined that additional information was required to enable the staffs continued review of the request. NextEra responded to the request for additional information in a letter dated September 8, 2010 (Reference 3). Via Reference (4), the NRC determined that additional clarification was needed for NextEra's Reference (3) response.

Enclosure 1 provides the NextEra response to the NRC staffs request for clarification. Clarification of the NextEra response to EMCB RAI 1 will be submitted in separate correspondence.

This letter contains no new Regulatory Commitments and no revisions to existing Regulatory Commitments.

The information contained in this letter does not alter the no significant hazards consideration contained in Reference (1) and continues to satisfy the criteria of 10 CFR 51.22 for categorical exclusion from the requirements of an environmental assessment.

In accordance with 10 CFR 50.91, a copy of this letter is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 15, 2010.

Very truly yours,

NextEra Energy Point Beach, LLC

Larry Meyer
Site Vice President

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE 1

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT REQUEST 261
EXTENDED POWER UPRATE
RESPONSE TO REQUEST FOR CLARIFICATION

The NRC staff determined that additional information was required (Reference 1) to enable the Mechanical and Civil Engineering Branch to complete the review of License Amendment Request (LAR) 261, Extended Power Uprate (EPU) (Reference 2). NextEra responded to the request for additional information in a letter dated September 8, 2010 (Reference 3). Via Reference (4), the NRC determined that additional clarification of NextEra's Reference (3) response was required. The following information is provided by NextEra in response to the NRC staff's request.

Response to EMCB RAI 7 (pg 15/41)

Response's letter Attachment 2 provides response to RAI 7 in reference to FW PMP Nozzle loads. It is not clear why calculated nozzle loads have been compared at some cases against generic API allowables and at other cases against vendor allowables. Why not use vendor supplied pump specific allowable nozzle loads for all cases? By examining the tables in attachment 2 it is indicated that comparison to API has been made where calculated loads exceeded the vendor allowables. Has the vendor been informed and been supplied with calculated pipe loads acting on its pump nozzles and has the vendor provide the plant with documentation which shows that the calculated loads that exceed the vendor allowable values (regardless whether or not meet the generic API allowable values) are acceptable for the structural integrity of the pump (flanges and flange bolting, casing, pump supports and hold-down anchors, etc) and acceptable for its intended function? (It is noted that WNES has approved such a case for one of the SG MS nozzles). It is noted that in one case a pipe induced FW pump nozzle load exceeded the vendor allowable by approximately 62%. What controlled documentation (such as calculations) exist that accept nozzle loads which have exceeded the vendor allowables?

NextEra Response

The main feed water pump and connecting piping are non-safety related ASME Class III components. The manufacturer has affirmed that the pump structural integrity is not challenged by any of the transient loading conditions.

The only acceptance criterion for static conditions is that applied moments not result in excessive displacement of the pump shaft. The only loading inputs used in determining the acceptability of applied moments under static conditions are the moments M_y and M_z applied to the suction and discharge nozzles. Therefore, the "Vendor Allowable" loads (F_x, F_y, and F_z) and moment about the axis of the shaft (M_x) in the previously provided responses are not applicable acceptance criteria for these loading cases.

Response to EMCB RAI 11 (pg 19/41)

a) *Is the 250 deltaT the greatest deltaT that can be experienced from all design transients?*
b) *From the explanation provided the primary plus secondary stress intensity value (Sn) for the outlet Rx Nozzle support pad should be 68ksi and not 67ksi (the difference though is insignificant).*
c) *Looking at results from other PWRs, the primary plus secondary Sn of the Inlet nozzle pad is greater than the Sn of the outlet nozzle pad. Therefore, it would be prudent to provide the range of primary plus secondary Sn values for the Inlet and Outlet nozzle support pads.*

NextEra Response

a) The 250°F ΔT is based on a reactor vessel outlet nozzle inside diameter temperature of 650°F and support shoe operating temperature of 400°F. The reactor vessel outlet nozzle normal water temperature (T_{hot}) for 100% EPU power conditions is 611.1°F. The 250°F ΔT would be a conservative maximum ΔT between the outlet nozzles and support shoes for all normal and upset transient conditions originating from 100% EPU power operation. The largest transient increase in the reactor vessel outlet nozzle water temperature (T_{hot}) is approximately 30°F for the loss-of-load transient. Thus, the largest transient increase in the reactor vessel outlet nozzle water temperature would be approximately 641°F, which is below the 650°F assumption. In the original NextEra response to this request for additional information, a primary plus secondary stress intensity range of 67 ksi was calculated using the 250°F ΔT. The original response also noted that the two transients producing this range were steam pipe break and reactor heatup with a concurrent no loss-of-function seismic event. The steam pipe break is a faulted condition transient, which is not required to be evaluated for primary plus secondary stress intensity range per the ASME Code. Therefore, the 67 ksi value using the 250°F ΔT is conservative.

For reactor heatup, the original stress reports list a ΔT of 341°F between the outlet nozzles and support shoes. The largest total stress range in the original stress reports occurs on the inside diameter of the outlet nozzle. The maximum total stress range is created from two transients: steam pipe break and reactor heatup with a concurrent no loss-of-function seismic event. As previously mentioned, the steam pipe break transient is a faulted condition transient, which is not required to be evaluated for primary plus secondary stress intensity range per the ASME Code. Therefore, the limiting total stress intensity range comes from two transients: primary side hydrostatic test and reactor heatup with a concurrent no loss-of-function seismic event.

The linear thermal gradient hoop and axial stress due to the ΔT of 341°F during reactor heatup is approximated by the equation:

$$\sigma_\theta = \sigma_x = \pm \frac{E\alpha\Delta T}{2(1-\upsilon)}$$

where:

E	= 26.05 x 10^6 psi
α	= 7.33 x 10^{-6} in/in/°F
ΔT	= 341°F
ν	= 0.3

The maximum linear thermal gradient stress would be approximately -47 ksi during reactor heatup including the no loss-of-function seismic event. Note that the reactor heatup transient did not change for the EPU and any changes to the no loss-of-function seismic event compressive stresses in the nozzle support pads would minimally impact the primary plus secondary stress intensity range since the thermal stress dominates this quantity. Recombining the given pressure, linear thermal gradient and external load stresses results in a primary plus secondary stress intensity range of approximately 62 ksi which is well under the 80.1 ksi 3Sm limit.

b) NextEra concurs with this statement. The 67 ksi stress intensity was calculated neglecting shear. When shear is included, the primary plus secondary stress intensity range for the outlet nozzle would be 68 ksi.

c) For all normal and upset transient conditions originating from 100% EPU power operation, the 250°F ΔT approximation is conservative considering the inlet nozzle normal water temperature (T_{cold}) for 100% EPU power conditions is a maximum of 542.9°F. Therefore, the thermal stress calculated for the outlet nozzle for normal and upset transient conditions originating from 100% EPU power operation would bound the inlet nozzle thermal stress. The inlet nozzle pressure and external load stresses are very similar to those for the outlet nozzle. Therefore, using the methodology presented in the original NextEra response to this request for additional information, the outlet nozzle primary plus secondary stress intensity range would also bound the inlet nozzle.

The original stress reports calculate a ΔT of 341°F between the inlet nozzles and support shoes during reactor heatup. As mentioned, the inlet nozzle pressure and external load stresses are very similar to those for the outlet nozzle. Therefore, using the methodology presented in (a) above, the outlet nozzle primary plus secondary stress intensity range considering the 341°F ΔT would bound the inlet nozzle.

Response to EMCB RAI 14 (pg 2/12)

The geometry that the SCF of 1.5 vs. 3.27 applies to is not clearly understood as well as whether a weld in that location is involved.

NextEra Response

The geometry of the external support brackets is shown in Figure 14-1. The external support brackets consist of two gusset plates and a support plate. The location where the 1.5 SCF has been applied is indicated in Figure 14-1. The external support brackets are attached to the vessel wall using a full penetration, double groove weld configuration as shown in Figure 14-2. The support plate has 45° grooves machined on each side. One side of the support plate is welded to the vessel. Machining is performed from the remaining groove until sound metal and weld is reached, and then the other side is welded. Finally, the fillet between the horizontal surfaces of the support plate to vessel wall are final machined to 1.125 inches, and the fillet between the vertical surfaces of the support plate to vessel wall are final machined to 2.0 inches. The solid model in Figure 14-1 does not include the blending of the 1.125 inch and 2.0 inch fillets at the corners.

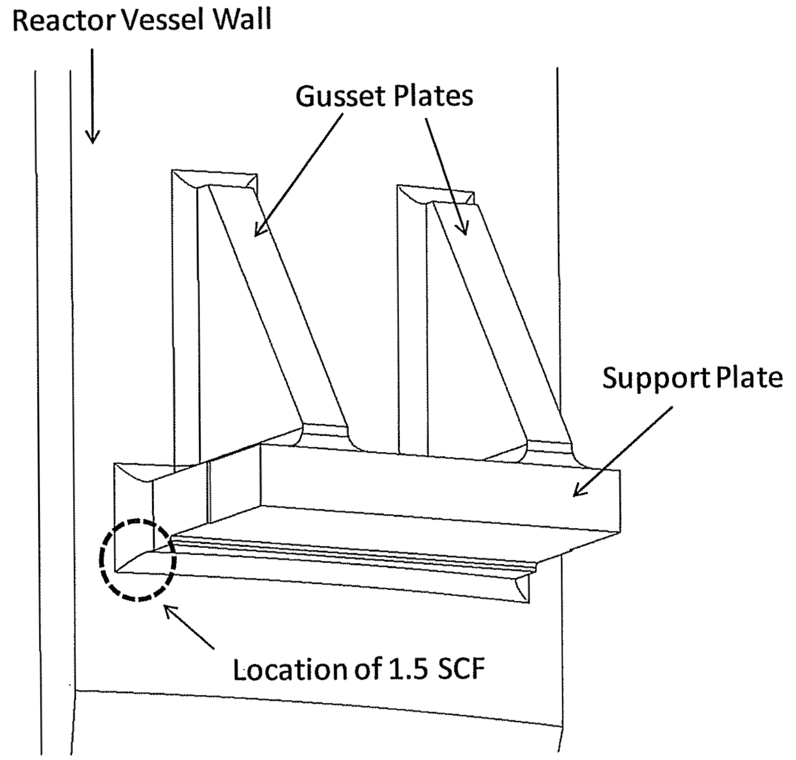

Figure 14-1: Point Beach Units 1 and 2 External Support Bracket Geometry

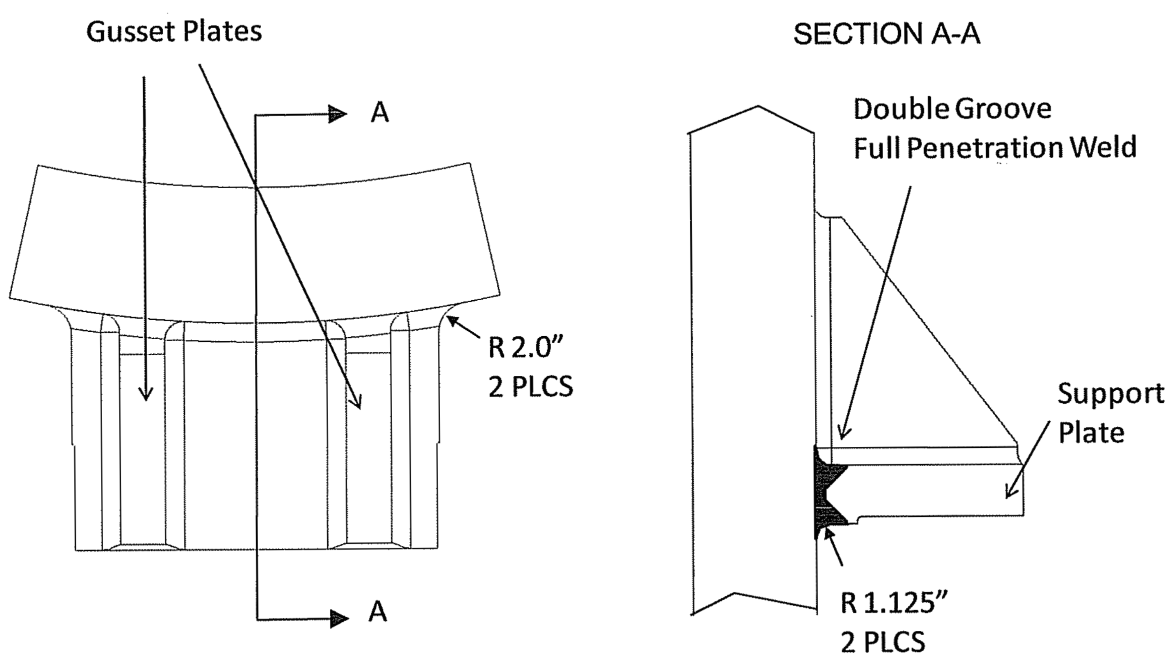

Figure 14-2: Point Beach Units 1 and 2 External Support Bracket Support Plate Weld Configuration

Response to EMCB RAI 15 (pg 23/41)

The RAI requested to verify whether or not the fatigue CUF values shown on Table 2.2.2.3-3, in reference to RPV components, reflect effects from environmentally assisted fatigue and If not to provide a justification. The response provides a short explanation and concludes that:

> *"[It] was not necessary to include the effects from environmentally assisted fatigue into the CUF values shown on LAR 261, Attachment 5, Table 2.2.2.3-3."*

EPU LR (LAR 261, Attachment 5) in reference to RPV components states that:

> *For the reactor vessel components determined to be potentially impacted by environmental fatigue, the environmental effects on fatigue were evaluated based on the updated fatigue usage factors determined from the EPU evaluations. The cumulative fatigue usage factors for the inlet nozzle, outlet nozzle, and bottom-head-to-shell juncture, with environmentally-assisted fatigue factors applied, are still below the ASME code limit of 1.0.*

The above two statements are not consistent. Has environmentally assisted fatigue been incorporated in the licensing renewal of the Point Beach Nuclear Plant, Units 1 and 2, which was approved by the NRC on December 2005?

NextEra Response

Environmentally assisted fatigue has been incorporated into the license renewal of PBNP, which was approved by the NRC in December 2005 (Reference 5).

Response to EMCB RAI 20 (pg 28/41)

RAI requested to discuss whether any acoustic resonance could be generated at EPU flow or during power ascension to EPU power in the feedwater (FW) and main steam lines and describe how the acoustics-driven dynamic pressure loading acting on the components inside the steam generator under EPU conditions will be estimated. The response to RAI 20 did not include a discussion pertaining to FW.

NextEra Response

Generation of acoustic resonances is not considered credible in the feedwater system of the PBNP steam generators (SGs) operating at EPU conditions including the power ascension transient. The BWR industry events discussed in the original NextEra response to EMCB RAI 20 pertained to conditions where acoustic resonances, generated in the steam lines and transmitted to the outlet nozzle, had detrimental effects on the internals of the reactor. A similar condition is not plausible in the Point Beach Nuclear Plant (PBNP) SG feedwater ring which remains submerged during normal operation and, by design either remains full or is slow to drain when SG water level is decreased to below the feedring. Since all the components in the region of the SG feedwater inlet remain submerged, significant amounts of vibration damping are provided by the liquid environment that is not available in a dry steam environment. Flow induced vibration (FIV) of the magnitude observed in certain BWR dryers experiencing the acoustic phenomenon in a steam environment would not be anticipated to develop in this region of the SGs as a result of this high level of damping.

Regarding the potential for fluctuating feedwater pressure loadings to have detrimental effects on the SG internals, the following can be stated:

- Many PWR units with similar feedwater distribution components have been visually inspected in this region of the SGs and have been found not to have degradation related to FIV. No industry experience has been documented to date with feedwater acoustic pressure fluctuations causing degradation to PWR SG internals. There is no history of vibration or significant flow spiking in the feedwater lines at PBNP.

- The feedwater inlet to the PBNP SGs consists of an inlet nozzle attached to a feed ring fabricated from 8.75 inch outer diameter pipe which uses 35 J-tubes at Unit 1 or 36 J-tubes at Unit 2, with a 2.0 inch outer diameter to relieve the feed ring internal pressure and evenly distribute the inlet feedwater into the SG. Although not specifically designed to mitigate pressure resonances in the feedwater line, the design features of the SG feedwater distribution path are such that pressure resonances would be substantially dissipated prior to entering a region of potential degradation to the SG internals.

- Internal components near the feedwater inlet to the SG, such as the primary moisture separators, wrapper and feed ring components are rigidly supported. As a result, these components are not considered susceptible to vibratory degradation, as supported by a history of Westinghouse PWR operational experience.

- High levels of damping associated with feed ring components submerged in liquid water effectively reduce the potential for significant displacements and stresses associated with acoustic resonance that could be generated in the feedwater system.

The Units 1 and 2 SG upper internals are periodically inspected in accordance with industry SG guidelines. As part of these planned inspections, the SG upper internals including the feedwater distribution system of both units will be inspected in each of the two subsequent cycles following EPU to monitor potential effects of increased flow. The only region of the SG internals where it has been considered appropriate to apply the theory behind the acoustic phenomenon observed in BWRs is the SG steam dryers. Such a condition has been evaluated at EPU and concluded not to be a concern within the operational bounds of the EPU as described in the original NextEra response to EMCB RAI 20.

<u>Response to EMCB RAI 25 (pg 35/41)</u>

RAI 25 in part requested that when a different (and reconciled) code than the original code of construction was used for evaluating the structural adequacy of SSCs to provide assurance that the allowable values from the original code of construction have been utilized with the reconciled (later) year code. While the response has provided such an assurance for the case of the SG section of the response, it is not clear that that is also the case with the remainder SSCs.

NextEra Response

Provided below is additional information regarding code reconciliation of the NSSS SSCs.

Reactor Vessels and Replacement Reactor Vessel Closure Heads

For EPU, the reactor vessels were evaluated to all the requirements and allowables of the original Construction Codes. Therefore, no code reconciliations were required.

For the replacement reactor vessel closure heads (RRVCHs), the code reconciliation was performed per the criteria and methodology of IWA-4220 in the ASME Code, Section XI, 1998 Edition through 2000 Addenda.

The Construction Code for both the PBNP Unit 1 and Unit 2 RRVCHs is the ASME Code, Section III, 1998 Edition through 2000 Addenda. The original Unit 1 reactor vessel closure head Construction Code was the ASME Code, Section III, 1965 Edition while the original Unit 2 reactor vessel closure head Construction Code was the ASME Code, Section III, 1968 Edition through Winter 1968 Addenda.

The PBNP Unit 1 and Unit 2 RRVCHs were analyzed and constructed in accordance with the requirements of the ASME Code, Section III, 1998 Edition through 2000 Addenda per the design requirements and parameters specified in the RRVCH design specification. The design specification required an ASME Code, Section XI reconciliation of the differences between the original reactor vessel closure head Construction Codes and the Construction Code for the RRVCHs that affect their design bases.

The reconciliation addressed the following paragraphs from Section XI: IWA-4221, IWA-4222, IWA-4223, IWA-4224, IWA-4225 and IWA-4226. While the applicable requirements in the aforementioned paragraphs were satisfied as part of the reconciliation, a detailed discussion of IWA-4224 is provided below.

IWA-4224.1 provides reconciliation requirements for identical replacement materials procured to a later version of Section III of the Construction Code. Materials, including welding materials, may meet the requirements of later dates of issue of the material specification and later Editions and Addenda of Section III, provided the materials are the same specification, grade, type, class, alloy and heat-treated condition. Differences in the specified material tensile and yield strength must be compared and evaluated. If the replacement material has a lower strength, a comparison must be made of the allowable stresses. If the tensile or yield strength is reduced and allowable stresses are reduced, the effect of the reduction on the design analysis must be reconciled. For welding materials, any reduction in specified tensile strength must be evaluated to ensure that the strength of the filler metal meets or exceeds the strength of the base materials.

IWA-4224.1 allows the use of higher strength identical material (and therefore potentially higher allowable stresses) from a later Edition or Addenda of the Construction Code. However, reconciliation is necessary if the identical material from a later Edition or Addenda of the Construction Code has a lower strength and potentially reduced allowable stresses. Changes in identical material tensile and yield strengths as well as allowable stresses were identified. The evaluations resulting from the differences in the material tensile and yield strengths as well as allowable stresses of identical materials were performed in the PBNP Unit 1 and Unit 2 RRVCH Design Report analyses and found to be acceptable.

IWA-4224.2 provides reconciliation requirements for identical replacement materials procured to an earlier version of Section III of the Construction Code. No identical material for the RRVCHs was procured to an earlier version of Section III of the Construction Code per design specification requirements.

The suitability of using different materials than the original materials for the specified design also requires evaluation per IWA-4224.3 and IWA-4311. The different materials used for the RRVCHs were identified. Evaluations of the suitability of the different materials were also performed in the PBNP Unit 1 and Unit 2 RRVCH Design Report analyses and found to be acceptable.

Reactor Vessel Internals

PBNP Units 1 and 2 were built before the implementation of Subsection NG of the ASME Code; therefore, no specific ASME Code year is applicable for Section 2.2.3 of LAR 261, Attachment 5. The ASME Boiler and Pressure Vessel Code Section III, Division 1, 1998 Edition with 2000 Addenda, "Code Section NG-3222 and NG-3228.3," was chosen to meet the intent of the ASME Code. The original analyses for the PBNP reactor internals adopted the allowable stress criteria of Article 4 of the ASME Boiler and Pressure Vessel Code Section III, Section 1, 1968 Edition with Addenda through Winter 1968.

Comparison of the material composition and fatigue criteria presented in Subsection NG of the 1998 Edition of the ASME Code Section III, Division I, including 2000 Addenda with Article 4 of the 1968 Edition with Addenda through Winter 1968, demonstrates that the criteria used in this analysis are reconciled with the requirements and the allowable stress limits. This has been documented and reconciled. Also, the fatigue usage criteria in the 1968 Edition with Addenda through Winter 1968 are reconciled to those in the 1998 Edition including 2000 Addenda of the ASME Code.

Reactor Coolant Pumps (RCPs)

The PBNP RCPs are not N-stamped and therefore do not have a particular code of construction. Stress analyses for the PBNP RCPs are based on the 1965 Edition of the ASME Code. No other code years were used for allowable stress values in the RCP analysis performed for EPU.

Pressurizer

The EPU calculation referenced ASME 1965-1966, Summer which is the original construction code for the Units 1 and 2 pressurizers. The pressurizer surge line thermal stratification report referenced ASME 1986, which was required by NRC Bulletin No. 88-11, Pressurizer Surge Line Thermal Stratification (Reference 6). The allowables for the surge nozzle referred to in both the EPU calculation and the pressurizer surge line thermal stratification report was 57.9 ksi for 3Sm. This corresponds to the Sm value for the design temperature of 680°F for SA-216 WCC for both ASME 1965-1966, Summer and ASME 1986.

Control Rod Drive Mechanisms (CRDMs)

A Section XI code reconciliation was performed for the CRDMs. The design report and CRDM calculation referenced this code reconciliation for the use of the ASME 1998-2000 Code. The allowables are virtually identical, with any differences being negligible, and attributed to different round off methods employed by the ASME Codes through the years.

RCS Supports (Excluding Reactor Vessel Supports)

RCS supports will be addressed in separate correspondence regarding EMCB RAI 1.

Reactor Vessel Supports

The original design basis requirements were used for the EPU project.

References

(1) NRC electronic mail to NextEra Energy Point Beach, LLC, dated August 26, 2010, Point Beach Nuclear Plant, Units 1 and 2 - Requests for Additional Additional Associated with Extended Power Uprate (TAC Nos. ME1044 and ME1045) (ML102440095)

(2) FPL Energy Point Beach, LLC letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)

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(4) NRC electronic mail to NextEra Energy Point Beach, LLC, dated November 18, 2010, Point Beach Nuclear Plant, Units 1 and 2 – Comments to NextEra's Request for Additional Information Responses (ML103330408)

(5) U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2," NUREG-1839, December 2005 (ML053420134)

(6) U.S. Nuclear Regulatory Commission, "Pressurizer Surge Line Thermal Stratification," NRC Bulletin No. 88-11, December 20, 1988 (ML031220290)