From: Poole, Justin
Sent: Thursday, March 04, 2010 3:21 PM
To: 'Hale, Steve'; 'COSTEDIO, JAMES'
Subject: DRAFT - Request for Additional Information from Reactor Systems RE: EPU

Steve

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated April 7, 2009 (Agencywide Documents Access and Management System Accession No. ML091250564), 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 Reactor Systems Branch has reviewed the information provided and determined that in order to complete its evaluation, additional information is required. We would like to discuss the questions, in draft form below, with you in a conference call.

This e-mail aims solely to prepare you and others for the proposed conference call. It does not convey a formal NRC staff position, and it does not formally request for additional information.

Justin C. Poole
Project Manager
NRR/DORL/LPL3-1
U.S. Nuclear Regulatory Commission
(301)415-2048
email: Justin.Poole@nrc.gov
~~~~~~

DRAFT

2.8.2-1. Regarding compliance with Point Beach Nuclear Plant (PBNP) General Design Criterion (GDC) 7, the licensing report (LR) states that the reactor core is designed such that mode operation under Conditions I and II events will not lead to thermo-hydrodynamic instabilities (LR Page 2.8.3-4, Third Bullet). Please explain how compliance with this design criterion is established, addressing the following items:

1. Fundamental mode total power oscillations
2. Xenon oscillations in radial, azimuthal, and diametral modes
3. First overtone mode xenon oscillations
4. Higher mode xenon oscillations

2.8.3-1. Section 2.8.3.2.3.2 states that the section supplements the methodology discussed in Section 2.8.5.3.1, Loss of Forced Reactor Coolant Flow. Please re-confirm this statement.

2.8.3-2. The acceptance criteria for the locked rotor transient are not a part of NRC's regulatory guidance. Please provide a brief statement regarding the technical basis for acceptable performance provided that hot spot peak cladding temperature (PCT) remains below 2700°F and local oxidation remains within 16-percent.
~~~~~~

2.8.3-3. The Safety Analysis Limit (SAL) departure from nuclear boiling ratio (DNBR) is 1.34; however, there is a transient-specific SAL DNBR for the rod withdrawal at power transient. The staff's understanding of the RTDP is that, when a specific transient cannot meet the SAL DNBR, DNBR margin is allocated for that transient. This assures that the available DNBR margin accurately describes safety margin relative to the limiting transient, rather than that relative to a non-limiting transient. Please explain why the development of a transient-specific, SAL DNBR is acceptable and how it conforms to the Revised Thermal Design Procedure (RTDP). If available, please provide examples of similar, precedent applications of the RTDP.

2.8.4-1. Explain how the fluence values, as a function of vessel exposure, for extended power uprate (EPU) without halfnium absorber rods, were determined. Omit any language regarding the use of approved methodology, adherence to regulatory standards, and previously referenced documentation, and address primarily the determination of vessel fluence as a function of exposure (Sections 2.8.4.3 and 2.8.2.2.1).

2.8.4-2. Section 2.8.4.4, Page 3, indicates that residual heat removal (RHR) evaluations were performed assuming that "Decay Heat curves bound current fuel cycles."

1. Do these decay heat curves correspond to a specific American Nuclear Society (ANS) standard? If so, please reference the standard.
2. Please explain the intent of the statement, "bound current fuel cycles," and explain in further detail what a "current fuel cycle" is.

2.8.4-3. RHR evaluations indicate that the plant can transition Modes within Technical Specification (TS)-required times based on an assumption that RHR operation is initiated 6 hours after reactor shutdown. Please explain how the ability to initiate RHR operation 6 hours following reactor shutdown was confirmed or evaluated. (Section 2.8.4.4)

2.8.5.4-1. The assumed maximum rod worth for the subcritical rod cluster control assembly (RCCA) withdrawal error accident is assumed to be 75 percent millirho per second (pcm/sec), while the maximum rod worth for the RCCA withdrawal at-power error (RWAP) event was determined to peak at 80.25 pcm/sec, with bounding analyses performed at 100 pcm/sec. Please explain this discrepancy, and specifically address why a 75 pcm/sec reactivity insertion rate is bounding for subcritical reactor operation. (Section 2.8.5.4.1)

2.8.5.4-2. Section 2.8.5.4.2.2.3 describes the RTDP analyses performed for the RWAP at-power cases, but does not address the analytic methodology for the RWAP pressure cases. Please provide similar information for the RWAP pressure analyses.

2.8.5.4-3. For the RWAP pressure analyses, please explain how VIPRE analyses resulted in a higher DNBR. In light of the differing results between RETRAN and VIPRE, please address also why both results could be considered acceptable.

2.8.5.4-4. DNBR evaluations have been carried out to four significant figures. Address the capability of the analytic methodology to yield credible results to this level of precision. Address the input process, any analytic calculations, treatment of uncertainties, and computer code precision.

2.8.5.4-5. For both units, “the maximum permissibile [reactivity] insertion rate was conservatively limited to 50 pcm/second,” to provide adequate overpressure margin for the rod withdrawal at power event. Please explain this statement. Ensure that the explanation addresses the following questions:

1. Is this an analytic limit?
2. Is this limit carried into the core design?
3. Does this limit constitute either of a lowest assumed functional capability of a structure, system, or component required to mitigate a design basis accident or transient, or an assumed initial condition of a design basis accident or transient analysis?
4. How is this limit enforced?
5. Is there a TS limit on possible reactivity insertion associated with the core or control rod nuclear design?

2.8.5.4-6. If the Dropped RCCA event is analyzed using the RTDP, explain why the LOFTRAN code was used to calculate transient system responses instead of RETRAN. (Section 2.8.5.4.3)

2.8.5.4-7. Explain why RHR flow rate is limited to 6000 gpm for the Mode 5 analysis. The response should address why the value of DLF[1] is capped at 6000 gallons per minute (gpm) for RHR flow rates greater than 6000 gpm. (Section 2.8.5.4.5)

2.8.5.4-8. The reviewer could not locate a disposition for boron dilution in Modes 3 and 4. Please explain. (Section 2.8.5.4.5)

2.8.5.4-9. Regarding the conservative boron dilution rate of 0.9 pcm/sec, please explain whether this is a conservatively high or conservatively low dilution rate. Also, if it is high, explain how the bank RCCA withdrawal analyses provide meaningful information regarding the progression of the dilution transient. (Section 2.8.5.4.5)

2.8.5.4-10. On Page 2.8.5.4.6-3, it is stated, “The ejected rod accident is sensitive to β [delayed neutron fraction] if the ejected rod worth is equal to or greater than β_{eff}, as in the zero-power transients.” Please clarify the reference to rod worth in terms of the delayed neutron fraction.

2.8.5.4-11. The section discussing reactivity weighting factor states, “Reactivity changes were compared and effective weighting factors determined… In this analysis…” Please clarify whether the analysis is a generic study, or a Point-Beach-specific analysis. If it is generic, please discuss its NRC approval status and provide reference information. If the reference predates 1999, please furnish the NRC with an electronic copy. (Section 2.8.5.4.6)

2.8.5.4-12. Provide an illustrative example of the determination of conservative moderator density coefficient curves. Discuss whether the curves are developed on a one-time or cycle-specific basis. If the curves are developed on a one-time basis, also discuss whether the conservatism of the curves is confirmed on a cycle-specific basis. (Section 2.8.5.4.6)

2.8.5.4-13. The licensing report states (Page 2.8.5.4.6-8) that “generic analyses demonstrate that the fission product release as a result of fuel rods entering

[1] DLF is a process variable, not an initialism.

DNB is limited to less than 10-percent of the fuel rods in the core." Please summarize these analyses and discuss their applicability to Point Beach Nuclear Plant.

2.8.5.5-1. Regarding the pressurization effects of the postulated chemical and volume control system (CVCS) malfunction, please provide the following additional information:

2.8.5.5-2. Provide a calculation to quantify the "several minutes" time frame available for operator action to terminate the CVCS flow.

2.8.5.5-3. Address the operator ability to respond within the calculated time frame.

2.8.5.5-4. Demonstrate the efficacy of automatic actions to respond to this event. If the power operated relief valves (PORVs) are anticipated to open, address the ability of the valves to re-seat, thus preventing this evolution from turning into a more serious transient. (Section 2.8.5.5)

2.8.5.6-7. Acceptance criteria discussed for the inadvertent pressurizer pressure relief valve opening include PBNP GDC 6, as it pertains to controls and protection systems allowing the core to function throughout its design lifetime without exceeding acceptable fuel damage limits. However, the disposition for the postulated event draws on the small break LOCA analysis. The small break LOCA analysis does not demonstrate conformance to commonly accepted fuel damage limits for Condition II events; rather it demonstrates conformance to acceptance criteria specified in 10 CFR 50.46. Please reconcile this difference, provide appropriately specified acceptable fuel damage limits, and provide a clarified analysis or disposition, as appropriate, for this transient.

2.8.5.6-8. NOTRUMP evaluations appear to have been carried out for extended periods of time, which provides assurance that all phases of the small break LOCA have been evaluated relative to the adequacy of the makeup systems to demonstrate that the core trends to, and remains in, a liquid covered configuration. Please provide similar assurances for the large break LOCA, which was evaluated only for the first 8-10 minutes of the transient.

2.8.5.6-9. Please provide the minimum capacity of the condensate storage tank.

2.8.5.6-10. Please provide the maximum capacity of the Boron Injection Tanks and maximum boric acid concentration.

[1] DLF is a process variable, not an initialism.