2807 West County Road 75
Monticello, MN 55362

January 11, 2024

L-MT-23-054
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 8

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

6) 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 5" dated August 28, 2023, ML23240A695

7) 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 Response to Request for Additional Information Set 2 and Supplement 6" dated September 05, 2023, ML23248A474

8) 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 7" dated November 30, 2023, ML23334A147

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 December of 2023 were provided in Supplements 2 through 7, listed in References 3 through 8, respectively. Note that Supplement 3 (Reference 4) did not make any changes to the SLRA. Additional clarifications discussed in the breakout audits occurring April through December are being provided in Supplement 8. The supplement is provided in the Enclosures.

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**. Changes incorporated from previous RAIs and supplements are provided by bold, black font and noted in enclosure.

Summary of Commitments

This letter makes new commitments and revisions to existing commitments as explained in the enclosures. Commitments 26, 30, and 36 include additions and revisions.

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

Executed on January 11, 2024.

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 01a

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:

- Reference the publicly available sources used to provide the generic, normalized curves for determining the variation of gamma flux along core height. These curves were developed using information from PWRs, but are deemed acceptable for evaluating the expected gamma flux for MNGP, a BWR.

- Clarify the publicly available sources used to quantify the reduction in concrete strength and material properties subjected to gamma dose exposure.

- State the specified minimum design compressive strength of concrete (f'c) for the original design of the bioshield wall, the degraded (reduced) strength that was used for the evaluation of gamma radiation, and the conservatisms applied.

- Indicate the controlling loading combination that was used for the evaluation of reduction in strength of the bioshield wall reinforced concrete, the code-of-record that governs the evaluation, and the maximum demand to capacity ratio(s) (D/C) for the degraded case for the controlling loading combination.

- Indicate how the plant specific analysis evaluated the potential for radiation induced volumetric expansion (RIVE) of the biological shield wall concrete thickness is also added.

Bold black font information in Section 3.5.2.2.2.6 on page 3.5-38 represents changes made in Enclosure 03c of Reference 1.

References:

1. L-MT-23-035, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 5, ML23240A695.

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 4.85×10^{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, 1×10^{10} rads, for radiation damage to concrete.

Recent research on the gamma dose limit of 1×10^{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 1×10^{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.

Report ORNL/TM-2018/769, Revision 0, *Expected Condition of Concrete Exposed to Radiation at Age 80 Years of Reactor Operation*, US Department of Energy, January 2018 and EPRI Report 3002011710, *Irradiation Damage of the Concrete Biological Shield: Basis for Evaluation of Concrete Biological Shield Wall for Aging Management*, May 2018 provide a normalized curve that relates the gamma flux to elevation for nuclear reactors. 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 35% of the peak gamma flux, or 1.7 x 10^{10} rads, was calculated using the the extrapolated flux that would occur at the bottom of the active core (approximately 6 feet below the core mid-plane; the top of the structural portion of the biological shield is approximately 13.5 feet below the core mid-plane). 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 mid-plane. However, 35% conservatively 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 (H. K. Hilsdorf, et. al., *The Effects of Nuclear Radiation on the Mechanical Properties of Concrete*, American Concrete Institute, Special Publications SP 55-10). In addition to the conservatism of using the lower bound curve, the gamma radiation dose used in the evaluation to determine the remaining concrete strength at the end of the SPEO used the 35% of the gamma flux at the core mid-plane as the dose, which is projected to be considerably higher than the dose at the elevation of the structural portion of the biological shield wall. The evaluation also conservatively used the specified minimum design compressive strength of concrete for the original design of the biological shield wall (4000 psi). The test strength of similar concrete pours for other containment elements are above the 4600 psi ACI 349-13 limit. Applying the most conservative curve for concrete compressive strength degradation and the conservative 35% of maximum gamma dose (1.7 x 10^{10} rads) results in a calculated degraded (reduced) strength of 3643 psi for the structural portion of the concrete in the biological shield wall. The compressive strength of the concrete (f'c) in the evaluation uses a further conservative value of 3500 psi for determining the demand to capacity ratios for the biological shield wall.

The biological shield wall concrete is encased in a 1/4 inch thick steel liner, providing additional structural support for the wall. The design of the biological shield wall used the Working Stress Method of ACI 318-63, American Concrete Institute, *Building Code Requirements for Reinforced Concrete*, 1963, which is the code of record. Controlling loads are evaluated per EPRI Report 3002011710. Allowable stresses were increased by a factor of 1.5 for controlling load cases involving jet forces. Main load combinations are seismic and jet forces, including:

- **Jet force of 393k at el. 953'-2**
- **Jet force of 666k at el. 957'-5**
- **Jet force of 127k at el. 962'-8**
- **Preload of 80k on stabilizers at el. 993'-0**
- **Seismic force of 600k on top cover ring at el. 993'-0**

The maximum demand/capacity ratio for the degraded case is 0.98 for the Transverse Shear load of the Concrete Ring.

Gamma dose, if calculated to be above the threshold (e.g., 2.3 x 10^{10} Rad) can be assessed in combination with the neutron fluence induced RIVE damage. The neutron fluence is significantly below 1.0E+19 n/cm^2. The

gamma dose was conservatively assumed to be above threshold but in reality gamma and neutron are below the threshold, and so the gamma information indicates current material properties are minimally affected.

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.

Enclosure 01b

SLRA Figure 3.5.2.2.2.6-1

SLRA Figure 3.5.2.2.2.6-1

Provided a legible SLRA Figure 3.5.2.2.2.6-1.

Affected SLRA Sections: 3.5.2.2.2.6

SLRA Page Numbers: 3.5-37

Description of Change:

SLRA Figure 3.5.2.2.2.6-1 was originally provided in Reference 1, Enclosure 3a. The figure did not reproduce legibly. This change will replace the original SLRA Figure 3.5.2.2.2.6-1 with a more legible figure.

References:

1. L-MT-23-035, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 5, ML23240A695.

SLRA Figure 3.5.2.2.2.6-1 on page 3.5-37 is revised as follows:

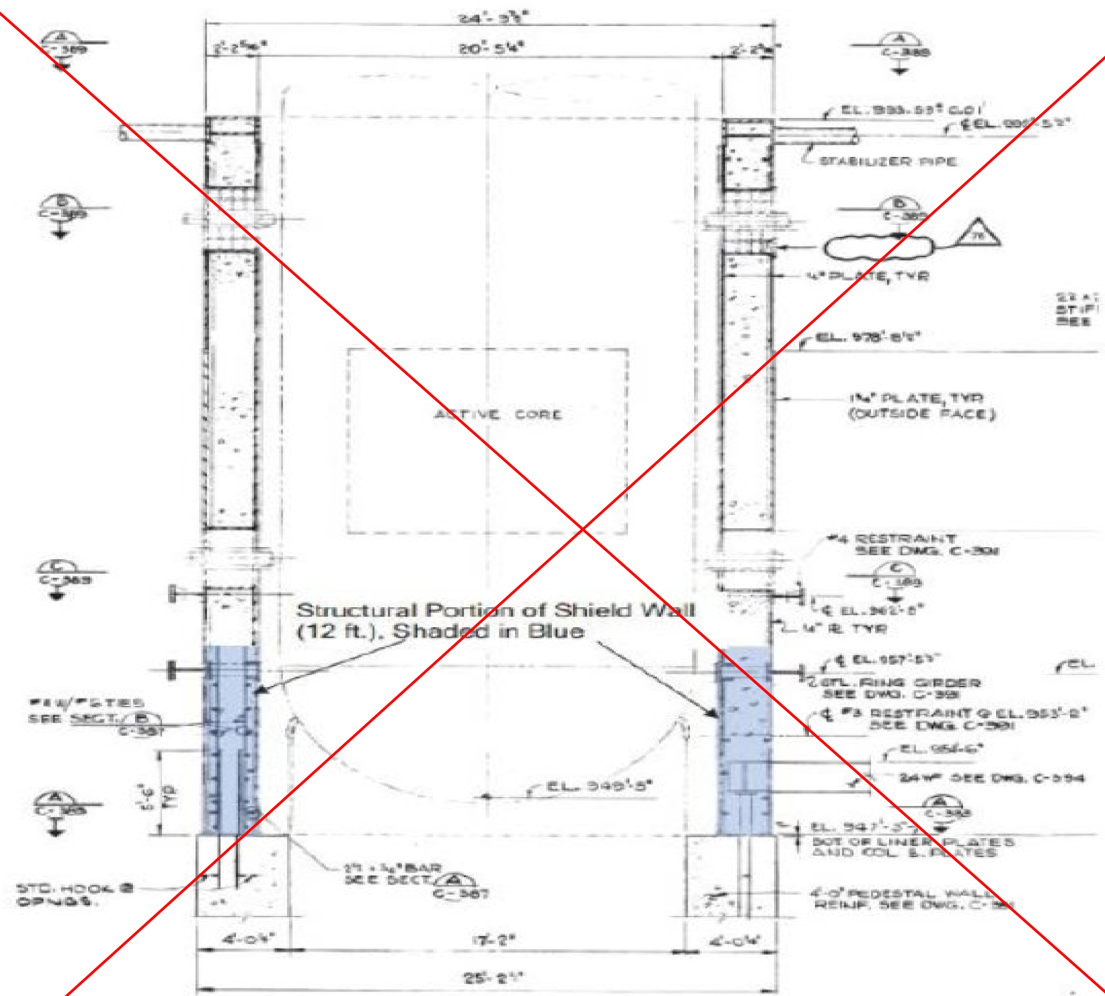

Figure 3.5.2.2.2.6-1: Section Through Monticello Biological Shield Wall

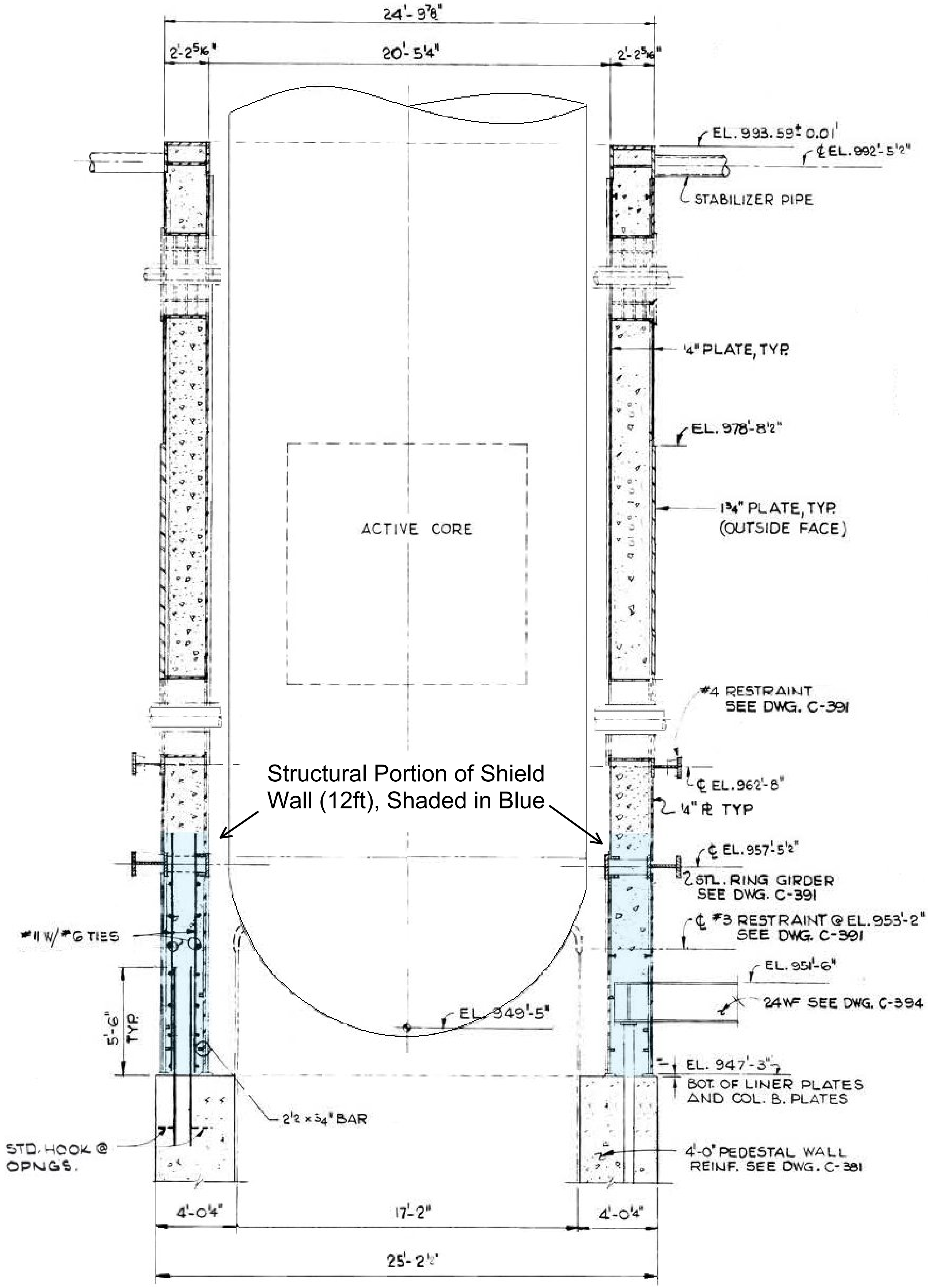

Figure 3.5.2.2.2.6-1: Section Through Monticello Biological Shield Wall

Enclosure 02a

Conservatisms for Gamma Heating of Concrete in the Biological Shield Wall

Conservatisms for Gamma Heating of Concrete in the Biological Shield Wall

Additional technical details (descriptions of assumptions, location of maximum cumulative temperature, and the basis for how the temperature increase in concrete and encapsulating steel due to heating effect of gamma radiation was determined) along with their effect is added to the SLRA.

Affected SLRA Sections: 3.5.2.2.2.6

SLRA Page Numbers: 3.5-38

Description of Change:

Update the SLRA Section 3.5.2.2.2.6 to include the following additional information:

- Description of the assumptions, and basis for determining the heating effect of gamma radiation that causes the temperature increase in the concrete and encapsulating steel as well as the resulting tensile stresses on both.
- Description of how the increase in stress impacts the strength reduction evaluation.

Bold black font information in Section 3.5.2.2.2.6 on page 3.5-38 represents changes made in Enclosure 03c of Reference 1.

References:

1. L-MT-23-035, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 5, ML23240A695.

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

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 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 mid-plane (top of structural concrete EL. 959'- 3", approximate core midplane EL. 972'- 8-1/2") to be approximately 13.5 feet difference in elevations. The gamma dose, and therefore the effects of gamma heating on the portion of the biological shield wall with a structural support intended function are considerably less than the values from this evaluation. The heating effect from gamma ray irradiation at the peak, core mid-plane dose has been determined to be limited to approximately 1.12°F of temperature change. The margin to general limit is 7 times and to local limits is 52 times 1.12°F of temperature increase due to gamma. This temperature change was calculated from extrapolation of the core mid-plane dose due to gamma irradiation (4.85×10^{10} rad) to the graphed information in RIL 2021-07 (specifically Figure 5-6 on page 5-14 and Figure 5-7 on page 5-15). The compressive test strength of similar concrete pours for other containment elements are above the 4600 psi ACI 349-13 limit (the higher ACI limit would allow for a 180ºF general concrete temperature limit in the calculation). Using the lower limit of 150ºF in the calculation is an additional conservatism that will bound any synergistic effect of gamma and neutron heating on concrete temperature limits.

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.

Because the concrete has a smaller thermal expansion coefficient than the steel liner for the biological shield wall, the inner liner thermal expansion is constrained by the concrete as it will try to expand more than the surrounding concrete material. Contact pressure will develop at the interface of the steel liner and the concrete. The expansion of the concrete from this calculated 1.12°F temperature increase results in a maximum of 0.94 ksi additional ~~tensile~~ compressive stress during operation. This additional stress caused by differential thermal expansion between the steel and concrete material and the heating from gamma radiation is shown to be inconsequential to the integrity of the MNGP Bioshield and that higher assumed gamma heating values would not challenge the integrity of the biological shield wall concrete. 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.

Enclosure 02b

Clarify the Description of the Concrete Temperature Increase Due to Gamma Heating

Clarify the Description of the Concrete Temperature Increase Due to Gamma Heating

Additional technical details including descriptions of assumptions and the basis for how the temperature increase in concrete due to the heating effect of gamma radiation was determined. This will reconcile any inconsistencies in SLRA Section 3.5.2.2.2.6 regarding the total increase temperature effect on the concrete.

Affected SLRA Sections: 3.5.2.2.2.6

SLRA Page Numbers: 3.5-38

Description of Change:

Update the SLRA Section 3.5.2.2.2.6 to include additional information to reconcile any apparent inconsistencies for determining the heating effect of gamma radiation that causes the temperature increase in the concrete.

Bold black font information in Section 3.5.2.2.2.6 on page 3.5-38 represents changes made in Enclosure 03c of Reference 1.

References:

1. L-MT-23-035, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 5, ML23240A695.

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

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 calculated threshold 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. The evaluation takes the Technical Specification limit for the average drywell air temperature of 135°F (this temperature is maintained by the primary containment cooling and ventilation system) and assumes a conservative range of +/- 10°F to account for the maximum expected high and low temperature variation of the air supplied to the air gap between the reactor and the biological shield wall to perform the upper and lower bound analysis for biological shield wall temperature. The heat transfer analysis determined that the upper and lower bound temperatures are 140.69°F and 120.97°F for the steel inner liner of the biological shield wall at the top of the structural concrete (EL. 959' - 3"). The steel inner liner mid-core level upper and lower bound temperatures are 140.66ºF and 120.94ºF (EL.972'- 8-1/2"). Based on the heat transfer analysis results, the maximum expected temperature on concrete surface of the MNGP Bioshield is 140.68~~9~~°F. Temperature increase due to gamma-heating (as described above) can be approximated to 1.12°F, which brings the maximum expected concrete surface temperature to approximately 142°F; below the ACI 349-13 limit of 150°F (using the lower ACI 349-13 general concrete temperature limit based on a compressive strength of 4000 psi adds conservatism for any synergistic effect of neutron and gamma heating of the concrete). Hence, thermal induced damage in the MNGP Bioshield concrete material is not a concern for the SPEO.

Enclosure 03

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:

- Clarify that the location evaluated for fracture mechanics of the biological shield structural steel conservatively used the maximum calculated dpa of 2.07×10^{-3} and a bounding stress.
- Delete the discussion on the use of the NDT methodology. The calculated dpa at MNGP exceeded the NDT criteria of NUREG-1509, and therefore, was not credited and will be deleted to avoid confusion.

Bold black font information in Section 3.5.2.2.2.6 on pages 3.5-40 and 3.5-41 represents changes made in Enclosure 03e of Reference 1.

References:

1. L-MT-23-035, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 5, ML23240A695.

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.

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 conservatively performed for all biological shield wall structural steel evaluations in accordance with NUREG-1509.

Criteria #2

NDT evaluations were not credited. A fracture mechanics evaluation in accordance with NUREG-1509 was credited instead.~~The 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 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~~ $\cancel{10^{-5}}$.

~~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~~ $\cancel{6.59 \times 10^{18}\ n/cm^2}$. ~~The peak fluence at the biological shield inner diameter for 72 EFPY equates to a displacement per atom =~~ $\cancel{2.07 \times 10^{-3}}$ ~~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 2.07×10^{-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

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 maximum residual weld stresses (36 ksi) 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.~~

The lower bound K_{IC} fracture toughness from industry literature (NUREG-3009 and *Fracture Toughness and CVN Data for A36 Steel with Wet Welding*, 2017, by Méndez, Gerardo Terán, et. al.) for the ASTM A36 steel (32 ksi-in$^{1/2}$) used for construction of the biological shield wall was used to provide a conservative and bounding evaluation of the ability of the biological shield wall to continue to perform its intended function through SPEO. ~~Plant-specific CMTR data has been assessed for evaluation of fracture toughness properties. With a KIC of 58.7 ksi-in$^{1/2}$, t~~The limiting stress intensity factor of 21.2 ~~19.4~~ ksi-in$^{1/2}$ (calculated in accordance with the guidance from NUREG-1509 and using conservative values for flaw size, flaw shape parameter [Q], and total stress) remains below the bounding lower 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. **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. 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.

Enclosure 04

Biological Shield Wall Structural Steel Clarifications

Biological Shield Wall Structural Steel Clarifications

The Structures Monitoring commitments/enhancements are revised to include the acceptance criteria for the Biological Shield Wall Structural Steel.

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

SLRA Page Numbers: A-92, B-240

Description of Change:

Revise the Structures Monitoring commitment 36 and the enhancement in Section B.2.3.33 to include the Biological Shield Wall Structural Steel in the list of components/commodities requiring acceptance criteria be added to the implementing procedure.

The information shown in bold black font in the mark-ups represent changes provided in Enclosures 31c of Reference 1 and align the information presented in Enclosure 01 of Reference 2.

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

2. L-MT-23-035, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Supplement 5, ML23240A695

SLRA Table A-3, Commitment 36 on page A-92 is revised as follows:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
36	Structures Monitoring (A.2.2.33)	XI.S6	**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** • **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**	

SLRA Section B.2.3.33 on page B-240 is revised as follows:

Element Affected	Enhancement
6. Acceptance Criteria	**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** • **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**

Enclosure 05

Underground Piping and Tanks Supplement

Underground Piping and Tanks Supplement

Identify the planned SPEO and pre-SPEO inspections and provide additional justification for the planned inspections.

Affected SLRA Sections: A.2.2.27, Table A-3 Commitment 30, and B.2.3.27

SLRA Page Numbers: A-26, A-27, A-84, B-197, B-198, B-199, B-200

Description of Change:

In-scope systems with pipes that are routed between the reactor and turbine buildings pass through a seismic gap space between the wall of the two buildings. Access to the portions of these pipes in the seismic gap space is limited because of the mechanical penetration seals required to maintain the integrity of the walls through which the pipes are routed. The actual environment of the piping in the seismic gap and the potential environment defined by GALL-SLR depends on the elevation of the penetrations. Of the 35 penetrations identified where pipes run between the reactor and turbine buildings, eight (8) penetrations are located below the grade elevation (931') of MNGP. The piping above the grade elevation of MNGP in the seismic gap space is exposed to an air-outdoor environment as defined by GALL-SLR and will have their aging effects managed using the External Surfaces Monitoring of Mechanical Components program during the SPEO. The two below-grade penetrations that are also located below or near the maximum elevation of the water table in the vicinity of MNGP (912') are considered to be exposed to an underground-groundwater environment for MNGP SLRA and will have their aging managed using the MNGP Buried and Underground Piping and Tanks program. The physical configuration of the remaining six penetrations that are below the grade elevation of MNGP but above the elevation of the water table could be considered to be in an underground-air environment or an air-outdoor environment based on interpretations of the definitions of those environments in GALL-SLR. The MNGP SLRA interprets the environment of the pipes to be best represented by the air-outdoor environment and proposes to manage the aging of these pipes using the External Surfaces Monitoring of Mechanical Components program.

This enclosure addresses the pipes located below or near the maximum elevation of the water table in the vicinity of MNGP in the seismic gap space between the reactor and turbine buildings. This enclosure adds a commitment to inspect a sample of these pipes that are most susceptible to corrosion for signs of degradation and provides a basis for not coating or wrapping these pipes. The inspections will occur at least once during the 6-year period prior to the SPEO and once in every 6-year or 3 refueling outage period afterward throughout the SPEO. The other aspects of these inspections and aging management of these pipes will be performed in accordance with the guidance in NUREG-2191, Chapter XI.M41. This will provide reasonable assurance that the intended function of the pipes will be maintained throughout the SPEO.

This enclosure also provides additional justification for not coating or wrapping the carbon steel Off-Gas piping in the underground environment located in the Recombiner Vault area beneath the Recombiner Building.

The information shown in bold black font in the mark-ups represent changes provided in Enclosure 06a of Reference 1, Enclosures 06a, b, and c of Reference 2, Enclosures 02 and 03 of Reference 3, and Enclosure 01 of Reference 4.

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
2. 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
3. L-MT-23-036, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Request for Additional Information Set 2 and Supplement 6, ML23248A474
4. L-MT-23-037, Monticello Nuclear Generating Plant, Docket No. 50-263, Renewed Facility Operating License No. DPR-22, Subsequent License Renewal Application Request for Additional Information Set 3, ML23265A158

SLRA Section A.2.2.27 on pages A-26 and A-27 are revised as follows:

The MNGP Buried and Underground Piping and Tanks AMP, previously known as the Buried Piping and Tanks Inspection Program, is an existing AMP that manages the aging effects associated with the external surfaces of buried and underground piping and tanks such as loss of material and cracking. This AMP addresses piping and tanks composed of metallic (steel and stainless steel) materials that are within the scope of SLR in the **CRD,** CST, EDGs, ESW, fire water, off-gas, **RAD,** secondary containment, service and seal water, and wells and domestic water systems. Loss of material is monitored by visual inspection of the exterior surface and wall thickness measurements of the piping. Wall thickness is determined by a non-destructive examination technique such as UT. **For steel components, where the acceptance criteria for the effectiveness of the cathodic protection is other than -850 mV instant off, loss of material rates are measured.**

This AMP also manages aging through preventive actions (e.g., coatings or wrapping, cathodic protection, and quality of backfill). Annual cathodic protection surveys are conducted. **MNGP will refurbish its Cathodic Protection System 5 years prior to the SPEO to meet the acceptance criteria of -850 mV relative to a copper/copper sulfate reference electrode (CSE) (instant off), or acceptance criteria alternatives as outlined in NUREG-2191, Section XI.M41, Subsection 6.m, for buried steel components. The intent is to satisfy conditions of Preventive Action Category C for inspections of buried steel and stainless steel piping during the SPEO, unless a reevaluation of future OE and soil conditions determines that another Preventive Action Category is more applicable.** The number of inspections for each 10-year inspection period, commencing 10 years prior to the SPEO, are based on the effectiveness of the preventive actions above.

Visual inspections of external surfaces of buried **and underground** components are performed to check for evidence of coating/wrapping damage, loss of material, and cracking. The selection of locations of the~~se~~ inspections **of buried components** will be based on plant OE and opportunities for inspection such as scheduled maintenance work; these inspections will occur once prior to the SPEO and at least every 10 years during the SPEO. **For underground stainless steel pipes located in the seismic gap space between the reactor and turbine buildings that are in the scope of this program, inspections will be performed on a sample of the pipes most susceptible penetration to corrosion during every 6-year or 3 refueling cycle period starting in the 6-year period before the SPEO and continuing throughout the SPEO. Underground Off-Gas steel piping located in the Recombiner Vault is inspected every 10-year or 5 refueling cycle period starting in the 10-year period before the SPEO and continuing throughout the SPEO.** Inspections are conducted by qualified individuals. Where the coatings, backfill or the condition of exposed piping does not meet acceptance criteria such That the depth or extent of degradation of the base metal could have resulted in a loss of pressure boundary function when the loss of material rate is extrapolated to the end of the SPEO, an increase in the sample size is conducted.

SLRA Table A-3 on page A-84 is revised as follows:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
30	Buried and Underground Piping and Tanks (A.2.2.27)	XI.M41	u) Underground stainless steel pipes located in the seismic gap space between the reactor and turbine buildings that are in the scope of this program, will have inspections performed on a sample of the pipes in the most susceptible penetration to corrosion at least once during every 6-year or 3 refueling outage period.	No later than 6 months prior to the SPEO, or no later than the last refueling outage prior to the SPEO. Implement the AMP and start 10-year interval inspections no earlier than 10 years prior to the SPEO. **Commitment 30q will be implemented 5 years prior to the SPEO in order to credit the system for pre-SPEO inspections.** Commitment 30u will be implemented no earlier than 6 years prior to the SPEO and no later than 6 months prior to the SPEO.

SLRA Section B.2.3.27 on page B-197 is revised as follows:

B.2.3.27 Buried and Underground Piping and Tanks

The MNGP Buried and Underground Piping and Tanks AMP, previously known as the Buried Piping and Tanks Inspection Program, is an existing condition monitoring program that manages the aging effects associated with the external surfaces of buried and underground piping and tanks such as loss of material and cracking. This program addresses piping and tanks composed of metallic (steel and stainless steel) materials that are within the scope of SLR in the **CRD,** Condensate Storage, DGN, ESW, Fire Water, Off-Gas, **RAD,** SCT, Service and Seal Water, and Wells and Domestic Water Systems. There are no polymeric, cementitious, or metallic materials other than those metals previously stated for the in-scope systems, therefore, the aging management of these materials is not applicable.

This AMP manages aging through preventive, mitigative, inspection and performance monitoring activities. The MNGP Buried and Underground Piping and Tanks AMP includes (a) preventive actions to mitigate degradation (e.g., external coatings or wrappings, cathodic protection and quality of backfill), (b) condition monitoring (inspections) (e.g., verification of cathodic protection effectiveness, nondestructive evaluation of pipe or tank wall thicknesses, and visual inspections of the external surfaces and coatings/wraps of pipe or tanks, and internal tank inspections capable of detecting loss of material on the external surface), and (c) performance monitoring activities (e.g., pressure testing of piping, performance monitoring of fire mains) to provide early warning of system leakage. The locations of these inspections will be based on plant OE and opportunities for inspection such as scheduled maintenance work. These inspections will occur once prior to the SPEO and at least every 10 years during the SPEO. **For underground stainless steel pipes located in the seismic gap space between the reactor and turbine buildings that are in the scope of this program, inspections will be performed on a sample of the pipes in the most susceptible penetration to corrosion during every 6-year or 3 refueling cycle period starting in the 6-year period before the SPEO and continuing throughout the SPEO. Underground Off-Gas steel piping located in the Recombiner Vault will be inspected every 10-year or 5 refueling cycle period starting in the 10-year period before the SPEO and continuing throughout the SPEO.** If an opportunity for inspection on non-leaking piping occurs prior to the scheduled inspection, the opportunistic inspection can be credited for satisfying the scheduled inspection. **The MNGP Fire Protection System Flow Test is performed annually which provides data on the Fire Water System more frequently to detect piping degradation of this buried piping. The annual testing is a credited alternative method used at MNGP and is used in lieu of performing two additional inspections of buried piping during each 10-year interval.**

SLRA Section B.2.3.27 on page B-198 is revised as follows:

The number of inspections for each 10 year inspection period, commencing 10 years prior to the start of SPEO, are based on the inspection quantities noted in NUREG-2191, Table XI.M41-2 for Category **C**, **except for underground stainless steel piping, which will have additional inspections due to the pipes not being coated/wrapped**. However, changes in plant specific conditions can result in transitioning to a **higher** number of inspections than originally planned at the beginning of a 10 year period. For example, **degradation** of the cathodic protection system**, coatings, backfill, or the condition of exposed piping that does not** meet acceptance criteria could result in transitioning from **Preventive Action** Category **C** to Preventive Action Category F**. If alternatives to visual inspections are performed, they will be performed in accordance with NUREG-2191, Section XI.M41, Subsection 4.e.**

Material	No. of Inspections	Notes
Steel piping (buried)[1]	**1** inspection~~s~~	The smaller of **0.5**% of the piping length or **1** inspection.
Steel piping (underground)[1]	2 inspections	The smaller of 2% of the piping length or 2 inspections.
Stainless steel piping (buried)	1 inspection	None
Stainless steel piping (underground)	~~1~~**2** inspection**s**	**The stainless steel pipes most susceptible underground penetration in the scope of this program will be inspected at least once during every 6-year or 3 refueling cycle period starting in the 6-year period before the SPEO and continuing throughout the SPEO.**~~None~~
Steel tank (buried)	1 inspection	Only one tank is buried at MNGP. If the diesel fuel oil storage tank is properly cathodically protected with a refurbishment to the system in the future, no inspections would be required per NUREG-2191 XI.M41 Section 4.b.vii.

Note 1: This AMP treats carbon steel as "steel" as the aging effects are identical for these materials. This includes buried and underground piping found in the Off-Gas systems.

SLRA Section B.2.3.27 on page B-199 is revised as follows:

New and replacement coatings for underground components are to be consistent with Table 1 of SP0169-2007 or Section 3.4 of NACE RP0285-2002.

Existing coatings for underground components at MNGP were installed per site design specifications. Site design specifications did not require coatings on underground components. This is an exception to what is required by Table 1 of SP0169-2007 or Section 3.4 of NACE RP0285-2002.

This exception is acceptable for the underground Off-Gas System piping located in the Recombiner Vault beneath the Recombiner Building because OE demonstrates that the lack of moisture intrusion does not lead to accelerated degradation of the uncoated underground steel components in that vault at MNGP. The vault contains a steam pipe which has steam at 540ºF and two condensate pipes which operate between approximately 100ºF to 130ºF. These pipes increase the temperature in the vault; helping to remove any moisture, if present, in the vault. Pictures taken when the vault was entered showed no evidence of degradation of the piping or that water intrusion was creating an aggressive environment. The vault is located under the Recombiner Building roof and is only partially underground (approximately the bottom 2 feet is below grade). The vault environment is more like an indoor environment than an underground vault, though an underground environment was selected to be conservative. The Off-Gas System piping in the vault that is in the scope of license renewal is located approximately one foot above the floor of the Recombiner Vault; minimizing the potential for sustained contact with water intrusion. All of these factors make the Recombiner Vault environment different from and not represented by the OE for the carbon steel pipe that was located in the seismic gap space between the reactor and turbine buildings, which was periodically exposed to groundwater due to the pipe's elevation with respect to the water table and because the seismic gap space was not enclosed to prevent communication with the external environment. Although the piping in the vault is not wrapped or coated as described in Element 2 of NUREG-2191, Chapter XI.M41, increased inspection of the piping is not needed because of the plant specific OE that has not observed any degradation of the Off-Gas piping in the vault below the Recombiner Building, the location of the in-scope piping above the vault floor, and because the environment in vault would prevent condensation and mitigate the ability of any moisture that entered the vault to remain present and cause significant corrosion. MNGP will manage the aging of the underground Off-Gas piping through the Buried and Underground Piping program.

This exception is acceptable for the stainless steel piping that is identified as being in an underground environment in the seismic gap space between the reactor and turbine buildings because the piping is newly installed and so is expected to have no age-related degradation at this time. The pipes are usually exposed to an underground (air – outdoor) environment and occasionally exposed to groundwater. The air – outdoor environment at MNGP has not been shown to have significant concentrations of halides to cause degradation of the stainless steel pipes. This is based on the 50+ years of operating experience at MNGP with no records of corrosion or cracking to stainless steel piping exposed to an air – outdoor environment (see SLRA Sections 3.3.2.2.3 and 3.3.2.2.4). The pipes are also occasionally exposed to an underground (groundwater) environment. Groundwater at MNGP over the last 4 years has shown an average chloride concentration of ~230 ppm, with maximum and minimum values of 384 ppm and 135

ppm, respectively. The trend of the chloride concentration in the groundwater over the last four years is slightly decreasing overall. Although NUREG-2191 indicates that SCC at ambient temperatures only occurs in the presence of significant concentrations of contaminants (Chapter IX, Section D, page IX D-1), it does not define a threshold value for significant. NUREG-2191 does define the aggressive environment for steel embedded in concrete as 500 ppm chlorides (Chapter IX, Table IX.D, page IX D-2; and Chapter IX, Table IX.F, page IX F-3), which is more than twice the average chloride concentration in the groundwater and over 100 ppm more than the maximum recorded chloride concentration in the groundwater. The stainless steel piping is only exposed to groundwater when the groundwater level rises to an elevation of ~910 feet above sea level and so it is not in continuous contact with groundwater. However, because the stainless steel pipes in an underground environment in the seismic gap space between the reactor and turbine building are not wrapped or coated and are exposed to the chlorides discussed above, MNGP will perform increased inspections beyond what is provided in the guidance in NUREG-2191, Chapter XI.M41 on these pipes. An enhancement will be made to inspect a sample of the pipes in the most susceptible penetrations to corrosion at least once during every 6-year or 3 refueling cycle period starting in the 6-year period before the SPEO and continuing throughout the SPEO.

An enhancement has been added to ensure compliance with Table 1 of NACE SP0169-2007 or Section 3.4 of NACE RP0285-2002 for new and replacement underground piping.

SLRA Section B.2.3.27 on page B-200 is revised as follows:

Element Affected	Enhancement
4. Detection of Aging Effects	Update MNGP BUPT AMP procedures as appropriate: • Clarify that inspections of buried and underground piping and tanks within the applicable plant systems will be conducted in accordance with NUREG-2191 Table XI.M41-2 Preventive Action Category **C** for buried steel and stainless steel piping, unless a reevaluation of cathodic protection performance, future OE, or soil conditions determines that another Preventive Action Category is more applicable. When the inspections for a given material type is based on percentage of length and results in an inspection quantity of less than 10 feet, then 10 feet of piping is inspected. If the entire run of piping of that material type is less than 10 feet in total length, then the entire run of piping is inspected. • Clarify that the visual inspections will be supplemented with surface and/or volumetric nondestructive testing if evidence of wall loss beyond minor surface scale is observed. • **Clarify that, if alternatives to visual inspections are performed, they will be performed in accordance with NUREG-2191, Section XI.M41, Subsection 4.e.** • Clarify the guidance for piping inspection location selection as follows: (a) a risk ranking system software incorporates inputs that include coating type, coating condition, cathodic protection efficacy, **cathodic protection overprotection history,** backfill characteristics, soil resistivity, pipe contents, and pipe function; (b) opportunistic examinations of nonleaking pipes may be credited toward examinations if the location selection criteria are met; and (c) the use of guided wave ultrasonic examinations may not be substituted for the required inspections. • **Clarify that, the underground stainless steel pipes located in the seismic gap space between the reactor and turbine buildings that are in the scope of this program, will have inspections performed on a sample of the pipes in the most susceptible penetration to corrosion at least once during every 6-year or 3 refueling outage period.**

Enclosure 06

External Surfaces Monitoring of Mechanical Components

External Surfaces Monitoring of Mechanical Components

Implement Follow-up Inspections of Two of the Six Penetrations With Below Grade Piping Going Between the Reactor and Turbine Buildings Using the External Surfaces Monitoring of Mechanical Components Program.

Affected SLRA Sections: A.2.2.23, Table A-3, Commitment 26, and B.2.3.23

SLRA Page Numbers: A-24, A-80, B-176, B-178, B-181

Description of Change:

In-scope systems with pipes that are routed between the reactor and turbine buildings pass through a seismic gap space between the wall of the two buildings. Access to the portions of these pipes in the seismic gap space is limited because of the mechanical penetration seals required to maintain the integrity of the walls through which the pipes are routed. The actual environment of the piping in the seismic gap and the potential environment defined by GALL-SLR depends on the elevation of the penetrations. Of the 35 penetrations identified where pipes run between the reactor and turbine buildings, eight (8) penetrations are located below the grade elevation (931') of MNGP. The piping above the grade elevation of MNGP in the seismic gap space is exposed to an air-outdoor environment as defined by GALL-SLR and will have their aging effects managed using the External Surfaces Monitoring of Mechanical Components program during the SPEO. The two below-grade penetrations that are also located below or near the maximum elevation of the water table in the vicinity of MNGP (912') are considered to be exposed to an underground-groundwater environment for MNGP SLRA and will have their aging managed using the MNGP Buried and Underground Piping and Tanks program. The physical configuration of the remaining six penetrations that are below the grade elevation of MNGP but above the elevation of the water table could be considered to be in an underground-air environment or an air-outdoor environment based on interpretations of the definitions of those environments in GALL-SLR. The MNGP SLRA interprets the environment of the pipes to be best represented by the air-outdoor environment and proposes to manage the aging of these pipes using the External Surfaces Monitoring of Mechanical Components program.

Because of the potential to interpret the environment of the pipes in these six penetrations as being in an underground-air environment, MNGP will add a commitment in this enclosure to enhance the External Surfaces Monitoring of Mechanical Components program to perform additional inspections on a sample of these pipes to provide reasonable assurance that the intended functions of the pipes in these six penetrations will be adequately aged managed throughout the SPEO.

An unrelated enhancement will also be added to the External Surfaces Monitoring of Mechanical Components program to explicitly include the in-scope portions of the Off-Gas System as a part of the program to ensure that aging is managed throughout the SPEO.

SLRA Section A.2.2.23 on page A-24 is revised as follows:

A sample of the below-grade pipes within the scope of the External Surfaces Monitoring of Mechanical Components AMP that are located in the seismic gap space between the reactor and turbine buildings will be inspected 10 years following the initial inspection that was performed prior to the SPEO. The sample will include the piping located in at least two of the six below-grade piping penetrations between the reactor and turbine buildings.

Inspections not conducted in accordance with ASME Code Section XI requirements are conducted in accordance with plant-specific procedures, including inspection parameters such as lighting, distance, offset, and surface coverage and presence of protective coatings.

Acceptance criteria are such that the component will meet its intended function until the next inspection or the end of the SPEO. Qualitative acceptance criteria are clear enough to reasonably assure a singular decision is derived based on observed conditions.

SLRA Table A-3 on page A-80 is revised as follows:

No.	Aging Management Program or Activity (Section)	NUREG-2191 Section	Commitment	Implementation Schedule
26	External Surfaces Monitoring of Mechanical Components (A.2.2.23)	XI.M36	n) Revise procedures to specify, where practical, acceptance criteria are quantitative. o) Revise procedures to specify 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. **p) Revise procedures to require a sample of the below-grade pipes within the scope of the External Surfaces Monitoring of Mechanical Components AMP that are located in the seismic gap space between the reactor and turbine buildings to be inspected 10 years following the initial inspection that was performed prior to the SPEO. The sample will include the piping located in at least two of the six below-grade piping penetrations between the reactor and turbine buildings.**	No later than 6 months prior to the SPEO, or no later than the last refueling outage prior to the SPEO. **Perform commitment p inspections no later than May 2033.**

SLRA Section B.2.3.23 on page B-176 is revised as follows:

A sample of the below-grade pipes within the scope of the External Surfaces Monitoring of Mechanical Components AMP that are located in the seismic gap space between the reactor and turbine buildings will be inspected 10 years following the initial inspection that was performed prior to the SPEO. The sample will include the piping located in at least two of the six below-grade piping penetrations between the reactor and turbine buildings.

Alternative methods for detecting moisture/corrosion inside piping insulation (such as thermography, neutron backscatter devices, and moisture meters) will be used for inspecting piping jacketing that is not installed in accordance with plant-specific procedures (such as no minimum overlap, wrong location of seams, etc.).

SLRA B.2.3.23 on page B-178 is revised as follows:

Element Affected	Enhancement
	○ hardening or loss of strength of internal surfaces for elastomeric components. ○ When credited, the program provides the basis to establish that the external and internal surface condition and environment are sufficiently similar. • **Revise procedures to include inspection and aging management of the in-scope portions of the Off-Gas System in the program.**

SLRA B.2.3.23 on page B-181 is revised as follows:

Element Affected	Enhancement
	○ Visual inspections will cover 100 percent of accessible component surfaces. ○ Manual or physical manipulation can be used to augment visual inspection to confirm the absence of hardening or loss of strength for elastomers and flexible polymeric materials (e.g., heating, ventilation, and air conditioning flexible connectors) where appropriate, and the sample size for manipulation is at least 10 percent of available surface area. • **Revise procedures to require a sample of the below-grade pipes within the scope of the External Surfaces Monitoring of Mechanical Components AMP that are located in the seismic gap space between the reactor and turbine buildings to be inspected 10 years following the initial inspection that was performed prior to the SPEO. The sample will include the piping located in at least two of the six below-grade piping penetrations between the reactor and turbine buildings.**