NEXTera ENERGY POINT BEACH

May 20, 2010

NRC 2010-0030
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 electronic mail to NextEra Energy Point Beach, LLC, dated February 25, 2010, DRAFT – Request for Additional Information From Reactor Systems Branch Re: Extended Power Uprate (ML100560283)

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. The Enclosure 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 May 20, 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

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 Reactor Systems 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 SRXB-LTT-1

Licensing Report Section 2.12.1.2.3.2, "EPU Power Ascension Test Plan and Test Plateaus," refers to "transient data gathered during the specified transient tests at low-power" (LR Page 2.12-6, Paragraph 2), and additional discussion refers the reader to Table 2.12-1 and Table 2.12-2 for additional details regarding the proposed EPU test plan. Table 2.12-1, however, does not refer to any transient testing other than data collection. Table 2.12-2, Item 13, by contrast, refers to "the planned load swing tests" that will "dynamically test the FW control system." This item refers the reader back to 2.12.1.2.3; however, the reviewer was unable to locate further discussion of any planned load swing tests. Describe the planned load swing tests in further detail.

NextEra Response

There are no planned load swing tests for the Point Beach Nuclear Plant (PBNP) EPU power ascension, as discussed in LAR 261, Attachment 5, Section 2.12.1.2.6, Justification for Exception – Specific, Electrical Load Loss and Load Swings. This is consistent with LAR 261, Attachment 5, Table 2.12-2, Item 35, Load Swing and Load Reduction Test. The references to "transient testing" in Section 2.12.1.2.2, "transient data gathered during the specified transient tests at low-power," in Section 2.12.1.2.3.2, "transient data collection" in Table 2.12-1, and the "planned load swing test" in Table 2.12-2, Item 13 were in error.

The feedwater regulating valves are being modified for EPU to replace the valve trim and actuators (including solenoid valves). However, the feedwater control function and operation of the feedwater control system is not being modified. Therefore, a load swing test is not required. Normal post-modification testing, surveillance testing, and inservice testing will be performed. The feedwater control system will be monitored during power ascension to ensure the feedwater controls are operating correctly and that steam generator level is automatically controlled within operating limits.

RAI SRXB-LTT-2

A modification is planned to the pressurizer heater system that will remove backup heater actuation on a pressurizer high level deviation signal. How are the effects of this modification tested or demonstrated against a transient that challenges the RCS inventory and pressure control?

NextEra Response

The original generic design basis for the Westinghouse plants that included backup heater actuation on pressurizer high level deviation was to increase operating margin to a variable low pressure reactor trip setpoint on a Condition I 10% step load decrease transient. Reactor coolant system (RCS) temperature initially increases on a 10% step load decrease transient causing an insurge of cooler water in the pressurizer. It was assumed at that time that the cooler insurge water would instantly mix with the warmer water in the pressurizer and would reduce the pressure. Under these circumstances, turning on the backup heaters when the water volume increases above the deviation limit would provide additional margin to a variable low pressure reactor trip setpoint.

Since the time of the original generic design, the variable low pressure reactor trip has been replaced with an over-temperature ΔT reactor trip function but the heater actuation on high level deviation was retained.

The 10% step load decrease transient was analyzed for PBNP at EPU conditions. This analysis did not take credit for the pressurizer backup heaters' actuation on pressurizer high level deviation. The results of this analysis were acceptable at EPU conditions. LAR 261, Attachment 5, Section 2.4.2.1, Plant Operability (Margin to Trip), describes this analysis. It should be noted that although the actuation of backup heaters on pressurizer high level deviation has been deleted, the alarm indicating that the pressurizer high level condition is reached is retained at EPU conditions for operator awareness.

As shown in LAR 261, Attachment 5, Table 2.8.5.0-9, Key Safety Analysis Input Changes Made in Support of the PBNP EPU Program, actuation of the backup heaters on pressurizer high level deviation is not modeled in the loss of normal feedwater event (Section 2.8.5.2.3) and the loss of AC event (Section 2.8.5.2.2) at EPU conditions. The results of these events met all acceptance criteria.

RAI SRXB-LTT-3

For the operability evaluations justifying exemptions from large transient testing, describe modeling guidelines followed to assure that LOFTRAN analyses were performed in such a manner as to assure that error attributable to the modeling approach has been minimized.

NextEra Response

As noted in LAR 261, Attachment 5, Section 2.12.1.2.6, Justification for Exception to Transient Testing, design basis normal condition transients such as a 10% step load increase and decrease, 50% load reduction (large load decrease transient) transients and unanticipated transients such as a normal reactor trip and a turbine trip without a reactor trip from the permissive P-9 setpoint transients were analyzed at EPU conditions. These analyses were performed using the LOFTRAN code at EPU conditions. Acceptable results were obtained from these analyses. LAR 261, Attachment 5, Section 2.4.2.1, Plant Operability (Margin to Trip), describes these analyses.

All design basis normal condition transients analyzed for EPU are "best estimate" analyses and modeled the transient in a way that represents expected plant actual operating conditions and configuration. In general, these analyses assumed plant parameters are at design nominal conditions, all nuclear steam supply system (NSSS) control systems are operable and in the automatic mode of operation, pressure control components are functional, reactivity feedbacks are as expected during normal plant operating conditions, etc. As indicated in Section 2.12.1.2.6, the LOFTRAN code has been verified against data from several plant transients and results showed good agreement between LOFTRAN results and the plant data. LOFTRAN has also been used for the operability as well as for the safety analyses for several other plants for various EPU programs, including the 2-loop, Westinghouse R.E. Ginna plant. The Ginna steam generator tube rupture (SGTR) event was simulated in a version of LOFTRAN and the comparison of the event data with LOFTRAN simulation results was good. The results from Ginna event simulation have also been used as justification for exemption of a large load decrease transient test. Differences between Ginna and PBNP have been included in the LOFTRAN analyses.

RAI SRXB-LTT-4

Licensing Report section 2.12.1.2.6, "Justification for Exception to Transient Testing," Page 2.12-12, Paragraph 4 (numbered list excluded), indicates that the LOFTRAN results are consistent with experience on several similar Westinghouse-designed, 2-loop nuclear power plants that use the LOFTRAN computer code for analysis of Condition I and II initiating events. Please provide additional information to describe how this conclusion was reached and validated: (1) How was the fact that the results are consistent with other Westinghouse plant models established, and (2) How was the assertion that this comparison justifies exception to large transient testing validated?

NextEra Response

As described in LAR 261, Attachment 5, Section 2.12.1.2.5, Transient Analytical Methodology, the NRC safety evaluation for Westinghouse topical report WCAP-7907-P-A describes the LOFTRAN verification process performed by Westinghouse for transients including reactor trip from 100% power, 100% load reduction, and step load changes up to 44% load. The verification process consisted of comparison of LOFTRAN results to actual plant data. As noted in the NextEra response to SRXB-LLT-3, an SGTR event at Ginna was simulated in a version of LOFTRAN and comparison of LOFTRAN results to available plant data from this event further demonstrated the ability of LOFTRAN to analyze the SGTR event.

A LOFTRAN computer model was developed for PBNP at the proposed EPU conditions for best estimate analyses. This computer model simulates the overall thermal-hydraulic and nuclear response of the NSSS as well as various control and protection systems.

The results of the Condition I transients analyses as described in LAR 261, Attachment 5 Section 2.4.2.1, Plant Operability (margin to trip), and Section 2.4.2.2, Pressure Control Component Sizing, indicate that the system dynamic behavior is satisfactory and that no new thermal-hydraulic phenomena or adverse system interactions are created by the proposed EPU. The LOFTRAN analyses used proposed EPU nuclear steam supply system (NSSS) and balance of plant (BOP) settings and setpoints from Section 2.4.1, Reactor Protection, Safety Features Actuation, and Control Systems. The EPU analyses confirmed that departure from nucleate boiling (DNB), RCS pressure and secondary system pressure remain within the allowable design margins and the response to design basis operational transients (i.e., Condition I, including the large load reduction transient) remain acceptable. Based on the above, large transient testing is not necessary since the LOFTRAN simulation showed acceptable plant behavior. Performance of the large load reduction testing would not be expected to provide additional information and would subject the plant to unnecessary perturbations.

Note that the responses to the Equipment Quality Vendor Branch RAIs, EQVB 2.12-2 and EQVB 2.12-3 (Reference 4) provided additional information on power ascension test results performed on a similar plant (Ginna) at EPU conditions. For Ginna, tests were performed at 30% and 100% of EPU power levels. Small load swings, small ramp load changes and a turbine trip test were performed at Ginna. These tests met applicable test criteria. The results from these tests also confirmed LOFTRAN predictions and setpoint studies. The load swing tests verified that pressurizer pressure and level control, rod control and steam generator level control all functioned properly and consistent with the analyses. The turbine trip test fulfilled the purpose of the various control systems including the steam dump control system test performed during original plant startup testing for Ginna. The Ginna tests concluded that analyses performed and resulting prediction for Ginna are confirmed by the test results.

References

(1) NRC electronic mail to NextEra Energy Point Beach, LLC, dated February 25, 2010, DRAFT – Request for Additional Information From Reactor Systems Branch Re: Extended Power Uprate (ML100560283)

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

(3) Westinghouse WCAP-7907 P-A, LOFTRAN Code Description, April 1984

(4) NextEra Energy Point Beach, LLC letter to NRC, dated May 6, 2010, License Amendment Request 261, Extended Power Uprate, Response to Request for Additional Information (ML101270061)