UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 22, 2009

Mr. Larry Meyer
Site Vice President
NextEra Energy Point Beach, LLC
6610 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION FROM REACTOR SYSTEMS BRANCH RE: EXTENDED POWER UPRATE (TAC NOS. ME1044 AND ME1045)

Dear Mr. Meyer:

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated April 7, 2009, as supplemented by letters dated September 11 and October 9, 2009 (Agencywide Documents Access and Management System Accession Nos. ML091250564, ML092570205, and ML092860098), FPL Energy Point Beach, LLC, submitted a request to increase each unit's licensed core power level from 1540 megawatts thermal (MWt) to 1800 MWt reactor core power, and revise the technical specifications to support operation at this increased core thermal power level.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. During a discussion with your staff on December 16, 2009, it was agreed that you would provide the additional information by January 10, 2010.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC's goal of efficient and effective use of staff resources. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-2048.

Sincerely,

Justin C. Poole, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosure:
Request for Additional Information

cc w/encl: Distribution via ListServ

REQUEST FOR ADDITIONAL INFORMATION

POINT BEACH NUCLEAR POWER PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

2.8.5.0-1. Explain why a +1.4°F reactor coolant system (RCS) T_{avg} bias was applied for revised thermal design procedure analyses rather than treated statistically. Explain the effect of this assumption and compare to a statistical treatment of the same conditions.

2.8.5.0-2. Please provide a copy of Reference 6 (page 2.8.5.0-13), Nuclear Safety Advisory Letter NSAL-07-10, "Loss-of-Normal Feedwater/Loss-of-Offsite AC Power Analysis PORV [Pressurizer Power-Operated Relief Valve] Modeling Assumptions," November 7, 2007.

2.8.5.1-1. How were the limiting break sizes, 0.59 ft2 (Unit 1) and 0.63 ft2 (Unit 2), determined for the analyses of Steam System Piping Failures at Full-Power?

2.8.5.1-2. Describe the reactor vessel inlet mixing assumptions, and their bases, used in the analyses of Steam System Piping Failures and other asymmetric cooldown events.

2.8.5.1-3. Since the OPΔT trip function is not qualified for a harsh environment caused by the steamline break, the applicant states that the Hi-1 containment pressure safety injection signal would generate a reactor trip signal before the time credited, in the analyses, for the OPΔT trip signal. What is the basis for this statement? What models and assumptions were used in containment pressure response analyses in order to yield conservatively late Hi-1 containment pressure safety injection signals?

2.8.5.1-4. What are the results of analyses for inside containment cases of Steam System Piping Failures at Full-Power that credit only the low steam line pressure safety injection signal?

2.8.5.1-5. During a full-power steamline rupture-core response event, the main feedwater system flow will increase to match the steam flow until feedwater isolation occurs. It appears that the main feedwater system flow does not increase to match the steam flow, during a no-load steamline rupture-core response event. Describe how such a feedwater system response would affect a 1.4 ft2 no-load steamline break.

2.8.5.1-6. The steamline break analysis discussion states that the core attains criticality before boron solution from the emergency core cooling system (ECCS) and accumulators enters the RCS (actually the core). Tables 2.8.5.1.2-1 and 2.8.5.1.2-2 indicate that the core attains criticality sometime after flow from the ECCS enters the RCS and shortly after the accumulators begin to inject. This indicates that the delivery of boron solution is determined by RCS pressure (not

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ECCS actuation signal delays or pump startup and valve opening times), and comes largely from the accumulators.

a) When does boron solution from the safety injection system enter the core?
b) What would be the result of a smaller steamline break that would not depressurize the RCS to the accumulator injection setpoint?
c) What would be the result of an even smaller steamline break that would not depressurize the RCS to the safety injection system shutoff head?

2.8.5.2-1. In the Loss of Load event discussion, one of the listed acceptance criteria is stated as follows, "An incident of moderate frequency in combination with any single active component failure, or single operator error, is considered an event for which an estimate of the number of potential fuel failures is provided for radiological dose calculations. For such accidents, fuel failure is assumed for all rods for which the departure from nucleate boiling ratio falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model, that fewer failures occur. There is no loss of function of any fission product barrier other than the fuel cladding." Why are there no analyses of anticipated operational occurrences combinations presented to show this criterion has been satisfied?

2.8.5.2-2. Please explain the assumptions regarding auxiliary feedwater (AFW) flow: The AFW flow was initiated 30 seconds after the low-low steam generator water level setpoint was reached; from 30 to 60 seconds, the AFW flowrate ramped from 0 percent to 80 percent of total flow; from 60 to 120 seconds, the AFW flowrate ramped from 80 percent to 100 percent of total flow; beyond 120 seconds, 100 percent of total flow (275 gpm) was maintained.

2.8.5.2-3. If the restrictive acceptance criterion that the pressurizer does not become water solid were used for the Loss of Feedwater event, then why were the PORVs not modeled?

2.8.5.4.5-1. Please explain how, and in which operating modes, the Chemical and Volume Control System is designed to prevent uncontrolled or inadvertent reactivity changes which might cause system parameters to exceed design limits.

2.8.5.6-1. Explain whether transient local and core-wide oxidation values calculated for the large and small break loss-of-coolant accident (LOCA) analyses include pre-transient oxidation. Discuss whether when considering pre-transient oxidation results remain within the 50.46 acceptance criteria.

2.8.5.6-2. Discuss whether oxidation models for ASTRUM and NOTRUMP calculate cladding oxidation on both inner and outer cladding surfaces.

2.8.5.6-3. Provide the result for core-wide oxidation for the small-break LOCA (SBLOCA) analysis.

2.8.5.6-4. The analysis of record SBLOCA analysis predicts a limiting peak cladding temperature (PCT) of 1205°F for the Unit 1 3-inch break. The extended power

uprate (EPU) SBLOCA analysis predicts a limiting PCT of 1103°F. Compare assumed initial conditions, equipment functional capabilities and actuation setpoints to identify the causes of this significant decrease.

2.8.5.6-5. In light of the fact that Nuclear Regulatory Commission (NRC) approval of the SBLOCA analysis approval is required prior to EPU implementation, provide information to justify that any analytic changes effecting the noted reduction in PCT are the direct result of modifications that will be implemented following interim approval of requested EPU-related modifications.

2.8.5.6-6. Explain what/how the SBLOCA analysis reflects revisions requested to Technical Specification (TS) 3.3.2, Function 6.e, TS 3.7.3, TS 3.7.5, and TS 3.7.6. Note that the original description of these change requests provided in the April 7, 2009, application for the EPU did not identify these TS Changes as affecting the SBLOCA analysis.

2.8.5.7-1. List all operator actions credited for the anticipated transient without scram (ATWS) analysis.

2.8.5.7-2. For the period of time 250-300s following the ATWS initiating event when primary system pressure and temperature are decreasing, explain what phenomena hold down reactivity.

2.8.5.7-3. Explain what provides long-term shutdown capability following the ATWS.

2.8.5.7-4. Standard Review Plan Chapter 15.8 indicates that ATWS analytic results should be compared to the LOCA acceptance criteria provided in 10 CFR 50.46 regarding coolable geometry, peak cladding temperature, cladding oxidation, and hydrogen generation. Demonstrate conformance to these acceptance criteria.

2.8.5.7-5. Describe how the loss of normal feedwater event is confirmed to be the limiting ATWS transient. Why is no other ATWS initiator evaluated?

2.8.5.7-6. Explain the RCS flow transient. Describe the cause of the reduction in flow from 50-140 seconds, and provide the cause of the reactor coolant pump trip that follows the initial flow decrease.

2.8.5.7-7. The Licensing Report states, "The ATWS evaluation for EPU assumed a [Point Beach Nuclear Plant] PBNP-specific [moderator temperature coefficient] MTC of -8pcm/°F that bounds 95 percent of the cycle. This value is consistent with that assumed in generic ATWS analyses."

Explain what compensating phenomena and operator actions provide acceptable reductions in risk from ATWS when the MTC is non-bounding of cycle operation.

2.8.5.7-8. Provide a list of initial conditions and plant parameters assumed in the ATWS analysis that differ from Point Beach-specific parameters. Provide the assumed value and compare it to the actual value that would be appropriate for analysis of Point Beach. Where any parameters are non-bounding, justify their use.

2.8.5.7-9. Verify that actual component and actuation setpoint testing supports TS actuation values, and that the TS actuation values are used in the ATWS analyses. Specifically identify any differences between analyzed equipment actuation setpoints and respective TS values.

2.8.5.7-10. Confirm whether TS-permitted equipment out of service assumptions are reflected in the ATWS analyses.

2.8.A-1. During its review of the RAVE implementation for Locked Rotor – Rods in departure from nucleate boiling (DNB) analyses, the NRC staff requested additional information (RAI) concerning the qualification of RAVE analysts (RAI 1), the nodalization of the VIPRE, RETRAN and SPNOVA models (RAI 2), and sensitivity analyses performed to demonstrate the conservatism of assumed voiding present in excess of 30 percent (RAI 7). Confirm that the information presented in response to the NRC staff's previous RAI is applicable to the EPU analyses as well. If this is not the case, justify any differences from the previous response.

2.8.A-2. Condition 3 for implementation of RAVE requires that a plant be licensed to use each of the three constituent codes – VIPRE, RETRAN, and SPNOVA – for safety analyses. Detailed justification was provided for EPU implementation of VIPRE and RETRAN; analogous information was not provided for SPNOVA. Please provide the following information:

a) Has SPNOVA been implemented previously at Point Beach for licensing applications? If so, please describe the application and reference its NRC approval.

b) If SPNOVA has not been implemented previously at Point Beach for licensing applications, provide justification for its implementation that is analogous to that provided for implementation of VIPRE and RETRAN.

T.S.-1. Low pressurizer pressure is credited to terminate the rod cluster control assembly (RCCA) drop accident. Note 2 of Table 2.8.5.0-5 states, "The generic two-loop RCCA drop analysis, which is applicable to PBNP, modeled the low pressurizer pressure reactor trip setpoint as a "convenience trip." The cases that actuated this function assumed dropped rod and control bank worth combinations that were non-limiting with respect to DNB. The fact that the plant-specific low pressurizer pressure reactor trip setpoint is lower than the value assumed in the generic analysis does not invalidate the applicability of the generic two-loop RCCA drop analysis to PBNP. Therefore, the low pressurizer pressure reactor trip setpoint value that was used in the generic two-loop RCCA drop analysis does not represent an analytical limit for this function for PBNP."

a) Please provide an electronic copy of WCAP-11394-P-A.
b) What trip is credited in the DNB-limiting case?
c) What limiting analysis is available to demonstrate the acceptability of a pressurizer low pressure trip setpoint of 1855 psia?

T.S.-2. Are ECCS subsystem boron concentration levels proposed to change as a part of the expedited modifications? If not, explain how compliance with 10 CFR 50.36(c)(2)(ii)(B) is maintained in light of post-LOCA subcriticality analyses that presumably credit the increased TS minimum ECCS boron concentrations.

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ADAMS Accession Number: ML093500203 *per memo dated December 15, 2009

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