August 25, 2021

L-2021-154
10 CFR 54.17

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
11545 Rockville Pike
One White Flint North
Rockville, MD 20852-2746

Point Beach Nuclear Plant Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

SUBSEQUENT LICENSE RENEWAL APPLICATION - AGING MANAGEMENT REQUESTS FOR CONFIRMATION OF/ADDITIONAL INFORMATION (RCI/RAI) SET 3 RESPONSES

References:

1. NextEra Energy Point Beach, LLC (NEPB) Letter NRC 2020-0032 dated November 16, 2020, Application for Subsequent Renewed Facility Operating Licenses (ADAMS Package Accession No. ML20329A292)
2. U.S. Nuclear Regulatory Commission (NRC) Letter dated January 15, 2021, Point Beach Nuclear Plant, Units 1 and 2 - Determination of Acceptability and Sufficiency for Docketing, Proposed Review Schedule, and Notice of Opportunity to Request a Hearing Regarding the NextEra Energy Point Beach, LLC Application for Subsequent License Renewal (EPID No. L-2020-SLR-0002) (ADAMS Accession No. ML21006A417)
3. NRC Letter dated January 15, 2021, Point Beach Nuclear Plant, Units 1 and 2 - Aging Management Audit Plan Regarding the Subsequent License Renewal Application Review (ADAMS Accession No. ML21007A260)
4. US Nuclear Regulatory Commission Meeting with NextEra Energy Concerning the Point Beach Subsequent License Renewal Application Review – June 3, 2021 Public Meeting (ADAMS Accession No. ML21148A116)
5. NRC Email and Attachment dated July 27, 2021, Point Beach SLRA RAIs Set 3 Final (ADAMS Accession Nos. ML21214A083, ML21214A084)

NEPB, owner and licensee for Point Beach Nuclear Plant (PBN) Units 1 and 2, has submitted a subsequent license renewal application (SLRA) for the Facility Operating Licenses for PBN Units 1 and 2 (Reference 1). On January 15, 2021, the NRC determined that NEPB's SLRA was acceptable and sufficient for docketing (Reference 2), and on January 15, 2021 issued the regulatory audit plan for the aging management portion of the SLRA review (Reference 3). Based on the information exchanged and discussions held during the public meeting held on June 3, 2021 (Reference 4), the NRC issued its Set 3 RAIs to NEPB (Reference 5). The attachments to this letter provide responses to those information requests.

For ease of reference, the index of attached information is provided on page 3 of this letter.

Should you have any questions regarding this submittal, please contact me at (561) 304-6256 or William.Maher@fpl.com.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 25th day of August 2021.

Sincerely,

William D. Maher
Licensing Director - Nuclear Licensing Projects

Cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
Public Service Commission Wisconsin

Attachments Index		
Attachment No.	RCI/RAI No.	Subject
1	RCI 3.5.2.2.2.7-3	Reactor Vessel Support Structures Fracture Mechanics Evaluation
2	RAI B.2.2.1-2	Fatigue Monitoring

SLRA Section 3.5.2.2.2.7, "Expected Further Evaluation for Loss of Fracture Toughness Due to Irradiation Embrittlement of Reactor Vessel (RV) Supports for NRC Review of the First Three SLRAs"

RCI 3.5.2.2.2.7-3

Information

Based on the audit review of Bechtel Specification No. 6118-C-10, "Specification for Detailing, Fabrication and Delivery of Major Component Support Structures for the Point Beach Nuclear Plant Unit 1. Wisconsin Michigan Power Company. Bechtel Corporation, San Francisco CA for Westinghouse Electric Corporation Atomic Power Divisions," Revision 2, the staff noted that the welding shall be performed in accordance with paragraph 14.7. The staff also noted that this would indicate that the weld strength is lower than the T-1 steel material strength.

Request

Confirm that all welding of T-1 steels analyzed in the fracture mechanics evaluation of the reactor vessel steel support structures in SLRA FE 3.5.2.2.2.7 were performed in accordance with paragraph 14.7 as specified in Bechtel Specification No. 6118-C-10.

NEPB Response:

Westinghouse confirms that all welding of T-1 steels analyzed in the fracture mechanics evaluation of the reactor vessel steel support structures in SLRA Section 3.5.2.2.2.7 was performed in accordance with paragraph 14.7 as stated above. For T-1 steels, Westinghouse also confirms that the weld strength is lower than the T-1 steel material strength.

References:

None.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.

SLRA Section B.2.2.1, "Fatigue Monitoring"

RAI B.2.2.1-2

Regulatory Basis

10 CFR 54.21(a)(3) requires an applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. One of the findings that the NRC staff must make to issue a renewed license (10 CFR 54.29(a)) is that actions have been identified and have been or will be taken with respect to managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis. In order to complete its review and enable it to make a finding under 10 CFR 54.29(a), the staff requires additional information in regard to the matters described below.

Background

The "monitoring and trending" program element of GALL-SLR AMP X.M1, in part, indicates that the monitoring of plant operating conditions or water chemistry parameter conditions is performed to provide inputs for components with stress-based fatigue calculations or environmental fatigue calculations.

In comparison, NextEra (the applicant) used the FatiguePro computer program to perform stress-based fatigue monitoring and analysis for certain ASME Code Class 1 and 2 components, as described in Structural Integrity Associates, Inc. (SIA) Calculation Package No. 2000088.303, Revision 0. The SIA report and implementing procedure (NP 7.7.19, Attachment B) for fatigue monitoring indicate that the stress-based method is used to monitor and evaluate the fatigue for certain Class 1 and 2 components such as steam generator feedwater nozzles, pressurizer shells, and hot leg nozzles for pressurizer surge lines. The method monitors operating conditions such as flow rates and temperatures.

SLRA Section B.2.2.1 also indicates that the applicant contracted with SIA to review the FatiguePro program and received an updated report that incorporated FatiguePro data through 2019 for 80-year life projections. The SLRA further states that, going forward, the applicant will no longer use FatiguePro and instead will use the manual count method.

Issue

The staff needs additional information to clarify why the manual count method is sufficient for the components, for which the stress-based fatigue monitoring has been used, given that the manual count method is mainly for counting certain pre-determined transient cycles (e.g., heatup and cooldown cycles) but may not be sufficient to monitor complex temperature or pressure variations for the components (e.g., thermal stratification transients).

Request

Discuss why the manual count method is sufficient for the components for which the stress-based fatigue monitoring has been used. As part of the response, clarify (1) whether the monitoring data previously collected from stress-based fatigue monitoring will be used as basis information for fatigue monitoring (e.g., monitoring data from pressurizer surge lines) and (2) whether the system delta T of the pressurizer surge line (temperature difference between pressurizer and reactor coolant system hot leg) will be monitored to ensure that the system delta T does not exceed a certain limit, consistent with the following reference (Reference: Section 2.4.2 of WCAP-13510, "Structural Evaluation of the Point Beach Units 1 & 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," October 1992, ADAMS Accession No. ML20198D998).

NEPB Response:

1. Monitoring data previously collected from stress-based fatigue (SBF) monitoring does not need to be used as basis information for fatigue monitoring. The monitoring data has been extensively reviewed and the idealized, design transient definitions have been demonstrated to be significantly more conservative. The idealized insurge/outsurge (I/O) and stratification transients that were developed in response to NRC Bulletin 88-11, and documented in the Analysis of Record (AOR) WCAP-13510, were used for the analysis of the hot leg surge nozzle in SIA calculation 2000088.310P. These transient definitions were conservatively defined to bound plant-specific operating practices. Since the I/O and stratification cycles with the most fatigue significance occurs during plant heatups and cooldowns when the pressurizer to hot leg temperature difference is the highest, the I/O and stratification sub-cycles from WCAP-13510 were prorated with the design number of RCS heatup and RCS cooldown transients for the analysis set used (details are described in SIA calculation 2000088.310P). Therefore, manual counts of the global RCS heatup and RCS cooldown transients will be compared to the analyzed number of these transients in the SIA calculation 2000088.310P to demonstrate acceptable fatigue margin.
2. The loading analyzed in SIA calculation 2000088.310P, which is consistent with the AOR documented in WCAP-13510, conservatively postulates a maximum system delta T of over 300°F. In practice, a review of plant data for Point Beach Units 1 and 2 collected over a period of approximately 12 years has demonstrated that the maximum system delta T during RCS heatups and cooldowns remains well below 300°F, as a benefit of precautions and limitations contained within plant start-up and shutdown operating procedures OP-3C Unit 1 and OP-3C Unit 2. The maximum delta T, as reflected by measured plant data, is typically in the range of 160°F to 220°F, resulting in very low usage values when analyzed using Stress-Based Fatigue methods. Also, the maximum temperature difference is uniformly and significantly less than the highest temperature differences (over 300°F) analyzed in the top four I/O and stratification transients used in SIA calculation 2000088.310P. Thus, it is unnecessary to monitor the system delta T, since plant operating procedures have been demonstrated to be effective in providing

reasonable assurance the AOR's conservative transient definitions will not be exceeded.

References:

None.

Associated SLRA Revisions:

None.

Associated Enclosures:

None.