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From: Tam, Peter
Sent: Thursday, June 07, 2012 8:54 AM
To: 'james.costedio@nexteraenergy.com'
Cc: Purtscher, Patrick; Beltz, Terry; Frankl, Istvan; Rosenberg, Stacey
Subject: Point Beach Units 1 and 2 - Draft RAI on the Reactor Vessel Internals Inspection Plan (TAC ME8235 and ME8236)

Mr. Costedio:

By letter dated December 19, 2011, NextEra Energy Point Beach, LLC, submitted an inspection plan for the reactor vessel internals (RVI) components at Point Beach Nuclear Plant (PBNP), Units 1 and 2. The Nuclear Regulatory Commission (NRC) staff has reviewed the inspection plan and found that additional information, as depicted in the draft RAI below, is needed for it to complete its review. You may accept this e-mail as formal RAI and formally respond to the following questions within 45 days, or you may request to discuss the questions and an alternative response date with the staff in a conference call. If you desire such a conference call, please arrange it with me, or arrange it with Terry Beltz, the Point Beach project manager, upon his return to the office.

Thank you..

Peter S. Tam, Senior Project Manager
(for D.C. Cook and Monticello)
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Tel. 301-415-1451

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RAI-1

Applicant/Licensee Action Item 1 from the NRC staff's final safety evaluation (SE) of MRP-227-A, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," requires that applicants/licensees submit an evaluation that demonstrates that their plant is bounded by the assumptions regarding plant design and operating history that were made in the failure modes, effects and consequences analyses (FMECA) and functionality analyses for reactors of their design.

The licensee's response to Applicant/Licensee Action Item 1 in the RVI inspection plan addresses the core loading assumptions (switch to a low-leakage core) and operational (base loaded plant) aspects of design and operation that are mentioned in MRP-227-A, Section 2.4. An additional assumption listed in Section 2.4 of MRP-227-A is that there have been no design changes to the RVI beyond those identified in general industry guidance or recommended by the original vendors. Section 2.4 of MRP-227-A indicated that these assumptions are considered to represent any U.S PWR operating plant provided that these three assumptions are met, given the information on design and operation known to the MRP as of May 2007.

MRP-191, Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR [pressurized water reactor] Designs," (proprietary document), documents the screening for susceptibility to aging effects, the FMECA results, and the categorization and ranking of the RVI components. In addition to the assumptions listed in Section 2.4 of MRP-227-A, MRP-191 documents additional assumptions that were used. In particular, neutron fluence range, temperature, and material grade for each generic component of the Westinghouse design internals were used for input to the screening process. These values were determined based on an "expert elicitation" process. Stress values were not explicitly tabulated, but were recorded as either above the stress threshold (>30 ksi) or not based on the expert interviews.

MRP-232, Revision 0, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals," (proprietary document) reported more specific stress, temperature and neutron fluence values based on finite element analyses for selected high consequence of failure components identified in MRP-191.

The EPRI-MRP did not verify that the values of fluence, temperature, stress, and material, documented in MRP-191 and MRP-232 were bounding for all individual plants, and in fact MRP-227 states, "These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter."

The NRC staff expects that the licensee should have access to design information enabling verification that the material for each RVI component is bounded by the design assumptions of the MRP. In this context, the NRC staff requests that the licensee provide the following information:

(1) Describe the process used to verify that the RVI components at PBNP, Units 1 and 2 are bounded by the assumptions regarding the variable (i.e., neutron fluence, temperature, stress values, and materials) that were made for each component in the FMECA and functionality analyses supporting the development of MRP-227-A.

(2) To provide reasonable assurance that the RVI components are bounded by assumptions in the FMECA and functionality analyses supporting the development of MRP-227-A, the licensee is requested to respond to either part a) or part b) of this RAI:

(a) Provide the plant-specific values of neutron fluence (n/cm^2, E>1.0 MeV), temperature, stress, and materials for a sample of RVI components. The components selected should represent a range of neutron fluences, and temperatures. This information should identify whether the stress is greater or less than 30 ksi. Values of neutron fluence and temperature may be estimated or analytical values. The values should be the peak values of each parameter for each component (e.g., peak end-of-life value for fluence). Provide the method used to estimate the values, or describe the analysis method. An acceptable sample of components is:

i) Lower Core Plate

ii) Core Barrel Flange

iii) Barrel-Former Bolts

iv) Upper Core Barrel Welds

v) Lower Core Barrel Welds

vi) Upper Core Plate Alignment Pins

(b) Provide a qualitative assessment regarding the differences between the plant-specific variables (neutron fluence, temperature, stress values, and materials) and the variables of a "representative" PWR vessel used in developing the MRP-227-A report, for those components listed in part a) or for those components that are either identified as "Expansion" or were scoped out in the FMECA.

(3) If there are any components at PBNP, Units 1 and 2 not bounded by assumptions regarding neutron fluence, temperature, stress or material used in the development of MRP-227-A, describe how the differences were addressed in the plant-specific RVI Inspection Plan. The NRC staff requests that the licensee, as a part of its demonstration, discuss whether there would be any changes to the screening, categorization, FMECA process and functionality analyses if the plant-specific variables (the neutron fluence, temperature, stress values, plant-specific operating experience, and materials) are used. This evaluation should address whether additional aging mechanisms would become applicable to the component.

(4) For any non-bounded components, determine if any changes to the inspection requirements of MRP-227-A are needed. Provide either plant-specific inspection requirements, an alternate aging management program (AMP), or if no changes to the inspection requirements are proposed, provide a justification for the adequacy of the existing MRP-227-A inspections for the unbounded components.

RAI-2

The licensee has listed several plant-specific examples of operating experience related to the aging degradation in the RVI components. The staff requests more specific information related to these examples. For the baffle-former bolts that were examined in 1997, what was the total number of baffle-former bolts in Unit 2, how many bolts were examined, and how many were cracked? Did the cracked bolts conform to any pattern related to neutron exposure? How many were replaced, and did the replacement guarantee the structural margins? What was the original material and the replacement material in 1997? Describe the planned inspections. Is the inspection procedure that is planned to be used in the future at PBNP the same as that used in the 1997 bolt inspections?

MRP-51 lists in Table 2-10 and 2-14 the measured irradiation conditions, temperature and fluence, for samples taken from the 1997 inspections. Given these values for the irradiation conditions in 1997, provide estimated irradiation conditions for the most susceptible baffle-former bolts from Unit 1 in 2013, and estimated irradiation conditions for 2015 that represent the most susceptible of the Unit 2 bolts that were replaced in 1997 as well as the most susceptible of the Unit 2 bolts that were not replaced in 1997.

RAI-3

The NRC staff requests the license discuss the results from the existing programs (Attachment D) and the extent of aging degradation (if any) that occurred thus far in the following components at PBNP:

(a) baffle-edge bolts,

(b) clevis insert bolts,

(c) flux thimble tubes,

(d) core barrel bolting, and

(e) thermal shields.

RAI-4

Historically, the following materials used in the PWR RVI components were known to be susceptible to some of the aging degradation mechanisms that are identified in the MRP-227-A report. In this context, the NRC staff requests that the licensee confirm that these materials are not currently used in the RVI components at PBNP. If they are used in any PBNP RVI components, please identify any service history associated with the components that is not captured in the response to RAI 2.

(1) Nickel base alloys—Inconel 600; Weld Metals—Alloy 82 and 182 and Alloy X-750

(2) Alloy A-286 ASTM A 453 Grade 660, Condition A or B

(3) Stainless steel type 347 material (excluding baffle-former bolts)

(4) Precipitation hardened (PH) stainless steel materials—17-4 and 15-5

(5) Type 431 stainless steel material

RAI-5

MRP-227-A provides general descriptions for the examination coverage for several primary components where the licensee must make specific decisions to complete the required exams. For example, the licensee is required to inspect 20% of the CRGT guide card assemblies per Attachment B of the submittal. For the CRGT lower flange welds, 100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies is the exam coverage. For the core barrel assembly welds, MRP-227-A requires 100% of one side of the accessible surfaces. The NRC staff requests that the licensee provide specifics on what will be included in the examination with an explanation for the selection process, which should include the following aspects:

(1) most susceptible areas to experience aging degradation,

(2) high stress areas,

(3) accessibility issues and,

(4) plant-specific operating experience.

RAI-6

In Attachment D for existing programs in the December 15, 2011, submittal, the licensee stated that the flux thimble tubes were last examined in the spring of 2010 for Unit 1 and the fall of 2009 for Unit 2, referencing NUREG-1801, Revision 1. The staff does not see any specific guidance in the reference for selecting the inspection frequency. What is the inspection frequency for the flux thimble tubes at Point Beach, Units 1 and 2?

The staff also requests that the licensee update their aging management program (AMP) to reference NUREG-1801, Revision 2 (GALL) and describe how their plant-specific AMP, LR-AMP-006-TTI, "Thimble Tube Inspection Program Basis Document" compares to the XI.M37, "Flux Thimble Tube Inspection" AMP in the GALL report.