POINT BEACH

December 21, 2010

NRC 2010-0195
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, FW: Comments to NextEra's Request for Additional Information Responses Via Letter NRC 2010-0136, dtd 11/8/10 (ML103330408)
(5) NextEra Energy Point Beach, LLC letter to NRC, dated December 15, 2010, License Amendment Request 261, Extended Power Uprate, Response to Request for Clarification

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 is required to enable the staff's 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 staff's request for clarification for EMCB RAI 1. NextEra responded to the remaining Reference (4) requests for clarification via Reference (5).

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 21, 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 Energy Point Beach, LLC (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 for clarification of EMCB RAI 1. The NextEra response to the remaining clarification requests was submitted in Reference (5)

<u>Response to EMCB RAI 1 (pg 2/41)</u>

In reference to BOP, response statement that "he requirements and intent stipulated in the alter codes of record have been satisfied" is not acceptable. Also, in reference to RCS supports, response states in two places that "Updated revisions to [AISC 6th Ed] can be used provided the requirements and intent stipulated in the later codes of record have been satisfied." This statement is also not acceptable. The original code of construction requirements and intent need to be satisfied when performing a code reconciliation and not the other way around.

Response also states that:

> *The EPU evaluations for BOP pipe supports were evaluated using the AISC code, Sixth Edition, including updated revisions through the Ninth Edition.*

Please be specific which code has been utilized for EPU pipe support evaluations and if different than the AISC 6th edition state whether the later edition used has been reconciled to the 6th edition and assure that 6th edition allowables have been utilized with the reconciled code.

FSAR Table 4.1-9 states that:

> *The code requirement is the USAS B31.I Power Piping Code and that the version of the Code which was in effect at the time the original component was ordered is applicable.*

In reference to NSSS piping and supports, response states that:

> *The PBNP NSSS piping evaluations and qualifications for EPU conditions also utilized the ANSI Code for Pressure Piping B31.1, 1973 Edition, as was used in the existing plant design basis AOR. Therefore, the code of record allowable values used in the existing plant design basis AOR are used for the EPU evaluation.*

Assure that reconciliation to later codes used exists and provide assurance that EPU evaluations have been performed utilizing allowables from the code of construction that the original component was ordered, as required by the plant FSAR.

NextEra Response

With regard to the comment in the first paragraph, NextEra agrees. The statement in NextEra response in EMCB RAI 1 in Reference (2) is in error and should read, "Updated revisions to [AISC Sixth Ed] can be used provided the requirements and intent stipulated in the original code of record has been satisfied."

Balance of Plant (BOP) Pipe Supports

The safety-related and seismically designed BOP pipe support evaluations for the EPU were performed in accordance with the American Institute of Steel Construction (AISC) Code, Sixth Edition, including updated revisions through the Ninth edition. The Final Safety Analysis Report (FSAR) does not specify which revision of the AISC code was used for the design of BOP pipe supports. The AISC Sixth Edition was in effect at the start of original plant design and construction. That Code edition has been inferred to be the design basis code of record for BOP pipe support design for Point Beach Nuclear Plant (PBNP).

For pipe supports that were modified or required structural reanalysis as a result of EPU, the detailed analysis was performed in accordance with the AISC Code editions Sixth, Seventh or Ninth. For those analyses that used any portion of the AISC Code Seventh or Ninth Editions, a comparison of the data/methods used from these later AISC Codes to the corresponding data/methods from the AISC Code Sixth Edition was performed to assure that the pipe support evaluations satisfy the requirements and intent of the AISC Code, Sixth Edition. Specific attributes that were evaluated in this comparison included 1) code equations, 2) threaded connections, 3) section properties and 4) allowable stresses. A summary of these evaluations/comparisons are as follows:

1) There are no significant differences between the code equations used in the BOP pipe support reanalyses and the AISC Sixth Edition.

2) There were no threaded connections governed by the AISC code within the scope of the BOP pipe supports reanalyzed for EPU.

3) The member section properties from later code editions used in the BOP pipe support reanalysis were compared to those provided in AISC, Sixth Edition, and the differences were evaluated. An evaluation determined that all supports remain acceptable.

4) Except as noted below, there are no significant adverse differences between allowable stresses used in the BOP pipe support reanalyses and the AISC Sixth Edition.

   a) The structural reanalysis of BOP pipe supports used a 1.5 allowable stress factor for the Faulted load condition. Section 1.5.6 of the AISC Codes allows a one-third increase in allowable stresses when calculating stresses that include wind or seismic loadings. The AISC codes do not address loadings to the piping fluid transient events nor does it differentiate between various levels of seismic excitation. With no specific guidance available within the AISC code, the allowable stress increase factor used for the Faulted load condition, which includes higher (safe shutdown earthquake) seismic levels and fluid transient loadings, was based on a factor of 1.5. This is consistent with, or more restrictive than, the methodology presented in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NF for Emergency and Faulted load conditions. Furthermore, Table A.5-3, Loading Conditions and Stress Limits, of the FSAR provides a 1.5 load factor for the Faulted condition for ASME Subsection NF component supports. Therefore, the use of the 1.5 allowable stress factor for Faulted load conditions is concluded to be acceptable.

b) The reanalysis of structural welds for BOP pipe supports uses allowable Normal condition weld stresses of 18 ksi and 21 ksi for E60xx and E70xx electrodes, respectively. This is and has been the industry standard since the construction of the PBNP. The AISC Sixth Edition specifies allowable weld stresses of 13.6 ksi and 15.8 ksi for E60xx and E70xx electrodes, respectively. No specific basis for the allowable values of 13.6 ksi and 15.8 ksi is provided and they appear to have been carried over since early editions of the Code. The AISC Seventh Edition changed the allowable weld stresses to the industry standards and the commentary in the Seventh Edition noted that weld stresses presented in earlier editions "were known to be overly conservative for their recommended use." Therefore, the use of the industry acceptable allowable weld stresses meets the intent of the FSAR.

In summary, the subject EPU BOP pipe support evaluations satisfy the intent of the requirements provided in the AISC Code, Sixth Edition.

Reactor Coolant System (RCS) Supports (Excluding Reactor Vessel Supports)

The original code of construction for PBNP, nuclear steam supply system (NSSS) RCS supports (steam generator (SG), reactor coolant pump (RCP) and pressurizer) was AISC, Sixth Edition. The existing plant design basis analysis of record (AOR) evaluation for PBNP, NSSS RCS supports (SG, RCP and pressurizer) performed during 1995 and 2001 timeframe (including the replacement SG program) utilized the AISC, Eighth Edition and ASME B&PV Code, Section III, Subsection NF, 1974 Edition as the code of record. With regard to reconciliation of the Eighth Edition to the Sixth Edition of the AISC Code, the allowable values, equations, and threaded connection information are the same for the materials used for the NSSS RCS supports. Additionally, the AOR and EPU utilized the section properties of the Sixth Edition of the AISC Code, so no reconciliation is required. With regard to the ASME B&PV Code, Section III, Subsection NF, 1974 Edition portion of the AOR and EPU, only limited analyses, utilizing the ASME Code, were performed to address certain components (attachment bolting, pins, etc.) that were not explicitly covered by AISC. Thus, no reconciliation is required.

The NSSS RCS support evaluations for EPU were performed consistent with the analysis of record which was appropriately reconciled to the original code of construction as described above.

NSSS Piping

For the NSSS piping (RCS piping), Table 4.1-9 of the FSAR states that the code requirement is the United States of America Standard (USAS) B31.1 Power Piping Code and that the version of the Code which was in effect at the time the original component was ordered is applicable.

Other sections of the FSAR (e.g., Section 4.2, Page 4.2-14) state that the piping and fittings are designed to the B31.1, 1955 Edition. Based on the review of the existing plant design basis AOR evaluations and qualifications performed for the steam generator replacement program and fuel upgrade program, NextEra concluded that the American National Standards Institute (ANSI) Code for Pressure Piping B31.1, 1973 Edition was utilized as the code of record. This is considered acceptable because the equations presented in the 1955 version of the code are identical to the equations presented in the 1973 version of the code. Additionally, the allowable stress definitions for normal operation, and variations from normal operation, are identical, and the material allowable is from the version of the Code which was in effect at the time when original components were designed and fabricated. Also, per the note in ANSI B31.1, 1973 Edition, the B31.1, 1967 Edition through B31.1, 1972 Edition, have been revised and consolidated into one publication and re-designated B31.1, 1973 Edition. The PBNP NSSS piping evaluations and qualifications for EPU conditions also utilized the ANSI Code for Pressure Piping B31.1, 1973 Edition, as was used in the existing plant design basis AOR. Therefore, the code of record allowable values used in the existing plant design basis AOR are used for the EPU evaluations.

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)

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