April 26, 2010

NRC 2010-0035
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 Additional Information

References:
(1) FPL Energy Point Beach, LLC letter to NRC, dated April 7, 2009, License Amendment Request 261, Extended Power Uprate (ML091250564)
(2) NRC letter to NextEra Energy Point Beach, LLC, dated March 25, 2010, Point Beach Nuclear Plant, Units 1 and 2 – Request for Additional Information from Piping and NDE Branch Re: Extended Power Uprate (TAC NOS. ME1044 and ME1045) (ML100780610)

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 license 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 staff's continued review of the request. Enclosure 1 provides the NextEra response to the NRC staff's request for additional information.

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 April 26, 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 ADDITIONAL INFORMATION

The NRC staff determined that additional information was required (Reference 1) to enable the Piping Integrity and NDE Branch to complete its review of License Amendment Request (LAR) 261, Extended Power Uprate (EPU) (Reference 2). The following information is provided by NextEra Energy Point Beach, LLC (NextEra) in response to the NRC staff's request.

RAI CPNB-1

On page 2.1.5-8, the licensee discussed the inspection requirements for the reactor pressure vessel (RPV) closure head based on the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-729-1. The staff notes that ASME Code Case N-729-1 has been incorporated by reference in Title 10 of The Code of Federal Regulations (10 CFR) 50.55a(g)(6)(ii)(D), Reactor vessel head inspections, with conditions.

a) *Discuss RPV head inspection including methods, results, and dates at PBNP since 2005.*
b) *Discuss whether the RPV closure head inspections will be conducted in accordance with the regulatory requirements of 10 CFR 50.55a(g)(6)(ii)(D).*
c) *Discuss whether 10 CFR 50.55a(g)(6)(ii)(E), Reactor coolant pressure boundary visual inspections, will be followed.*

NextEra Response

Unit 1 and Unit 2 reactor pressure vessel (RPV) closure heads at Point Beach Nuclear Plant (PBNP) were replaced in 2005. Both RPV closure heads underwent a complete baseline pre-service inspection at Mitsubishi Heavy Industries facilities in Japan. These inspections included visual, liquid penetrant, ultrasonic, and eddy current testing. No rejectable indications were found.

The Unit 1 RPV closure head has undergone a visual examination for leakage (VT-2) each refueling outage since installation in 2005. During the spring 2010 Unit 1 refueling outage, a bare metal visual inspection was performed per American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-729-1. No indications of leakage from the RPV head penetrations were noted in the exams.

The Unit 2 RPV closure head has undergone a visual examination for leakage (VT-2) each refueling outage since installation in 2005. During the fall 2009 Unit 2 refueling outage, a bare metal visual inspection was performed per ASME Code Case N-729-1. No indications of leakage from the RPV head penetrations were noted in the exams.

PBNP will comply with the requirements of 10 CFR 50.55a(g)(6)(ii)(D) and Code Case N-729-1. The new requirements have been incorporated in the fourth interval Inservice Inspection (ISI) Program for both units. Bare metal examinations required by ASME Code Case N-729-1 have been completed.

PBNP will comply with the requirements of 10 CFR 50.55a(g)(6)(ii)(E) and Code Case N-722 as written. The new requirements have also been incorporated in the ISI Program.

RAI CPNB-2

On page 2.1.5-9, the licensee referenced ASME Code Case N-481 for the inspection of primary loop pump casings that are fabricated with cast austenitic stainless steel (CASS). ASME Code Case N-481 was previously approved for use in Regulatory Guide (RG) 1.147, Revision 14, but ASME annulled Code Case N-481 as of March 28, 2004 and this annulment is reflected in Revision 15 of RG 1.147. Justify the use of Code Case N-481 or propose alternative examinations for pump casings and valve bodies that are fabricated with CASS.

NextEra Response

The evaluations to Code Case N-481 were performed prior to N-481 being annulled. Code Case N-481 was annulled because it was incorporated into the ASME Code. Code Case N-481 is not applicable to the year of the ASME Section XI currently being used by PBNP (1998 Edition with Addenda through 2000). Alternative pump casing and valve body exams are not required because PBNP performs examinations as specified by ASME Section XI.

RAI CPNB-3

Table 2.1.5-1 on page 2.1.5-5 summarizes service temperature changes in the RPV closure head and bottom-mounted instrumentation (BMI) penetrations due to the proposed EPU. On page 2.1.5-4, the licensee did not clearly describe how the maximum temperature at the RPV head is used to determine the maximum change in the primary water stress corrosion cracking (PWSCC) susceptibility and whether the maximum temperatures are appropriate or conservative.

a) Clarify the use of the maximum temperature at the RPV head and BMI penetrations to determine the PWSCC susceptibility.
b) Clarify whether the inspection of the RCS components will be affected by the EPU conditions.

NextEra Response

In response to Question a), considering that susceptibility to primary water stress corrosion cracking (PWSCC) is proportional to temperature, the maximum temperature is used to determine PWSCC susceptibility. Refer to LAR 261, Attachment 5, Table 2.1.5-1, Summary of Service Temperature Changes in the RV Closure Head and Bottom-Mounted Instrumentation (BMI) Penetrations Due to the Proposed EPU. As stated in LAR 261, Attachment 5, Section 2.1.5, Reactor Coolant Pressure Boundary Materials, the increase in temperature in the closure head would not result in a significant increase in the potential for PWSCC to occur. In addition, the closure head penetrations are made of Alloy 690. Since the EPU will result in a net decrease in temperature in the bottom-mounted instruments BMIs, the lower temperature would result in a decrease in PWSCC susceptibility.

In response to Question b), based on the completed reviews of material degradation mechanisms as shown in EPU LAR 261, Attachment 5, Section 2.1.5, inspection of reactor coolant system (RCS) components will not be affected.

RAI CPNB-4

Pages 2.1.5-6 through 2.1.5-8 discussed PWSCC of Alloy 600/82/182 and replacement efforts using Alloy 690/52/152 materials. However, the licensee did not address the impact of the EPU on those Alloy 600 components that have not been replaced with Alloy 690. Discuss the impact of EPU on Alloy 600 components and any programs or procedures to monitor the degradation of Alloy 600 components.

NextEra Response

The components containing Alloy 600/82/182 are listed on Page 2.1.5-6 in LAR 261, Attachment 5, Section 2.1.5, Reactor Coolant Pressure Boundary Materials. As discussed above in the response to RAI CPNB-3, at EPU conditions the BMIs would experience a decrease in temperature which represents a reduction in PWSCC susceptibility. In Unit 2, the steam generator hot leg and cold leg safe end welds were made with Alloy 82/182 weld deposits that were inlaid with Alloy 152. The Alloy 152 seals the Alloy 82/182 from reactor coolant, thus protecting the Alloy 82/182 from PWSCC. In Unit 1, the steam generator tubing is thermally treated Alloy 600 which provides about a four-fold decrease in PWSCC susceptibility compared to non-thermally treated Alloy 600 tubing. The temperature increase at the Unit 1 tubing and other Alloy 600 components (clevis inserts and clevis insert lock keys in Units 1 and 2, steam generator channel head drains in Unit 1, and steam generator divider plate in Unit 1) represents a slight increase in PWSCC susceptibility. The inspection program remains adequate to monitor the degradation of Alloy 600 components at EPU conditions.

RAI CPNB-5

On page 2.1.5-9, the licensee stated that the EPU will not increase the susceptibility of Alloy 600/82/182 components to PWSCC at PBNP. Discuss why and how the EPU will not increase the susceptibility of Alloy 600/82/182 components to PWSCC.

NextEra Response

See response to RAI CPNB-4 above.

RAI CPNB-6

On page 2.1.5-9, the licensee briefly mentioned that a separate flaw tolerance evaluation was done to manage thermal aging of CASS material of the reactor coolant system (RCS) piping components as a part of its license renewal application (LRA). Discuss the details of the flaw tolerance evaluation that was performed to manage the effect of EPU for the RCS piping components.

NextEra Response

The reactor coolant loop A376 TP316 piping material is not cast austenitic stainless steel (CASS) and is not susceptible to thermal aging. However, some of the A351 CF8M piping elbow material is susceptible due to the δ-ferrite content level. The susceptible piping locations in the reactor coolant loop piping system were determined based on the molybdenum content and casting method as well as ferrite content using the guidelines given in the U.S. Nuclear Regulatory Commission letter dated May 19, 2000, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components." Using the same guidelines, flaw tolerance evaluations were performed for the susceptible locations in accordance with the evaluation procedures and acceptance criteria in Paragraph IWB-3640 Section XI, of the ASME Code.

The results of the flaw tolerance evaluations demonstrated that even with thermal aging in the susceptible reactor coolant loop CASS piping material, the susceptible piping locations are tolerant of large flaws. The maximum acceptable flaw size for a range of aspect ratios (flaw length/flaw depth) at the susceptible CASS piping locations in the hot leg, crossover leg and cold leg were determined in the flaw tolerance evaluations. For example, an axial flaw in the susceptible hot leg location with a flaw depth of 28% of the wall thickness and an aspect ratio of six (6) would not exceed the ASME Section XI acceptance criteria for the next 30 years, which represents the remaining plant life for PBNP units. The acceptable flaw depths with the same aspect ratio are even larger for other flaw orientations at other susceptible piping locations in the reactor coolant loop.

RAI CPNB-7

Discuss whether the RCS water chemistry program (e.g. chemistry limits and monitoring parameters) needs to be changed as a result of EPU.

NextEra Response

The RCS primary water chemistry is rigorously controlled, particularly with regards to oxygen, chlorides, and other halogens, in accordance with the requirements of the PBNP chemistry control program. The proposed PBNP EPU lithium, boron, and pH management program based on an 18 month cycle recommended operating at pH levels between 7.4 and 6.9 while the Lithium level is maintained between 2.35 and 2.05 parts per million (ppm). These conditions are identical to current operating parameters and are bounded by the proposed Electric Power Research Institute (EPRI) chemistry guidelines.

The EPU does not require a change to the RCS water chemistry program.

RAI CPNB-8

Discuss the impact of EPU on neutron irradiation induced embrittlement of the reactor vessel.

NextEra Response

In general, an EPU will result in increased embrittlement of the reactor vessel materials due to increased fast neutron fluence from the increased reactor core power level. Therefore, analyses were performed to determine the integrity of the reactor vessel materials under EPU conditions through the end of life extension (EOLE), 53 effective full power years (EFPY).

Certain analyses conclude that action will be required prior to a specific EFPY. These are summarized below:

- Pressurized Thermal Shock (PTS): The screening criteria for PTS are contained in 10 CFR 50.61. PBNP Unit 1 has acceptable PTS calculations through the renewed license period. PBNP Unit 2 is projected to exceed the screening criteria at 39.5 EFPY. Therefore, a separate license amendment request will be submitted in accordance with regulatory requirements for the Unit 2 PTS criteria.

- Emergency Response Guideline (ERG) Limits: The ERG Limits were evaluated to determine if changes are required as a result of the EPU. The Unit 1 ERG category is III.b and it will remain unchanged through the period of extended operation. The Unit 2 ERG category must be changed from Category III.b to plant-specific prior to 39.5 EFPY.

- Pressure-Temperature (P-T) Limit Curves: P-T limit curves have recently been updated and apply to both PBNP units. These curves were evaluated for applicability under the uprated conditions. New applicability dates were calculated to account for the increased rate of embrittlement due to the EPU. The end of life (EOL) curves are now applicable through 35.9 EFPY (previously 36.9 EFPY for current licensed power level). The EOLE curves are now applicable through 53 EFPY (previously 56.9 EFPY).

In summary, there will be increased fast neutron fluence on the reactor vessel materials as a result of the EPU. This increased embrittlement has been evaluated for the reactor vessel materials, and has been determined to have the impact as stated above. For more information, refer to LAR 261, Attachment 5, Section 2.1.1, Materials and Chemical Engineering, Section 2.1.2, Pressure-Temperature Limits and Upper Shelf Energy, and Section 2.1.3, Pressurized Thermal Shock, which discuss the impact in greater detail.

RAI CPNB-9

Page 2.1.5-5 discussed the boric acid corrosion control (BACC) program in terms of LRA. Discuss the impact of the EPU on the BACC program for RCPB and whether the BACC program will be changed as a result of EPU.

NextEra Response

As stated in LAR 261, Attachment 5, Section 2.1.5.2.5, Results, "No new material degradation issues of carbon steel boric acid corrosion are expected due to the EPU water chemistry." On Page 2.1.5-11, the last paragraph after the bullet states, "The results of the reactor coolant pressure boundary material degradation assessment showed that no new material issues will result from the proposed power uprating at PBNP. On this basis it is concluded that the new EPU environmental conditions (chemistry, temperature, and fluence) will not introduce any new aging effects on their components during 60 years of operation, nor will the EPU change the manner in which the component aging will be managed by the aging management program credited in the LRA and accepted by the NRC in the SER."

There is no impact of the EPU on the boric acid corrosion control (BACC) program and no changes to the program are required.

RAI CPNB-10

Discuss the impact of EPU on the integrity of reactor vessel internals.

NextEra Response

Reactor vessel internals are discussed in LAR 261, Attachment 5, Section 2.1.4, Reactor Internals and Core Support Materials. A number of potential age-related degradation mechanisms may affect reactor vessel internals materials during a period of life extension, including stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking, gamma heating, void swelling, and thermal aging. The proposed EPU will not significantly change the identified aging mechanisms or the ability of programs to manage aging effects.

RAI CPNB-11

On page 2.1.5-3, the licensee briefly mentioned about absorption of energy within the elastic strain energy range and absorption of energy by plastic deformation. Discuss in detail the impact of EPU on the absorption of energy within the elastic strain energy range and absorption of energy by plastic deformation.

NextEra Response

The reactor vessel integrity analyses contained in LAR 261, Attachment 5, Sections 2.1.2, Pressure-Temperature Limits and Upper Shelf Energy, and Section 2.1.3, Pressurized Thermal Shock, encompass the requirements for the reactor vessel in the elastic and plastic ranges. The most limiting condition for the reactor vessel beltline region is in the transition region between brittle and ductile behavior and energy absorbed is analyzed utilizing linear elastic fracture mechanics (LEFM) in this region. The PTS and P-T limits evaluations (LAR 261, Attachment 5, Sections 2.1.3 and 2.1.2, respectively) address the transition region.

The basis for the PTS Screening Criteria and the P-T limits is LEFM. The upper shelf energy (USE) evaluation (LAR 261, Attachment 5, Section 2.1.2) is focused on the ductile region where high temperatures are reached and plastic behavior is encountered. LAR 261, Attachment 5, Sections 2.1.2 and 2.1.3 discuss the impact of the EPU on these evaluations.

RAI CPNB-12

Discuss the impact of EPU on the piping loads and resulting stresses for the RCS piping and whether safety margins in the ASME Code, Section III, NB-3200 and NB-3600 are satisfied.

NextEra Response

The impact of the EPU on the RCS piping loads and resulting stresses are discussed in LAR 261, Attachment 5, Section 2.2.2.1, NSSS Piping, Components and Supports, and primarily in the Subsection 2.2.2.1.2, Technical Analysis, subsection titled Description of Analyses and Evaluations, and subsection titled NSSS Piping, Components, and Support Results located on Pages 2.2.2-8 through 2.2.2-10.

The acceptance criteria for the PBNP RCS piping is based on ANSI B31.1 Code criteria and satisfies the safety margins and complies with the acceptance requirements of ANSI B31.1 code criteria for the EPU evaluations. Refer to LAR 261, Attachment 5, Table 2.2.2.1-1 on Page 2.2.2-13.

RAI CPNB-13

Discuss whether the EPU will result in degradation mechanisms (i.e. steam/water hammer, low and high cycle fatigue, creep damage, erosion, general corrosion, and other environmental conditions) which would lead to increased degradation of RCPB systems.

NextEra Response

EPU conditions will not cause a change in degradation mechanisms nor will they add additional degradation mechanisms that could lead to increased degradation of the reactor coolant pressure boundary systems. LAR 261, Attachment 5, Section 2.1.5, Reactor Coolant Pressure Boundary Materials, provides additional detail on potential reactor coolant pressure boundary degradation mechanisms.

RAI CPNB-14

Pages 2.1.5-9 and 2.1.5-10 discussed that the EPU will not affect thermal aging of CASS. It is not clear to the staff why the high temperature (611.1 °F) from the EPU would not have any effect on thermal aging of CASS. Discuss in details why the fracture toughness of CASS will not be affected by the EPU.

NextEra Response

Fracture toughness values considering the effect of thermal aging of CASS were calculated in the existing leak-before-break (LBB) analysis (see Reference 1 of LAR 261, Attachment 5, Section 2.1.6, Leak-Before-Break) using the temperature values (pre-EPU) shown in the NextEra response to RAI CPNB-15, below. The change in temperatures (5.6°F for the hot leg, 1.6°F for the cross-over leg and 1.5°F for the cold leg) due to the EPU conditions will not have a significant impact on the fracture toughness values used in the existing LBB analysis. The EPU evaluation demonstrated that a significant margin exists between detected flaw size and flaw instability.

RAI CPNB-15

Section 1.1 does not clearly state the differences in pressure and temperature between the values used in the original LBB evaluation and the values as a result of the EPU. Provide the pressure and temperature used in the original LBB analysis and the values used to assess the original LBB evaluation under the EPU conditions.

NextEra Response

The table below summarizes the pressure and temperature used in the original LBB analysis and the values used to assess the original LBB evaluation under the EPU conditions for the primary loop piping. LAR 261, Attachment 5, Section 2.1.6, Leak-Before-Break, provides the analysis under EPU conditions.

Parameter	Existing LBB Analysis	EPU LBB Analysis
Pressure	2250 psia	2250 psia
Hot Leg Temperature	605.5°F	611.1°F
Crossover Leg Temperature	541.1°F	542.7°F
Cold Leg Temperature	541.4°F	542.9°F

For other lines such as the pressurizer surge line, residual heat removal (RHR) line, and accumulator line, there are no changes in temperature and pressure due to EPU conditions.

RAI CPNB-16

Operating experience has shown that Alloy 82/182 dissimilar metal (DM) butt welds are susceptible to PWSCC.

a) Discuss whether Alloy 82/182 DM butt welds exist in the LBB piping (e.g. primary loop piping, pressurizer surge line piping, accumulator lines, and residual heat removal (RHR) lines).
b) Discuss whether any mitigation has been implemented at these welds. If not, discuss plans for mitigation.
c) Discuss whether the original LBB analysis is affected by EPU.

NextEra Response

In response to Questions a) and b), neither PBNP unit utilizes Alloy 82/182 dissimilar metal welds in the reactor vessel hot or cold legs. In Unit 2, the steam generator hot leg and cold leg safe end welds were made with Alloy 82/182 weld deposits that were inlaid with Alloy 152. The Alloy 152 seals the Alloy 82/182 from reactor coolant thus protecting the Alloy 82/182 from PWSCC. Therefore, mitigation is not applicable.

In response to Question c), the impact of temperature changes in primary loop piping as a result of EPU on the existing LBB analysis is not significant. The EPU evaluation results demonstrate that all the LBB margins for the primary loop piping continue to be satisfied for the EPU conditions. For the pressurizer surge line, accumulator lines and the RHR lines, the existing LBB analyses remain valid for EPU conditions, as discussed in LAR 261, Attachment 5, Section 2.1.6, Leak-Before-Break.

RAI CPNB-17

In page 2.1.6-5, the licensee stated that based on the evaluations documented in LRA Section 2.2.2.1 the current design basis loads and results for the pressurizer surge line piping, accumulator lines, and RHR lines remain unchanged. However, for the primary loop piping (discussion on page 2.1.6-4), the licensee did not mention LRA Section 2.2.2.1 to evaluate the primary loop piping. Discuss how the primary loop piping was evaluated for the EPU conditions.

NextEra Response

For the primary loop piping LBB evaluation, the input from LAR 261, Attachment 5, Section 2.2.2.1, NSSS Piping, Components, and Supports, analysis was also used. However, the impact was not significant and the LBB margins for the primary loop piping continue to be satisfied for the EPU conditions.

References

(1) NRC letter to NextEra Energy Point Beach, LLC, dated March 25, 2010, Point Beach Nuclear Plant, Units 1 and 2 – Request for Additional Information from Piping and NDE Branch Re: Extended Power Uprate (TAC NOS. ME1044 and ME1045) (ML100780610)

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