November 22, 2010

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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 No. 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Amendment Request 265
Revision to the Reactor Vessel Head Drop Methodology
Response to Request for Additional Information

References:
(1) NextEra Energy Point Beach, LLC letter to NRC, dated June 1, 2010, License Amendment Request 265, Revision to the Reactor Vessel Head Drop Methodology (ML101520200)
(2) NRC letter to Nuclear Energy Institute, dated September 5, 2008, Safety Evaluation Regarding NEI 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0 (ML082410532)
(3) NRC letter to NextEra Energy Point Beach, LLC, dated October 25, 2010, Point Beach Nuclear Plant, Units 1 and 2 - Request for Additional Information Re: License Amendment Request Associated with a Revision to the Reactor Vessel Head Drop Analysis Methodology (TAC Nos. ME4006 and ME4007) (ML102870204)

NextEra Energy Point Beach, LLC (NextEra) submitted License Amendment Request 265, Revision to the Reactor Vessel Head Drop Methodology (Reference 1), pursuant to 10 CFR 50.90. The proposed amendment consists of revising the current license basis regarding a postulated reactor vessel head (RVH) drop event to conform to the NRC-endorsed guidance of Nuclear Energy Institute (NEI) 08-05 (Reference 2).

Via Reference (3), the NRC staff determined that additional information was required to enable the staff's continued review of the request. Enclosure 1 provides the NextEra response to the staff's request.

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 application with enclosures is being provided to the designated Wisconsin Official.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on this day, November 22, 2010.

Very truly yours,

NextEra Energy Point Beach, LLC

FVR

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 265
REVISION TO THE REACTOR VESSEL HEAD DROP METHODOLOGY
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

The NRC staff determined that additional information was required (Reference 1) to enable the Mechanical and Civil Engineering Branch to complete the review of License Amendment Request 265, Revision to the Reactor Vessel Head Drop Methodology (Reference 2). The following information is provided by NextEra Energy Point Beach, LLC (NextEra) in response to the NRC staff's questions.

EMCB RAI 1

As described in Section 4.3 of Enclosure 2 to Reference 2, bottom mounted instrumentation (BMI) conduits 32 and 29 are the only conduits modeled in the ANSYS analyses used to support the conclusion in this license amendment request (LAR) that the BMI conduits will maintain their structural integrity under a postulated reactor vessel head (RVH) drop. While it is recognized by the U.S. Nuclear Regulatory Commission (NRC) staff that BMI conduits 32 and 29 geometrically bound the other conduits (i.e., they are the shortest and longest conduits, respectively), no information was provided which indicates that there would be no adverse affects on these conduits due to the structural interactions with the other 34 conduits attached to the reactor vessel. By analyzing conduits 32 and 29 individually, other interactions between the 34 conduits absent in the analyses may not be adequately accounted for in the present structural analyses.

Please provide justification which demonstrates that modeling only these two conduits, without loads resulting from possible interactions from adjacent conduits, provides a bounding structural evaluation.

NextEra Response

The bend radii of all of the bottom-mounted instrument (BMI) conduits, including conduits 32 and 29, is specified to be 8'-0". The conduits, which are depressurized at the time of a reactor vessel head (RVH) lift/set evolution, have a nominal inside diameter of 3/8", and a wall thickness of 0.281 to 0.344". This is comparable to 3/4" double extra strong piping. As such, the thick-walled conduits are not susceptible to pressure boundary damage from incidental conduit-to-conduit contact.

The conduits are each supported by common racks that maintain conduit-to-conduit spacing at periodic intervals. The support nodes shown in Reference (3), Enclosure 2, Figure 4-2 (conduit 32) have corresponding support nodes on conduit 29 and each of the other conduits. These support racks were conservatively assumed to be rigid in the analysis to maximize the reaction loads and moments. The support frames for the conduits that are located in the vertical rise furthest from the reactor vessel are rigidly mounted to the containment structure. The

support frames represented by the other nodes in the model are mounted on spring supports that provide flexibility and accommodate limited relative motion between the reactor vessel and the seal table. This was conservatively neglected in the stress analysis model, and all support frames were assumed to be rigidly mounted. This support system ensures that all conduits will move in phase in response to a common excitation, such as downward displacement of the RVH in response to a reactor head drop impact.

As depicted in Reference (3), Enclosure 2, Figure 4-1, and as confirmed in plant configuration control documentation, conduits 29 and 32 represent both the longest and shortest conduits, respectively. Given the common routing and supporting, the maximum stresses generated in these two conduits reasonably bound the stresses that can be expected to be generated in each of the conduits, regardless of total length.

The high stress locations correspond to the point where the conduits are welded to the reactor vessel nozzles. The conduits are constrained against rotation in all axes, and against translation in the two horizontal axes at this node. Only vertical translation is permitted. This results in relatively high bending moments and gives rise to the limiting results.

In contrast, the conduits are free to rotate at the other support points, as discussed in Reference (3), Enclosure 2, Assumption 3, thus alleviating potentially high bending moments.

Based on the above, NextEra concluded that incidental conduit-to-conduit contact, as may be postulated to occur due to mid-span vibration, will not result in unacceptable stresses in the thick-walled conduits.

EMCB RAI 2

Section 4.6 of Enclosure 2 to Reference 2 indicates that the material data used as an input to the ANSYS model used to demonstrate the structural integrity of the BMI conduits is based on true stress-strain data. The guidance found in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and in Section 2.3.2 of Nuclear Energy Institute (NEI) Report 08-05, "Industry Initiative on Control of Heavy Loads," stipulates that the use of true stress-strain data is appropriate for these analyses. However, the construction of true stress-strain curves used in these analyses can be based on a number of factors; these are also discussed in Section 2.3.2 of NEI 08-05.

Please address the following items regarding the true stress-strain curve used for the Point Beach Nuclear Plant RVH drop methodology LAR under consideration:

a) *A temperature of 70 degrees Fahrenheit has been assumed to develop the true stress-strain data curve for the material model.*

   Please provide justification for the use of this temperature as it relates to actual operating parameters when the accident is assumed to occur.

b) In Reference 2, Enclosure 2, the Table 4-1 data points used to construct the true stress-strain curve in Figure 4-6 were gathered from Westinghouse Calculation Note, CN-RCDA-04-46, Rev. 1, "Weld Overlay -Material Properties," dated June 19, 2006 (Reference 5 of Enclosure 2).

Please discuss the methodology used in this calculation to develop these data points. This discussion should focus on the development of the material model curve and the justification for the use of this model as it relates to the guidance found in NEI 08-05.

c) Please indicate whether a dynamic increase factor (DIF) has been incorporated into the true stress-strain curve found in Figure 4-6 of Enclosure 2 to Reference 1. If a DIF has been applied to the curve, please provide technical justification for the DIF(s) applied to the curve.

d) Please confirm that the curve found in Figure 4-6 is not extrapolated further for the ANSYS analysis and that ultimate failure strain for this model is 0.26 in/in, as indicated in Table 4-1.

NextEra Response

a) During head lift/set activities, the reactor is in MODE 6, requiring the reactor coolant system (RCS) temperature to be less than 200°F. Temperatures of approximately 100°F are typical to keep the work area temperatures and humidity tolerable. This has been confirmed by a review of the RCS temperatures during the last refueling outage for each unit.

Minimum specified material properties are typically documented in codes and standards at 70°F, with maximum allowable stresses and modulus of elasticity tabulated in 100°F increments starting at 200°F. At the low temperatures prevailing during RVH lift/set, the modulus of elasticity, yield strength and ultimate tensile strength (all used in deriving the true stress-strain curve as described below) would not vary significantly from those published for 70°F.

b) The true stress-strain data from Westinghouse Calculation Note CN-RCDA-04-46, Revision 1, "Weld Overlay -Material Properties," was used to develop the stress-strain curve shown in Figure 4-6 of Reference (3), Enclosure 2. The methodology used to develop these data points follows.

Development of True Stress-Strain Data Points

Engineering stress-strain data points at room temperature (70°F) were established first. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, 2001 Edition through 2003 Addenda, was used to determine the ASME Code minimum yield (Sy) and ultimate strength (Su) values of 30 ksi and 75 ksi, respectively. To provide typical yield stress (YS) and ultimate strength (TS) values, the Code minimum yield strength value was increased by 20% to 36 ksi and the Code minimum ultimate strength value of 75 ksi was increased by 10% to 82.5 ksi. Those typical values were used as the basis for the engineering stress-strain data points.

The first engineering stress-strain data point is 90% of the typical yield stress, or 0.9(YS), defined at a strain of e_{PL} = 0.9(YS)/E. The second engineering stress-strain data point is the yield stress (YS), defined at a strain = 0.002 + (YS)/E. An elastic modulus of 28,300 ksi at 70°F was selected from the ASME B&PV Code. The last point is the typical ultimate stress (TS), defined at a strain = uniform elongation. Uniform elongation trends were established using elongation data for annealed Type 316 stainless steel from the Nuclear Systems Materials Handbook, Volume 2, Part I. An engineering uniform elongation of 40% (0.40) at 70°F was selected. Interim points that define the stress-strain curve were selected using engineering judgment to fit the general profile of stress-strain curves in the plastic region for ductile metals.

Once engineering stress-strain data points were determined, the true stress-strain data points were calculated using the following equations:

True Strain = ln (1+Engineering Strain)
True Stress = Engineering Stress x (1+Engineering Strain)

Justification for Use of True Stress-Strain Data Relative to NEI 08-05

Nuclear Energy Institute (NEI) 08-05 (Reference 4) requires that minimum code/specification, representative or actual test data yield and ultimate strength values be used to develop the material stress-strain curves. The NEI guidelines state, "It is acceptable to use curves developed from test data, which have similar engineering strengths and elongation as the average of engineering strengths and elongation from the component code or specification. As an alternative, it is acceptable to use true stress-strain curves for similar materials that have been modified to match the code or specification minimum properties for yield stress, ultimate stress and minimum elongation. . . . Where multiple tests are available, the minimum values for both stress and strain should be used."

As stated above, the subject BMI analysis used a true stress-strain curve that was developed with typical yield and ultimate strength values for Type 304 stainless material. These typical values are based on a large sampling of actual test data. ASME Code materials typically exhibit yield and ultimate strengths that are at least 5 ksi higher than the Code minimum values. Therefore, although the curve used in the analysis is not based on certified material test reports specific to the PBNP BMI conduit material, it is a very reasonable representation of the actual material behavior. In addition, the faulted stress intensity limits used to evaluate the stresses resulting from the analysis were conservatively based on Code minimum strength values.

c) As described in the NextEra response to EMCB RAI 2.b) above, a dynamic increase factor (DIF) was not incorporated into the true stress-strain curve depicted in Reference (3), Enclosure 2, Figure 4-6.

d) The curve described by Table 4-1 and Figure 4-6 of Reference (3), Enclosure 2, is confirmed to be the true stress-strain curve used in the ANSYS analysis. Based on the engineering stress-strain data and equations provided in the NextEra response to EMCB RAI 2.b) above, the final point of the curve depicted in Figure 4-6 of the analysis should actually terminate at a true ultimate strain of 0.337 in/in and a corresponding true ultimate stress of 115.5 ksi. However, the curve in Figure 4-6 is a truncated version of the actual stress-strain curve, ending at a true strain of 0.26 in/in and a true stress of 104.64 ksi because of an ANSYS limitation on the number of data points.

Based on a maximum allowable stress (Membrane plus Bending) from the analysis of 67.5 ksi, the corresponding strain is ~0.08 in/in, which is well within the limits of the portrayed curve.

References

(1) NRC letter to NextEra Energy Point Beach, LLC, dated October 25, 2010, Point Beach Nuclear Plant, Units 1 and 2 - Request for Additional Information Re: License Amendment Request Associated with a Revision to the Reactor Vessel Head Drop Analysis Methodology (TAC Nos. ME4006 and ME4007) (ML102870204)

(2) NextEra Energy Point Beach, LLC letter to NRC, dated June 1, 2010, License Amendment Request 265, Revision to the Reactor Vessel Head Drop Methodology (ML101520200)

(3) NextEra Energy Point Beach, LLC letter to NRC, dated July 9, 2010, License Amendment Request 265, Revision to the Reactor Vessel Head Drop Methodology, Supplement 1 (ML102030115 and ML102030116)

(4) NRC letter to Nuclear Energy Institute, dated September 5, 2008, Safety Evaluation Regarding NEI 08-05, "Industry Initiative on Control of Heavy Loads," Revision 0 (ML082410532)