November 15, 2010

NRC 2010-0171
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 Clarification Request

References:

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

(2) NextEra Energy Point Beach, LLC letter to NRC, dated January 13, 2010, License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information (ML100140163)

(3) NextEra Energy Point Beach, LLC letter to NRC dated July 8, 2010, License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information (ML101890788)

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.

During a discussion with the Reactor Systems Branch on October 27, 2010, and a follow-up telephone conference on November 8, 2010, the NRC staff requested clarification of information provided in References (2) and (3). Enclosure 1 provides the NextEra response to the NRC staff's request for clarification.

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 November 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 CLARIFICATION REQUEST

During a discussion with the Reactor Systems Branch on October 27, 2010, and a follow-up telephone conference on November 8, 2010, NRC staff requested clarification of information provided in References (1) and (2) to enable the Reactor Systems Branch to complete the review of License Amendment Request (LAR) 261, Extended Power Uprate (EPU) (Reference 3). The requested clarification is described below.

<u>Clarification Request</u>

The NRC staff requested additional information, to that provided in Reference (2), regarding the trends and effects which would support application of WCAP-9226-P-A, Revision 1 to two-loop Westinghouse Plants.

<u>NextEra Response</u>

NextEra recognizes that the current two-loop Westinghouse plant design applicable to the Point Beach Nuclear Plant (PBNP) EPU analysis differs in some respects from the three-loop plant design that was used in the steam line break (SLB) topical report analyses. Some of the key differences are discussed below, along with their potential impact on the analyses. Although this clarification request specifically concerned with the hot full power (HFP) SLB analysis, differences affecting both HFP and hot zero power (HZP) SLB cases are included in the discussion below.

1) Variations in the core design affect the shutdown margin, post-trip reactivity coefficients, location of the limiting stuck rod, and peaking factors that occur during a SLB transient. The methodology specifically models these effects and their influence on the subsequent reactivity and power transients and peak power reached as a result of the cooldown. In addition, these variations do not affect the assumption that the end-of-life core conditions result in the largest moderator density reactivity coefficients and result in the highest power level during the post-trip transient.

2) Both the two-loop and three-loop plant designs have safety injection actuation signals on Low Pressurizer Pressure and Low Steam Pressure. Setpoints and the actuation logics are similar, although differences in the dynamic compensation terms exist. However, due to the rapid depressurization resulting from the break, the timing for the start of safety injection is not significantly different. Finally, the actuated equipment is similar for both designs being compared, and any variations in the capacity and response times of the equipment are explicitly considered as analysis inputs.

3) For post-trip analyses, appropriate conservative design reactor vessel mixing coefficients corresponding to the number of coolant loops are applied in the plant-specific analyses, including the PBNP EPU analysis. The vessel mixing coefficients are based on the data taken from the Indian Point Nuclear Generating Plant 1/7th scale tests, which specifically addressed two-loop, three-loop, and four-loop vessel configurations. Differences in the vessel mixing coefficients may affect the temperature asymmetries seen at the core inlet (cross-loop mixing) and locations of the limiting stuck rod (due to lower plenum mixing). However, this does not affect the overall behavior of the reactor coolant system (RCS) that the faulted loop will cool down farther and faster than the intact loops. The faulted loop provides the forcing function for the cooldown of the intact loops. Again, the transient response of the RCS will be similar regardless of the number of coolant loops (two, three, or four). More information on the vessel mixing coefficients is presented in the response to NRC RAI "Question 4" documented in letter NSD-NRC-98-5765 (Appendix B of WCAP-14882-P-A).

4) A highly borated boron injection tank (BIT), included in the post-trip analyses in the SLB topical report, has been either reduced in boron concentration or physically eliminated from the piping for virtually all Westinghouse-designed plants. This change allows the transient to progress to higher power levels, since injection of borated water is delayed. However, these differences do not affect the key trends of this event, specifically, the depressurization and cooldown behavior following this transient. The original design with a highly borated BIT was necessary in order to support a conservative licensing criterion (i.e., that the transient should not return to critical conditions) that was applied to the credible SLB analysis (inadvertent opening of a steam relief or dump valve). Subsequently, this conservative criterion was eliminated and the relaxed criterion that the departure from nucleate boiling (DNB) design basis must be met (the same criterion conservatively applied in the more limiting hypothetical SLB case) was approved in the licensing of plant-specific licensing amendment requests for BIT elimination.

References

(1) NextEra Energy Point Beach, LLC letter to NRC, dated January 13, 2010, License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information (ML100140163)

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