US Nuclear Regulatory Commission Meeting with NextEra Energy Concerning the Point Beach Subsequent License Renewal Application Review – June 3, 2021, Public Meeting

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Discussion Questions Regarding Irradiated Reactor Pressure Vessel Concrete and Steel Supports and Associated Ageing Management Activities

Regulatory Basis (applies to all questions)

Title 10 of the Code of Federal Regulations (CFR) Section 54.21(a)(1) requires license renewal applicants to perform an integrated plant assessment (IPA) and their application to identify and list systems, structures, and components (SSCs) that are within the scope of license renewal and subject to aging management review (AMR). Further, 10 CFR 54.21(a)(3) requires for the SSCs identified to be subject to AMR, the applicant demonstrate that the effects of aging will be adequately managed such that their intended functions are maintained consistent with the current licensing basis (CLB) for the subsequent period of extended operation. To complete its review and enable the staff making a reasonable assurance finding on functionality of reviewed SSCs for the subsequent period of extended operation consistent with 10 CFR 54.21, the staff requires under 10 CFR 54.29(a) additional information be provided regarding the matters described below.

Question 1

Background

Subsequent License Renewal Application (SLRA) Section 3.5.2.2.2.6 of the Point Beach Nuclear Plant Units 1 and 2 (PBN), NextEra Energy Point Beach, LLC (NextEra or the applicant) presents an evaluation of the irradiation effects of the biological shield wall for the specified period of extended operation to ensure it will maintain structural integrity and not affect the primary shield wall under design basis loading conditions. However, NRC staff observed that this evaluation is based on neutron fluence and gamma dose results for which the uncertainty has not been assessed. The effect of neutron fluence and gamma dose on the biological shield wall concrete may in fact be greater with consideration of this uncertainty. While no method, generic or specific to Point Beach Nuclear, has been approved by the NRC for calculations of exposure for the biological shield wall and primary shield wall concrete, the calculations for neutron fluence and gamma dose have generally been found acceptable in prior reviews on the basis that the uncertainty in the calculations necessary for the results to exceed the exposure levels of concern identified in NUREG-2192 is substantial (e.g., 200%). In the present evaluation, the neutron fluence and gamma dose for concrete already exceed the NUREG-2192 damage thresholds. The staff is not able to determine whether reasonable assurance exists that the limiting neutron fluence and gamma dose values for concrete were identified with sufficient margin and conservatism to accommodate uncertainties in the fluence analysis methodology associated with calculating exposure at an ex-vessel location.

<u>Issue</u>

In order to ensure the reduction of strength and mechanical properties of concrete due to irradiation are adequately managed, it is necessary to assess the uncertainty associated with the neutron fluence and gamma dose results.

<u>Request</u>

Provide an estimate of the uncertainty associated with the neutron fluence and gamma dose results at the surface of the biological shield wall.

Question 2

<u>Background</u>

SLRA Section 3.5.2.2.2.7 presents an evaluation of the irradiation effects on loss of fracture toughness of the reactor vessel (RV) support steel (ring girder and support columns) for the specified period of extended operation to ensure it will maintain its structural integrity. However, NRC staff observed this evaluation is based on neutron fluence and displacements per atom (dpa) results for which the uncertainty has not been assessed. The results of the structural analysis and fracture mechanics evaluation indicate that several RV steel support structure components possess a small amount of margin to interaction ratios and allowable flaw sizes, respectively. The effect of neutron fluence and dpa on the RV support steel with consideration of the uncertainty may cause margins to be exceeded. The staff is not able to determine whether reasonable assurance exists that the limiting neutron fluence and dpa values for the RV support steel were identified with sufficient margin and conservatism to accommodate uncertainties in the fluence analysis methodology associated with calculating exposure at an ex-vessel location.

Issue

In order to ensure the loss in fracture toughness of steel due to the effects of irradiation are adequately managed, it is necessary to assess the uncertainty associated with neutron fluence and dpa calculation results.

Request

Provide an estimate of the uncertainty associated with the neutron fluence and dpa results for the RV support steel (ring girder and support columns).

Question 3

Background

SLRA Section 3.5.2.2.2.6 (Supplement 1), states that the "reactor coolant piping which penetrates the PSW [primary shield wall] is insulated to ensure ambient temperatures remain within design limits." Information Notice (IN) 2007-21, Supplement 1, "Pipe Wear due to Interaction of Flow-Induced Vibration and Reflective Metal Insulation," states that the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code contains no specific requirements for licensees to remove insulation periodically for visual inspections assessing the integrity of stainless steel reflective metal insulation (RMI) and wear of encapsulated piping. The IN also states that this type of wear, if unchecked, could result in a small break loss of coolant accident and challenge the plant emergency core cooling systems.

To maintain the integrity of insulation, the GALL-SLR AMP XI.M36, "External Surfaces Monitoring of Mechanical Components," provides measures for monitoring, inspecting, and detecting age-related degradation of stainless steel or aluminum insulation at a frequency not to exceed one refueling cycle. Section B.2.3.23, "External Surfaces Monitoring of Mechanical Components," of the PBN SLRA states that the program monitors the reduction in thermal insulation resistance and evidence of insulation damage.

Issue

The staff reviewed item 3.5-1, 048 of SLRA Table 3.5-1, "Containment Building Structure and Internal Structural Components - Summary of Aging Management Programs," which states:

> There have been no instances of elevated temperatures for PBN plant structures other than containment (which is addressed in item 3.5-1, 003 and Section 3.5.2.2.1.2). In addition, insulation for high-temperature piping (> 200°F) is in scope to assist in maintaining local primary auxiliary building and turbine building concrete temperatures and is managed by the External Surfaces Monitoring of Mechanical Components (B.2.3.23) AMP.

The staff reviewed Item 3.5-1, 003 of SLRA Table 3.5-1, "Containment Building Structure and Internal Structural Components - Summary of Aging Management Programs," and noted that it addresses temperatures of containment penetrations, including the insulated "Main Steam and Feedwater penetrations that experienced elevated temperatures prior to initial license renewal." The staff also reviewed SLRA Table 3.1-1, "Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System, and noted that "non-metallic thermal insulation associated with reactor coolant piping and piping components does not perform a SLR intended function and is therefore not in scope." The staff further reviewed the PBN "External Surfaces Monitoring of Mechanical Components," program, but it is not clear whether its "scope of program," program element includes the reactor coolant piping thermal insulation. The aforementioned do not discuss how aging effects of the reactor coolant piping insulation that penetrates the PSW to ensure ambient temperatures remain within design limits are to be managed.

The staff is also not clear what type (e.g., rigid, flexible, made of stainless steel or other material) of thermal insulation PBN has used where the reactor coolant piping penetrates the primary shield wall (PSW) and whether such insulation aligns with the RMI description provided in Information Notice 2007-21, "Pipe Wear Due to Interaction of Flow-Induced Vibration and Reflective Metal Insulation." In addition, the staff is not clear whether the frequency of its inspection is consistent with the guidance provided in the "detection of aging effects" program element of GALL-SLR AMP XI.M36.

<u>Request</u>

a. Clarify the type of thermal insulation used on the reactor coolant piping in areas penetrating the PSW and its adequacy of protecting the PSW concrete from potential exposures to radiation and abnormal temperatures.

b. If it is a RMI and IN 2007-21 was applicable to PBN, discuss the IN inspection results, identified problems, and actions taken.
c. Clarify whether the PBN “External Surfaces Monitoring of Mechanical Components,” program includes in its “scope of program” program element this particular reactor coolant piping insulation.
d. If so, discuss whether accessibility and inspectability of the reactor coolant piping insulation is consistent with guidance provided in GALL-SLR AMP XI.M36, “detection of aging effects,” other applicable program elements, or other applicable PBN programs, so that it can fulfill its intended function (i.e., protection of the PSW concrete to abnormal temperature exposure) during the period of extended operation.
e. Augment SLRA Section 3.5.2.2.2.6 to indicate that the effects of aging for the encapsulating insulation to the reactor coolant piping that penetrates the PSW are managed by the “External Surfaces Monitoring of Mechanical Components” program so that the surrounding PSW concrete ambient temperature remains within design limits.

Question 4

Background

SLRA Section 3.5.2.2.2.6 states that both Units 1 and 2 have a ¼ inch thick steel liner plates installed at the inner face of the biological shield wall (BSW) and that the “liner plates are welded to each other and are anchored to the concrete with steel angle sections, thus enabling composite action with the concrete wall.” Attachment 1, “Reactor Vessel Supports, and Concrete Bioshield Exposure Data in Support of the Point Beach Unit 2 Subsequent License Renewal (SLR) Time-Limited Aging Analysis (TLAA),” to Westinghouse LTR-REA-20-28-NP, Revision 0, dated July 31, 2020 and submitted as Attachment 1 to Enclosure 4 of the PBN SLRA, “provides select exposure data applicable to the Point Beach Unit 2 reactor pressure vessel (RPV), RPV supports, concrete bioshield, and in-vessel and ex-vessel dosimetry.” The data indicates that the maximum fluence for E > 0.1 MeV with a 10% bias is 5.23E+19 n/cm^2 for 72 EFPY. The staff notes that the maximum fluence occurs on the BSW at an azimuthal angle of 90°and at an approximate elevation of 43.0 cm above the bottom of active fuel. Table 3.5.2-1, “Containment Building Structure and Internal Structural Components - Summary of Aging Management Evaluation,” of SLRA Supplement 1, includes an AMR item for loss of material, distortion of the reactor cavity liner (BSW steel liner) with a radiation shielding intended function. Neither the SLRA or its attachments, however, discuss an evaluation of the liner loss of fracture toughness, so that its radiation shielding and structural support intended functions as noted in

SLRA Table 2.4-1 and Table 3.5.2-1, are maintained to the end of the subsequent period of extended operation. .

Issue

The staff, using information contained on the applicant ePortal, calculated the gap between RV and the ¼ inch thick steel liner to be 6.5". Based on the information provided above, the most severe radiation exposure to the BSW steel liner will occur at or about its mid-height. Given the close proximity of BSW steel liner to the RV, it is not clear whether PBN calculated the harming energy of neutron fluence in terms of dpas (atoms permanently displaced from their position) at the mid-height of the steel liner and associated weldments, if any, at that location. Although PBN in SLRA Supplement 1 addressed the effects of radiation-induced volumetric expansion (RIVE) of concrete on the BSW steel liner and provided measures to identify its deformation, if any, it is not clear whether such loading coupled to the effects of streaming radiation on the ¼ inch thick steel liner would be factors to its potential cracking. If so, it is not clear what methodology PBN has used to evaluate the liner integrity, its welds, and attachments to concrete for effects of aging due to RIVE of concrete and liner embrittlement due streaming radiation for 72 EFPY of operation.

Requests

a. Discuss whether the RIVE effects on concrete compounded by potential liner embrittlement due to streaming radiation were considered in the evaluation of BSW steel liner integrity including its welds and attachments to concrete for projected 72 EFPY of PBN plant operation.
b. If so, discuss, the methodology used to evaluate the structural integrity of the BSW steel liner its weldments, including its anchorage to concrete, with respect to streaming radiation for the subsequent period of extended operation.
c. If not, justify why an evaluation of the BSW steel liner for loss of/reduction to fracture toughness was not necessary for the subsequent period of extended operation.

Question 5

Background

The SLRA states that the PBN Structures Monitoring program is consistent with enhancements to the GALL-SLR AMP, XI.S6. As such, its scope of program includes non-ASME Code related steel structural elements and steel liners. Item 3.5-1, 097 of SLRA Table 3.5-1, “Containment

Building Structure and Internal Structural Components - Summary of Aging Management Programs," states that the program manages the reactor cavity liner condition. SLRA Table 3.5.2-1, "Containment Building Structure and Internal Structural Components - Summary of Aging Management Evaluation," of the SLRA Supplement 1, includes an AMR item for loss of material, the distortion of the reactor cavity liner (BSW steel liner) with the intended function of radiation shielding.

SLRA Section 3.5.2.2.2.6 (Supplement 1), states:

> Therefore, the BSW and PSW will continue to satisfy the design criteria considering the long term radiation effects and a plant specific AMP or enhancements to an existing AMP are not required. The BSW and PSW will continue to be inspected as part of the Structures Monitoring (B.2.3.34) AMP, with specific attention to the potential for localized distortion of the cavity liner plate as a result of the RIVE effect on the underlying concrete.

The SRP-SLR, in Generic Branch Technical Position RLSB-1, identifies the GALL-SLR Report as an approved topical report (TR) for evaluating existing programs generically to document conditions under which they are considered adequate or when they need to be augmented to manage identified effects of aging. It states:

> If it is determined that the response to a specific applicant action item will result in the need for augmentation of specific programmatic criteria beyond those activities recommended in the applicable TR, the applicant should define the AMP accordingly to identify the AMP program element or elements that are impacted by the basis for responding to the applicable action item and the adjustments that will need to be made to the TR guidance recommendations, as defined in the impacted program elements for the AMP and applicable to the CLB and design basis for the facility.

<u>Issue</u>

Although the RV structural steel support assembly, as noted in WCAP-18554-P/NP, was inspected to minimize the possibility of flaws, and the liner is to be inspected for potential distortion for the RIVE effect during the subsequent period of extended operation, there is no discussion whether potential cracking of the BSW (reactor cavity) steel liner was included in such inspections. Neither the Westinghouse attachments to the SLRA or the SLRA discuss the effects of radiation on the liner, liner weldments, or its anchorage to concrete. NUREG/CR-5320, “Impact of Radiation Embrittlement on Integrity of Pressure Vessel Supports for Two PWR Plants,” states that “[t]he concern over radiation embrittlement is that it increases the potential for propagation of flaws that might exist.”

The staff could not locate an AMR item in Table 3.5.2-1, of the SLRA or in SLRA Supplement 1 addressing loss of/reduction in fracture toughness or cracking due to irradiation embrittlement for managing the effects of aging for the BSW (reactor cavity) steel liner with radiation shielding intended function. Similarly, the staff could not locate an enhancement to the Structures Monitoring program for managing such an aging effect or liner cracking due to radiation embrittlement. It is not clear how PBN will manage the effects of aging for loss/reduction of fracture toughness/cracking of the reactor cavity liner during the subsequent period of extended operation.

<u>Request</u>

a. Discuss how PBN plans to manage loss of/reduction in fracture toughness/cracking effects of aging of the (BSW) reactor cavity steel liner during the subsequent period of extended operation.
b. Discuss why the PBN SLRA and SLRA Supplement 1 do not provide a Table 2, AMR item and corresponding enhancements to the PBN Structures Monitoring program, applicable program elements for managing loss of/reduction in fracture toughness/cracking aging effect(s) due to radiation embrittlement of the (BSW) reactor cavity steel liner during the subsequent period of extended operation.
c. As an alternative to Request b. above, provide appropriate enhancements to the PBN Structures Monitoring program and include corresponding AMR item(s) and SLRA commitments that demonstrate adequate management of loss of/reduction in fracture toughness or cracking due to irradiation embrittlement of the reactor cavity (BSW) liner. Update the PBN SLRA Basis Document(s) and UFSAR supplement for the Structures Monitoring program as needed.

Question 6

Background

SLRA Supplement 1, Section 3.5.2.2.2.2.6, states that “[t]emperature assumptions were based on the normal operating temperature of the fluid in the RV nozzle of approximately 613°F and cooling of approximately 100°F for each inch away from the heat source. These temperature assumptions are consistent with previous structural analyses.”

Issue

The SLRA Supplement 1, does not reference the “previous structural analyses,” and the staff is unable to verify the statement.

Request

State and provide the technical reference(s) to the “previous structural analyses.”

Question 7

Background

Attachment 5 to “Point Beach Nuclear Plants Unit 1 and 2 License Amendment Request 261, Extended Power Uprate Licensing Report (EPU LAR),” (ADAMS Accession Nos. ML091250566, ML091250569) states that the “revised RPV supports loads and load combinations were found to be less than the appropriate allowable load limits with stress interaction ratios (IRs) indicating that “adequate design margins exist for support loads resulting from EPU conditions.” The Staff’s safety evaluation (ADAMS Accession Nos. ML110880039, ML110450159), for the EPU LAR, of the RPV supports “concludes that the licensee has demonstrated that the PBNP’s RPV and supports will remain structurally adequate to perform their function at EPU conditions and will continue to meet the requirements of PBNP GDCs 1, 2, 9 and 40 and the ASME Code Section III, Division 1, following implementation of the proposed EPU.”

Table 3.5.2.2-4 of SLRA Supplement 1 presents an update to the EPU IRs for RPV support components and states that “[t]hese interaction ratios have been updated from the ones in [SLRA] References 3.5.4.7 and 3.5.4.8 based on an issue identified and corrected by Westinghouse when performing the critical flaw size analyses for SLR.”

Issue

A comparison of the listed IR values in Table 2.2.2.3-5 of the PBN EPU LAR with those of Table 3.5.2.2-4 of the SLRA Supplement 1, indicates that IRs have increased approximately from 10 to 70 percent. It is not clear what methodology was used in calculating the IRs and where in each of the reported components they occur. It is also not clear whether the newly reported IR values in the Table 3.5.2.2-4 of the SLRA Supplement 1, for each of the reported RV structural steel support components have considered the corresponding Certification of Materials Testing Result (CMTR) reported strength values or the minimum applicable ASTM material strength values, and the effects of radiation. Given the uncertainty in the fluence and the values provided in the SLRA table, it is not clear what are the actual margins in the critical support components, particularly those components with IRs approximately equal to 1.

Requests

a. Clarify the methodology used in calculating the IRs and the calculated location.
b. Clarify the origin (e.g., CMTRs, ASTM) of the strength values used in calculations of the IRs listed in Table 3.5.2.2.-4 of the SLRA Supplement 1.
c. Clarify whether the values in the SLRA Supplement 1, Table 3.5.2.2-4 consider the effects of radiation. If so, define the projected margins for the RV steel support structure components (i.e., in girders and columns with and without the effects of radiation).
d. Given the uncertainty involved in the definition of fluence, discuss how likely is for the margins to increase/decrease.

Question 8

Background

SLRA Section 3.5.2.2.2.7, as amended by Supplement 1, defines the loading conditions for the RV structural assembly to be as follows:

Normal = Deadweight + Thermal
Upset = Normal + OBE Seismic
Faulted-1 = Normal + SSE Seismic
Faulted-2 = Normal + SSE Seismic + LOCA

According to Table 3.5.2.2-4, "Summary of RPV Support Component Stress Interaction Ratios [IRs]," of the SLRA Supplement 1, the IRs for the pipe column supports are the highest, having values 0.9954 and 0.9986, for the Upset and Faulted-2 conditions, respectively. The table also lists an IR of 0.7582 for the Faulted-1 condition. An IR value of 1.0 indicates that applied load stresses equal those that are allowed by the applicable design codes.

Issue

The Upset loading condition includes Normal loads plus seismic loads associated with the OBE. The Faulted-1 loading condition includes Normal loads plus the SSE seismic loads. It is noted that Upset and Faulted-1 loading conditions differ only in seismic forces. For pipe column supports, it is not clear why IRs for Faulted-1 loading condition are less than those for Upset, when SSE seismic forces are greater than OBE seismic forces. It is also not clear why the IRs for the Faulted-2 loading condition that includes increased seismic and LOCA loads over those of Upset loading condition are incremented only by 0.0032.

Request:

Discuss the apparent inconsistency in calculations of pipe column IRs for Upset, Faulted 1, and Faulted 2 loading conditions. Clarify why the IRs for the:

i. Faulted-1 loading condition are less than those reported for the Upset loading condition.
ii. Faulted-2 loading condition that includes increased seismic and LOCA loads over those of Upset loading condition are incremented only by 0.0032.

Question 9

Background

Table 3.5.2.2-6, "Summary of Postulated Critical Flaw Sizes for 72 EFPY," of SLRA Section 3.5.2.2.2.7, as amended by Supplement 1, itemizes postulated critical flaw sizes for Normal, Upset, Faulted-1, and Faulted-2 loading conditions. A footnote to the aforementioned SLRA table states that the "postulated critical flaw sizes are determined by setting [the] applied stress intensity factor equal to [the] fracture toughness and back-calculating [the] flaw size." The staff also notes that for the definition of fracture toughness, ASME Section III and Section XI require consideration of stresses from applicable loadings, including the effects of aging due to irradiation.

Table 3.5.2.2-4, “Summary of RPV Support Component Stress Interaction Ratios [IRs],” of the SLRA Supplement 1, indicates that the IRs for the pipe column supports for the Upset and Faulted-2 loading conditions are approaching unity, while those of the box ring girder subject to similar loading conditions are much less. Section 3.5.2.2.2.7 of SLRA Supplement 1, also states that ASME Section XI, Subsection IWF AMP will perform visual inspections of the RV steel support structure.

Issue

The staff noted that applicant ePortal documents (e.g., Standards for Welding T-1 Material Specifications) indicate that a lower yield strength electrode was used for the T-1 weldments, resulting in “undermatched” welds. Although weld fracture toughness is not addressed in the WCAP-18554-P/NP, weld flaws has been addressed through NDTs as part of the RV structural steel assembly fabrication. For those IRs approaching unity in Table 3.5.2.2-4 of the SLRA Supplement 1, the inference is that design stresses approach the controlling material yield stress without considering the effects of potential undetected flaws. Conservatively assuming that such material discontinuities exist, it is not clear whether the stress analysis methodology used in the definition of IRs considered potential undetected flaws in the irradiated welded structural steel RV support assembly, potentially resulting in IRs > 1.0. Furthermore, it is not clear: (a) where in the columns the IRs are maximized; (b) whether the maximized IR locations represent welded joints, and (c) if so whether residual stresses were considered in the calculation of the IRs.

Requests

a. Clarify whether the effects of potential undetected flaws, if any, have been considered in the calculated IR values in Table 3.5.2.2-4 of the SLRA Supplement 1.
b. If not, discuss the adequacy of the ASME Section XI, Subsection IWF AMP to examine the RV steel support assembly and reasonably assure that potential undetected flaws, including those in welds, would not affect its structural integrity during the subsequent period of extended operation.
c. Clarify: (a) where in the columns the IRs are maximized; (b) whether the maximized IR locations represent welded joints, and (c) if so whether residual stresses were considered in the calculation of the IRs.

Question 10

<u>Background</u>

SLRA Section 3.5.2.2.2.7, as amended by SLRA Supplement 1, states in part that "a plant specific AMP or enhancements to an existing AMP are not required to manage loss of fracture toughness due to irradiation embrittlement of the RV supports at PBN." Revision 0 of the audited PBN ASME Section XI, Subsection IWF AMP Basis Document, FPLCORP00036-REPT-059, reinforces this notion and states that "[f]urther evaluation determined that a plant-specific AMP or enhancements to an existing AMP are not required to manage the aging effect of loss of fracture toughness due to irradiation embrittlement of the RV supports at PBN." Revision 0 of the audited "Primary Shield Wall and Reactor Vessel Support Irradiation Evaluation," FPLCORP00036-REPT-035, basis document also states that:

> A review of the aging effects of loss of fracture toughness due to irradiation embrittlement on the PBN supports for SLR was performed … [and] a plant-specific AMP or enhancements to an existing AMP to manage the effects of concrete and RV support irradiation are not expected to be necessary to ensure the components perform their intended function consistent with the CLB through the SPEO.

The SRP-SLR in its Generic Branch Technical Position RLSB-1, identifies the GALL-SLR Report as an approved topical report (TR) for evaluating existing programs generically to document conditions under which they are considered adequate or when they need to be augmented to manage identified effects of aging. It states:

> If it is determined that the response to a specific applicant action item will result in the need for augmentation of specific programmatic criteria beyond those activities recommended in the applicable TR, the applicant should define the AMP accordingly to identify the AMP program element or elements that are impacted by the basis for responding to the applicable action item and the adjustments that will need to be made to the TR guidance recommendations, as defined in the impacted program elements for the AMP and applicable to the CLB and design basis for the facility.

Issue

SLRA Supplement 1, Table 3.5.2-1: Containment Building Structure and Internal Structural Components - Summary of Aging Management Evaluation," for the RV supports and bolting states that loss of fracture toughness aging effect is managed by the ASME Section XI, Subsection IWF (B.2.3.31) AMP. The staff reviewed the PBN ASME Section XI, Subsection IWF (B.2.3.31) AMP, but could not identify any enhancements associated with loss of fracture toughness aging effect for the RV steel structural supports. It is not clear what program elements of the ASME Section XI, Subsection IWF (B.2.3.31) AMP PBN plans to augment or make adjustments to, so that loss of fracture toughness aging effect due to radiation embrittlement of the RV steel structural support assembly is adequately managed during the subsequent period of extended operation.

Requests

a. Provide measures that would enhance the PBN ASME Section XI, Subsection IWF (B.2.3.31) AMP and corresponding SLRA commitment(s) and updated UFSAR supplement description, to manage loss of fracture toughness/cracking aging effects due to radiation embrittlement for the RV supports and bolting during the subsequent period of extended operation.
b. If none intended, state why. Otherwise, update PBN SLRA Basis Document(s) for the ASME Section XI, Subsection IWF program as needed.

Question 11

Background

The postulated critical flaw sizes in SLRA Table 3.5.2.2-6 are based on the fracture mechanics analysis in report WCAP-18554-NP, Revision 1, which the applicant included as Attachment 2 of Enclosure 4 to the SLRA (ADAMS Accession No. ML20329A264). These postulated critical flaw sizes were determined by setting the applied stress intensity factor equal to the fracture toughness and back-calculating flaw size. Fracture toughness depends on the temperature at each of the limiting locations shown in SLRA Table 3.5.2.2-6. In Section 5.1.2 of WCAP-18554-NP, Revision 1, the applicant stated that the vertical legs of the supports and the corners of the hexagonal ring-beam support (box ring girder) are exposed to considerable movement of ambient temperature air and are, therefore, close to ambient temperatures (~65°F-100°F).

Issue

The bolts at the ring girder are one of the limiting locations identified in the table of postulated critical flaw sizes in SLRA Table 3.5.2.2-6. Based on information in WCAP-18554-P, the staff noted that the appropriate temperature of the bolts at the ring girder appeared to be in the range of 65°F to 100°F. However, the temperature for the bolts at the ring girder indicated in Table 5-2 of WCAP-18554-NP, Revision 1 is higher than 65°F to 100°F. A higher temperature means a higher fracture toughness value for the bolts at the ring girder, which results in a larger, less conservative postulated critical flaw size for the bolts at the ring girder.

Requests

Either

a. Recalculate the postulated critical flaw size for the bolts at the ring girder using the ambient temperature at the corners of the hexagonal box ring girder (65°F to 100°F), which as the staff noted, appeared to be the appropriate temperature for the bolts at the ring girder,

or

b. Given that the temperature used for the fracture toughness calculation of the bolts at the ring girder was higher than 65°F to 100°F, justify how the postulated critical flaw size for the bolts at the ring girder shown in SLRA Table 3.5.2.2-6 is adequate or explain how the postulated critical flaw size would be affected by using the ambient temperature of 65°F to 100°F at the corners of the hexagonal box ring girder.