Xcel Energy®

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August 28, 2023

L-MT-23-035
10 CFR 54.17

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Monticello Nuclear Generating Plant
Docket No. 50-263
Renewed Facility Operating License No. DPR-22

Subsequent License Renewal Application Supplement 5

References:

1) Letter from Northern States Power Company, a Minnesota corporation (NSPM), d/b/a Xcel Energy to Document Control Desk, "Monticello Nuclear Generating Plant Docket No. 50-263, Renewal License Number DPR-22 Application for Subsequent Renewal Operating License" dated January 9, 2023, ML23009A353

2) Letter from Northern States Power Company, a Minnesota corporation (NSPM), d/b/a Xcel Energy to Document Control Desk, "Monticello Nuclear Generating Plant Subsequent License Renewal Application Supplement 1" dated April 3, 2023, ML23094A136

3) Letter from Northern States Power Company, a Minnesota corporation (NSPM), d/b/a Xcel Energy to Document Control Desk, "Monticello Nuclear Generating Plant Subsequent License Renewal Application Supplement 2" dated June 26, 2023, ML23177A218

4) Letter from Northern States Power Company, a Minnesota corporation (NSPM), d/b/a Xcel Energy to Document Control Desk, "Monticello Nuclear Generating Plant Subsequent License Renewal Application Supplement 3" dated July 11, 2023, ML23193B026

5) Letter from Northern States Power Company, a Minnesota corporation (NSPM), d/b/a Xcel Energy to Document Control Desk, "Monticello Nuclear Generating Plant Subsequent License Renewal Application Supplement 4 and Responses to Request for Confirmation of Information - Set 1" dated July 18, 2023, ML23199A154

Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy hereafter "NSPM", is submitting a supplement to the Subsequent License Renewal Application, listed in Reference 1.

Clarifying information regarding Tables 4.2.3-1 and 4.2.3-2 and an updated reference was provided in Supplement 1, listed in Reference 2. Clarifications to sections of the SLRA discussed in the breakout audits occurring April through June of 2023 were provided in Supplements 2, 3, and 4, listed in References 3, 4, and 5, respectively. Additional clarifications discussed in the breakout audits occurring April through June of 2023 are being provided in the Enclosures of Attachment 1 of this Supplement.

In the enclosures, changes are described along with the affected section(s) and page number(s) of the docketed SLRA (Reference 1) where the changes are to apply. For clarity, revisions to the SLRA are provided with deleted text by ~~strikethrough~~ and inserted text by **bold red underline**. Previous changes incorporated as a result of other Supplements are provided by bold, black font and noted in the description of the Enclosure.

Summary of Commitments

This letter makes new commitments and revisions to existing commitments as explained in the enclosures. Commitments 33 and 36 include additions and revisions.

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

Executed on August 28, 2023.

Shawn Hafen
Site Vice President, Monticello Nuclear Generating Plant
Northern States Power Company – Minnesota

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

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Enclosure 01

Irradiation Effects on Biological Shield Structural Steel Components

Irradiation Effects on Biological Shield Structural Steel Components

The aging effects due to irradiation embrittlement (loss of fracture toughness) will be managed by the Structures Monitoring (B.2.3.33) AMP that is credited for the biological shield wall steel components.

Affected SLRA Sections: 3.5.2.1.1, Table 3.5.2-1, Table A-3, Commitment 36, B.2.3.33.

SLRA Page Numbers: 3.5-3, 3.5-76, 3.5-83, 3.5-84, A-92, B-239

Description of Change:

Updates the SLRA for the Structures Monitoring, B.2.3.33 AMP to include managing the aging effects of loss of fracture toughness for the biological shield steel components for which the Structures Monitoring AMP is credited. Also, corresponding Table 2 AMR items and commitment table additions are provided.

The Information shown in bold black font in the mark-ups for Table A-3, Commitment 36, represent changes provided in Enclosures 31c, 31d, and 35b of Reference 1. The Information shown in bold black font in the mark-ups for Section B.2.3.33 represent changes provided in Enclosure 31e of Reference 1.

References:

1. L-MT-23-025, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 2, ML23177A218

SLRA Section 3.5.2.1.1 on page 3.5-3 is revised as follows:

Aging Effects Requiring Management

The following aging effects associated with the PCT structure and internal structural components require management:

- Cracking
- Cumulative Fatigue Damage
- Loss of Bond
- Loss of Coating or Lining Integrity
- **Loss of Fracture Toughness**
- Loss of Leak Tightness
- Loss of Material
- Loss of Mechanical Function
- Loss of Mechanical Properties
- Loss of Preload
- Loss of Sealing
- Reduction of Strength

SLRA Table 3.5.2-1 on page 3.5-76 is revised to add the following:

Table 3.5.2-1: Primary Containment – 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**
Biological Shield Wall (Columns, Beams, Liner, Doors)	**Structural Support**	**Steel**	**Air - indoor Uncontrolled**	**Loss of Fracture Toughness**	**Structures Monitoring (B.2.3.33)**	**None**	**None**	**H, 12**

SLRA Table 3.5.2-1 on pages 3.5-83 and 3.5-84 are revised to add the following:

General Notes

H. Aging effect not in NUREG-2191 for this component, material, and environment combination.

Plant-Specific Notes

12. The Structures Monitoring (B.2.3.33) AMP is used to manage loss of fracture toughness. No additional aging management of the biological shield wall structural steel beyond the current Structures Monitoring (B.2.3.33) AMP is necessary for aging effects due to irradiation. Further evaluation is documented in the Biological Shield Structural Steel Evaluation in Section 3.5.2.2.2.6.

SLRA Table A-3, Commitment 36 on page A-92 is revised to add the following:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
36	Structures Monitoring (A.2.2.33)	XI.S6	f) Revise the implementing procedure to include qualification requirements for both inspection and evaluation personnel that are in accordance with ACI 349.3R-02. g) Revise the implementing procedure to explicitly include inspection of the following components and commodities: • Expansion plugs • Fuel Storage Racks (New Fuel) • Manhole covers, supports • Supports • <u>**Biological Shield Wall Structural Steel**</u> • Concrete Diesel Fuel Oil Storage Tank Deadmen • Vibration Isolation Elements • Electrical Enclosures • RPV to Drywell Refueling Seal • Exterior Surfaces of Roofing h) Revise the implementing procedure to include acceptance criteria for concrete surfaces based on the "second-tier" evaluation criteria provided in ACI 349.3R-02. i) Revise the implementing procedure to include that if any projected inspection results will not meet acceptance criteria prior to the next scheduled inspection, inspection frequencies are adjusted as determined by the CAP. j) **Revise the implementing procedure to include acceptance criteria for inspections of the following components and commodities:** • **Expansion plugs** • **Fuel Storage Racks (New Fuel)** • **Manhole covers, supports** • **Supports** • **Concrete Diesel Fuel Oil Storage Tank Deadmen**	

			• Vibration Isolation Elements **• Electrical Enclosures** **• RPV to Drywell Refueling Seal** **• Exterior Surfaces of Roofing** k) **Ensure that the implementing procedure states that visual inspections of inaccessible concrete for evidence of leaching of calcium hydroxide and carbonation are performed if the area becomes accessible or if inspections in an accessible area identifies a condition that would be a leading indicator for the inaccessible area.** **l) Include trending of quantitative measurements and qualitative information for findings exceeding the acceptance criteria for all applicable parameters monitored or trended.** **m) Revise the implementing procedure to include enhanced acceptance criteria for detection of alkali-silica reactions in concrete to include:** **• Alkali-silica gel exudations** **• Surface staining** **• Expansion causing structural deformation, relative movement or displacement, or misalignment/distortion of attached components** **n) Revise the implementing procedure to include monitoring for irradiation embrittlement during existing structures monitoring inspections of the Biological Shield wall Structural Steel.**	

SLRA Section B.2.3.33 on page B-239 is revised to add the following changes:

Element Affected	Enhancement
1. Scope of Program	**Revise the implementing procedure to explicitly include inspection of the following components and commodities:** • **Expansion plugs** • **Fuel Storage Racks (New Fuel)** • **Manhole covers, supports** • **Supports** • **Biological Shield Wall Structural Steel** • **Concrete Diesel Fuel Oil Storage Tank Deadmen** • **Vibration Isolation Elements** • **Electrical Enclosures** • **RPV to Drywell Refueling Seal** • **Exterior Surfaces of Roofing**
1. Scope of Program **3. Parameters Monitored or Inspected**	**Revise the implementing procedure to include monitoring for irradiation embrittlement during existing structures monitoring inspections of the Biological Shield wall Structural Steel.**

Enclosure 02

Loss of Fracture Toughness of RV Supports

Loss of Fracture Toughness of RV Supports

Loss of fracture toughness due to irradiation embrittlement is conservatively added as an applicable aging effect for the RV steel supports.

Affected SLRA Sections: 3.5.2.1.7, Table 3.5.2-7, Table A-3, B.2.3.30

SLRA Page Numbers: 3.5-8, 3.5-98, 3.5-103, A-89, B-224, B-225

Description of Change: Update the ASME Section XI, Subsection IWF (B.2.3.30) AMP to manage the aging effects due to irradiation embrittlement for the RV steel support assembly components. Revise Section 3.5.2.1.7 and Table 3.5.2-7 to include the aging effect “loss of fracture toughness” (due to neutron flux). Revise Table A-3, Commitment 33 to include monitoring the RV steel supports for the aging effect “loss of fracture toughness”.

The Information shown in bold black font in the mark-ups for Section B.2.3.30 represent changes provided in Enclosure 22 of Reference 1.

References:

1. L-MT-23-025, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 2, ML23177A218.

SLRA Section 3.5.2.1.7 on page 3.5-8 is revised as follows:

Aging Effects Requiring Management

The following aging effects associated with the Hangers and Supports structural components require management:

- Cracking
- Crazing
- Dimensional Change
- Discoloration
- Hardening
- Increase in Porosity and Permeability
- Loss of Bond
- **Loss of Fracture Toughness**
- Loss of Material
- Loss of Mechanical Function
- Loss of Preload
- Loss of Strength
- Reduced Thermal Insulation Resistance
- Reduction In Concrete Anchor Capacity
- Reduction or Loss of Isolation Function
- Scuffing
- Shrinkage
- Surface Cracking

SLRA Table 3.5.2-7 on page 3.5-98 is revised to insert the following:

Table 3.5.2-7: Hangers and Supports Commodity Group – 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**
ASME Class 1 Supports	Structural Support	Steel	Air - Indoor Uncontrolled	Loss of Fracture Toughness	ASME Section XI, Subsection IWF (B.2.3.30)	None	None	H, 4

SLRA Table 3.5.2-7 general notes and plant specific notes on page 3.5-103 are revised to include the following:

General Notes

H. Aging effect not in NUREG-2191 for this component, material, and environment combination.

Plant Specific Notes

4. The ASME Section XI, Subsection IWF (B.2.3.30) AMP is used to manage loss of fracture toughness. No additional aging management of the ASME Class I Supports beyond the current IWF (B.2.3.30) AMP is necessary for aging effects due to irradiation. Further evaluation is documented in the Reactor Vessel Support Steel Irradiation Evaluation in Section 3.5.2.2.2.6.

SLRA Table A-3, Commitment 33 on page A-89 is revised as follows:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
33	ASME Section XI, Subsection IWF (A.2.2.30)		**k. Revise procedures to include monitoring for irradiation embrittlement during existing IWF inspections of the reactor vessel support steel.**	

SLRA Section B.2.3.30, first paragraph, on page B-224 is revised as follows:

B.2.3.30 ASME Section XI, Subsection IWF

The ASME Section XI, Subsection IWF AMP is an existing AMP that consists of periodic visual examination of supports for ASME Class 1, 2, 3, and MC piping and components for signs of degradation such as corrosion; cracking; deformation; misalignment of supports; missing, detached, or loosened support items; loss of integrity of welds; improper clearances of guides and stops; and improper hot or cold settings of spring supports and constant load supports. **The AMP also visually monitors RV supports for loss of fracture toughness.** Bolting for Class 1, 2, 3**, and MC** piping and component supports is also included and inspected for corrosion, loss of integrity of bolted connections due to self-loosening, and material conditions that can affect structural integrity.

SLRA Section B.2.3.30, Enhancement Table, Element 1 (Scope of Program) and Element 3 (Parameters Monitored or Inspected) on page B-225 is revised to insert the following:

Element Affected	Enhancement
1. Scope of Program	Revise procedures to evaluate the acceptability of inaccessible areas (e.g., portions of ASME Class 1, 2, 3**, and MC** supports encased in concrete, buried underground, or encapsulated by guard pipe) when conditions are identified in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.
1. Scope of Program **3. Parameters Monitored or Inspected**	**Revise procedures to include monitoring for irradiation embrittlement during existing IWF inspections of the reactor vessel support steel.**

Enclosure 03a

Reduction of Strength and Mechanical Properties of Structural Steel and Concrete Due to Irradiation

Reduction of Strength and Mechanical Properties of Structural Steel and Concrete Due to Irradiation

Add discussion of the locations, methodologies and conclusions along with a supporting figure used in the calculations to determine the effects of radiation on the biological shield/pedestal concrete and RPV steel supports.

Affected SLRA Sections: List of Figures,3.5.2.2.2.6

SLRA Page Numbers: xx, 3.5-36, and 3.5-37

Description of Change:

A narrative to explain the corresponding locations for the bounding fluence and gamma dose is being added. A figure that includes shading to show the region of bioshield concrete that is considered in further evaluation section 3.5.2.2.2.6 for the effects of irradiation is also being added to the SLRA. A summary of the methodology and conclusions in calculations supporting the effects of radiation on the biological shield/pedestal concrete and RPV steel supports is also added.

SLRA List of Figures on page xx is revised as follows:

Figure 3.5.2.2.2.6-1

Section Through Monticello Biological Shield Wall 3.5-37

SLRA Section 3.5.2.2.2.6 on pages 3.5-36 and 3.5-37 is revised as follows:

3.5.2.2.2.6 Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation

Reduction of strength, loss of mechanical properties, and cracking due to irradiation could occur in PWR and BWR Group 4 concrete structures that are exposed to high levels of neutron and gamma radiation. These structures include the reactor (primary/biological) shield wall, the sacrificial shield wall, and the reactor vessel support/pedestal structure. Data related to the effects and significance of neutron and gamma radiation on concrete mechanical and physical properties is limited, especially for conditions (dose, temperature, etc.) representative of light-water reactor (LWR) plants. However, based on literature review of existing research, radiation fluence limits of 1×10^{19} neutrons/cm^2 neutron radiation and 1×10^8 Gy (1×10^{10} rad) gamma dose are considered conservative radiation exposure levels beyond which concrete material properties may begin to degrade markedly (Ref. 17, 18, 19).

Further evaluation is recommended to determine the need for a plant-specific AMP or plant-specific enhancements to selected existing AMPs to manage aging effects of irradiation if the estimated (calculated) fluence levels or irradiation dose received by any portion of the concrete from neutron (fluence cutoff energy E >0.1 MeV) or gamma radiation exceeds the respective threshold level during the subsequent period that could affect intended functions. Higher fluence or dose levels may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and/or loss of mechanical properties of concrete from those fluence levels, at or above the operating temperature experienced by the concrete, and the effects are applied to the design calculations. Supporting calculations/analyses, test data, and other technical basis are provided to estimate and evaluate fluence levels and the plant-specific program. The acceptance criteria are described in BTP RLSB-1 (Appendix A.1 of this SRP-SLR).

As summarized in Table 3.5-1, item 3.5.1-097, the potential for reduction of strength, loss of mechanical properties, and cracking due to irradiation of reinforced concrete is a concern for the biological shield around the reactor vessel and its support pedestal inside the drywell through the SPEO. Surrounding the reactor vessel and supported on the reactor vessel pedestal at elevation 947-ft 2-in is the biological shield whose primary function is to protect equipment inside the drywell against radiation and thermal effects. The biological shield is composed of two steel cylinders interconnected with columns and filled with concrete. Only the lower 12 feet of concrete, up to the 959-ft elevation, has been designed as structural concrete capable of resisting forces and shears. Above the

959-ft elevation the two steel cylinders and columns are structurally adequate, and the concrete fill has not been considered as adding to the support. The biological shield extends from elevation 947-ft 2-in to 993-ft 7-in (from RPV support pedestal to the seismic restraint above the RPV nozzle penetration). The biological shield concrete, like the reactor building concrete, is reinforced Type II Portland cement with a total air content of not less than 3 percent and not more than 5 percent by volume. The biological shield is approximately 26-in thick and consists of 27-in wide flange columns tied together by horizontal wide flange beams and ¼” steel plates. These plates are welded to the column flanges, both inside and outside, thereby forming a double walled shell.

The portion of the biological shield directly across from the active core will receive the maximum irradiation, which drops off significantly above and below the core. Irradiation effects on the biological shield concrete, the biological shield structural steel, and reactor vessel support structure inside the reactor cavity are evaluated below. **The evaluation starts with determining the cumulative neutron fluence and gamma dose in reactor cavity locations at 72 effective full power years (EFPY). Those values are then used in evaluating the strength and mechanical properties of the bioshield concrete. The values and associated displacements per atom (dpa) are also used regarding the potential irradiation embrittlement (ductility reduction) of the biological shield structural steel, including seismic restraints, and of the reactor vessel support structure, which comprises a steel and concrete pedestal, well below the height of the active fuel.**

The location of the bounding 72 EFPY fluence and gamma dose corresponds roughly to the core mid-plane. Figure 3.5.2.2.2.6-1 shows the elevation of structural concrete relative to the core and core mid-plane.

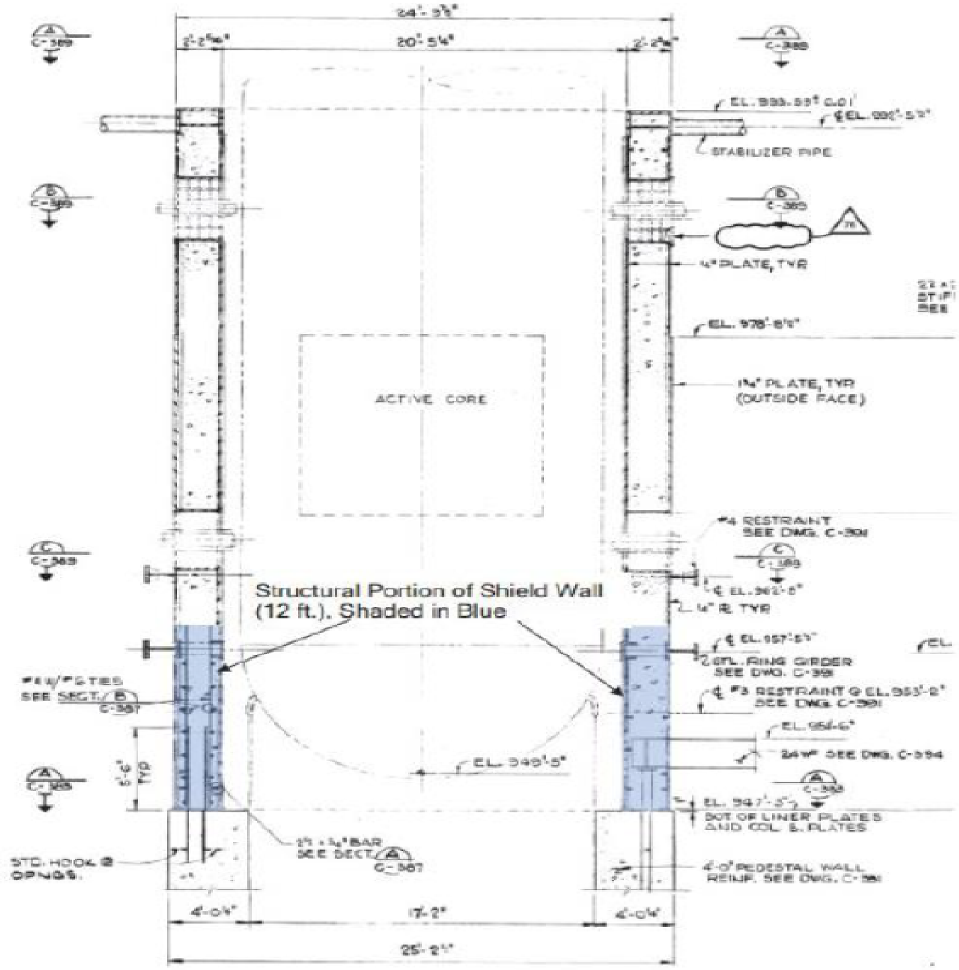

Figure 3.5.2.2.2.6-1: Section Through Monticello Biological Shield Wall

Bounding 72 EFPY radiation levels at the biological shield wall were projected. As discussed in more detail in the subsections below, the cumulative 72 EFPY neutron fluence for both epi-thermal and fast energies (cutoff energy E> 0.1 MeV) is below the recommended threshold limit of 1 x 10^{19} neutrons per square centimeter (n/cm^2), but the calculated gamma dose exceeds the recommended threshold limit for concrete degradation due to irradiation of 1 x 10^{10} rads. The reduction in strength and loss of mechanical properties of concrete from the expected irradiation levels for the biological shield wall and concrete portion of the reactor vessel support pedestal at 72 EFPY, with the effects of irradiation-related strength reduction applied to the design calculation of record, was projected. The evaluation also includes a review of the operating temperature experienced by the concrete to assess the potential for temperature-related (e.g., gamma heating) degradation.

Although the gamma dose on the biological shield wall at 72 EFPY is calculated to exceed the recognized irradiation damage threshold, the effect from the gamma dose on the capacity of the biological shield wall was calculated and determined to be negligible. The reduction in concrete compressive strength at 72 EFPY is less than 10% due to the control provided by the tensioned steel, which does not experience degradation of its mechanical properties at the calculated radiation levels. Additionally, the demand-to-allowable capacity ratios changed by 2% or less. Maximum tensile stress (including membrane, bending, and peak stress) in the steel inner liner plate is 4.49 ksi operational plus 36 ksi weld residual stress or 40.49 ksi, which is greater than the 6 ksi threshold requiring additional fracture toughness consideration in NUREG-1509. That fracture mechanics evaluation of stress intensity factors passes the NUREG-1509 acceptance criterion. As such, the properties of the tensioned steel are adequate to compensate for the change in concrete compressive strength at 72 EFPY. Therefore, the integrity of the biological shield is assured throughout the SPEO and no additional aging management of the concrete beyond the Structures Monitoring (B.2.3.33) AMP is necessary for aging effects due to irradiation.

Enclosure 03b

RAMA Fluence Methodology for Biological Shield Irradiation Evaluation

RAMA Fluence Methodology for Biological Shield Irradiation Evaluation

Summary of RAMA is Added to Neutron Fluence Biological Shield Irradiation Evaluation.

Affected SLRA Sections: 3.5.2.2.2.6

SLRA Page Numbers: 3.5-37

Description of Change:

Section 3.5.2.2.2.6, Neutron Fluence Biological Shield Irradiation Evaluation is being revised to add a summary for the TransWare Radiation Analysis Modeling Application (RAMA) Fluence Methodology used to perform the biological shield calculation.

SLRA Section 3.5.2.2.2.6 on page 3.5-37 is revised as follows:

Neutron Fluence Biological Shield Irradiation Evaluation

Relative to neutron fluence, the maximum fluence level is 5.94 x 10^{18} n/cm^2 neutron radiation (fluence cutoff energy E >1 MeV) at the inside surface of the RPV at the beltline for the 72 ~~Effective Full Power Years (~~EFPY~~)~~ projected for the SPEO. The maximum estimated fluence levels at the biological shield concrete are based upon determining the attenuation through the intervening reactor vessel shell, air gap, and inner biological shield plate thickness and determining the neutron fluence levels at the energy levels of interest regarding potential concrete damage. Neutrons of sufficient energy collide with atoms in aggregates causing atomic displacement cascades, resulting in point defects that agglomerate to cause amorphization and nanovoid formation in mineral solids and lead to swelling after a nominal threshold of fluence is reached. The interaction of neutrons with crystalline solids is typically quantified as displacements per atom (dpa), which is a measure of the number of atomic displacements from equilibrium positions in the lattice per atom in the irradiated material due to elastic atomic collisions. In concrete, swelling of the aggregates puts the cement matrix in local tensile fields, adversely affecting the mechanical properties of the concrete. Furthermore, at high enough fluence, macroscopic swelling of the concrete occurs—which may affect the internal stress morphologies in structural elements. The majority of dpa damage (to concrete) is from neutrons with energies above 0.1 ~~mega electron volts (~~MeV~~)~~.

As such, it was necessary to determine the E >0.1 MeV fluence incident on the inner surface of the concrete. A bounding neutron fluence (E >0.1 MeV) was determined for the MNGP reactor biological shield concrete at 72 EFPY. The neutron source that was used to calculate the neutron fluence, as well as the gamma dose at 72 EFPY, for the biological shield concrete is the maximum-power reactor statepoint condition that was determined to occur in Cycle 28. The computational model that was used to perform the biological shield calculation was derived from the MNGP reactor fluence model, **which used the TransWare Radiation Analysis Modeling Application (RAMA) Fluence Methodology.** ~~and has been validated in~~**In accordance** with ~~the required benchmarks cited in~~ U.S. NRC RG 1.190**, TransWare has benchmarked the RAMA Fluence Methodology as discussed in Section 4.2.1.1. Neutrons and contributing gamma rays were accounted for in the radiation analysis of the MNGP bioshield concrete. In a multi-step process, TransWare's TRANSFX radiation transport software, which supports a coupled 52-neutron / 42-gamma ray group nuclear data library, was used to calculate the initial spatial and spectral distribution of neutron and prompt gamma rays in the bioshield concrete. The neutron and prompt gamma ray spectra were then used in the ORIGEN computer code to calculate delayed and secondary gamma ray source terms for reactor materials. The ORIGEN results were then used in a second transport calculation to calculate the delayed and secondary gamma rays in the bioshield concrete. The neutron and gamma ray spectra calculated in the transport calculations were then**

assimilated to produce the final neutron fluence and gamma ray dose in the bioshield concrete.

The bounding fluence (E >0.1 MeV) incident in the inner surface of the biological shield concrete at 72 EFPY was determined to be 6.59 x 10^{18} neutrons per square centimeter (n/cm^2). In addition, Figure 5 of EPRI report 3002008128, Revision 0, July 2016, *Structural Disposition of Neutron Radiation Exposure in BWR Vessel Support Pedestals* records an 80-year reactor vessel outer diameter fluence (E >0.1 MeV) for MNGP of approximately 9.0 x 10^{18} n/cm^2. As such, the MNGP biological shield concrete fluence (E >0.1 MeV) through the SPEO is less than the recommended radiation fluence threshold of 1 *x* 10^{19} n/cm^2 for radiation damage to concrete.

Enclosure 03c

Gamma Dose Biological Shield Irradiation Evaluation Clarification

Gamma Dose Biological Shield Irradiation Evaluation Clarification

Gamma Dose Biological Shield Irradiation Evaluation Subsection is updated for clarifications.

Affected SLRA Sections: 3.5.2.2.2.6

SLRA Page Numbers: 3.5-38

Description of Change:

SLRA Section 3.5.2.2.2.6 is being revised to clarify:

- The analysis of the potential reduction in concrete strength due to gamma radiation.
- The use of a generic normalized curve for variation of gamma flux along core height for a PWR for MNGP BWR.
- The results of the separate analysis that was performed of the effects of potential reduction in concrete strength due to gamma radiation that would demonstrate structural integrity is assured for all relevant structural components.

SLRA Section 3.5.2.2.2.6 on page 3.5-38 is revised as follows:

Gamma Dose Biological Shield Irradiation Evaluation

Relative to the gamma dose incident on the biological shield concrete, the same calculation that determined the neutron fluence addressed above also determined the total gamma dose incident on the inner surface of the biological shield concrete. The bounding gamma dose for the MNGP biological shield concrete through 72 EFPY was determined to be 4.85 x 10^{10} rads. As such, the estimated 72 EFPY gamma dose incident on the inner surface of the biological shield concrete is greater than the recommended gamma radiation threshold, 1 x 10^{10} rads, for radiation damage to concrete.

Recent research on the gamma dose limit of 1 x 10^{10} rads reveals that this value may be overly conservative after subsequent reviews of previous test data. A recent paper published by I. Maruyama et al, Journal of Advanced Concrete Technology, Volume 15, 440-523 (2017), funded by the Japanese Regulator, concluded that there is no direct effect of gamma dose on concrete strength and recommends removing gamma dose limits. This paper concludes that previous studies that showed a decrease in concrete strength as a function of gamma dose were seeing an elevated temperature effect due to the high gamma flux in accelerated aging tests. Similar issues with the gamma dose limit of 1 x 10^{10} rad were also identified in NUREG/CR-7171, November 2013, *A Review of the Effects of Radiation on Microstructure and Properties of Concrete Used in Nuclear Power Plants*.

However, a separate analysis of the potential reduction in concrete strength due to gamma radiation above the recommended threshold has been completed for MNGP. **The methodology followed the evaluation procedure in industry literature for the effects on concrete properties due to gamma radiation. Controlling loads, configuration and load path were identified for the biological shield wall.** This analysis considered attenuation through the concrete, and the potential for radiation induced volumetric expansion (RIVE) of the biological shield concrete thickness that is above the damage threshold, as well as the impact to gamma heating considerations.

It was assumed that gamma flux varied along the height of the shield wall, normalized to the flux at core mid-plane, is consistent with industry literature; and that gamma dose is proportional to gamma flux. The variation along the height of the active core region in the industry literature is for a typical PWR. However, the studies reviewed by EPRI in producing the literature found that gamma dose from BWR plants are not expected to be greater than the PWRs. Based on similarity in axial gamma flux shaping data for PWR and BWR designs, the evaluation for the MNGP structural concrete conservatively assumed a strength reduction from irradiation based on 35% of the plant-specific maximum absorbed dose from gamma irradiation. The actual absorbed dose value from these profiles drops to approximately 0% of the maximum absorbed dose at the height of the structural portion of the biological shield wall relative to that of the active core and its midplane. However, 35% bounds the evaluation based on available data. Furthermore, the change in compressive strength of

concrete with gamma dose was assumed to be consistent with the lower bound curve in the pertinent industry literature.

Acceptance criteria as applied in the evaluation procedure were selected to conform to the guidelines in BRP RLSB-1, Paragraph A.1.2.3.6 of Appendix A.1 of NUREG-2192. Original design basis calculation methodologies and code of construction allowable stress levels were maintained in the design check of the biological shield wall at the end of the SPEO. Demand-to-capacity (D/C) ratios for the degraded concrete components under evaluation were determined to be less than 1.0, indicating capacity is greater than demand at 72 EFPY.

For the pedestal below the shield wall anchorage, the concrete is sufficiently remote from the active core region such that gamma dose is less than the threshold of concern, and concrete mechanical properties are not affected.

NUREG/CR-7171 (ML13325B077) and RIL 2021-07 (ML21238A064) contain equations backed by test results that show the cumulative effect of heating due to irradiation. The heating effect from gamma ray irradiation has been determined to be limited to approximately 1.12°F of temperature change due to gamma irradiation. The basis assumptions for the gamma irradiation dose (integrated energy absorption of the bioshield) evaluation at MNGP aligns with the irradiation demonstrated and discussed in RIL 2021-07.

The expansion of the concrete from this temperature increase results in a maximum of 0.9 ksi additional tensile stress during operation. Even in conservative and bounding scenarios, this level of additional stress will not affect conclusions of the basis bioshield or piping analysis or result in any additional age-related degradation mechanisms that must be addressed, monitored, or assessed.

Concrete elements are not subject to elevated temperatures in excess of 150°F weighted average (general area/bulk) and 200°F local area. Plant areas that bound high temperature considerations are the drywell general area and biological shield wall piping penetration local area, which experience temperatures of 135°F and 179°F, respectively. Insulation is credited with maintaining the penetration temperatures below the local limits of 200°F, as described in Section 3.5.2.2.1.2.

In addition, at mid-core height, where peak radiation is expected, thermal concrete expansion might induce additional stress on the steel liner and a finite element model was used to evaluate the radial temperature profile (of the bioshield wall) from heat loads resulting from the reactor vessel temperature under operating conditions. To evaluate thermal-induced damage, the derived concrete temperature in the reinforced concrete was compared to the American Concrete Institute (ACI) temperature limits. The effects of gamma-heating in the concrete were also considered. Based on the heat transfer analysis results, the maximum expected temperature on concrete surface of the MNGP bioshield is 140.69°F. Temperature increase due to gamma-heating can be approximated to 1.12°F, which brings the maximum expected concrete surface temperature to approximately 142°F;

below the ACI limit of 150°F. Hence, thermal induced damage in the MNGP bioshield concrete material is not of concern for the SPEO.

As a result, the integrity of the biological shield is assured, and no additional aging management of the biological shield concrete beyond the current Structures Monitoring (B.2.3.33) AMP is necessary for aging effects due to irradiation during the SPEO. As such, there is reasonable assurance that a loss (or reduction) of concrete strength, loss of mechanical properties, and cracking will not affect the ability of the biological shield concrete to perform its component intended functions through the SPEO.

Enclosure 03d

Reactor Vessel Support Steel Irradiation Evaluation Clarifications

Reactor Vessel Support Steel Irradiation Evaluation Clarifications

SLRA is modified to clarify that BWRVIP-342 is utilized as a reference only and an MNGP specific analysis was performed to assess the effect of irradiation.

Affected SLRA Sections: 3.5.2.2.2.6

SLRA Page Numbers: 3.5-38 and 3.5-39

Description of Change:

Revise Section 3.5.2.2.2.6 to clarify how MNPG plant specific parameters are bounded by the evaluations in BWRVIP-342.

SLRA Section 3.5.2.2.2.6, subsection “Biological Shield Structural Steel Evaluation” is modified to include an explanation and a further discussion (with the parameter values from evaluation 2200285.302.R0) to demonstrate Initial NDT + shift < lowest service temperature of the steel is subject to with a sufficient margin (using the guidance on sufficient margin in NUREG-1509).

Update the SLRA Section 3.5.2.2.2.6, to include additional information describing the evaluation of irradiation effects on the steel RV seismic restraint and stabilizer structure components (brackets, tension rods, couplings, trusses, etc.) for loss of fracture toughness due to irradiation embrittlement effects.

SLRA Section 3.5.2.2.2.6 on pages 3.5-38 and 3.5-39 is revised as follows:

Reactor Vessel Support Steel Irradiation Evaluation

In addition to the potential aging effects due to irradiation of reinforced concrete, a loss (or reduction) in fracture toughness due to irradiation embrittlement of the reactor vessel support steel is a potential aging effect considered. The reactor vessel is shown in USAR Figures 3.6-2 and 4.2-1. The reactor vessel support structures at MNGP are described in USAR Section 4.2.2. The reactor vessel is supported by a steel skirt. The top of the skirt is welded to the bottom of the vessel. The skirt is then supported by a concrete and steel pedestal, which carries the load through the drywell to the reactor building foundation slab. Stabilizer brackets, located below the vessel flange, **and well above the active core region,** are connected to tension bars with flexible couplings. The bars are then connected to stabilizer brackets located on top of the biological shield wall to limit horizontal vibration and to resist seismic and jet reaction forces. The reactor pedestal is concrete with a ¼ in. steel liner on the exterior face. The reactor support skirt is bonded to the inside face. A 3-inch layer of pneumatically applied mortar covers the inside face of the skirt. As listed in Table 3.5.2-1, the reactor vessel support skirt is fabricated from steel. Also, the USAR Section 4.2.4.1 notes that the initial NDT temperature for the reactor vessel bottom head to which the support skirt is welded, is no higher than 40°F.

NUREG-1509, May 1996, *Radiation Effects on Pressure Vessel Supports*, is a resource for addressing irradiation embrittlement for SLR. NUREG-1509, Section 4.2.1 notes that radiation embrittlement is not an issue for reactor vessel support skirts. **In addition to NUREG-1509,** BWRVIP-342, *Aging Management of Reactor Vessel (RV) Supports for Extended operations*, 20222 (EPRI Report 3002020999), has recently been prepared to address irradiation of the RV support using the methodologies described in NUREG-1509. **Therefore, BWRVIP-342 was also used to provide additional clarification. BWRVIP-342 was referenced for guidance in interpreting the effects of irradiation on hardening and embrittlement of steel supports in the calculation of record for MNGP. The information referenced in BWRVIP-342 is independent of the MNGP RV support structure configuration. Data cited in BWRVIP-342 has no bearing on actual design basis transients and calculated design loads used in the analysis for MNGP.**

As listed in Figure 3-1 of the EPRI document, MNGP is within the locus of GE-designed BWRs for which bounding design transients, maximum design loads and operating conditions are evaluated in the report. Furthermore, the 72 EFPY fast fluence (E >1 MeV) for the reactor vessel support skirt located well below the active fuel, as well as for the lateral supports **(which includes the seismic restraint and stabilizer structural components),** located well above the active fuel, is estimated to remain below the 1 x 10^{17} n/cm^2 threshold for embrittlement of steel. Table 4.2.1.1-2 shows the reactor vessel beltline region. USAR Figure 4.2-1 shows the length of the reactor vessel and location of the active core. The radiation fields are significantly reduced above and below this region considering distance correction factors using the inverse square law as described in EPRI 3002008128. MNGP reactor vessel fluence calculations, projected for 72 EFPY, were performed using TransWare ~~Radiation Analysis Modeling Application (~~RAMA~~)~~ Fluence Methodology. In compliance with

RG 1.190, TransWare has benchmarked the RAMA Fluence Methodology against industry standard benchmarks and plant-specific dosimetry measurements for BWRs and PWRs. The results of the benchmarking show that the fluence methodology implemented by TransWare is capable of predicting specimen activities with no discernable bias in the computed fluence.

The fluence value (E >1 MeV) at 72 EFPY reported near the core mid-height is 5.94 x 10^{18} n/cm^2, which exceeds the steel embrittlement threshold of 1 x 10^{17} n/cm^2 (E >.1 MeV). The RPV beltline region is defined as the region in which fast neutron fluence exceeds 1 x 10^{17} n/cm^2 (E >1 MeV). The MNGP beltline region is a total of 16.1-ft. The RPV lateral supports, seismic restraint, and stabilizer structure are located approximately 11-ft above the reactor beltline, and therefore, are sufficiently remote from the active core and not subject to fluence (E >1 MeV) above the 1 x 10^{17} n/cm^2 (E >1 MeV) steel embrittlement threshold.

The fluence value (E >1 MeV) at 72 EFPY reported at a nozzle location below the reactor beltline is 3.25 x 10^{16} n/cm^2. The top portion (knuckle region) of the support skirt is approximately 11 feet below the bottom of active fuel. As such, the fluence (E >1 MeV) at the MNGP reactor vessel skirt is below the 1 x 10^{17} n/cm^2 (E >1 MeV) embrittlement threshold. ~~Therefore, the conclusions of EPRI 300202099 are applicable to MNGP.~~

~~The EPRI document evaluates the estimated maximum fluence levels and degree of embrittlement that was projected for the high stress (knuckle) region of the BWR reactor vessel supports. Also, the temperatures and loading conditions in the knuckle region were examined to determine whether irradiation induced embrittlement of the reactor vessel support steel could reduce the level of toughness and affect the margins against brittle fracture. The EPRI document concludes that the predicted level of embrittlement is minimal, using the appropriate embrittlement trend curve model for the BWR vessel supports after 80 years of plant operation. The predicted level of embrittlement is minimal since the fluence is low, the operating temperature is high, and the ductility of the skirt knuckle region is high. Therefore,~~ **Although** the integrity of the reactor vessel support is assured, **with fluence below the threshold at both the knuckle and seismic stabilizers,** ~~and no additional aging management of reactor vessel supports beyond~~ the current ASME Section XI, Subsection IWF (B.2.3.30) AMP **inspection of the RV supports will also confirm there is no visible evidence of a loss of fracture toughness (e.g., cracking).** ~~is necessary for aging effects due to irradiation~~ during the MNGP SPEO. ~~Accordingly, there is reasonable assurance that a loss (or reduction) of fracture toughness due to irradiation embrittlement will not affect the ability of the RV support steel to perform its component intended functions through the SPEO.~~

Enclosure 03e

Biological Shield Structural Steel Evaluation Clarifications

Biological Shield Structural Steel Evaluation Clarifications

Revise SLRA Section 3.5.2.2.2.6 to provide clarifications to the Biological Shield Structural Steel Evaluation.

Affected SLRA Sections: 3.5.2.2.2.6

SLRA Page Numbers: 3.5-40 and 3.5-41

Description of Change:

SLRA Section 3.5.2.2.2.6 is revised to provide the following:

- an explanation and further discussion demonstrating Initial NDT + shift is less than the lowest service temperature the steel is subject to with a sufficient margin.
- additional information describing the stress analysis load combinations, the controlling load combination, and the limiting load combinations (5, 6, and 7).
- additional information to update the conclusions for the Structures Monitoring AMP that includes managing the aging effects of loss of fracture toughness for the biological shield steel components for which the Structures Monitoring AMP is credited.

The original design bioshield stress analysis uses the 1969 AISC to analyze the allowable stress. This is documented in the USAR (Reference 1) as well as the June 8, 1972 letter from Northern States Power to the U.S. Atomic Energy Commission (Reference 2). The original design bioshield stress analysis was compared to the concrete degradation evaluation to validate that the allowable capacity is greater than the applied demand.

References:

1. Monticello Updated Safety Analysis Report.
2. Letter, Monticello Nuclear Generating Plant E-5979 AEC Operating License - Reporting, June 8, 1972, ML112970770

SLRA 3.5.2.2.2.6 on pages 3.5-40 and 3.5-41 is revised as follows:

Biological Shield Structural Steel Evaluation

As described above, the biological shield is approximately 26-inch thick and consists of 27-inch wide flange columns tied together by horizontal wide flange beams and steel plates. These plates are welded to the column flanges, both inside and outside, thereby forming an interior and exterior steel liner.

Similar to the reactor vessel support steel addressed above, the potential effects of irradiation on the steel elements (wide flange columns, liner, and welds) of the biological shield across from the active core height are addressed.

NUREG-1509 maps an approach for evaluating radiation embrittlement of RV support steel using the following key criteria. If this criteria are met, radiation embrittlement would be considered negligible and its integrity can be reasonably assured with no need for further investigation.

- **Criterion 1: The end-of-life radiation exposure at the biological shield wall is low (2.0 x 10^{-5} displacements per atom (dpa) or less).**
- **Criterion 2: The nil-ductility transition (NDT) temperature of the biological shield wall steel is below the minimum operating temperature.**
- **Criterion 3: The peak tensile stresses are 6 ksi, or less.**

In the event radiation exposure of the steel exceeds the embrittlement threshold (i.e., criteria 1 is not met), NUREG-1509 recommends a fracture mechanics evaluation also be performed.

The same logic was used to assess radiation embrittlement of the biological shield wall steel. ~~However, the conclusions of EPRI 300202099 are not applicable to the MNGP biological shield structural steel as the 72 EFPY fluence (E >1 MeV) across from the active fuel is above the 1 x 10^{17} n/cm^2 threshold for embrittlement of steel. As described above, the biological shield is approximately 26-in thick and consists of 27-in wide flange columns tied together by horizontal wide flange beams and ¼” steel plates. These plates are welded to the column flanges, both inside and outside, thereby forming a double walled shell.~~

Criteria #1

The maximum dpa, occurring at the mid-height of the active fuel, is 2.07 x 10^{-3}; which is above the 2 x 10^{-5} threshold for embrittlement of steel. Therefore, a fracture mechanics evaluation was performed in accordance with NUREG-1509.

Criteria #2

~~Similar to NUREG-1509, t~~**T**he reduction in fracture toughness assessment of the biological shield structural steel can be based on a transition temperature analysis, wherein a demonstration is made that there is adequate margin between the normal operating temperature and the ductile-to-brittle fracture mode transition temperature (commonly known as the nil-ductility transition (NDT) temperature for end-of-life/license (EOL) conditions) or fracture toughness evaluations. The transition temperature approach is based on the proposition that catastrophic failure by brittle fracture can be avoided by maintaining the normal operating biological shield service temperature above the NDT temperature of the steel. When using the transition temperature to evaluate the biological shield integrity, the NDT temperature at EOL should include the irradiation induced shift.

MNGP normal operating temperatures range from 100°F to 136°F inside the drywell. Section 5.2 of the MNGP USAR states that the primary containment cooling and ventilation system consists of four air coolers, ductwork, fans, and controls which maintain the drywell atmosphere below a 135°F bulk average temperature. Within the biological shield wall annulus the normal operating temperature ranges from 112°F to 141°F. At the locations of penetrations through the biological shield wall, local concrete temperatures do not exceed 179°F.

As described in the original construction specifications and confirmed in the material receipt records, the steel elements of the biological shield wall, consisting of the columns, 1/4-inch thick steel liner plates, and transfer beams, are fabricated from steel conforming to ASTM A36 low carbon steel. ~~The assumed initial (unirradiated) NDT temperature, plus 1.3σ, provided in NUREG-1509 Table 4-1 and Table 4-2 for this carbon-manganese material is 39°F.~~ The original specification did not specify that any additional copper or nickel be incorporated into the ASTM A36 material and there are no chemical measurements for copper or nickel in material receipt records for the MNGP biological shield structural steel made from ASTM A36 low carbon steel.

NUREG-1509 provides a method for approximating the NDT shift by determining exposure in terms of displacements per atom (dpa), and then using Figure 3-1 of that reference to establish the irradiation induced shift of the NDT. By fitting the experimental data in NUREG-1509, a trend curve prediction model was developed for embrittlement shift versus dpa that incorporated the effects of flux and fluence, irradiation temperature, and gamma heating as shown by the upper bound line in Figure 3-1 in NUREG-1509. That model included an upper bound transition temperature shift that was adjusted with zero-degree shift at a dpa of 10^{-5}.

For the purpose of this evaluation of the biological shield structural steel, use of the NUREG-1509 trend curve model for NDT shift versus dpa is conservative since there is little copper in the ASTM A36 materials and because the ratio of low energy neutrons to fast neutrons in the biological shield is much smaller than that used in the test reactor experiments. Fluence calculations were performed to confirm the attenuation effects through the reactor vessel and outward to the biological shield. The bounding fluence (E >0.1 MeV) incident on the inner surface of the biological shield at 72 EFPY was determined to be 6.59×10^{18} n/cm^2. The peak fluence at the biological shield inner diameter for 72 EFPY equates to a displacement per atom = 2.07×10^{-3} dpa.

The potential irradiation induced NDT is a function of the dpa fluence shown in NUREG-1509 Figure 3-1. The dashed upper bound curve is based on the fit to the experimental test data for reactor vessel carbon steel support materials (which did not include ASTM A36 materials) under low temperature, low flux neutron exposure conditions. As a result, the weld materials are similar to the ASTM A36 materials for the purposes of this further evaluation, and the same conclusions are made regarding the potential effects of irradiation induced embrittlement for the weld materials incorporated into the biological shield wall as were made regarding the biological shield wall steel elements.

Summary of Transition Temperature Evaluation

The maximum dpa of 2.07 x 10^{-3} is utilized as a limiting value for all steel within the bioshield. As part of NUREG-1509 guidance, ASTM A36 steel is a carbon-manganese steel, and therefore has an initial NDT of -28°F. Using the NUREG-1509 Figure 3-1 upper-bound curve, for the given dpa value, the NDT shift is equivalent to approximately 129.6°F (72°C).

Adjusted NDT = Initial NDT + NDT Shift

101.6°F = -28°F + 129.6°F

The adjusted NDT of 101.6°F is below the plants operating temperature of 111.9°F (lower bound temperature of conservative thermal analysis).

Criteria #3 and Fracture Mechanics

~~An evaluation of the steel elements of the MNGP biological shield wall was performed by identifying the region of the shield wall subject to high fluence levels and to NDT temperature plus shift near the range of expected operating temperatures. Stress analysis of the area of interest of the shield wall was performed using finite element analysis.~~ To evaluate the stress levels in the biological shield wall, the entire shield wall structure was modeled in ANSYS, including portions of the liner**, stabilizers, and restraints. The finite element model was constructed in stages. The first evolution of the analysis was modeled after the original design basis stress analysis model, consisting of beam elements in a 3-D frame analysis. Design basis loads for the shield wall space frame, documented in the original design calculations, were applied to the ANSYS frame model and combined into design basis load combinations. Results were compared to resulting member forces and moments reported in the original design calculations to ensure conservatism and modeling accuracy. Loading consists of the controlling design basis loads and load combinations applied in the original calculations. The controlling load combinations presented in the design basis stress analysis model and implemented in the SLRA stress analysis were as follows:**

Load 5: Jet Force P=127 kip at El. 962-8
Load 6: Preload P=80 kip at Vessel Stabilizers
Load 7: Seismic Force P=600 kip at Vessel Stabilizers

The controlling load case is Load 5 + Load 6 + Load 7.

Maximum tensile stress in the area of interest (adjacent to the active core region) was determined to be 4.49 ksi, which is less than the 6 ksi set NUREG-1509 where fluence levels and NDT temperature plus shift warrant consideration of tensile stress levels through more-detailed fracture mechanics analysis.

Although the 6 ksi criterion from NUREG-1509 is satisfied for operational stresses, the conservative inclusion of residual weld stresses pushes the stress level above the 6 ksi criterion, although basis design criteria remain satisfied. The residual weld stress can be up to the yield strength of the material. Therefore, additional evaluation for fracture mechanics and the transition temperature approach have been assessed.

For the transition temperature approach, a plant-specific thermal analysis has been performed for MNGP, based on temperature values from instrumentation and Technical Specification Limits of bulk and insulation temperature. The bounding operating temperature of the assembly material is 111.9°F. Contributions to temperature from gamma heating have been ignored for the purposes of transition temperature and fracture mechanics evaluation. The maximum delta NDTT temperature of the material is 121.04°F, and therefore the transition temperature is 101.6°F. This material property limitation is exceeded by the bounding bioshield operating temperature of 111.9°F, and therefore the change in transition temperature will not affect the stability and operability of the bioshield and liner.

Plant-specific CMTR data has been assessed for evaluation of fracture toughness properties. With a K_{IC} of 58.7 ksi-in$^{1/2}$, the limiting stress intensity factor of 19.4 ksi-in$^{1/2}$ remains below the material fracture toughness value. Therefore, the decrease in fracture toughness to the end of the extended period of operation will not affect the stability and operability of the bioshield and liner.

Accordingly, the potential effects of irradiation on the steel elements of the biological shield, including the welding material, are not significant. ~~As a result,~~ **While** the integrity of the biological shield is assured, **conservatively the current Structures Monitoring (B.2.3.33) AMP will serve to ensure there is not a loss of fracture toughness for the biological shield wall structural steel. The Structures Monitoring (B.2.3.33) AMP manages loss of material for the accessible portions of the biological shield wall steel liners. The condition of the liners will be used to indicate the condition of the remaining biological shield wall steel.** ~~and n~~**No** additional aging management of the biological shield **wall** structural steel beyond the current Structures Monitoring (B.2.3.33) AMP is necessary for aging effects due to irradiation during the SPEO. ~~As such, there is reasonable assurance that a loss (or reduction) of fracture toughness due to irradiation embrittlement will not affect the ability of the biological shield structural steel to perform its component intended functions through the SPEO~~.

Enclosure 04

Concrete Aging Management Review – Containment Temperature Control Clarification

Concrete Aging Management Review – Containment Temperature Control Clarification

Clarification of concrete temperature controls

Affected SLRA Sections: 3.5.2.2.1.2

SLRA Page Numbers: 3.5-20

Description of Change:

Section 3.5.2.2.1.2 is revised to add additional details related to the control of temperature within the containment.

Section 3.5.2.2.1.2 on page 3.5-20 is revised to as follows:

3.5.2.2.1.2 Reduction of Strength and Modulus Due to Elevated Temperature (as supplemented by SLR-ISG-2021-03-STRUCTURES)

Reduction of strength and modulus of concrete due to elevated temperatures could occur in PWR and BWR concrete and steel containments. The implementation of 10 CFR 50.55a and ASME Code Section XI, Subsection IWL would not be able to identify the reduction of strength and modulus of concrete due to elevated temperature. Subsection CC-3440 of ASME Code Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. Further evaluation is recommended to determine the need for a plant-specific AMP or plant-specific enhancements to ASME Code Section XI Subsection IWL and/or Structures Monitoring AMPs, essential to manage these aging effects for portion of the concrete containment components that exceed specified temperature limits {i.e., general area temperature greater than 66 degrees Celsius (150 degrees Fahrenheit) and local area temperature greater than 93 degrees Celsius (200 degrees Fahrenheit). Higher temperatures may be allowed if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations. Acceptance criteria are described in Branch Technical Position (BTP) RLSB (License Renewal and Standardization Branch)-1, "Aging Management Review – Generic, July 2017" (Appendix A.1 of this SRP-SLR).

Elevated temperature impacts on concrete were addressed during the initial license renewal. This aging effect mainly concerns PWR and BWR Mark II and III concrete containments; however, the temperature criteria presented in this section apply to all concrete. Plant documents confirm that concrete elements are not subject to elevated temperatures in excess of 150°F generally and 200°F locally. Plant areas that bound high temperature considerations are the drywell general area and biological shield wall piping penetration local area, which experience temperatures of 135°F and 179°F, respectively. Additionally, normal temperature, pressure, and humidity conditions either do not significantly change due to the EPU or remain bounded by values used in the current analysis

As summarized in item 3.5.1-003, reduction of strength and modulus of concrete due to elevated temperatures is not applicable to the MNGP Mark I steel containment. The bulk drywell temperature is maintained by the primary containment ventilating and cooling system. ~~The average air temperature inside the drywell during normal plant operation is limited to 135°F.~~ **MNGP Technical Specification Limiting Condition for Operation 3.6.1.4 requires the drywell average air temperature to be less than or equal to 135°F. Technical Specification Surveillance requirement SR 3.6.1.4.1 requires periodic verification that the containment temperatures remain within the limit. Monitoring and maintaining drywell average air temperature within limits ensures aging effects related to elevated general area concrete temperature are not present.** Therefore, concrete structural components located inside the drywell are not subject to general area temperatures greater than 150°F.

Surrounding the reactor vessel and supported on the reactor vessel pedestal is the biological shield whose primary function is to protect equipment inside the drywell against radiation and thermal effects. Local area temperature in the biological shield wall due to hot reactor REC System penetrations is calculated at 179°F; less than the concrete degradation threshold of 200°F. **The calculation of local concrete temperatures conservatively assumed a 25°F increase in temperature due to nuclear heating effects. This bounds the anticipated gamma heating effects for concrete discussed in Section 3.5.2.2.2.6.** Consistent with NUREG-1865, thermal insulation is credited with maintaining the temperatures in the bioshield wall below 200°F and are therefore within the scope of license renewal and subject to an AMR as described in Table 3.5.2-7.

Enclosure 05

Correction of the Intended Functions Associated with the CRD System

Correction of the Intended Functions Associated with the CRD System

Revisions to the Intended Functions for the Control Rod Drive System

Affected SLRA Sections: Table 2.3.3-4, Table 3.3.2-4

SLRA Page Numbers: 2.3-32, 3.3-111 through 3.3-118

Description of Change:

This supplement corrects the component types and intended functions associated with the Control Rod Drive System. Primarily, the leakage boundary intended function is identified where appropriate and replaces either the structural integrity (attached) or pressure boundary intended function. The throttle function of the orifices is removed as the orifices are corrected to only be in scope for SLR in accordance with 10 CFR 54.4(a)(2). Similarly, the heat transfer intended function for the CRD PMP Thrust BRG CLR heat exchanger tubes is removed as they are also now correctly identified as in scope for 10 CFR 54.4(a)(2).

There is no carbon steel piping exposed to treated water associated with the scram discharge volume and as such the fatigue line item for this material and environment is removed.

Black bold font information in Table 2.3.3-4 and Table 3.3.2-4 represents changes made in Attachment 1, Enclosure 06b and Enclosure 12 of Supplement 4 (Reference 1). Table 3.3.2-4 on pages 3.3-111 through 3.3-118 is replaced in whole and supersedes the changes made in Attachment 1, Enclosure 06b and Enclosure 12 of Supplement 4 (Reference 1). Table 3.3.2-4 on page 3.3-110 is not changed by this Enclosure and so it was not included with the replacement of the remainder of the table.

References:

1. L-MT-23-031, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 4 and Response to Request for Confirmation of Information - Set 1, ML23199A154

Table 2.3.3-4 on page 2.3-32 is revised as follows:

Table 2.3.3-4
Control Rod Drive System Components Subject to Aging Management Review

Component Type	Component Intended Function(s)
Accumulator (Scram)	Pressure Boundary
Bolting (Closure)	Mechanical Closure
Heat Exchanger (CRD PMP Thrust BRG CLR) Shell **Side Components**	~~Pressure Boundary~~ **Leakage Boundary**
Heat Exchanger (CRD PMP Thrust BRG CLR) Tube~~s~~ **Side Components**	~~Heat Transfer~~ ~~Pressure Boundary~~ **Leakage Boundary**
Orifice	~~Pressure Boundary~~ **Leakage Boundary** ~~Throttle~~
Piping, Piping Components	**Leakage Boundary** Pressure Boundary ~~Structural Integrity (Attached)~~
Pump Casing (CRD)	~~Pressure Boundary~~ **Leakage Boundary**
Pump Casing (Lubricating Oil)	~~Pressure Boundary~~ **Leakage Boundary**
Speed Increaser Assembly	~~Pressure Boundary~~ **Leakage Boundary**
Tanks (Scram Discharge)	Pressure Boundary
Valve Body	**Leakage Boundary** Pressure Boundary

Table 3.3.2-4 on pages 3.3-111 through 3.3-118 is revised as follows:

Table 3.3.2-4: Control Rod Drive – 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
~~Bolting (Closure)~~	~~Mechanical Closure~~	~~Stainless Steel Bolting~~	~~Air - Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~Bolting Integrity (B.2.3.10)~~	~~VII.I.A-03~~	~~3.3.1-012~~	~~A~~
~~Bolting (Closure)~~	~~Mechanical Closure~~	~~Stainless Steel Bolting~~	~~Air - Indoor Uncontrolled (External)~~	~~Loss of Preload~~	~~Bolting Integrity (B.2.3.10)~~	~~VII.I.AP-124~~	~~3.3.1-015~~	~~A~~
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Shell Side Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.3.23)~~	~~VII.I.AP-41~~	~~3.3.1-080~~	~~A~~
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Shell Side Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~Lubricating Oil Analysis (B.2.3.25)~~	~~VII.H2.AP-131~~	~~3.3.1-098~~	~~A~~
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Shell Side Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.H2.AP-131~~	~~3.3.1-098~~	~~A~~
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Tubes~~	~~Heat Transfer~~	~~Carbon Steel~~	~~Closed Cycle Cooling Water (Internal)~~	~~Reduction of Heat Transfer~~	~~Closed Treated Water Systems (B.2.3.12)~~	~~VII.F1.AP-204~~	~~3.3.1-050~~	~~A~~
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Tubes~~	~~Heat Transfer~~	~~Carbon Steel~~	~~Lubricating Oil (External)~~	~~Reduction of Heat Transfer~~	~~Lubricating Oil Analysis (B.2.3.25)~~	~~VII.E4.A-791~~	~~3.3.1-257~~	~~A~~

Table 3.3.2-4: Control Rod Drive – 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
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Tubes~~	~~Heat Transfer~~	~~Carbon Steel~~	~~Lubricating Oil (External)~~	~~Reduction of Heat Transfer~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.A-791~~	~~3.3.1-257~~	~~A~~
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Tubes~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Closed Cycle Cooling Water (Internal)~~	~~Loss of Material~~	~~Closed Treated Water Systems (B.2.3.12)~~	~~VII.C2.AP-189~~	~~3.3.1-046~~	~~A~~
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Tubes~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (External)~~	~~Loss of Material~~	~~Lubricating Oil Analysis (B.2.3.25)~~	~~VII.H2.AP-131~~	~~3.3.1-098~~	~~A~~
~~Heat Exchanger – (CRD PMP Thrust BRG CLR) Tubes~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (External)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.H2.AP-131~~	~~3.3.1-098~~	~~A~~
~~Orifice~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.3.23)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
~~Orifice~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~A~~
~~Orifice~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~B~~
~~Orifice~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-209a~~	~~3.3.1-004~~	~~A~~

Table 3.3.2-4: Control Rod Drive – 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
~~Orifice~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.I.A-751b~~	~~3.3.1-222~~	~~A~~
~~Orifice~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~A~~
~~Orifice~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~B~~
~~Orifice~~	~~Throttle~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~A~~
~~Orifice~~	~~Throttle~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~A~~
~~Orifice~~	~~Throttle~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~B~~
~~Orifice~~	~~Throttle~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~B~~
~~Piping, Piping Components~~	**~~Leakage Boundary~~**	**~~Stainless Steel~~**	**~~Underground (External)~~**	**~~Loss of Material~~**	**~~Buried and Underground Piping and Tanks (B.2.3.27)~~**	**~~VII.I.A-775b~~**	**~~3.3.1-246~~**	**~~B~~**
~~Piping, Piping Components~~	**~~Leakage Boundary~~**	**~~Stainless Steel~~**	**~~Underground (External)~~**	**~~Cracking~~**	**~~Buried and Underground Piping and Tanks (B.2.3.27)~~**	**~~VII.I.A-714b~~**	**~~3.3.1-146~~**	**~~B~~**

Table 3.3.2-4: Control Rod Drive – 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
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air - Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.3.23)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
~~**Piping, Piping Components**~~	~~**Pressure Boundary**~~	~~**Carbon Steel**~~	~~**Condensation (Internal)**~~	~~**Cumulative Fatigue Damage**~~	~~**TLAA – Section 4.3, Metal Fatigue**~~	~~**VII.E3.A-34**~~	~~**3.3.1-002**~~	~~**A**~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Condensation (Internal)~~	~~Loss of Material~~	~~Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.3.24)~~	~~VII.D.A-26~~	~~3.3.1-055~~	~~A~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~Lubricating Oil Analysis (B.2.3.25)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~
~~**Piping, Piping Components**~~	~~**Pressure Boundary**~~	~~**Carbon Steel**~~	~~**Treated Water (Internal)**~~	~~**Cumulative Fatigue Damage**~~	~~**TLAA – Section 4.3, Metal Fatigue**~~	~~**VII.E3.A-34**~~	~~**3.3.1-002**~~	~~**A**~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~A~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~B~~

Table 3.3.2-4: Control Rod Drive – 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
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Copper Alloy with 15% Zinc or Less~~	~~Air – Dry (Internal) (added Internal)~~	~~Loss of Material~~	~~Compressed Air Monitoring (B.2.3.14)~~	~~VII.D.A-764~~	~~3.3.1-235~~	~~A~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Copper Alloy with 15%~~	~~Air – Indoor Uncontrolled (External)~~	~~None~~	~~None~~	~~VII.J.AP-144~~	~~3.3.1-114~~	~~A~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-209a~~	~~3.3.1-004~~	~~A~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Condensation (Internal)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-209a~~	~~3.3.1-004~~	~~A~~
~~Piping, Piping Components~~	**~~Pressure Boundary~~**	**~~Stainless Steel~~**	**~~Condensation (Internal)~~**	**~~Cumulative Fatigue Damage~~**	**~~TLAA – Section 4.3, Metal Fatigue~~**	**~~VII.E3.A-62~~**	**~~3.3.1-002~~**	**~~A~~**
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Condensation (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.I.A-751b~~	~~3.3.1-222~~	~~A~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Gas (Internal)~~	~~None~~	~~None~~	~~VII.J.AP-22~~	~~3.3.1-120~~	~~A~~
~~Piping, Piping Components~~	**~~Pressure Boundary~~**	**~~Stainless Steel~~**	**~~Treated Water (Internal)~~**	**~~Cumulative Fatigue Damage~~**	**~~TLAA – Section 4.3, Metal Fatigue~~**	**~~VII.E3.A-62~~**	**~~3.3.1-002~~**	**~~A~~**
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~A~~

Table 3.3.2-4: Control Rod Drive – 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
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~B~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Treated Water >140 F (Internal)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.A-773~~	~~3.3.1-244~~	~~A~~
~~Piping, Piping Components~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Treated Water >140 F (Internal)~~	~~Cracking~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.A-773~~	~~3.3.1-244~~	~~B~~
~~Piping, Piping Components~~	**~~Pressure Boundary~~**	**~~Stainless Steel~~**	**~~Treated Water >140 F (Internal)~~**	**~~Cumulative Fatigue Damage~~**	**~~TLAA – Section 4.3, Metal Fatigue~~**	**~~VII.E3.A-62~~**	**~~3.3.1-002~~**	**~~A~~**
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Carbon Steel~~	~~Air - Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.3.23)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Carbon Steel~~	~~Condensation (Internal)~~	~~Loss of Material~~	~~Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.3.24)~~	~~VII.D.A-26~~	~~3.3.1-055~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~Lubricating Oil Analysis (B.2.3.25)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~

Table 3.3.2-4: Control Rod Drive – 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
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~B~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Copper Alloy with 15% Zinc or Less~~	~~Air – Indoor Uncontrolled (External)~~	~~None~~	~~None~~	~~VII.J.AP-144~~	~~3.3.1-114~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Copper Alloy with 15% Zinc or Less~~	~~Gas (Internal)~~	~~None~~	~~None~~	~~VII.J.AP-9~~	~~3.3.1-114~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Stainless Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-209a~~	~~3.3.1-004~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Stainless Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.I.A-751b~~	~~3.3.1-222~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Stainless Steel~~	~~Condensation (Internal)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-209a~~	~~3.3.1-004~~	~~A~~

Table 3.3.2-4: Control Rod Drive – 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
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Stainless Steel~~	~~Gas (Internal)~~	~~None~~	~~None~~	~~VII.J.AP-22~~	~~3.3.1-120~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~A~~
~~Piping, Piping Components~~	~~Structural Integrity (Attached)~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~B~~
~~Pump Casing (CRD)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.3.23)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
~~Pump Casing (CRD)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~A~~
~~Pump Casing (CRD)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~B~~
~~Pump Casing (Lubricating Oil)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.3.23)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
~~Pump Casing (Lubricating Oil)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~Lubricating Oil Analysis (B.2.3.25)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~

Table 3.3.2-4: Control Rod Drive – 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
~~Pump Casing (Lubricating Oil)~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~
~~Speed Increaser Assembly~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.3.23)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
~~Speed Increaser Assembly~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~Lubricating Oil Analysis (B.2.3.25)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~
~~Speed Increaser Assembly~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~
~~Tanks (SCRAM Discharge)~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.I.A-751b~~	~~3.3.1-222~~	~~A~~
~~Tanks (SCRAM Discharge)~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Condensation (Internal)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-209a~~	~~3.3.1-004~~	~~C~~
~~Tanks (SCRAM Discharge)~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Condensation (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.I.A-751b~~	~~3.3.1-222~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Aluminum~~	~~Air – Indoor Uncontrolled (External)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.A4.A-451a~~	~~3.3.1-189~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Aluminum~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.A4.A-763a~~	~~3.3.1-234~~	~~A~~

Table 3.3.2-4: Control Rod Drive – 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
~~Valve Body~~	~~Pressure Boundary~~	~~Aluminum~~	~~Gas (Internal)~~	~~None~~	~~None~~	~~VII.J.AP-37~~	~~3.3.1-113~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.3.23)~~	~~VII.I.A-77~~	~~3.3.1-078~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~Lubricating Oil Analysis (B.2.3.25)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Lubricating Oil (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-127~~	~~3.3.1-097~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Carbon Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-106~~	~~3.3.1-021~~	~~B~~
~~Valve Body~~	~~Pressure Boundary~~	~~Copper Alloy with 15% Zinc or Less~~	~~Air – Dry (Internal)~~	~~Loss of Material~~	~~Compressed Air Monitoring (B.2.3.14)~~	~~VII.D.A-764~~	~~3.3.1-235~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Copper Alloy with 15% Zinc or Less~~	~~Air – Indoor Uncontrolled (External)~~	~~None~~	~~None~~	~~VII.J.AP-144~~	~~3.3.1-114~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Air – Indoor Uncontrolled (External)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-209a~~	~~3.3.1-004~~	~~A~~

Table 3.3.2-4: Control Rod Drive – 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
~~Valve Body~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Air - Indoor Uncontrolled (External)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.I.A-751b~~	~~3.3.1-222~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Condensation (Internal)~~	~~Cracking~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-209a~~	~~3.3.1-004~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Condensation (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.I.A-751b~~	~~3.3.1-222~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Gas (Internal)~~	~~None~~	~~None~~	~~VII.J.AP-22~~	~~3.3.1-120~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~One-Time Inspection (B.2.3.20)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~A~~
~~Valve Body~~	~~Pressure Boundary~~	~~Stainless Steel~~	~~Treated Water (Internal)~~	~~Loss of Material~~	~~Water Chemistry (B.2.3.2)~~	~~VII.E4.AP-110~~	~~3.3.1-203~~	~~B~~
Bolting (Closure)	**Mechanical Closure**	**Stainless Steel Bolting**	**Air - Indoor Uncontrolled (External)**	**Loss of Material**	**Bolting Integrity (B.2.3.10)**	**VII.I.A-03**	**3.3.1-012**	**A**
Bolting (Closure)	**Mechanical Closure**	**Stainless Steel Bolting**	**Air - Indoor Uncontrolled (External)**	**Loss of Preload**	**Bolting Integrity (B.2.3.10)**	**VII.I.AP-124**	**3.3.1-015**	**A**
Heat Exchanger – (CRD PMP Thrust BRG CLR) Shell Side Components	**Leakage Boundary**	**Carbon Steel**	**Air – Indoor Uncontrolled (External)**	**Loss of Material**	**External Surfaces Monitoring of Mechanical Components (B.2.3.23)**	**VII.I.AP-41**	**3.3.1-080**	**A**

Table 3.3.2-4: Control Rod Drive – 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
Heat Exchanger – (CRD PMP Thrust BRG CLR) Shell Side Components	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	Lubricating Oil Analysis (B.2.3.25)	VII.H2.AP-131	3.3.1-098	A
Heat Exchanger – (CRD PMP Thrust BRG CLR) Shell Side Components	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.H2.AP-131	3.3.1-098	A
Heat Exchanger – (CRD PMP Thrust BRG CLR) Tube Side Components	Leakage Boundary	Carbon Steel	Air – Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.3.23)	VII.I.AP-41	3.3.1-080	A
Heat Exchanger – (CRD PMP Thrust BRG CLR) Tube Side Components	Leakage Boundary	Carbon Steel	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.3.12)	VII.C2.AP-189	3.3.1-046	A
Orifice	Leakage Boundary	Carbon Steel	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.3.23)	VII.I.A-77	3.3.1-078	A
Orifice	Leakage Boundary	Carbon Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-106	3.3.1-021	A
Orifice	Leakage Boundary	Carbon Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-106	3.3.1-021	B

Table 3.3.2-4: Control Rod Drive – 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
Orifice	Leakage Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Orifice	Leakage Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C
Orifice	Leakage Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-110	3.3.1-203	A
Orifice	Leakage Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-110	3.3.1-203	B
Piping, Piping Components	Leakage Boundary	Carbon Steel	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.3.23)	VII.I.A-77	3.3.1-078	A
Piping, Piping Components	Leakage Boundary	Carbon Steel	Condensation (Internal)	Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.3.24)	VII.D.A-26	3.3.1-055	A
Piping, Piping Components	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	Lubricating Oil Analysis (B.2.3.25)	VII.E4.AP-127	3.3.1-097	A
Piping, Piping Components	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-127	3.3.1-097	A

Table 3.3.2-4: Control Rod Drive – 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
Piping, Piping Components	Leakage Boundary	Carbon Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-106	3.3.1-021	A
Piping, Piping Components	Leakage Boundary	Carbon Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-106	3.3.1-021	B
Piping, Piping Components	Leakage Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Piping, Piping Components	Leakage Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C
Piping, Piping Components	Leakage Boundary	Stainless Steel	Condensation (Internal)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Piping, Piping Components	Leakage Boundary	Stainless Steel	Condensation (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C
Piping, Piping Components	Leakage Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-110	3.3.1-203	A
Piping, Piping Components	Leakage Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-110	3.3.1-203	B
Piping, Piping Components	Leakage Boundary	Stainless Steel	Underground (External)	Cracking	Buried and Underground Piping and Tanks (B.2.3.27)	VII.I.A-714b	3.3.1-146	B

Table 3.3.2-4: Control Rod Drive – 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
Piping, Piping Components	Leakage Boundary	Stainless Steel	Underground (External)	Loss of Material	Buried and Underground Piping and Tanks (B.2.3.27)	VII.I.A-775b	3.3.1-246	B
Piping, Piping Components	Pressure Boundary	Carbon Steel	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.3.23)	VII.I.A-77	3.3.1-078	A
Piping, Piping Components	Pressure Boundary	Carbon Steel	Condensation (Internal)	Cumulative Fatigue Damage	TLAA – Section 4.3, Metal Fatigue	VII.E3.A-34	3.3.1-002	A
Piping, Piping Components	Pressure Boundary	Carbon Steel	Condensation (Internal)	Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.3.24)	VII.D.A-26	3.3.1-055	A
Piping, Piping Components	Pressure Boundary	Copper Alloy with 15% Zinc or Less	Air - Indoor Uncontrolled (External)	None	None	VII.J.AP-144	3.3.1-114	A
Piping, Piping Components	Pressure Boundary	Copper Alloy with 15% Zinc or Less	Air - Dry (Internal)	Loss of Material	Compressed Air Monitoring (B.2.3.14)	VII.D.A-764	3.3.1-235	A

Table 3.3.2-4: Control Rod Drive – 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
Piping, Piping Components	Pressure Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Piping, Piping Components	Pressure Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C
Piping, Piping Components	Pressure Boundary	Stainless Steel	Condensation (Internal)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Piping, Piping Components	Pressure Boundary	Stainless Steel	Condensation (Internal)	Cumulative Fatigue Damage	TLAA – Section 4.3, Metal Fatigue	VII.E3.A-62	3.3.1-002	A
Piping, Piping Components	Pressure Boundary	Stainless Steel	Condensation (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C
Piping, Piping Components	Pressure Boundary	Stainless Steel	Gas (Internal)	None	None	VII.J.AP-22	3.3.1-120	A
Piping, Piping Components	Pressure Boundary	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage	TLAA – Section 4.3, Metal Fatigue	VII.E3.A-62	3.3.1-002	A
Piping, Piping Components	Pressure Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-110	3.3.1-203	A
Piping, Piping Components	Pressure Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-110	3.3.1-203	B

Table 3.3.2-4: Control Rod Drive – 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
Piping, Piping Components	Pressure Boundary	Stainless Steel	Treated Water >140 F (Internal)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.A-773	3.3.1-244	A
Piping, Piping Components	Pressure Boundary	Stainless Steel	Treated Water >140 F (Internal)	Cracking	Water Chemistry (B.2.3.2)	VII.E4.A-773	3.3.1-244	B
Piping, Piping Components	Pressure Boundary	Stainless Steel	Treated Water >140 F (Internal)	Cumulative Fatigue Damage	TLAA – Section 4.3, Metal Fatigue	VII.E3.A-62	3.3.1-002	A
Pump Casing (CRD)	Leakage Boundary	Carbon Steel	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.3.23)	VII.I.A-77	3.3.1-078	A
Pump Casing (CRD)	Leakage Boundary	Carbon Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-106	3.3.1-021	A
Pump Casing (CRD)	Leakage Boundary	Carbon Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-106	3.3.1-021	B
Pump Casing (Lubricating Oil)	Leakage Boundary	Carbon Steel	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.3.23)	VII.I.A-77	3.3.1-078	A
Pump Casing (Lubricating Oil)	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	Lubricating Oil Analysis (B.2.3.25)	VII.E4.AP-127	3.3.1-097	A

Table 3.3.2-4: Control Rod Drive – 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
Pump Casing (Lubricating Oil)	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-127	3.3.1-097	A
Speed Increaser Assembly	Leakage Boundary	Carbon Steel	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.3.23)	VII.I.A-77	3.3.1-078	A
Speed Increaser Assembly	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	Lubricating Oil Analysis (B.2.3.25)	VII.E4.AP-127	3.3.1-097	A
Speed Increaser Assembly	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-127	3.3.1-097	A
Tanks (SCRAM Discharge)	Pressure Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	A
Tanks (SCRAM Discharge)	Pressure Boundary	Stainless Steel	Condensation (Internal)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	C
Tanks (SCRAM Discharge)	Pressure Boundary	Stainless Steel	Condensation (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	A
Valve Body	Leakage Boundary	Carbon Steel	Air - Indoor Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.3.23)	VII.I.A-77	3.3.1-078	A

Table 3.3.2-4: Control Rod Drive – 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
Valve Body	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	Lubricating Oil Analysis (B.2.3.25)	VII.E4.AP-127	3.3.1-097	A
Valve Body	Leakage Boundary	Carbon Steel	Lubricating Oil (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-127	3.3.1-097	A
Valve Body	Leakage Boundary	Carbon Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-106	3.3.1-021	A
Valve Body	Leakage Boundary	Carbon Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-106	3.3.1-021	B
Valve Body	Leakage Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Valve Body	Leakage Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C
Valve Body	Leakage Boundary	Stainless Steel	Condensation (Internal)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Valve Body	Leakage Boundary	Stainless Steel	Condensation (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C
Valve Body	Leakage Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-110	3.3.1-203	A

Table 3.3.2-4: Control Rod Drive – 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
Valve Body	Leakage Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-110	3.3.1-203	B
Valve Body	Pressure Boundary	Aluminum	Air - Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B.2.3.20)	VII.A4.A-451a	3.3.1-189	A
Valve Body	Pressure Boundary	Aluminum	Air - Indoor Uncontrolled (External)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.A4.A-763a	3.3.1-234	A
Valve Body	Pressure Boundary	Aluminum	Gas (Internal)	None	None	VII.J.AP-37	3.3.1-113	A
Valve Body	Pressure Boundary	Copper Alloy with 15% Zinc or Less	Air - Dry (Internal)	Loss of Material	Compressed Air Monitoring (B.2.3.14)	VII.D.A-764	3.3.1-235	A
Valve Body	Pressure Boundary	Copper Alloy with 15% Zinc or Less	Air - Indoor Uncontrolled (External)	None	None	VII.J.AP-144	3.3.1-114	A
Valve Body	Pressure Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Valve Body	Pressure Boundary	Stainless Steel	Air - Indoor Uncontrolled (External)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C

Table 3.3.2-4: Control Rod Drive – 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
Valve Body	Pressure Boundary	Stainless Steel	Condensation (Internal)	Cracking	One-Time Inspection (B.2.3.20)	VII.E4.AP-209a	3.3.1-004	A
Valve Body	Pressure Boundary	Stainless Steel	Condensation (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.I.A-751b	3.3.1-222	C
Valve Body	Pressure Boundary	Stainless Steel	Gas (Internal)	None	None	VII.J.AP-22	3.3.1-120	A
Valve Body	Pressure Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	One-Time Inspection (B.2.3.20)	VII.E4.AP-110	3.3.1-203	A
Valve Body	Pressure Boundary	Stainless Steel	Treated Water (Internal)	Loss of Material	Water Chemistry (B.2.3.2)	VII.E4.AP-110	3.3.1-203	B

Enclosure 06

Supplement for AMR Items that Do Not Cite Applicable SRP-SLR and GALL-SLR Item Numbers within Their IPA Group

Supplement for AMR Items that Do Not Cite Applicable SRP-SLR and GALL-SLR Item Numbers within Their IPA Group

Revise Table 3.1-1, Table 3.1.2-3, and Table 3.4-1 item numbers to ones associated with their IPA group.

Affected SLRA Sections: Table 3.1-1, Table 3.1.2-3 and Table 3.4-1

SLRA Page Numbers: 3.1-36, 3.1-40, 3.1-66, 3.1-68, 3.1-75, and 3.4-38

Description of Change:

The following changes are made:

- Table 3.1-1 is revised to modify items 3.1.1-107 and 3.1.1-137 to reflect revised alignment to the GALL-SLR requirements.

- Table 3.1.2-3 is revised to remove 3.3.1-120 and replace with 3.1.1-107 to align with RCS components. Also, the NUREG-2191 item was changed from VII.J.AP-22 (for Auxiliary Systems) to IV.E.RP-07 for Reactor Vessel, Internals, and Reactor Coolant System components.

- Table 3.1.2-3 is revised to remove 3.3.1-114 and replace with 3.1.1-107 to align with RCS components. Also, the NUREG-2191 item was changed from VII.J.AP-22 (for Auxiliary Systems) to IV.E.RP-07 for Reactor Vessel, Internals, and Reactor Coolant System components.

- Table 3.4-1, Item 3.4.1-106 is revised to remove reference to “Reactor Vessel, Internals and Reactor Coolant.”

SLRA Table 3.1-1 on page 3.1-36 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 / TLAA	Further Evaluation Recommended	Discussion
3.1.1-107	Stainless steel piping, piping components exposed to gas, air with borated water leakage	None	None	No	~~Not applicable.~~ ~~There are no stainless steel components exposed to gas or air with borated water leakage in the Reactor Vessels, Internals, and Reactor Coolant System.~~**Consistent with NUREG-2191.** **There are no aging effects that require management for stainless steel piping and piping components exposed to gas.**

SLRA Table 3.1-1 on page 3.1-40 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 / TLAA	Further Evaluation Recommended	Discussion
3.1.1-137	Copper alloy piping, piping components exposed to air, condensation, gas	None	None	No	Not applicable. There are no copper alloy ~~with 15% zinc or less~~ piping or piping components in the MNGP RCS. ~~Cracking in copper alloy >15% Zinc (Zn) exposed to air indoor uncontrolled is addressed in item 3.4.1-106~~.

SLRA Table 3.1.2-3 on page 3.1-66 is revised as follows:

Table 3.1.2-3 – Reactor Coolant Pressure Boundary and Connected Piping - 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**
Piping, Piping Components	Pressure Boundary	Stainless Steel	Gas (Internal)	None	None	~~VII.J.AP-22~~ **IV.E.RP-07**	~~3.3.1-120~~ **3.1.1-107**	A

SLRA Table 3.1.2-3 on page 3.1-68 is revised as follows:

Table 3.1.2-3 – Reactor Coolant Pressure Boundary and Connected Piping - 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**
Piping, Piping Components	Structural Integrity (Attached)	Stainless Steel	Gas (Internal)	None	None	~~VII.J.AP-22~~ **IV.E.RP-07**	~~3.3.1-120~~ **3.1.1-107**	A

SLRA Table 3.1.2-3 on page 3.1-75 is revised as follows:

Table 3.1.2-3 – Reactor Coolant Pressure Boundary and Connected Piping - 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**
Valve Body	Pressure Boundary	Stainless Steel	Gas (Internal)	None	None	~~VII.J.AP-22~~ **IV.E.RP-07**	~~3.3.1-114~~ **3.1.1-107**	A

SLRA Table 3.4-1 on page 3.4-38 is revised as follows:

Table 3.4-1: Summary of Aging Management Evaluations for the Steam and Power Conversion Systems					
Item Number	**Component**	**Aging Effect / Mechanism**	**Aging Management Program / TLAA**	**Further Evaluation Recommended**	**Discussion**
3.4.1-106	Copper alloy (>15% Zn or >8% Al) piping, piping components exposed to air, condensation	Cracking due to SCC	AMP XI.M36, "External Surfaces Monitoring of Mechanical Components"	No	Consistent with NUREG-2191. This item is also applied to heat exchanger components. The External Surfaces Monitoring of Mechanical Components (B.2.3.23) AMP is used to manage cracking of copper alloy >15% Zn piping, piping components, and heat exchanger components exposed to condensation, air indoor uncontrolled, or air outdoor. This line item is also applied to components in the ~~Reactor Vessel, Internals, and Reactor Coolant,~~ ESF and Auxiliary Systems.

Enclosure 07

Supplement 2 Editorial Corrections

Supplement 2 Editorial Corrections

The SLRA is corrected to update the editorial mistakes from Supplement 2.

Affected SLRA Sections: Table 2.3.4-6, Table 3.4.2-6, A.2.2.33, and A.3.5.5

SLRA Page Numbers: 2.3-81, 3.4-103, A-30, and A-51

Description of Change:

These revisions are being made to provide clarity to the updates that were made in Supplement 2 (Reference 1). Table 2.3.4-6 and Table 3.4.2-6 were updated to show the combined changes from Enclosure 28a and 28b. The update to A.3.5.5 shows the combined changes from Enclosure 33b and 33a. A misspelling is being corrected in Enclosure 35c.

The changes made to the SLRA in the above listed Enclosures are reflected in bold, black font. Corrections to the information in the Enclosures are shown as new changes (i.e., strikeout for deletions and bold, red, underlined font for insertions).

References:

1. L-MT-23-025, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 2, ML23177A218

Table 2.3.4-6 on page 2.3-81 of the SLRA was revised in both Enclosures 28a and 28b of Supplement 2 (Reference 1). The table below captures both revisions on page 2.3-81 in a single location to clarify that both component types are incorporated and the first two components listed in the table should be as follows (table below shows only an excerpt of the added components and the next six components in Table 2.3.4-6, and is not a complete recreation of the table):

Table 2.3.4-6
Turbine Generator System Components Subject to Aging Management Review

Component Type	Component Intended Function
Accumulator (EPR)	**Leakage Boundary**
Blower Housing (Vapor Extractor)	**Leakage Boundary**
Bolting (Closure)	Mechanical Closure
Heat Exchanger (Exciter Air Cooler) Shell Side Components	Leakage Boundary
Heat Exchanger (Exciter Air Cooler) Tube Side Components	Leakage Boundary
Heat Exchanger (Generator Hydrogen Cooler) Shell Side Components	Leakage Boundary
Heat Exchanger (Generator Hydrogen Cooler) Tube Side Components	Leakage Boundary
Heat Exchanger (Isophase Bus Cooler) Shell Side Components	Leakage Boundary

Table 3.4.2-6 on page 3.4-103 of the SLRA was revised in both Enclosures 28a and 28b of Supplement 2 (Reference 1). The table below captures both revisions on page 3.4-103 in a single location to clarify that both component types are included, and that the first six rows of the table following incorporation of the changes from Enclosures 28a and 28b should look as follows:

Table 3.4.2-6: Turbine Generator – 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
Accumulator (EPR)	**Leakage Boundary**	**Carbon Steel**	**Air – Indoor Uncontrolled (External)**	**Loss of Material**	**External Surfaces Monitoring of Mechanical Components (B.2.3.23)**	**VIII.H.S-29**	**3.4.1-034**	**A**
Accumulator (EPR)	**Leakage Boundary**	**Carbon Steel**	**Lubricating Oil (Internal)**	**Loss of Material**	**Lubricating Oil Analysis (B.2.3.25)**	**VIII.A.SP-91**	**3.4.1-040**	**C**
Accumulator (EPR)	**Leakage Boundary**	**Carbon Steel**	**Lubricating Oil (Internal)**	**Loss of Material**	**One-Time Inspection (B.2.3.20)**	**VIII.A.SP-91**	**3.4.1-040**	**C**
Blower Housing (Vapor Extractor)	**Leakage Boundary**	**Carbon Steel**	**Air-Indoor Uncontrolled (External)**	**Loss of Material**	**External Surfaces Monitoring of Mechanical Components (B.2.3.23)**	**VIII.H.S-29**	**3.4.1-034**	**A**
Blower Housing (Vapor Extractor)	**Leakage Boundary**	**Carbon Steel**	**Condensation (Internal)**	**Loss of Material**	**Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.3.24)**	**VIII.E.SP-60**	**3.4.1-037**	**A**
Bolting (Closure)	Mechanical Closure	Carbon and Low Alloy Steel Bolting	Air - Indoor Uncontrolled (External)	Loss of Material	Bolting Integrity (B.2.3.10)	VIII.H.S-02	3.4.1-009	A

Section A.2.2.33 on page A-30 of the SLRA with changes made in Supplement 2, Enclosure 35c (Reference 1) contained a misspelling, which is revised as follows:

A.2.2.33 Structures Monitoring

The MNGP Structures Monitoring AMP is an existing AMP that consists of periodic visual inspection and monitoring of the condition of concrete and steel structures, structural components, component supports, and structural commodities to ensure that aging degradation (such as those described in ACI 349.3R, ACI 201.1R, SEI/ASCE 11, and other documents) will be detected, the extent of degradation determined and evaluated, and corrective actions taken prior to loss of intended functions. Structures are monitored on an interval not to exceed 5 years. Inspections also include seismic joint fillers, elastomeric materials; steel edge supports, and bracings associated with masonry walls, and periodic evaluation of ground water chemistry and opportunistic inspections for the condition of below grade concrete. **The program includes annual survey measurement of settlement for the Diesel Fuel Oil Transfer House, the Diesel Fuel Oil Storage Tank and Offgas ~~Stroage~~Storage Building HTV exhaust pipe to provide early indication of potential stress increases that could result in cracking or deflection of the structural components associated with these structures.** Quantitative results (measurements) and qualitative information from periodic inspections are trended with sufficient detail, such as photographs and surveys for the type, severity, extent, and progression of degradation, to ensure that corrective actions can be taken prior to a loss of intended function. The acceptance criteria are derived from applicable consensus codes and standards. For concrete structures, the program includes personnel qualifications and quantitative evaluation criteria of ACI 349.3R.

Section A.3.5.5 on page A-51 of the SLRA was revised in both Enclosures 33a and 33b of Supplement 2 (Reference 1). Both revisions are shown here to clarify that both changes are included in the SLRA, and that the changes in Enclosure 33b did not eliminate the changes indicated in Enclosure 33a. Section A.3.5.5 with all changes incorporated appears as follows:

A.3.5.5 Primary Containment Process Penetration Bellows Fatigue Analysis

Containment pipe penetrations that are required to accommodate thermal movement have expansion bellows. The bellows are designed for a minimum number of **equivalent full temperature thermal** cycles over the design life of the plant. Consequently, the primary containment process penetrations bellows cycle basis is a TLAA.

This evaluation was performed as part of the ASME Section III, Class 2 and 3 and ANSI B31.1 fatigue evaluation and is described in Section **A.3.3.6**. **The limiting pipe penetration expansion bellows was evaluated and was determined to have a thermal cycle count below the 7000 cycle limit with considerable margin.**

The containment penetration bellows fatigue design criteria remains valid for the SPEO in accordance with 10 CFR 54.21(c)(1)(i).

Enclosure 08

Supplement 4 Editorial Correction

Supplement 4 Editorial Correction

The SLRA is corrected to update the editorial mistake from Supplement 4.

Affected SLRA Sections: Table 3.5.2-13

SLRA Page Numbers: 3.5-123 and 3.5-124

Description of Change:

Supplement 4, Enclosure 02 showed incorrect information in one of the cells in a row that was being deleted from SLRA Table 3.5.2-13 as part of the Enclosure (Reference 1). The incorrect NUREG-2191 item is corrected in this Enclosure with bold, red, and underlined font to show the correct information that should have been displayed in Reference 1. The corrected NUREG-2191 item is then also shown deleted in this Enclosure with a strikeout through the entire row, including the corrected NUREG-2191 item. The other insertions made to SLRA Table 3.5.2-13 in Supplement 4, Enclosure 02 are reflected in this Enclosure in bold, black font.

References:

1. L-MT-23-031, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 4 and Responses to Request for Confirmation of Information - Set 1, ML23199A154

Table 3.5.2-13 on pages 3.5-123 and 3.5-124 of the SLRA with changes made in Supplement 4, Enclosure 02 (Reference 1) contained an incorrect NUREG-2191 Item, which is revised (change shown in bold, red, underlined font to show the correct information that should have been displayed in Supplement 4, Enclosure 02) and then deleted (strikethrough the bold, red, underlined font to indicate that this information is still to be deleted from the SLRA) as follows:

Table 3.5.2-13: Plant Control and Cable Spreading Structure – 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
~~Concrete: Basemat, Foundation (Accessible)~~	~~Structural support~~	~~Concrete (reinforced)~~	~~Groundwater/soil~~	~~Cracking~~	~~Structures Monitoring (B.2.3.33)~~	~~III.A3.TP-204~~ ~~**<u>III.A3.TP-25</u>**~~	~~3.5.1-054~~	~~A~~
Concrete: Basemat, Foundation (Accessible)	**Structural Support**	**Concrete (Reinforced)**	**Air – Outdoor**	**Cracking Increase in Porosity and Permeability Loss of Material**	**Structures Monitoring (B.2.3.33)**	**III.A3.TP-28**	**3.5.1-067**	**A**
Concrete: Exterior Walls and Roof (Accessible)	**Flood Barrier Missile Barrier Pressure Boundary Shelter, Protection Structural Support**	**Concrete (Reinforced)**	**Air-Indoor Uncontrolled Air – Outdoor**	**Cracking Increase in Porosity and Permeability Loss of Material**	**Structures Monitoring (B.2.3.33)**	**III.A3.TP-28**	**3.5.1-067**	**A**

Enclosure 09

Supplement 4 Enclosure 06b Administrative Correction

Supplement 4 Enclosure 06b Administrative Correction

The SLRA is corrected to update the administrative mistake from Supplement 4 Enclosure 06b.

Affected SLRA Sections: Table 3.3-1

SLRA Page Numbers: 3.3-65 and 3.3-84

Description of Change:

Supplement 4, Enclosure 06b left out part of the discussion for Item Numbers 3.3.1-146 and 3.3.1-246 in Table 3.3-1 (Reference 1). The discussion is corrected to state “Consistent with NUREG-2191 with exception for the Buried and Underground Piping and Tanks (B.2.3.27) AMP.” The exception to the Buried and Underground Piping and Tanks AMP was added by Enclosure 06b of Supplement 2 (Reference 2). The correction is reflected in bold, red and underlined font. The changes made to the SLRA in Supplement 4, Enclosure 06b are reflected in bold, black font.

References:

1. L-MT-23-031, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 4 and Responses to Request for Confirmation of Information - Set 1, ML23199A154

2. L-MT-23-025, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 2, ML23177A218

Table 3.3-1 on page 3.3-65 of the SLRA, with changes provided in Supplement 4, Enclosure 06b (Reference 1) incorporated, is revised as follows:

Table 3.3-1: Summary of Aging Management Evaluation for the Auxiliary Systems

Item Number	Component	Aging Effect / Mechanism	Aging Management Program / TLAA	Further Evaluation Recommended	Discussion
3.3.1-146	Stainless steel underground piping, piping components, tanks	Cracking due to SCC	AMP XI.M32, “One-Time Inspection,” AMP XI.M41, "Buried and Underground Piping and Tanks," or AMP XI.M42, “Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks”	Yes (SRP-SLR Section 3.3.2.2.3)	**Consistent with NUREG-2191 with exception for the Buried and Underground Piping and Tanks (B.2.3.27) AMP.** **Buried and Underground Piping and Tanks (B.2.3.27) AMP is used to manage cracking of stainless steel piping and piping components exposed to underground in the Auxiliary Systems.** Further evaluation is documented in Section 3.3.2.2.3.

Table 3.3-1 on page 3.3-84 of the SLRA, with changes provided in Supplement 4, Enclosure 06b (Reference 1) incorporated, is revised as follows:

Table 3.3-1: Summary of Aging Management Evaluation for the Auxiliary Systems

Item Number	Component	Aging Effect / Mechanism	Aging Management Program / TLAA	Further Evaluation Recommended	Discussion
3.3.1-246	Stainless steel, nickel alloy underground piping, piping components, tanks	Loss of material due to pitting, crevice corrosion	AMP XI.M32, "One-Time Inspection," AMP XI.M41, "Buried and Underground Piping and Tanks," or AMP XI.M42, "Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks"	Yes (SRP-SLR Section 3.3.2.2.4)	**Consistent with NUREG-2191 with exception for the Buried and Underground Piping and Tanks (B.2.3.27) AMP.** **Buried and Underground Piping and Tanks (B.2.3.27) AMP is used to manage loss of material of stainless steel piping and piping components exposed to underground in the Auxiliary Systems.** Further evaluation is documented in Section 3.3.2.2.4.

Enclosure 10

Table 3.5.2-1 Line Item and Note 7 Correction

Table 3.5.2-1 Line Item and Note 7 Correction

Revise Table 3.5.2-1 Line Item for Biological Shield Wall and clarify Plant Specific Note 7 to be consistent with the evaluation in SLRA Section 3.5.2.2.2.6.

Affected SLRA Sections: Table 3.5.2-1

SLRA Page Numbers: 3.5-76 and 3.5-84

Description of Change:

Table 3.5.2-1 is revised to show that the line item associated with the Biological Shield Wall has both a radiation shielding function and a structural support function. Table 3.5.2-1, plant specific Note 7 is revised to reflect that gamma radiation impacts have been evaluated separately and the conclusions summarized in Section 3.5.2.2.2.6.

SLRA Table 3.5.2-1 on Page 3.5-76 is revised as follows:

Table 3.5.2-1 Primary Containment – 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
Biological Shield Wall	Radiation Shielding **Structural Support**	Concrete (Reinforced)	Air – Indoor Uncontrolled	Reduction of Strength, Loss of Mechanical Properties	Structures Monitoring (B.2.3.33)	III.A4.T-35	3.5.1-097	A, 6, 7

SLRA Table 3.5.2-1 on Page 3.5-84 is revised as follows:

7. Consistent with SLR-ISG-2021-03-STRUCTURES, which allows a plant-specific AMP, or a selected AMP enhanced as necessary; the Structures Monitoring (B.2.3.33) AMP will be used to manage the potential for reduction in strength, loss of mechanical properties, or cracking of the biological shield due to irradiation near the reactor vessel, as the projected value~~s~~ for neutron ~~and gamma~~ radiation incident on the shield wall ~~are~~**is** less than the threshold value~~s~~ of 1×10^{19} n/cm^2 and **the gamma radiation impacts have been evaluated separately as summarized in Section 3.5.2.2.2.6**~~1×10^{10} rads, respectively~~. **Structural support is provided at the bottom portion of the biological shield wall as shown in Section 3.5.2.2.2.6, Figure 3.5.2.2.2.6-1.**