NEXTera ENERGY
POINT BEACH

May 27, 2021

L-2021-113
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 SUPPLEMENT 3**

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. NEPB Letter L-2021-081 dated April 21, 2021, Subsequent License Renewal Application - Aging Management Supplement 1 (ADAMS Accession No. ML21111A155)
5. NEPB Letter L-2021-102 dated May 6, 2021, Subsequent License Renewal Application - Aging Management Supplement 2 (ADAMS Accession No. ML21126A239)

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 and issued the regulatory audit plan for the aging management portion of the SLRA review (References 2 and 3). During this audit conducted between January 19, 2021 to March 26, 2021, NEPB agreed to supplement the SLRA (Enclosure 3, Attachment 1 of Reference 1) with new or clarifying information. The attachment to this letter provides that information and does not incorporate or otherwise affect any new or clarifying information provided in References 4 and 5.

For ease of reference, the attachment topic index is provided on page 3 of this letter. In the attachment, changes are described along with the affected section(s) and page number(s) of the docketed SLRA

(Enclosure 3 Attachment 1) where the changes are to apply. For clarity, revisions to the SLRA are provided with deleted text by ~~strikethroughs~~ and inserted text by **bold red underline**.

Pursuant to 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the State of Wisconsin.

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 27th day of May 2021.

Sincerely,

William Maher Digitally signed by William Maher DN: cn=William Maher, o=Nuclear, ou=Nuclear Licensing Projects, email=william.maher@fpl.com, c=US Date: 2021.05.27 09:39:02 -04'00'

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

| Attachment Index | |
|---|---|
| Attachment No. | PBN SLRA Enclosure 3 Attachment 1 Topic |
| -- | Incorporation of Interim Staff Guidance SLR-ISG-2021-01-PWRVI |

**Incorporation of SLR-ISG-2021-01-PWRVI**

**Affected SLRA Sections/Tables: Section 2.1.6, Section 2.1.6.4 (new), Table 2.3.1-2, Section 3.1.2.1.2, Section 3.1.2.2.9, Table 3.1-1, Table 3.1.2-2, Table 4.1.5-2, Table 16-3, Section B.1.1, Table B-4, Section B.2.3.7, Section C.1.0**

**SLRA Page Numbers: 2.1-31, 2.1-32, 2.1-35, 2.3-7, 3.1-3, 3.1-14, 3.1-15, 3.1-32, 3.1-33, 3.1-41 through 3.1-43, 3.1-46, 3.1-56 through 3.1-58, 3.1-75 through 3.1-86, 4.1-6, A-67, B-5, B-18, B-73, C-3**

**Description of Change:**

The SLR-ISG provides interim guidance to subsequent license renewal applicants which is incorporated into the following SLRA Sections.

- Section 2

  Section 2 is revised to identify the Interim Staff Guidance which is being incorporated by this Supplement and outline the changes made to NUREG-2191 and NUREG-2192. Table 2.3.1-2 is updated to reflect accurate component names.

- Section 3

  Section 3.1.2.2 is updated to reflect accurate aging effects and aging management programs.

  Section 3.1.2.2.9 is updated to incorporate the changes made to the further evaluation by the Interim Staff Guidance.

  Table 3.1-1 is revised to account for changes in inspection and examination (I&E) criteria for PWR reactor vessel internals (RVI) components made in MRP-227, Revision 1-A, and in other relevant industry documents.

  Table 3.1.2-2 is revised to incorporate the Interim Staff Guidance and make editorial changes.

- Section 4

  Table 4.1.5 2 is revised to incorporate the Interim Staff Guidance.

- Appendix A

  Table 16-3 row 11 is revised to make editorial changes.

- Appendix B

  Section B.1.1 is revised to reflect that there is no longer an exception to the Reactor Vessel Internals AMP due to incorporation of the Interim Staff Guidance.

  Table B-4 is revised to reflect that there is no longer an exception to the Reactor Vessel Internals AMP due to incorporation of the Interim Staff Guidance.

  Section B.2.3.7 is revised to reflect that there is no longer an exception to the Reactor Vessel Internals AMP due to incorporation of the Interim Staff Guidance.

- Appendix C

  Section C.1.0 is revised to state that the Interim Staff Guidance has been incorporated.

SLRA Enclosure 3 Attachment 1 Section 2.1.6, Pages 2.1-31 and 2.1-32, is revised as follows:

**2.1.6. Interim Staff Guidance Discussion**

As discussed in NEI 17-01, the NRC has encouraged applicants to address Subsequent License Renewal Interim Staff Guidance (SLR-ISG) documents in the Subsequent License Renewal Applications (SLRA). The following final SLR-ISGs have been issued for use and comment but have not been incorporated in NUREG-2191 or NUREG-2192 at the time of submittal:

- SLR-ISG-Electrical-2020-XX (Reference ML20156A324) — Updated Aging Management Criteria for Electrical Portions of Subsequent License Renewal Guidance
- SLR-ISG-Structures-2020-XX (Reference ML20156A338) — Updated Aging Management Criteria for Structures Portions of Subsequent License Renewal Guidance
- SLR-ISG-Mechanical-2020-XX (Reference ML20156A330) — Updated Aging Management Criteria for Mechanical Portions of Subsequent License Renewal Guidance
- **SLR-ISG-2021-01-PWRVI (Reference ML20217L203)** — **Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized-Water Reactors**

The following sub-sections provide summaries of how each of the SLR-ISGs are addressed in the SLRA.

New Section 2.1.6.4 is added on SLRA Page 2.1-35:

### 2.1.6.4 Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized-Water Reactors (SLR-ISG-2021-01-PWRVI)

This SLR-ISG provides interim guidance to subsequent license renewal applicants for the following NUREG-2191 and NUREG-2192 Sections:

- NUREG-2192, Table 3.1-1

  The SLR-ISG revises NUREG-2192, Table 3.1-1 to account for changes in inspection and examination (I&E) criteria for PWR reactor vessel internals (RVI) components made in MRP-227, Revision 1-A, and in other relevant industry documents. The PBN RVI further evaluation items in Table 3.1-1 incorporate the guidance presented in this SLR-ISG.

- NUREG-2191, Tables IV.B2, IV.B3 and IV.B4

  The SLR-ISG revises NUREG-2191, Tables IV.B2, IV.B3 and IV.B4 to update the staff's guidance for RVI components to account for changes in I&E criteria for PWR RVI components made in MRP-227, Revision 1-A, and in other relevant industry documents. Tables IV.B3 and IV.B4 are revised to reflect changes for Combustion Engineering and Babcock & Wilcox designed RVI components, respectively, and are not applicable to PBN. Table IV.B2 is revised to reflect changes for Westinghouse designed RVI components and is applicable to PBN. The revisions in Table IV.B2 have been incorporated into the PBN RVI AMR in Table 3.1.2-2.

- NUREG-2192 Further Evaluation items 3.1.2.2.9 and 3.1.3.2.9

  The SLR-ISG revises NUREG-2192 Further Evaluation items 3.1.2.2.9 and 3.1.3.2.9 to provide staff guidance for the acceptance criteria and review procedures, respectively, related to aging management of PWR RVI components. The revisions to item 3.1.2.2.9 have been incorporated into the PBN RVI AMR. Item 3.1.3.2.9 is not applicable to the PBN SLRA. Item 3.1.3.2.9 provides NRC staff review procedures and is not meant to be incorporated into an application.

- NUREG-2191 AMP XI.M16A, PWR Vessel Internals

  The SLR-ISG revises the AMP to incorporate the changes included in MRP-227, Revision 1-A. The PBN PWR Vessel Internals AMP (B.2.3.7) incorporates the guidance presented in this SLR-ISG.

- NUREG-2191, Table IX.C

  The SLR-ISG revises NUREG-2191, Table IX.C to add "Stellite" material and its usage. This revision has been incorporated into the PBN RVI AMR, as appropriate.

- NUREG-2192, Table 4.7-1

  The SLR-ISG revises NUREG-2192, Table 4.7-1 to add "EPRI MRP cycle-based and fluence-based analyses in support of MRP-227" as an example of a plant-specific TLAA topic. Cycle-based fatigue for the PBN RVI is

**included with the generic industry TLAA "Metal Fatigue of Class 1 Components" in SLRA Table 4.1.5.3 and Section 4.3.1. A PBN plant-specific RVI fluence-based analysis is not part of the PBN CLB and therefore does not meet the TLAA definition for SLR**

SLRA Enclosure 3 Attachment 1 Table 2.3.1-2, Page 2.3-7, is revised as follows:

**Table 2.3.1-2**
**Reactor Vessel Internals Components Subject to Aging Management Review**

| Component Type | Component Intended Function(s) |
|---|---|
| Alignment and interfacing components (clevis bearing Stellite wear surfaces) | Structural support |
| Alignment and interfacing components (clevis insert bolts) | Structural support |
| Alignment and interfacing components (clevis insert dowels) | Structural support |
| Alignment and interfacing components (upper core plate alignment pins) | Structural support |
| Baffle-former assembly (baffle plates, baffle edge bolts, former plates) | Structural support<br>Flow distribution |
| Baffle-former assembly (baffle plates, former plates) | Structural support<br>Flow distribution |
| Baffle-former assembly (baffle-edge bolts) | Structural support |
| Baffle-former assembly (baffle-former bolts) | Structural support |
| Bottom mounted instrumentation (column bodies) | Structural support |
| Bottom mounted instrumentation (flux thimble tubes) | Structural support<br>Pressure boundary |
| Control rod guide tube assembly (guide cards) | Structural support |
| Control rod guide tube assembly (lower flange welds in peripheral assemblies) | Structural support |
| Control rod guide tube assembly (lower flange welds in non-peripheral assemblies) | Structural support |
| Core barrel assembly (barrel former bolts) | Structural support |
| Core barrel assembly (core barrel flange) | Structural support<br>Flow distribution |
| ~~Core barrel assembly (core barrel outlet nozzle weld)~~ | ~~Structural support~~ |
| Core barrel assembly (lower axial welds) | Structural support |
| Core barrel assembly (lower flange weld) | Structural support |
| Core barrel assembly (lower girth weld) | Structural support |
| Core barrel assembly (middle axial welds) | Structural support |
| Core barrel assembly (upper axial weld) | Structural support |
| Core barrel assembly (upper flange weld) | Structural support |
| Core barrel assembly (upper girth weld) | Structural support |
| Lower core plate (fuel alignment pins) | Structural support |
| Lower internals assembly (lower core plate) | Structural support<br>Flow distribution |
| Lower internals assembly (lower support forging) | Structural support |
| Lower support assembly (lower support column bodies) | Structural support |
| Lower support assembly (lower support column bolts) | Structural support |

SLRA Enclosure 3 Attachment 1 Section 3.1.2.1.2, page 3.1-3, is revised as follows:

**Aging Effects Requiring Management**

The following aging effects associated with the reactor vessel internals require management:

- Changes in dimensions
- Cracking
- Cumulative fatigue damage
- Loss of fracture toughness
- Loss of material
- Loss of preload
- ~~Wear~~

**Aging Management Programs**

The following AMPs manage the aging effects for the reactor vessel internals components:

- ~~ASME Section XI Inservice Inspection (B.2.3.1)~~
- Flux Thimble Tube Inspection (B.2.3.24)
- Reactor Vessel Internals (B.2.3.7)
- Water Chemistry (B.2.3.2)

SLRA Enclosure 3 Attachment 1 Section 3.1.2.2.9, pages 3.1-14 through 3.1-15, is revised as follows:

3.1.2.2.9 Aging Management of PWR Reactor Vessel Internals (Applicable to Subsequent License Renewal Periods Only)

*Electric Power Research Institute (EPRI) Topical Report (TR)-1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)" (Agency wide Documents Access and Management System (ADAMS) Accession Nos. ML12017A191 through ML12017A197 and ML12017A199), ~~provides~~* ***provided*** *the industry's ~~current aging management~~* ***initial set of aging management inspection and evaluation (I&E)*** *recommendations for the reactor vessel internal (RVI) components that are included in the design of a PWR facility.* ***Since the issuance of MRP-227-A on January 9, 2012, EPRI updated its I&E guidelines for the PWR RVI components in Topical Report No. 3002017168, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Revision 1-A)" (ADAMS Accession No. ML20175A112). MRP-227, Revision 1-A, incorporated the industry's bases for resolving operating experience and industry lessons learned resulting from component-specific inspections performed since the issuance of MRP-227-A in January 2012. The staff found the guidelines in MRP-227, Revision 1-A, acceptable, as documented in a staff-issued safety evaluation dated April 25, 2019 (ADAMS Accession No. ML19081A001) and approved the topical report for use as documented in the staff's letters to the EPRI Materials Reliability Program (MRP) dated February 19, 2020 and July 7, 2020 (ADAMS Accession Nos. ML20006D152 and ML20175A149).***

*In ~~this report~~* ***MRP-227, Revision 1-A****, the EPRI ~~Materials Reliability Program (MRP)~~ identified that the following aging mechanisms may be applicable to the design of the RVI components in these types of facilities: (a)* ***stress corrosion cracking (****SCC****)****, (b) irradiation-assisted stress corrosion cracking (IASCC), (c) fatigue, (d) wear, (e) neutron irradiation embrittlement, (f) thermal aging embrittlement, (g) void swelling and irradiation growth* ***or component distortion****, ~~or~~* ***and*** *(h) thermal or irradiation-enhanced stress relaxation or irradiation enhanced creep. ~~The methodology in MRP-227-A was approved by the NRC in a safety evaluation dated December 16, 2011 (ADAMS Accession No. ML11308A770), which includes those plant-specific applicant/licensee action items that a licensee or applicant applying the MRP-227-A report would need to address and resolve and apply to its licensing basis.~~*

*The EPRI MRP's functionality analysis and failure modes, effects, and criticality analysis bases for grouping Westinghouse-designed, B&W-designed and Combustion Engineering (CE)-designed RVI components into ~~these~~* ***the applicable*** *inspection categories* ***(as evaluated in MRP-227, Revision 1-A)*** *~~was~~* ***were*** *based on an assessment of aging effects and relevant time-dependent aging parameters through a cumulative 60-year licensing period (i.e., 40 years*

SLRA Enclosure 3 Attachment 1 Section 3.1.2.2.9, pages 3.1-14 through 3.1-15, is revised as follows:

*for the initial operating license period plus an additional 20 years during the initial period of extended operation). The EPRI MRP***'s** ~~*has not assessed*~~ **assessment in MRP-227, Revision 1-A, did not evaluate** *whether operation of Westinghouse-designed, B&W-designed and CE-designed reactors during an SLR operating period* **(60 to 80 years)** *would have any impact on the existing susceptibility rankings and inspection categorizations for the RVI components in these designs, as defined in MRP-227***, Revision 1***-A or* ~~*it's*~~ **the** *applicable MRP background documents (e.g., MRP-191***, Revision 1,** *for Westinghouse-designed or CE-designed RVI components or MRP-189***, Revision 2,** *for B&W-designed components).*

*As described in GALL-SLR Report AMP XI.M16A, the applicant may use the MRP-227***, Revision 1***-A based AMP as an initial reference basis for developing and defining the AMP that will be applied to the RVI components for the subsequent period of extended operation. However, to use this alternative basis, GALL-SLR Report AMP XI.M16A recommends that the MRP-227***, Revision 1***-A based AMP be enhanced to include a gap analysis of the components that are within the scope of the AMP. The gap analysis is a basis for identifying and justifying* ~~*any potential*~~ *changes to the MRP-227***, Revision 1***-A based program that* ~~*may be*~~ **are** *necessary to provide reasonable assurance that the effects of age-related degradation will be managed during the subsequent period of extended operation. The criteria for the gap analysis are described in GALL-SLR Report AMP XI.M16A.*UU **If a gap analysis is needed to establish the appropriate aging management criteria for the RVI components, the applicant has the option of including the gap analysis in the SLRA** ~~**for its reactor unit(s)**~~ **or making the gap analysis and any supporting gap analysis documents available in the in-office audit portal for the SLRA review.**

**Subsequent license renewal (SLR) applicants for units of a PWR design will no longer need to include separate SLRA Appendix C section responses in resolution of the A/LAIs previously issued on MRP-227-A because the A/LAIs were resolved and closed by the staff in the April 25, 2019, safety evaluation for MRP-227, Revision 1-A. The sole A/LAI issued by the staff in the safety evaluation dated April 25, 2019, relates to an applicant's methods and timing of inspections that will be applied to the baffle-to-former bolts or core shroud bolts in the plant design. Since an applicant's resolution of this A/LAI can be appropriately addressed in the "Operating Experience" program element discussion for the AMP and in the applicant's basis document for the AMP, a separate SLRA Appendix C response for the A/LAI is unnecessary.**

*Alternatively, the PWR SLRA may define a plant-specific AMP for the RVI components to demonstrate that the RVI components will be managed in accordance with the requirements of 10 CFR 54.21(a)(3) during the proposed*

SLRA Enclosure 3 Attachment 1 Section 3.1.2.2.9, pages 3.1-14 through 3.1-15, is revised as follows:

*subsequent period of extended operation. Components to be inspected, parameters monitored, monitoring methods, inspection sample size, frequencies, expansion criteria, and acceptance criteria are justified in the SLRA.* ~~*The*~~ ***If the AMP is a plant-specific program, the*** UU*NRC staff will assess the adequacy of the plant-specific AMP against the criteria for the 10 AMP program elements that are defined in Section A.1.2.3 of SRP-SLR Appendix A.1.*

The PBN Reactor Vessel Internals AMP is based on the current MRP-227 Revision 1-A framework modified by an 80-year gap analysis. Appendix C of this application provides a detailed discussion of the RVI gap analysis. As enhanced, this program will continue to manage the effects of stress corrosion cracking, irradiation-assisted stress corrosion cracking, wear, fatigue, thermal aging embrittlement, irradiation embrittlement, void swelling, thermal and irradiation-induced stress relaxation, and irradiation creep, including any combined effects. As a condition monitoring program, the PBN Reactor Vessel Internals AMP specifies inspection methods that are sufficient to detect aging effects, such as cracking, whether from a single aging mechanism or combination of mechanisms, prior to a component approaching a condition in which it may not be able to fulfill its intended functions; and if such aging effects are detected, the evaluation and corrective action is required to consider the effects from any applicable mechanism in order to provide reasonable assurance that the component will continue to perform its intended function.

SLRA Enclosure 3 Attachment 1 Table 3.1-1, pages 3.1-32 and 3.1-33, is revised as follows:

| Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System | | | | | |
|---|---|---|---|---|---|
| **Item Number** | **Component** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation Recommended** | **Discussion** |
| 3.1-1, 025 | Steel (with nickel alloy cladding) or nickel alloy steam generator primary side components: divider plate and tube-to-tube sheet welds exposed to reactor coolant | Cracking due to primary water SCC | AMP XI.M2, "Water Chemistry," and AMP XI.M19, "Steam Generators." In addition, a plant- specific program is to be evaluated. | Yes (SRP-SLR Sections 3.1.2.2.11.1 and 3.1.2.2.11.2) | Not applicable. Further evaluation is documented in subsection 3.1.2.2.11. |
| 3.1-1, 028 | **Westinghouse-specific** "Existing Programs" components: Stainless steel, nickel alloy ~~Westinghouse~~**, and X-750** control rod guide tube support pins **(split pins)**~~, and Combustion Engineering thermal shield positioning pins; Zircaloy-4 Combustion Engineering incore instrumentation thimble tubes~~ exposed to reactor coolant and neutron flux | Loss of material due to wear; cracking due to SCC, ~~irradiation-assisted SCC~~**IASCC**, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | ~~Consistent with NUREG-2191. The PBN Reactor Vessel Internals (B.2.3.7) and Water Chemistry (B.2.3.2) AMPs are used to manage the reactor vessel internals upper core plate and alignment pins.~~ **Not applicable. The control rod guide tubes are not an "Existing Programs" component. Further evaluation is documented in subsection 3.1.2.2.11.** |
| 3.1-1, 029 | Not applicable. This line item only applies to BWRs. | | | | |
| 3.1-1, 030 | Not applicable. This line item only applies to BWRs. | | | | |
| 3.1-1, 031 | Not applicable. This line item only applies to BWRs. | | | | |

SLRA Enclosure 3 Attachment 1 Table 3.1-1, pages 3.1-32 and 3.1-33, is revised as follows:

**Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System**

| Item Number | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| ~~3.1-1, 032~~ | ~~Stainless steel, nickel alloy, or CASS reactor vessel internals, core support structure (not already referenced as ASME Section XI Examination Category B- N-3 core support structure components in MRP-227- A), exposed to reactor coolant and neutron flux~~ | ~~Cracking, loss of material due to wear~~ | ~~AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD"~~ | ~~No~~ | ~~Consistent with NUREG-2191. The PBN ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1) AMP is used to manage cracking and loss of material in reactor vessel internal core support structures exposed to reactor coolant and neutron flux.~~ |
| 3.1-1, 033 | Stainless steel, steel with stainless steel cladding Class 1 reactor coolant pressure boundary components exposed to reactor coolant | Cracking due to SCC | AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and AMP XI.M2, "Water Chemistry" | No | Consistent with NUREG-2191. The PBN ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.17) and Water Chemistry (B.2.3.2) AMPs are used to manage SCC in Class 1 reactor coolant pressure boundary components exposed to reactor coolant. |
| 3.1-1, 034 | Stainless steel, steel with stainless steel cladding pressurizer relief tank (tank shell and heads, flanges, nozzles) exposed to treated borated water >60°C (>140°F) | Cracking due to SCC | AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and AMP XI.M2, "Water Chemistry" | No | Not applicable. The PBN pressurizer relief tank is not an ASME Section XI component. Cracking due to SCC in the stainless steel pressurizer relief tank exposed to treated borated water >140°F is managed with item number 3.1-1, 080. |

SLRA Enclosure 3 Attachment 1 Table 3.1-1, pages 3.1-41 through 3.1-43, is revised as follows:

**Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System**

| Item Number | Component | Aging Effect/Mechanism | Aging Management Program (AMP)/TLAA | Further Evaluation Recommended | Discussion |
|---|---|---|---|---|---|
| 3.1-1, 053a | Stainless steel, nickel alloy Westinghouse reactor internal "Primary" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | Consistent with NUREG-2191. The Reactor Vessel Internals (B.2.3.7) and Water Chemistry (B.2.3.2) AMPs are used to manage cracking due to SCC, irradiation assisted SCC and fatigue in reactor vessel internals "Primary" components exposed to reactor coolant and neutron flux. Note that many aging effects managed by the Reactor Vessel Internals (B.2.3.7) AMP are dispositioned through FMECA analysis and not inspected. Further evaluation is documented in subsection 3.1.2.2.9. |

SLRA Enclosure 3 Attachment 1 Table 3.1-1, pages 3.1-41 through 3.1-43, is revised as follows:

| **Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System** | | | | | |
|---|---|---|---|---|---|
| **Item Number** | **Component** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation Recommended** | **Discussion** |
| 3.1-1, 053b | Stainless steel Westinghouse reactor internal "Expansion" components exposed to reactor coolant and neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~IASCC, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | Consistent with NUREG-2191. The Reactor Vessel Internals (B.2.3.7) and Water Chemistry (B.2.3.2) AMPs are used to manage cracking due to SCC, irradiation assisted SCC and fatigue in reactor vessel internals "Expansion" components exposed to reactor coolant and neutron flux. Note that many aging effects managed by the Reactor Vessel Internals (B.2.3.7) AMP are dispositioned through FMECA analysis and not inspected. Further evaluation is documented in subsection 3.1.2.2.9. |

SLRA Enclosure 3 Attachment 1 Table 3.1-1, pages 3.1-41 through 3.1-43, is revised as follows:

| Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System | | | | | |
|---|---|---|---|---|---|
| **Item Number** | **Component** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation Recommended** | **Discussion** |
| 3.1-1, 053c | Stainless steel, nickel alloy**, or stellite** Westinghouse reactor internal "Existing Programs" components exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~**IASCC**, fatigue | AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (for SCC mechanisms only) | Yes (SRP-SLR Section 3.1.2.2.9) | Consistent with NUREG-2191. The Reactor Vessel Internals (B.2.3.7) and Water Chemistry (B.2.3.2) AMPs are used to manage cracking due to SCC, irradiation assisted SCC and fatigue in reactor vessel internals "Existing Programs" components exposed to reactor coolant and neutron flux. Note that many aging effects managed by the Reactor Vessel Internals (B.2.3.7) AMP are dispositioned through FMECA analysis and not inspected. Further evaluation is documented in subsection 3.1.2.2.9. |
| 3.1-1, 054 | Stainless steel **Westinghouse-design** bottom mounted instrument system flux thimble tubes (with or without chrome plating) exposed to reactor coolant and neutron flux | Loss of material due to wear | AMP XI.M37, "Flux Thimble Tube Inspection" | No | Consistent with NUREG-2191. The Flux Thimble Tube Inspection (B.2.3.24) AMP is used to manage loss of material due to wear in stainless steel bottom mounted instrument system flux thimble tubes exposed to reactor coolant and neutron flux. |
| 3.1-1, 055a | Not applicable. This line item only applies to Babcock and Wilcox designs. | | | | |
| 3.1-1, 055b | Not applicable. This line item only applies to Combustion Engineering designs. | | | | |

SLRA Enclosure 3 Attachment 1 Table 3.1-1, page 3.1-46, is revised as follows:

| Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System | | | | | |
|---|---|---|---|---|---|
| **Item Number** | **Component** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation Recommended** | **Discussion** |
| 3.1-1, 059c | Stainless steel (SS, including CASS, PH SS or martensitic SS), ~~or~~ nickel alloy, or stellite Westinghouse reactor internal "Existing Programs" components exposed to reactor coolant and neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; changes in dimensions due to void swelling, distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation, creep; loss of material due to wear | AMP XI.M16A, "PWR Vessel Internals" | Yes (SRP-SLR Section 3.1.2.2.9) | The Reactor Vessel Internals (B.2.3.7) AMP is used to manage reactor vessel internals "Expansion" components exposed to reactor coolant and neutron flux. Note that many aging effects managed by the Reactor Vessel Internals (B.2.3.7) AMP are dispositioned through FMECA analysis and not inspected. Further evaluation is documented in subsection 3.1.2.2.9. |
| 3.1-1, 060 | Not applicable. This line item only applies to BWRs. | | | | |
| 3.1-1, 061 | Steel steam generator steam nozzle and safe end, feedwater nozzle and safe end, AFW nozzles and safe ends exposed to secondary feedwater/steam | Wall thinning due to flow-accelerated corrosion | AMP XI.M17, "Flow-Accelerated Corrosion" | No | Consistent with NUREG-2191. The Flow-Accelerated Corrosion (B.2.3.8) AMP is used to manage wall thinning due to flow accelerated corrosion in the steam generator feedwater nozzle and steam outlet nozzle exposed to secondary feedwater/steam. |

SLRA Enclosure 3 Attachment 1 Table 3.1-1, pages 3.1-56 through 3.1-58, is revised as follows:

| **Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System** | | | | | |
|---|---|---|---|---|---|
| **Item Number** | **Component** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation Recommended** | **Discussion** |
| 3.1-1, 113 | Not applicable. This line item only applies to BWRs. | | | | |
| 3.1-1, 114 | Reactor coolant system components defined as ASME Section XI Code Class components (ASME Code Class 1 reactor coolant pressure boundary components, reactor vessel interior attachments, or core support structure components~~,~~; or ASME Class 2 or 3 components - including ASME defined appurtenances, component supports, and associated pressure boundary welds, or components subject to plant-specific equivalent classifications for these ASME code classes) | Cracking due to SCC, IGSCC, PWSCC, IASCC (SCC mechanisms for stainless steel, nickel alloy components only), fatigue, or cyclic loading; loss of material due to general corrosion (steel only), pitting corrosion, crevice corrosion, or wear | AMP XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," and AMP XI.M2, "Water Chemistry" (water chemistry- related or corrosion- related aging effect mechanisms only) | No | Not used.<br>All relevant aging mechanisms requiring management by ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1) or Water Chemistry (B.2.3.2) are recognized using line items more specific to the individual component type. |
| 3.1-1, 115 | Stainless steel piping, piping components exposed to concrete | None | None | Yes (SRP-SLR Section 3.1.2.2.15) | Not applicable.<br>There are no PBN stainless steel reactor coolant system piping or piping components exposed to concrete. Further evaluation is documented in subsection 3.1.2.2.15. |

SLRA Enclosure 3 Attachment 1 Table 3.1-1, pages 3.1-56 through 3.1-58, is revised as follows:

| Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System | | | | | |
|---|---|---|---|---|---|
| **Item Number** | **Component** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation Recommended** | **Discussion** |
| 3.1-1, 116 | Nickel alloy control rod drive penetration nozzles exposed to reactor coolant | Loss of material due to wear | Plant-specific aging management program | Yes (SRP-SLR Section 3.1.2.2.10.1) | Consistent with NUREG-2191. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.2) AMP is used to manage loss of material due to wear in the control rod drive mechanism head penetration housings exposed to reactor coolant. Further evaluation is documented in subsection 3.1.2.2.10. |
| 3.1-1, 117 | Stainless steel, nickel alloy control rod drive penetration nozzle thermal sleeves exposed to reactor coolant | Loss of material due to wear | Plant-specific aging management program | Yes (SRP-SLR Section 3.1.2.2.10.2) | Not applicable. Further evaluation is documented in subsection 3.1.2.2.10.2. |
| 3.1-1, 118 | Stainless steel, nickel alloy PWR reactor vessel internal components **or LRA/SLRA-specified reactor vessel internal component** exposed to reactor coolant, neutron flux | Cracking due to SCC, ~~irradiation-assisted SCC~~**IASCC**, cyclic loading, fatigue | Plant-specific aging management program **or AMP XI.M16A, "PWR Vessel Internals," and AMP XI.M2, "Water Chemistry" (SCC and IASCC only), with an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component** | Yes (SRP-SLR Section 3.1.2.2.9) | Not applicable. Cracking due to SCC, irradiation-assisted SCC, cyclic loading, and fatigue of stainless steel, nickel alloy PWR reactor vessel internal components exposed to reactor coolant, neutron flux is addressed in rows 3.1-1, 053a, 3.1-1, 053b, and 3.1-1, 053c. The associated NUREG-2191 aging items are not used. |

SLRA Enclosure 3 Attachment 1 Table 3.1-1, pages 3.1-56 through 3.1-58, is revised as follows:

| **Table 3.1-1: Summary of Aging Management Evaluations for the Reactor Vessel, Internals, and Reactor Coolant System** | | | | | |
|---|---|---|---|---|---|
| **Item Number** | **Component** | **Aging Effect/Mechanism** | **Aging Management Program (AMP)/TLAA** | **Further Evaluation Recommended** | **Discussion** |
| 3.1-1, 119 | Stainless steel, nickel alloy**,** **stellite** PWR reactor vessel internal components **or LRA/SLRA-specified reactor vessel internal component** exposed to reactor coolant, neutron flux | Loss of fracture toughness due to neutron irradiation embrittlement or thermal aging embrittlement; changes in dimensions due to void swelling or distortion; loss of preload due to thermal and irradiation-enhanced stress relaxation or creep; loss of material due to wear | Plant-specific aging management program **or AMP XI.M16A, "PWR Vessel Internals," with an adjusted site-specific or component-specific aging management basis for a specified reactor vessel internal component** | Yes (SRP-SLR Section 3.1.2.2.9) | Consistent with NUREG-2191 for ~~loss of fracture toughness and changes in dimensions~~ **loss of material in the stainless steel upper and lower core plate fuel alignment pins as well as the stellite radial support keys and upper core plate alignment pins.** ~~Loss of fracture toughness and changes in dimension for stainless steel reactor vessel internals components is managed by the Reactor Vessel Internals (B.2.3.7) AMP. Loss of preload is not applicable. Loss of material is addressed with item number 3.1-1, 054.~~ Note that many aging effects managed by the Reactor Vessel Internals (B.2.3.7) AMP are dispositioned through FMECA analysis and not inspected. Further evaluation is documented in subsection 3.1.2.2.9. |
| 3.1-1, 120 | Not applicable. This line item only applies to BWRs. | | | | |
| 3.1-1, 121 | Not applicable. This line item only applies to BWRs. | | | | |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| ~~Alignment and interfacing components (clevis bearing Stellite wear surfaces)~~ | ~~Structural support~~ | ~~Stellite~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~None~~ | ~~None~~ | ~~J, 3~~ |
| Alignment and interfacing components (clevis bearing Stellite wear surfaces) | Structural support | Stellite | Reactor coolant Neutron flux | Loss of material | Reactor Vessel Internals (B.2.3.7) | ~~None~~IV.B2.RP-285 | ~~None~~3.1-1, 059c | ~~J, 3~~A |
| Alignment and interfacing components (clevis insert bolts) | Structural support | Nickel alloy | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-399 | 3.1-1, 053c | ~~B~~A, 1<br><br>B |
| ~~Alignment and interfacing components (clevis insert bolts)~~ | ~~Structural support~~ | ~~Nickel alloy~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| Alignment and interfacing components (clevis insert bolts) | Structural support | Nickel alloy | Reactor coolant Neutron flux | Loss of material<br>Loss of preload | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-285 | 3.1-1, 059c | ~~B~~A, 1 |
| ~~Alignment and interfacing components (clevis insert dowels)~~ | ~~Structural support~~ | ~~Nickel alloy~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| Alignment and interfacing components (clevis insert dowels) | Structural support | Nickel alloy | Reactor coolant Neutron flux | Loss of material | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-399 | 3.1-1, 053c | ~~B~~A, 1 |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| Alignment and interfacing components (upper core plate alignment pins) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-~~355~~**301** | 3.1-1, 053c | ~~D~~**A**<br><br>**B** |
| Alignment and interfacing components (upper core plate alignment pins) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Loss of material | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-299 | 3.1-1, 059c | ~~B~~**A**<br><br>**B** |
| Alignment and interfacing components (upper core plate alignment pins) | Structural support | Stellite | Reactor coolant<br>Neutron flux | Loss of material | Reactor Vessel Internals (B.2.3.7) | ~~NoneIV.B2.R-424~~ | ~~None~~**3.1-1, 119** | ~~J, 3~~**A** |
| Baffle-former assembly (baffle plates, baffle edge bolts, former plates) | Structural support<br>Flow distribution | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-270a | 3.1-1, 053a | ~~B~~**A**<br><br>**B** |
| Baffle-former assembly (baffle plates, baffle edge bolts, former plates) | Structural support<br>Flow distribution | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-387 | 3.1-1, 053a | ~~DC~~<br><br>~~D~~ |
| ~~Baffle-former assembly (baffle plates, baffle edge bolts, former plates)~~ | ~~Structural support~~<br>~~Flow distribution~~ | ~~Stainless steel~~ | ~~Reactor coolant~~<br>~~Neutron flux~~ | ~~Cracking~~<br>~~Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | A |
| Baffle-former assembly (baffle plates, former plates) | Structural support<br>Flow distribution | Stainless steel | Reactor coolant<br>Neutron flux | Changes in dimensions<br>**Loss of fracture toughness** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-270 | 3.1-1, 059a | ~~B~~**A** |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| ~~Baffle-former assembly (baffle plates, former plates)~~ | ~~Structural support~~<br>~~Flow distribution~~ | ~~Stainless steel~~ | ~~Reactor coolant~~<br>~~Neutron flux~~ | ~~Loss of fracture toughness~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-388~~ | ~~3.1-1, 059a~~ | ~~D~~**C** |
| Baffle-former assembly (baffle-edge bolts) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Loss of fracture toughness<br>Changes in dimensions<br>Loss of preload<br>**Loss of material** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-354 | 3.1-1, 059a | ~~B~~**A** |
| Baffle-former assembly (baffle-edge bolts) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Loss of material | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-296 | 3.1-1, 059a | ~~D~~**C** |
| Baffle-former assembly (baffle-former bolts) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-271 | 3.1-1, 053a | **A**<br>B |
| ~~Baffle-former assembly (baffle-former bolts)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant~~<br>~~Neutron flux~~ | ~~Cracking~~<br>~~Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | A |
| Baffle-former assembly (baffle-former bolts) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Loss of fracture toughness<br>Changes in dimensions<br>Loss of preload<br>**Loss of material** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-354 | 3.1-1, 059a | ~~B~~**A** |
| ~~Baffle-former assembly (baffle-former bolts)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant~~<br>~~Neutron flux~~ | ~~Loss of material~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-296~~ | ~~3.1-1, 059a~~ | D |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| Bottom mounted instrumentation (column bodies) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-293 | 3.1-1, 053b | ~~B~~A<br><br>B |
| Bottom mounted instrumentation (column bodies) | Structural support | Stainless steel | Reactor coolant Neutron flux | **Loss of fracture toughness**<br>Loss of material | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-292~~0b~~ | 3.1-1, 059b | ~~D~~A |
| ~~Bottom mounted instrumentation (flux thimble tubes)~~ | ~~Structural support Pressure boundary~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking~~ | ~~Reactor Vessel Internals (B.2.3.7)~~<br>~~Water Chemistry (B.2.3.2)~~ | ~~IV.B2.RP-355~~ | ~~3.1-1, 053c~~ | ~~D~~ |
| ~~Bottom mounted instrumentation (flux thimble tubes)~~ | ~~Structural support Pressure boundary~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Loss of fracture toughness Changes in dimensions~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.R-424~~ | ~~3.1-1, 119~~ | ~~E, 2~~ |
| Bottom mounted instrumentation (flux thimble tubes) | Structural support Pressure boundary | Stainless steel | Reactor coolant Neutron flux | Loss of material | Flux Thimble Tube Inspection (B.2.3.24) | IV.B2.RP-284 | 3.1-1, 054 | A |
| Control rod guide tube assembly (guide cards) | Structural support | Stainless steel Cast austenitic stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-296a~~8~~ | 3.1-1, 053a | ~~D~~A<br><br>B |
| ~~Control rod guide tube assembly (guide cards)~~ | ~~Structural support~~ | ~~Cast austenitic stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Loss of fracture toughness~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-297~~ | ~~3.1-1, 059a~~ | ~~D~~ |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| Control rod guide tube assembly (guide cards) | Structural support | **Stainless steel** Cast austenitic stainless steel | Reactor coolant Neutron flux | Loss of material **Loss of fracture toughness** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-296 | 3.1-1, 059a | ~~B~~**A** |
| ~~Control rod guide tube assembly (guide cards)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking~~ | ~~Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2)~~ | ~~IV.B2.RP-298~~ | ~~3.1-1, 053a~~ | ~~D~~ |
| ~~Control rod guide tube assembly (guide cards)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Loss of material~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-296~~ | ~~3.1-1, 059a~~ | ~~B~~ |
| Control rod guide tube assembly (lower flange weld**s in peripheral assemblies**) | Structural support | **Stainless steel** Cast austenitic stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2) | IV.B2.RP-298 | 3.1-1, 053a | ~~B~~**A** **B** |
| Control rod guide tube assembly (lower flange weld**s in peripheral assemblies**) | Structural support | **Stainless steel** Cast austenitic stainless steel | Reactor coolant Neutron flux | Loss of fracture toughness | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-297 | 3.1-1, 059a | ~~B~~**A** |
| ~~Control rod guide tube assembly (lower flange weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking~~ | ~~Reactor Vessel Internals (B.2.3.7) Water Chemistry (B.2.3.2)~~ | ~~IV.B2.RP-298~~ | ~~3.1-1, 053a~~ | ~~B~~ |
| ~~Control rod guide tube assembly (lower flange weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Loss of fracture toughness~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-297~~ | ~~3.1-1, 059a~~ | ~~B~~ |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| **Control rod guide tube assembly (lower flange welds in non-peripheral assemblies)** | **Structural support** | **Stainless steel**<br>**Cast austenitic stainless steel** | **Reactor coolant**<br>**Neutron flux** | **Loss of fracture toughness** | **Reactor Vessel Internals (B.2.3.7)** | **IV.B2.RP-297a** | **3.1-1, 059b** | **A** |
| Core barrel assembly (barrel former bolts) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-273 | 3.1-1, 053b | ~~B~~**A**<br>**B** |
| Core barrel assembly (barrel former bolts) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Loss of fracture toughness<br>Changes in dimensions<br>Loss of preload<br>**Loss of material** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-274 | 3.1-1, 059b | ~~B~~**A** |
| ~~Core barrel assembly (barrel former bolts)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Loss of material~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-296~~ | ~~3.1-1, 059a~~ | ~~D, 1~~ |
| Core barrel assembly (core barrel flange) | Structural support<br>Flow distribution | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-~~289~~**345a** | 3.1-1, 053c | ~~D~~**A**<br>**B** |
| ~~Core barrel assembly (core barrel flange)~~ | ~~Structural support Flow distribution~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | A |
| Core barrel assembly (core barrel flange) | Structural support<br>Flow distribution | Stainless steel | Reactor coolant<br>Neutron flux | Loss of material | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-345 | 3.1-1, 059c | ~~B~~**A** |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

| **Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation** | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **Component Type** | **Intended Function** | **Material** | **Environment** | **Aging Effect Requiring Management** | **Aging Management Program** | **NUREG-2191 Item** | **Table 1 Item** | **Notes** |
| ~~Core barrel assembly (core barrel outlet nozzle weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking~~ | ~~Reactor Vessel Internals (B.2.3.7)~~<br>~~Water Chemistry (B.2.3.2)~~ | ~~IV.B2.RP-278~~ | ~~3.1-1, 053b~~ | ~~B~~ |
| ~~Core barrel assembly (core barrel outlet nozzle weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| ~~Core barrel assembly (core barrel outlet nozzle weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Loss of material~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-290b~~ | ~~3.1-1, 059b~~ | ~~D~~ |
| Core barrel assembly (lower axial welds) | Structural support | Stainless steel | Reactor coolant Neutron flux | ~~Changes in dimensions~~<br>**Loss of material** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-274 | 3.1-1, 059b | ~~D~~**C** |
| Core barrel assembly (lower axial welds) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-387a | 3.1-1, 053b | ~~B~~**A**<br>**B** |
| ~~Core barrel assembly (lower axial welds)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| Core barrel assembly (lower axial welds) | Structural support | Stainless steel | Reactor coolant Neutron flux | Loss of fracture toughness<br>**Changes in dimensions** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-388a | 3.1-1, 059b | ~~B~~**A** |
| Core barrel assembly (lower flange weld) | Structural support | Stainless steel | Reactor coolant Neutron flux | ~~Changes in dimensions~~<br>**Loss of material** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-274 | 3.1-1, 059b | ~~D~~**C** |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| Core barrel assembly (lower flange weld) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-280 | 3.1-1, 053~~a~~**b** | ~~B, 1~~**A**<br>**B** |
| **Core barrel assembly (lower flange weld)** | **Structural support** | **Stainless steel** | **Reactor coolant Neutron flux** | **Loss of fracture toughness Changes in dimensions** | **Reactor Vessel Internals (B.2.3.7)** | **IV.B2.RP-280a** | **3.1-1, 059b** | **A** |
| ~~Core barrel assembly (lower flange weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| ~~Core barrel assembly (lower flange weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Loss of fracture toughness~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-388a~~ | ~~3.1-1, 059b~~ | ~~D~~ |
| ~~Core barrel assembly (lower girth weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Changes in dimensions~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-270~~ | ~~3.1-1, 059a~~ | ~~D~~ |
| Core barrel assembly (lower girth weld) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-387 | 3.1-1, 053a | ~~B~~**A**<br>**B** |
| ~~Core barrel assembly (lower girth weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| Core barrel assembly (lower girth weld) | Structural support | Stainless steel | Reactor coolant Neutron flux | Loss of fracture toughness **Changes in dimensions** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-388 | 3.1-1, 059a | ~~B~~**A** |
| ~~Core barrel assembly (middle axial welds)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Changes in dimensions~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-274~~ | ~~3.1-1, 059b~~ | ~~D~~ |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| Core barrel assembly (middle axial welds) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-387a | 3.1-1, 053b | ~~B~~A<br><br>B |
| ~~Core barrel assembly (middle axial welds)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | A |
| Core barrel assembly (middle axial welds) | Structural support | Stainless steel | Reactor coolant Neutron flux | Loss of fracture toughness<br>**Changes in dimensions** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-388a | 3.1-1, 059b | ~~B~~A |
| Core barrel assembly (upper axial weld) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-280~~387a~~ | 3.1-1, 053b | ~~B~~A<br><br>B |
| ~~Core barrel assembly (upper axial weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | A |
| Core barrel assembly (upper flange weld) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-276 | 3.1-1, 053a | ~~B~~A<br><br>B |
| ~~Core barrel assembly (upper flange weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | A |
| Core barrel assembly (upper girth weld) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-~~387~~280 | 3.1-1, 053b~~a~~ | A~~B, 1~~<br><br>B |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| ~~Core barrel assembly (upper girth weld)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| ~~Lower core plate (fuel alignment pins)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Changes in dimensions~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-270~~ | ~~3.1-1, 059a~~ | ~~D, 1~~ |
| ~~Lower core plate (fuel alignment pins)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking~~ | ~~Reactor Vessel Internals (B.2.3.7)~~<br>~~Water Chemistry (B.2.3.2)~~ | ~~IV.B2.RP-289~~ | ~~3.1-1, 053c~~ | ~~D~~ |
| Lower core plate (fuel alignment pins) | Structural support | Stainless steel | Reactor coolant Neutron flux | ~~Loss of fracture toughness~~ Loss of material | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-~~288~~**424** | 3.1-1, **119**~~059c~~ | ~~D~~**A** |
| ~~Lower internals assembly (lower core plate)~~ | ~~Structural support Flow distribution~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Changes in dimensions~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-270~~ | ~~3.1-1, 059a~~ | ~~D, 1~~ |
| Lower internals assembly (lower core plate) | Structural support Flow distribution | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-289 | 3.1-1, 053c | ~~B~~**A**<br>**B** |
| ~~Lower internals assembly (lower core plate)~~ | ~~Structural support Flow distribution~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| Lower internals assembly (lower core plate) | Structural support Flow distribution | Stainless steel | Reactor coolant Neutron flux | Loss of fracture toughness Loss of material **Changes in dimensions** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-288 | 3.1-1, 059c | ~~D~~**A** |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| Lower internals assembly (lower support forging) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-291a | 3.1-1, 053b | ~~B~~A<br>B |
| ~~Lower internals assembly (lower support forging)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| Lower support assembly (lower support column bodies) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | **Loss of fracture toughness**<br>Changes in dimensions | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-~~274~~**295** | 3.1-1, 059b | ~~D~~A |
| Lower support assembly (lower support column bodies) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-29~~1a~~**4** | 3.1-1, 053b | ~~D~~A |
| ~~Lower support assembly (lower support column bodies)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| ~~Lower support assembly (lower support column bodies)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Loss of fracture toughness~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-290a~~ | ~~3.1-1, 059b~~ | ~~B~~ |
| Lower support assembly (lower support column bolts) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-286 | 3.1-1, 053b | ~~B~~A<br>B |
| ~~Lower support assembly (lower support column bolts)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| Lower support assembly (lower support column bolts) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Loss of fracture toughness<br>Loss of preload<br>**Changes in dimensions**<br>**Loss of material** | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-287 | 3.1-1, 059b | ~~B~~A |
| ~~Lower support assembly (lower support column bolts)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant~~<br>~~Neutron flux~~ | ~~Loss of material~~ | ~~Reactor Vessel Internals (B.2.3.7)~~ | ~~IV.B2.RP-290b~~ | ~~3.1-1, 059b~~ | ~~D~~ |
| No additional measures components | Structural support<br>Flow distribution | Nickel alloy<br>Stainless steel | Reactor coolant<br>Neutron flux | None | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-265 | 3.1-1, 055c | ~~B~~A |
| ~~Radial support keys~~ | ~~Structural support~~ | ~~Stellite~~ | ~~Reactor coolant~~<br>~~Neutron flux~~ | ~~Cracking~~<br>~~Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~F~~ |
| Radial support keys | Structural support | Stellite | Reactor coolant<br>Neutron flux | ~~Wear~~**Loss of material** | Reactor Vessel Internals (B.2.3.7) | ~~None~~**IV.B2.R-424** | ~~None~~**3.1-1, 119** | ~~J, 3~~A |
| Reactor vessel internal components with a fatigue analysis | Structural support | Cast austenitic stainless steel<br>Nickel alloy<br>Stainless steel | Reactor coolant<br>Neutron flux | Cumulative fatigue damage | TLAA – Section 4.3.1, Metal Fatigue of Class 1 Components | IV.B2.RP-303 | 3.1-1, 003 | ~~B~~A |
| Thermal shield assembly (thermal shield flexures) | Structural support | Stainless steel | Reactor coolant<br>Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-302 | 3.1-1, 053a | ~~B~~A<br>B |

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

**Table 3.1.2-2: Reactor Vessel Internals – Summary of Aging Management Evaluation**

| Component Type | Intended Function | Material | Environment | Aging Effect Requiring Management | Aging Management Program | NUREG-2191 Item | Table 1 Item | Notes |
|---|---|---|---|---|---|---|---|---|
| ~~Upper core plate (fuel alignment pins)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking~~ | ~~Reactor Vessel Internals (B.2.3.7)~~<br>~~Water Chemistry (B.2.3.2)~~ | ~~IV.B2.RP-289~~ | ~~3.1-1, 053c~~ | ~~D~~ |
| Upper core plate (fuel alignment pins) | Structural support | Stainless steel | Reactor coolant Neutron flux | ~~Loss of fracture toughness~~<br>Loss of material | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP~~-288~~424 | 3.1-1, ~~059c~~119 | ~~D~~A |
| Upper internals assembly (upper core plate) | Structural support | Stainless steel | Reactor coolant Neutron flux | Cracking | Reactor Vessel Internals (B.2.3.7)<br>Water Chemistry (B.2.3.2) | IV.B2.RP-291b | 3.1-1, 053b | ~~B~~A<br>B |
| ~~Upper internals assembly (upper core plate)~~ | ~~Structural support~~ | ~~Stainless steel~~ | ~~Reactor coolant Neutron flux~~ | ~~Cracking~~<br>~~Loss of material~~ | ~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.3.1)~~ | ~~IV.B2.RP-382~~ | ~~3.1-1, 032~~ | ~~A~~ |
| Upper internals assembly (upper core plate) | Structural support | Stainless steel | Reactor coolant Neutron flux | Loss of material<br>~~Loss of fracture toughness~~ | Reactor Vessel Internals (B.2.3.7) | IV.B2.RP-290b~~5~~ | 3.1-1, 059b | ~~D~~A |

**Generic Notes**

A. Consistent with component, material, environment, aging effect and aging management program listed for NUREG-2191 line item. AMP is consistent with NUREG-2191 AMP description.
B. Consistent with component, material, environment, aging effect and aging management program listed for NUREG-2191 line item. AMP has exceptions to NUREG-2191 AMP description.
C. **Component is different, but consistent with material, environment, aging effect and aging management program listed for NUREG-2191 line item. AMP is consistent with NUREG-2191 AMP description.**
D. Component is different, but consistent with material, environment, aging effect and aging management program listed for NUREG-2191 line item. AMP has exceptions to NUREG-2191 AMP description.
~~E. Consistent with NUREG-2191 material, environment, and aging effect but a different aging management program is credited or NUREG-2191~~

SLRA Enclosure 3 Attachment 1 Table 3.1.2-2, pages 3.1-75 through 3.1-86, is revised as follows:

~~identifies a plant-specific aging management program.~~

~~F. Material not in NUREG-2191 for this component.~~

~~J. Neither the component nor the material and environment combination is evaluated in NUREG-2191.~~

**Plant Specific Notes**

1. Component inspection category is not consistent with the inspection category cited in Table 3.1-1.
2. ~~The PWR Vessel Internals program manages loss of fracture toughness and changes in dimension for stainless steel flux thimble tubes through FMECA analysis described further in Appendix C. Loss of preload is not applicable to flux thimble tubes, and loss of material is addressed by NUREG-2191 item IV.B2.RP-284. Flux thimble tubes are existing program components.~~
3. ~~Wear surfaces for the upper core plate alignment pins, clevis inserts, and radial support keys are Stellite. Aging effects identified in the Appendix C RVI gap analysis for these components are managed by the Reactor Vessel Internals program.~~

SLRA Enclosure 3 Attachment 1 Table 4.1.5-2, page 4.1-6, is revised as follows:

**Table 4.1.5-2**
**Review of Plant-Specific TLAAs Listed in NUREG-2192, Table 4.7-1**

| Table 4.7-1 Examples of Potential Plant-Specific TLAA Topics | Applies to PBN | SLRA Section |
|---|---|---|
| **PWRs** | | |
| Reactor pressure vessel underclad cracking | No (Note 1) | N/A |
| Leak-before-break | Yes | 4.7.1<br>4.7.2 |
| Reactor coolant pump flywheel fatigue crack growth | Yes | 4.7.4 |
| Response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification" | Yes | 4.3.1 |
| Response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems" | Yes | 4.3.1 |
| **EPRI MRP cycle-based and fluence-based analyses in support of MRP-227** | **No (Note 3)** | **N/A** |
| **BWRs and PWRs** | | |
| Fatigue of cranes (crane cycle limits) | Yes | 4.7.6 |
| Fatigue of the spent fuel pool liner | No (Note 2) | N/A |
| Corrosion allowance calculations | No (Note 2) | N/A |
| Flaw growth due to stress corrosion cracking | No (Note 2) | N/A |
| Predicted lower limit | Yes | 4.3.5 |

Note 1: Refer to Section 3.1.2.2.5.

Note 2: Refer to Notes 3, 4, and 5 of Table 4.1.5-1.

**Note 3: Cycle-based fatigue for the PBN RVI is included with the generic industry TLAA "Metal Fatigue of Class 1 Components" in SLRA Table 4.1.5.3 and Section 4.3.1. A PBN plant-specific RVI fluence-based analysis is not part of the PBN CLB and therefore does not meet the TLAA definition for SLR.**

SLRA Enclosure 3 Attachment 1 Appendix A Table 16-3, page A-67, is revised as follows:

**Table 16-3**
**List of SLR Commitments and Implementation Schedule**

| No. | Aging Management Program or Activity (Section) | NUREG-2191 Section | Commitment | Implementation Schedule |
|---|---|---|---|---|
| 11 | Reactor Vessel Internals (16.2.2.7) | XI.M16A | Continue the existing PBN Reactor Vessel Internals AMP, including enhancement to:<br>a) Implement the guidance in MRP 227 Rev. 1-A as supplemented by the gap analysis, or the latest NRC approved version of MRP 227 which addresses 80 years of operation if one is available prior to the subsequent period of extended operation.<br>b) Implement the results of the gap analysis in the Reactor Vessel Internals Program unless it is superseded by the latest NRC approved version of MRP 227 which addresses 80 years of operation. If so, the AMP may be implemented directly without the use of a gap analysis.<br>c) Incorporate the updated examination acceptance criteria, Primary / Expansion links, expansion criteria, and expansion item examination criteria in MRP 227 Rev. 1-A as supplemented by the gap analysis. | No later than 6 months prior to the SPEO, i.e.:<br>PBN1: 04/05/2030<br>PBN2: 09/08/2032 |

SLRA Enclosure 3 Attachment 1 Appendix B Section B.1.1, page B-5, is revised as follows:

These new AMPs will be consistent with the 10 elements of their respective NUREG-2191 AMPs. The following programs each have exception(s) justified by technical data:

- the PBN Water Chemistry AMP (Section B.2.3.2),
- the PBN Reactor Head Closure Stud Bolting AMP (Section B.2.3.3),
- ~~the PBN Reactor Vessel Internals AMP (Section B.2.3.7),~~
- the PBN Steam Generators AMP (Section B.2.3.10),
- the PBN Open-Cycle Cooling Water System AMP (Section B.2.3.11),
- the PBN Closed Treated Water Systems AMP (Section B.2.3.12),
- the PBN Fuel Oil Chemistry AMP (Section B.2.3.18),
- the PBN Reactor Vessel Material Surveillance AMP (Section B.2.3.19),
- the PBN Buried and Underground Piping and Tanks AMP (Section B.2.3.27),
- the PBN Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks AMP (Section B.2.3.28),
- the PBN ASME Section XI, Subsection IWF AMP (Section B.2.3.31).

SLRA Enclosure 3 Attachment 1 Appendix B Table B-4, page B-18, is revised as follows:`

**Table B-4**
**Point Beach Aging Management Program Consistency with NUREG-2191**

| PBN Aging Management Program | Section | PBN Plant-Specific? | NUREG-2191 Comparison | | |
|---|---|---|---|---|---|
| | | | NUREG-2191 Section | Enhancements? | Exceptions? |
| Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components | B.2.3.5 | No | XI.M11B | Yes | No |
| Thermal Aging Embrittlement of Cast Austenitic Stainless Steel | B.2.3.6 | No | XI.M12 | New | No |
| Reactor Vessel Internals | B.2.3.7 | No | XI.M16A | Yes | ~~Yes~~No |
| Flow-Accelerated Corrosion | B.2.3.8 | No | XI.M17 | Yes | No |
| Bolting Integrity | B.2.3.9 | No | XI.M18 | Yes | No |
| Steam Generators | B.2.3.10 | No | XI.M19 | Yes | Yes |
| Open-Cycle Cooling Water System | B.2.3.11 | No | XI.M20 | Yes | Yes |
| Closed Treated Water Systems | B.2.3.12 | No | XI.M21A | Yes | Yes |
| Inspection of Overhead Heavy Load Handling Systems | B.2.3.13 | No | XI.M23 | Yes | No |

SLRA Enclosure 3 Attachment 1 Appendix B Section B.2.3.7, page B-73, is revised as follows:

**NUREG-2191 Consistency**

The PBN RVI AMP, with enhancements, will be consistent ~~with an exception~~ with the **10 elements** ~~program described in~~ **of** NUREG-2191, Section XI.M16A**, "PWR Vessel Internals" as modified by the Interim Staff Guidance SLR-ISG-2021-01-PWRVI**.

**Exceptions to NUREG-2191**

~~The program described in NUREG-2191, Section XI.M16A provides MRP-227-A as the basis for a site specific RVI program. The scope of the PBN Reactor Vessel Internals AMP applies the methodology and guidance in MRP-227 Revision 1-A (as supplemented by a gap analysis). MRP-227 Revision 1-Ais the most recent NRC approved guidance for managing PWR vessel internals and incorporates significant recent operating experience.~~**None.**

SLRA Enclosure 3 Attachment 1 Appendix C Section C.1.0, page C-3, is revised as follows:

> **In accordance with Interim Staff Guidance SLR-ISG-2021-01-PWRVI,** ~~T~~the PBN subsequent license renewal (SLR) RVI gap analysis uses the most recent guidelines provided in EPRI Technical Report No. 3002017168, MRP-227 Rev. 1-A (Reference C.9.1) as the baseline to address an 80-year operating period, consistent with the NRC SE dated April 15, 2019 (Reference ML19081A001) indicating that MRP-227 Rev. 1 can be used as a starting point for performing a gap analysis in order to develop an RVI AMP for the 60-80-year subsequent period of extended operation (SPEO), and the NRC SE dated February 19, 2020 (Reference ML20006D152) indicating that MRP-227 Rev. 1-A is acceptable to the extent delineated in the April 15 2019 SE. Revision 1 of the guidelines provides updates based on Revision 1 of the NRC SE for MRP-227 Revision 0 (Reference ML11308A770) and includes operating experience and new knowledge gained from materials testing, modeling, and research. MRP-227 Rev. 1-A is the acceptance version incorporating changes from the NRC SE approving MRP-227 Revision 1. Note that MRP-227 Rev. 1-A still only addresses an operating period of 60 years and will be implemented at PBN for the current period of extended operation by January 1, 2022.