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NMP1L3711

10 CFR 50
10 CFR 51
10 CFR 54

September 21, 2026

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

Nine Mile Point Nuclear Station, Unit 1
Renewed Facility Operating License No. DPR-63
NRC Docket No. 50-220

Subject: Supplement No. 2 - Changes to the Nine Mile Point Nuclear Station, Unit 1,
Subsequent License Renewal Application

- References:
1. Letter from Christopher D. Wilson (Constellation Energy Generation, LLC) to U.S. NRC, "Application for Subsequent Renewed Operating License," dated March 25, 2026 (ML26084A179)
 2. Letter from Christopher D. Wilson (Constellation Energy Generation, LLC) to U.S. NRC, "Supplement 1 – Changes to the Nine Mile Point Nuclear Station, Unit 1, Subsequent License Renewal Application," dated July 30, 2026 (ML26211A185)

In Reference 1, Constellation Energy Generation, LLC (CEG) submitted a Subsequent License Renewal Application (SLRA) for Nine Mile Point Nuclear Station, Unit 1 (NMP1). In Reference 2, CEG submitted Supplement No. 1 to the SLRA for NMP1. The purpose of this letter is to provide Supplement No. 2 to the SLRA for NMP1. Supplement No. 2 includes 18 changes to the SLRA which provide additional information and clarifications to address NRC Safety Review Audit information needs, as well as CEG self-identified changes.

Enclosure A to this letter provides a description of each change, and corresponding mark-ups to affected portions of the SLRA, thereby supplementing the NMP1 SLRA.

Enclosure B to this letter provides an update to the Subsequent License Renewal Commitment List (SLRA Appendix A, Table A.5). There are no other new or revised regulatory commitments contained in this letter.

This submittal has been discussed with the NRC License Renewal Project Manager for the NMP1 Subsequent License Renewal project.

Should you have any questions regarding this submittal, please contact Ms. Lydia S. Dworakowski, Licensing Lead, NMP1 Subsequent License Renewal Project, by email at Lydia.Dworakowski@constellation.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 16th day of September 2026.

Respectfully,

Wilson, Christopher D Digitally signed by Wilson, Christopher D
Date: 2026.09.21 08:48:20 -04'00'

Christopher D. Wilson
License Renewal Director
Constellation Energy Generation, LLC

Enclosures:

- A. Changes to the Nine Mile Point Nuclear Station, Unit 1, Subsequent License Renewal Application
- B. Nine Mile Point Nuclear Station, Unit 1, Subsequent License Renewal Commitment List Update

cc: NRC Regional Administrator, Region I
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Enclosure A

**Changes to the Nine Mile Point Nuclear Station, Unit 1,
Subsequent License Renewal Application
Supplement 2**

Introduction

This enclosure contains 18 changes that are being made to the Subsequent License Renewal Application (SLRA) that were identified after submittal of the SLRA. For each item, the change is described and the affected page number(s) and portion(s) of the SLRA are provided. For clarity, entire sentences or paragraphs from the SLRA are provided. Previously submitted information is shown in normal font. Additions due to this submittal are highlighted with red underlined font for inserted text, and ~~strikethroughs~~ for deleted text. Revisions to SLRA tables are shown by providing excerpts from the affected tables.

INDEX

Change Number	Breakout Topic	Title
1	TRP-27	Clarify Fire Water System Program Exceptions #2 and #3 and Enhancement #6
2	TRP-27	Revise Table 3.3.2-9 and Appendix A, Appendix B, and Table A.5 to Clarify Fire Water System Program Requirements
3	TRP-76	Revise Section 2.4.2 to Clarify Reactor Vessel Support Skirt Evaluation Boundary
4	TRP-26	Revisions to Tables 3.3.1 and 3.3.2-9 in Response to Fire Protection Program Breakout Questions
5	TRP-27	Clarify Testing Requirements for Sprinklers
6	TRP-76	Add AMR Items Associated with the Reduction in Fracture Toughness Aging Effect and Revise SLRA Further Evaluation Sections Related to Irradiation of Concrete and Steel Structures and Components
7	TRP-142.1	Clarification Regarding Use of NRC Approved DORT Methodology to Perform Fluence Calculations
8	TRP-143.5	Clarification on Environmentally Assisted Fatigue Consideration for Components Located in the Dry Steam Environment
9	TRP-9	Clarification on Core Shroud Tie Rod X-750 Inconel Component Service Life Enhancement
10	Aux System	Revisions to Turbine Building HVAC System and Reactor Building HVAC System Aging Management Evaluations to Include Air, Outdoor (External) Environment for Galvanized Steel Ducting and Components
11	TRP-143.5	Clarification on Average Temperature Approach and Feedwater Nozzle Inspections
12	TRP-143.5	Table 4.3.1-2 Component Location Clarifications and Addition of an Operating Experience Example Related to Inspection of FW/RWCU Mixing Region
13	Aux System	Turbine Building HVAC System Scoping and Screening Results Clarification Regarding Battery Room Exhaust Ducting
14	TRP-27	Revise Fire Water System Program Description in Appendix A and Appendix B for Consistency with NUREG-2191
15	Self-Identified	Add Exception to One-Time Inspection Program to Specify Inspection Methods for Flow Blockage Due to Fouling
16	Self-Identified	Add Enhancement to Lubricating Oil Analysis Aging Management Program
17	Self-Identified	Correct Component Types and Materials in the Control Room HVAC System
18	Self-Identified	Correct the Material Type for Turbine Building Closed Loop Cooling System Piping Components

Change #1 – Clarify Fire Water System Program Exceptions #2 and #3 and Enhancement #6

Affected SLRA Sections: Appendix A, Section A.2.1.16; Appendix A, Table A.5; and Appendix B, Section B.2.1.16

SLRA Page Numbers: A-22, A-81, B-93, B-94, B-95, and B-101

Description of Change:

SLRA Appendix B, Section B.2.1.16 Justification for Exception #2 is being revised to clarify that the components applicable to the exception are the private service main fire pump discharge strainers and a discussion of relevant operating experience related to maintenance and monitoring of the private service main fire pump discharge strainers is being added. Exception #3 is being revised to clarify that the exception does not apply to the private service main fire pump discharge strainers. SLRA Appendix B, Section B.2.1.16 Enhancement #6 is being revised to clarify that the enhancement does not apply to the private service main fire pump discharge strainers. Corresponding changes are also being made to Appendix A, Section A.2.1.16 and Appendix A, Table A.5 to clarify Enhancement #6. These changes are in response to a Fire Water System Program Breakout Question (i.e., TRP 27-1).

SLRA Appendix B, Section B.2.1.16 Justification for Exception #3 is being revised to correctly reflect the maximum interval of the private service main mainline strainer inspection frequencies of 6, 8, and 10 years. Justification for Exception #3 is also being revised to include details regarding the aging mechanisms reviewed for justifying strainer inspection frequency extensions. These changes are in response to a Fire Water System Program Breakout Question (i.e., TRP 27-2).

Accordingly, Appendix A, Section A.2.1.16; Appendix A, Table A.5; and Appendix B, Section B.2.1.16 are revised.

SLRA Appendix A, Section A.2.1.16, Fire Water System, page A-22, Enhancement #6 is revised as shown below.

A.2.1.16 Fire Water System

The Fire Water System aging management program will be enhanced to:

6. Perform a system mainline strainer inspection and cleaning after system flow exceeding that of a nominal two-inch orifice. This enhancement does not apply to the private service main fire pump discharge strainers.

SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, Commitment 16, Fire Water System, page A-81, Enhancement #6 is revised as shown below.

A.5 Subsequent License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
16	Fire Water System	Fire Water System is an existing program that will be enhanced to: 6. Perform a system mainline strainer inspection and cleaning after system flow exceeding that of a nominal two-inch orifice. <u>This enhancement does not apply to the private service main fire pump discharge strainers.</u>	Program will be enhanced no later than six months prior to the subsequent period of extended operation. One-time inspections will be completed within the five years prior to the subsequent period of extended operation and no later than the last refueling outage prior to the subsequent period of extended operation.	Section A.2.1.16

SLRA Appendix B, Section B.2.1.16, Fire Water System, pages B-93 and B-94, Justification for Exception #2 is revised as shown below. This SLRA markup includes changes submitted in Supplement 1, dated July 30, 2026, which revised the number of annual inspections performed between 2009 and 2014 via Change #5 and is duplicated here as an aid to the reviewer.

B.2.1.16 Fire Water System

Exceptions to NUREG-2191

2. NUREG-2191, AMP XI.M27, Table XI.M27-1 recommends compliance with the requirements of NFPA 25, Section 7.2.2.3 performed annually and after each significant flow. NFPA 25, Section 7.2.2.3 requires the station perform private service main mainline strainer inspections.

Nine Mile Point Nuclear Station, Unit 1 does not perform annual or post significant flow mainline strainer inspections of the private service main fire pump discharge strainers. **Program Element Affected: Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4).**

Justification for Exception

The NMP1 Fire Protection Program is a risk-informed performance based program that is licensed in accordance with the requirements of NFPA 805. Annual and post significant flow inspections of the private service main fire pump mainline discharge strainers are not required for an effective Fire Water System aging management program due to operating experience, the periodic performance of other tests and inspections, and the configuration of the private service main fire pump discharge strainers.

An annual inspection of the private service main fire pump mainline discharge strainers was historically performed at NMP1. However, the station retired the inspection in 2014. Between the years of 2009 and 2014, the station performed six annual inspections of the stainless steel private service main fire pump mainline discharge strainers. These inspections resulted in the identification of a positive trend in results indicating no failures (e.g., no fouling and no loss of material) and lack of need for the inspection. The basis for retiring the inspection, which also addresses the private service main fire pump discharge strainer configuration, was as follows:

The strainers are duplex strainers with the strainer remaining aligned to one basket all year long. Issues would be seen during the monthly surveillance run which records the strainer [differential pressure]. Opening and cleaning the strainer every year under preventive maintenance does not provide value. The NEIL commitment is met by documenting the differential pressure during monthly tasks.

A 10-year search of plant-specific operating experience was performed to determine if NMP1 has noted any age-related degradation. No failure was identified. Specifically, no flow blockage, loss of material, or any other age-related degradation of the private service main fire pump discharge strainers was identified.

In addition, the station performs several inspections and tests that would identify flow blockage, loss of material, or fouling in the system due to private service main fire pump discharge strainer aging. The observation of foreign material and organic matter downstream of the private service main fire pump discharge strainer would trigger an obstruction investigation of the system which would include consideration of the source of the material, such as loss of material within the private service main fire pump discharge strainer allowing the strainer element to be bypassed, per NFPA 25 2011 Edition, Annex D guidance. The station performs annual fire pump performance tests where the differential pressure across the private service main fire pump discharge strainers is measured against acceptance criteria and the station performs system flow tests that measure and trend hydraulic conditions through the various flow loops. The private service main fire pump discharge strainers are located where the external conditions of the strainer body can be monitored during testing for any signs of leakage due to material loss. Deviations from acceptance criteria and adverse trends are captured and evaluated in CAP. In addition, the station performs monitoring and removal of sediment in the fire pump suction bays each refueling outage in accordance with the requirements of NRC Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

Finally, the suction source of the fire pump does not necessitate additional inspections of the private service main fire pump discharge ~~mainline~~ strainers. The fire pumps take suction from the fire pump suction bay below the operating floor of the screen house through suction strainers that are inspected every refueling outage. This bay is part of the larger network of suction bays consisting of reinforced concrete walls, floor, and a ceiling that is sealed off from sources of debris. Water from the lake is filtered through traveling screens before entering the suction bays which reduces debris.

Therefore, annual and post significant flow inspections of the private service main fire pump ~~mainline~~ discharge strainers are not required for an effective Fire Water System aging management program due to operating experience history that shows no age-related degradation, the configuration of the strainers where they are rotated to ensure they are not fouled, and the periodic performance of other tests and inspections that would be capable of detecting a change in condition of the strainers.

SLRA Appendix B, Section B.2.1.16, Fire Water System, pages B-94 and B-95, Exception #3 and Justification for Exception #3 are revised as shown below.

B.2.1.16 Fire Water System

Exceptions to NUREG-2191

3. NUREG-2191, AMP XI.M27, Table XI.M27-1 recommends compliance with the requirements of NFPA 25, Section 7.2.2.3 performed annually and after each significant flow. NFPA 25, Section 7.2.2.3 requires the station perform annual private service main mainline strainer inspections.

Private service main mainline strainer inspections are not performed at a fixed annual frequency but instead are conducted at a frequency based on evaluation. This exception does not apply to the private service main fire pump discharge strainers. **Program Element Affected: Detection of Aging Effects (Element 4)**

Justification for Exception

Surveillance frequencies are maintained in accordance with the NRC-approved fire protection program (ADAMS Accession No. ML14126A003). All strainer inspections were initially performed at an annual frequency. The frequencies were extended to two years after no change in inspection results and, after additional inspections, extended to ~~five~~ six years as results remained unchanged. One strainer inspection was extended a third time to ~~nine~~ ten years, and another was extended a third time to eight years. These extensions were performed over approximately the last 25 years, and the inspection results justified the frequency extensions. The previous inspection results reviewed as the basis for the extensions include visual inspection for conditions associated with both flow blockage and loss of material aging effects. Examples of inspection conditions include:

- Oxide Formation
- Scaling
- Pitting
- Nodules
- Tubercles
- Slime Layers
- Mussel Evidence
- Odors
- Cracking
- Crevice Corrosion
- Flow Accelerated Corrosion (FAC)
- Galvanic Corrosion
- General Corrosion

- Coating Integrity (if a coating is present)
- Erosion (loss of material)

Imposing an annual inspection frequency on these strainers would not improve aging management of the fire water system if the current inspection frequency is justified. The NFPA 25 recommended inspection intervals apply equal to new systems and existing systems where the interval is justified. The current extended intervals are justified by inspection results.

SLRA Appendix B, Section B.2.1.16, Fire Water System, page B-101, Enhancement #6 is revised as shown below.

B.2.1.16 Fire Water System

Enhancements

6. Perform a system mainline strainer inspection and cleaning after system flow exceeding that of a nominal two-inch orifice. This enhancement does not apply to the private service main fire pump discharge strainers. **Program Element Affected: Detection of Aging Effects (Element 4)**

Change #2 - Revise Table 3.3.2-9 and SLRA Appendix A, Appendix B, and Table A.5 to Clarify Fire Water System Program Requirements

Affected SLRA Sections: Table 3.3.2-9; Appendix A, Section A.2.1.16; Appendix A, Table A.5; and Appendix B, Section B.2.1.16

SLRA Page Numbers: 3.3-227, A-22, A-81, B-97, B-98, B-99 and B-101

Description of Change:

SLRA Table 3.3.2-9 is being revised to include spray nozzles in an air – outdoor (external) environment to represent the hydrogen rack deluge system located outdoors.

SLRA Appendix B, Section B.2.1.16 Exception #5 is being revised to remove the three turbine building curtain deluge systems as systems that perform functional testing at an interval that exceeds the maximum frequency allowed by note 'n' in NUREG-2191 Table XI.M27-1.

SLRA Appendix B, Section B.2.1.16 Justification for Exception #5 is being revised to remove the three turbine building curtain deluge systems and specify the testing frequency for the two reserve transformer's deluge systems. The Justification for Exception #5 is also being revised to indicate that a representative sample of each type of in-scope suppression system will be inspected on a five-year frequency. Exception #5 is being revised to address operating experience related to hydrogen rack deluge system external inspection results. The Justification for Exception #5 is being revised to include the operating experience review for the two reserve transformers testing frequency.

SLRA Appendix B, Section B.2.1.16 Exception #6 is being revised to state that the corrective action program will be used to perform an extent of condition assessment.

SLRA Appendix B, Section B.2.1.16 Enhancement #4 is being revised to specify that a representative sample of no less than 50 percent of each type of suppression system is to be tested in each five-year interval and that all systems shall be assessed through an extent of condition assessment if foreign organic and/or inorganic material is found in any system during the five-year internal inspection of piping. Corresponding changes associated with the revision to Enhancement #4 are also being made to Appendix A, Section A.2.1.16 and Appendix A, Table A.5.

These changes are in response to Fire Water System Breakout Questions (i.e., TRP 27-4 and TRP 27-5).

Accordingly, SLRA Table 3.3.2-9; Appendix A, Section A.2.1.16; Appendix A, Table A.5, and Appendix B, Section B.2.1.16 are revised.

SLRA Table 3.3.2-9, Fire Detection and Protection System Summary of Aging Management Evaluation, page 3.3-227 is revised as shown below:

Table 3.3.2-9 Fire Detection and Protection System Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
36512	Spray Nozzles	Spray	Copper Alloy with less than 15% Zinc	Air - Indoor, Uncontrolled (External)	None	None	VII.J.AP-144	3.3.1-114	A
<u>39289</u>	<u>Spray Nozzles</u>	<u>Spray</u>	<u>Copper Alloy with less than 15% Zinc</u>	<u>Air - Outdoor (External)</u>	<u>Loss of Material</u>	<u>Fire Water System (B.2.1.16)</u>	<u>VII.G.A-403</u>	<u>3.3.1-130</u>	<u>D</u>
<u>39290</u>	<u>Spray Nozzles</u>	<u>Spray</u>	<u>Copper Alloy with less than 15% Zinc</u>	<u>Air - Outdoor (External)</u>	<u>Flow Blockage</u>	<u>Fire Water System (B.2.1.16)</u>	<u>VII.G.A-403</u>	<u>3.3.1-130</u>	<u>D</u>
36513	Spray Nozzles	Spray	Copper Alloy with less than 15% Zinc	Condensation (Internal)	Flow Blockage	Fire Water System (B.2.1.16)	VII.G.A-403	3.3.1-130	D

SLRA Appendix A, Section A.2.1.16, Fire Water System, page A-22, Enhancement #4 is revised as shown below:

A.2.1.16 Fire Water System

The Fire Water System aging management program will be enhanced to:

4. Perform, on a five-year frequency, internal visual inspections of a representative sample of each type of every other in-scope suppression system. The representative sample will consist of no less than 50 percent of the in-scope areas for each suppression system type. The inspections will be performed on a rotating basis such that all in-scope suppression systems will be inspected at least once during each 10-year period during the subsequent period of extended operation. These inspections will include nozzle inspections when performed on deluge systems. ~~The systems not inspected in the first five year interval will be inspected in the second five year interval.~~ In dry pipe systems and pre-action systems, the sprinkler removed for inspection shall be from the most remote branch line from the source of water that is not equipped with the inspector's test valve. Tubercules or slime, if found, shall be tested for indications of microbiologically influenced corrosion (MIC). If the presence of foreign organic and/or inorganic material is found in any system in a building during the five-year internal inspection of piping, all systems shall be assessed. This extent of condition assessment shall determine the appropriate scope and method of follow-up actions that will be performed through the corrective action program. If the presence of sufficient foreign organic and/or inorganic material is found to obstruct pipe or sprinklers, an obstruction investigation will be conducted as described in Section 14.3 of NFPA 25. Alternatively, nondestructive examinations capable of ensuring flow blockage in piping and nozzles will not occur will be performed. The population of suppression systems eligible for inspection consists of the in-scope wet pipe sprinkler systems, dry-pipe systems, preaction systems, and deluge systems that are not functionally tested.

SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, Commitment 16, Fire Water System, page A-81, Enhancement #4 is revised as shown below:

A.5 Subsequent License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
16	Fire Water System	Fire Water System is an existing program that will be enhanced to: 4. Perform, on a five-year frequency, internal visual inspections of <u>a representative sample of each type of</u> ~~every other~~ in-scope suppression system. <u>The representative sample will consist of no less than 50 percent of the in-scope areas for each suppression system type. The inspections will be performed on a rotating basis such that all in-scope suppression systems will be inspected at least once during each 10-year period during the subsequent period of extended operation.</u> These inspections will include nozzle inspections when performed on deluge systems. ~~The systems not inspected in the first five-year interval will be inspected in the second five-year interval.~~ In dry pipe systems and pre-action systems, the sprinkler removed for inspection shall be from the most remote branch line from the source of water that is not equipped with the inspector's test valve. Tubercules or slime, if found, shall be tested for indications of microbiologically influenced corrosion (MIC). If the presence of foreign organic and/or inorganic material is found in any system in a building during the five-year internal inspection of piping, all systems shall be assessed. <u>This extent of condition assessment shall determine the appropriate scope and method of follow-up actions that will be performed through the corrective action program.</u> If the presence of sufficient foreign organic and/or inorganic material is found to obstruct pipe or sprinklers, an obstruction investigation will be conducted as described in Section 14.3 of NFPA 25. Alternatively, nondestructive examinations capable of ensuring flow blockage in piping and nozzles will not occur will be performed. The population of suppression systems eligible for inspection consists of the in-scope wet pipe sprinkler systems, dry-pipe systems, preaction systems, and deluge systems that are not functionally tested.	Program will be enhanced no later than six months prior to the subsequent period of extended operation. One-time inspections will be completed within the five years prior to the subsequent period of extended operation and no later than the last refueling outage prior to the subsequent period of extended operation.	Section A.2.1.16

SLRA Appendix B, Section B.2.1.16, Fire Water System, pages B-97 and B-98, Exception #5 and Justification for Exception #5 are revised as shown below:

B.2.1.16 Fire Water System

Exceptions to NUREG-2191

5. NUREG-2191, AMP XI.M27, Table XI.M27-1 recommends compliance with the requirements of NFPA 25, Section 10.3.4.3 which requires the operational testing of water spray systems with either water or air to be performed on a frequency that is allowed by Table XI.M27-1 to be up to three years.

The NMP1 Fire Water System aging management program does not perform functional tests of the following deluge and spray systems:

- The two ventilation system charcoal filter spray systems.
- The 13 indoor spray systems.
- The west yard hydrogen rack deluge system.

The NMP1 Fire Water System aging management program performs a functional test of the following systems at an interval that exceeds the maximum frequency allowed by note 'n' in Table XI.M27-1.

- ~~Three turbine building curtain deluge systems.~~
- Two transformer deluge systems.

Program Element Affected: Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4), Acceptance Criteria (Element 6)

Justification for Exception

The two transformer deluge systems are functionally tested every other refueling outage as part of the divisional outage cycles to maintain outage electrical scope divisional separation. Surveillance frequencies for the two transformer deluge systems are established by the NRC-approved fire protection program (ADAMS Accession No. ML14126A003). A review of the last ten years within the corrective action program database identified no IRs or deficient conditions related to the two transformer deluge systems during testing performed every other refueling outage. ~~The program will be enhanced to perform annual functional tests of the: 1) turbine building west wall water curtain, 2) turbine building south wall water curtain, and 3) turbine building south and east control building wall exposure above administration building roof deluge systems.~~

The NMP1 Fire Protection program does not functionally test the indoor spray systems, the ventilation system charcoal filter spray systems, or the hydrogen rack deluge system. The program does not need to be further enhanced

because the configuration of these systems, the existing inspections, and other enhancements ensure the effectiveness of the aging management program in managing the aging effects of these systems.

The indoor spray systems are constructed of galvanized steel pipe. There are no aging effects in NUREG-2191 for galvanized steel pipe exposed to indoor air. Aging effects are assumed due to the prescriptive nature of the NUREG-2191 AMP XI.M27 which requires the internal environment to be considered as condensation. These systems are not periodically wetted. There is no station OE of inadvertent action of these systems. External visual inspections are performed every two years which inspect the external conditions of the pipe and verify nozzle orientation. These external visual inspections would also identify leak-by in the control valve. Finally, these systems are included in the population of systems associated with the enhancement to require internal visual and nozzle inspections or alternative nondestructive examinations. The internal visual inspection of a representative sample of the in-scope suppression systems at a five-year interval, with the remaining population inspected during the subsequent five-year interval, provides assurance that the internal condition of the entire population is acceptable and no blockage exists. The construction of these systems, operating experience, external visual inspections, and the enhancements to the program provide reasonable assurance that the AMP is effective in managing aging effects in these systems.

The NMP1 ventilation system charcoal filter spray systems are designed with at least two closed valves isolating the systems from the supply header. In each system, a tell-tale drain with a normally open drain valve is positioned between the two closed valves. The isolation valves are verified closed, the tell-tale drain valves are verified open, and the tell-tales are inspected for leakage quarterly. The systems are not subject to periodic wetting. A review of station operating experience identified no instances of inadvertent system actuation. The system is not subject to significant internal corrosion or flow blockage due to fouling since it is maintained dry and is not subject to periodic wetting during testing. The systems are designed and installed to allow for complete drainage. The nozzles are located inside filter housings which protect the nozzles from unplanned obstructions. In addition, these systems are included in the population of systems associated with two program enhancements, the one-time volumetric examination, and periodic internal inspections of a representative sample on a five-year interval. The internal visual inspection of a representative sample of the in-scope suppression systems at a five-year interval, with the remaining population inspected during the subsequent five-year interval, provides assurance that the internal condition of the entire population is acceptable and no blockage exists. A similar exception was previously reviewed and approved by the NRC as documented in the Safety Evaluation Report Related to the Subsequent License Renewal of Dresden Nuclear Power Station, Units 2 and 3 (ADAMS Accession No. ML25251A150). Finally, these systems are associated with two program enhancements. Periodic internal visual inspections or alternative NDE examinations will be performed and these systems are included in the enhancement to perform a one-time volumetric exam of normally dry systems.

The hydrogen rack deluge system is located outdoors and constructed of galvanized pipe. The system design and installation allow for the system to be completely drained. The system is not periodically wetted. There is no station OE of inadvertent action of the system. External visual inspections are performed every two years which inspect the external conditions of the pipe and verify nozzle orientation. These external visual inspections would also identify control valve leak-by. Review of the last ten years of external inspection reports identified no cases of degraded conditions on the deluge system piping, valves, or nozzles. Finally, this system is associated with the enhancement to perform periodic internal visual inspections or alternative NDE examinations. The internal visual inspection of a representative sample of the in-scope suppression systems at a five-year interval, with the remaining population inspected during the subsequent five-year interval, provides assurance that the internal condition of the entire population is acceptable and no blockage exists.

Therefore, the program does not need to be further enhanced because the configuration of these systems, the existing inspections, and other enhancements ensure the effectiveness of the aging management program in managing the aging effects of these systems.

SLRA Appendix B, Section B.2.1.16, Fire Water System, page B-99, Exception #6 is revised as shown below.

B.2.1.16 Fire Water System

Exceptions to NUREG-2191

6. NUREG-2191, AMP XI.M27, Table XI.M27-1 recommends compliance with the requirements of NFPA 25, Section 14.2 which requires the performance of internal visual inspections and promulgates corrective actions to take when internal visual inspections identify foreign organic or inorganic material. Specifically, Section 14.2.2.2 of the 2011 Edition of NFPA 25 requires: "If the presence of foreign organic and/or inorganic material is found in any system in a building during the 5-year internal inspection of piping, all systems shall have an internal inspection."

The NMP1 Fire Water System program will be enhanced to perform the internal visual inspections. However, an extent of condition assessment to determine the appropriate scope and method of follow-up actions will be performed through the corrective action program instead of internal visual inspections of every system if a periodic internal visual inspection identifies foreign organic and/or inorganic material. **Program Element Affected: Detection of Aging Effects (Element 4)**

SLRA Appendix B, Section B.2.1.16, Fire Water System, page B-101, Enhancement #4 is revised as shown below:

B.2.1.16 Fire Water System

Enhancements

4. Perform, on a five-year frequency, internal visual inspections of a representative sample of each type of ~~every other in-scope~~ suppression system. The representative sample will consist of no less than 50 percent of the in-scope areas for each suppression system type. The inspections will be performed on a rotating basis such that all in-scope suppression systems will be inspected at least once during each 10-year period during the subsequent period of extended operation. These inspections will include nozzle inspections when performed on deluge systems. ~~The systems not inspected in the first five-year interval will be inspected in the second five-year interval.~~ In dry pipe systems and pre-action systems, the sprinkler removed for inspection shall be from the most remote branch line from the source of water that is not equipped with the inspector's test valve. Tubercules or slime, if found, shall be tested for indications of microbiologically influenced corrosion (MIC). If the presence of foreign organic and/or inorganic material is found in any system in a building during the five-year internal inspection of piping, all systems shall be assessed. This extent of condition assessment shall determine the appropriate scope and method of follow-up actions that will be performed through the corrective action program. If the presence of sufficient foreign organic and/or inorganic material is found to obstruct pipe or sprinklers, an obstruction investigation will be conducted as described in Section 14.3 of NFPA 25. Alternatively, nondestructive examinations capable of ensuring flow blockage in piping and nozzles will not occur will be performed. The population of suppression systems eligible for inspection consists of the in-scope wet pipe sprinkler systems, dry-pipe systems, preaction systems, and deluge systems that are not functionally tested.
Program Element Affected: Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4), Monitoring and Trending (Element 5), Acceptance Criteria (Element 6)

Change #3 – Revise Section 2.4.2 to Clarify Reactor Vessel Support Skirt Evaluation Boundary

Affected SLRA Section: Section 2.4.2

SLRA Page Number: 2.4-10

Description of Change:

SLRA Section 2.4.2 is being revised to state that the reactor pressure vessel support skirt is evaluated with the Component Supports Commodity.

This change is in response to a Breakout Question (i.e., TRP 76-11).

Accordingly, SLRA Section 2.4.2 is revised.

SLRA Section 2.4.2, Primary Containment, page 2.4-10, last paragraph is revised as shown below.

2.4.2 Primary Containment

Not included in the evaluation boundary of the Primary Containment are mechanical and electrical systems and components housed within the structure; fire barriers; and component supports. Mechanical components associated with the primary containment (e.g., vacuum relief valves, containment inerting subsystem, and drywell cooling subsystem) are evaluated with the Containment Systems. The SRV discharge lines and y-quenchers are evaluated with the Main Steam System. The RPV support skirt is evaluated with the Component Supports Commodity ~~Reactor Pressure Vessel~~. Other mechanical and electrical systems and components housed inside the structure are separately evaluated with their respective mechanical systems, electrical systems, or commodities. Fire protection components are evaluated separately within the Fire Detection and Protection System. Foundation concrete supporting the drywell bottom, the biological shield surrounding the drywell, and shield plugs at the top of the reactor head cavity are evaluated with the Reactor Building. Support structures for the torus and the vent system, including their respective bolting, are evaluated with Component Supports Commodity. The stabilizer supports from the reactor pressure vessel to the reactor shield wall are evaluated with Component Supports Commodity. Structural commodities, including their associated structural bolting and concrete anchors, are evaluated with the Structural Commodity Group. The Structural Commodity Group evaluates electrical enclosures and raceways (including cable trays and tube track); hazard barriers and elastomers; insulation; and structural miscellaneous components (e.g., metal platforms, grating, stairs, and ladders).

Change #4 – Revisions to Tables 3.3.1 and 3.3.2-9 in Response to Fire Protection Program Breakout Questions

Affected SLRA Sections: Tables 3.3.1 and 3.3.2-9

SLRA Page Numbers: 3.3-140, 3.3-214, 3.3-215, 3.3-216 and 3.3-233

Description of Change:

A line item is being created to address separation as an aging effect for polymer Fire Barriers (Penetration Seals) using an H note in SLRA Table 3.3.2-9. Table 3.3.1 is being updated to reflect that the Fire Protection program will be used to manage separation of the polymer fire barrier penetration seals exposed to air - indoor, uncontrolled in the Fire Detection and Protection System.

A line item is being created to address change in material properties due to gamma irradiation exposure not being an applicable aging effect for polymer Fire Barriers (Penetration Seals) using an H note in SLRA Table 3.3.2-9.

A Plant Specific Note is being added to clarify the material and justification for each new item regarding polymer Fire Barriers (Penetration Seals).

A Plant Specific Note is also being added to 12 line items to clarify that grout, used to repair masonry and concrete walls, is managed under Fire Barriers (Masonry Walls) and Fire Barriers (Concrete Walls, Ceilings, Floors).

These changes are in response to Fire Protection program Breakout Questions (i.e., TRP 26-1 and TRP 26-2).

Accordingly, SLRA Tables 3.3.1 and 3.3.2-9 are revised.

SLRA Table 3.3.1, Summary of Aging Management Evaluations for the Auxiliary Systems, page 3.3-140 is revised as shown below.

Table 3.3.1 Summary of Aging Management Evaluations for the Auxiliary Systems

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-267	Subliming compound fireproofing/fire barriers (Thermo-lag®, Darmatt™, 3M™ Interam™, and other similar materials) exposed to air	Loss of material due to abrasion, flaking, vibration; cracking/delamination due to chemical reaction, settlement; change in material properties due to gamma irradiation exposure; separation	AMP XI.M26, "Fire Protection"	No	~~Not Applicable.~~ ~~There are no subliming compound fireproofing/fire barriers (Thermo-lag®, Darmatt™, 3M™ Interam™, and other similar materials) exposed to air in Auxiliary Systems.~~ <u>Consistent with NUREG-2191. The Fire Protection (B.2.1.15) program will be used to manage separation of the polymer fire barrier penetration seals exposed to air - indoor, uncontrolled in the Fire Detection and Protection System.</u>

SLRA Table 3.3.2-9, Fire Detection and Protection System Summary of Aging Management Evaluation, page 3.3-214 is revised as shown below.

Table 3.3.2-9 Fire Detection and Protection System Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
35175	Fire Barriers (Concrete Walls, Ceilings, Floors)	Fire Barrier	Reinforced Concrete	Air - Indoor, Uncontrolled (External)	Cracking	Fire Protection (B.2.1.15)	VII.G.A-90	3.3.1-060	A, <u>13</u>
45175	Fire Barriers (Concrete Walls, Ceilings, Floors)	Fire Barrier	Reinforced Concrete	Air - Indoor, Uncontrolled (External)	Cracking	Structures Monitoring (B.2.1.33)	VII.G.A-90	3.3.1-060	B, <u>13</u>
35176	Fire Barriers (Concrete Walls, Ceilings, Floors)	Fire Barrier	Reinforced Concrete	Air - Indoor, Uncontrolled (External)	Loss of Material	Fire Protection (B.2.1.15)	VII.G.A-90	3.3.1-060	A, <u>13</u>
45176	Fire Barriers (Concrete Walls, Ceilings, Floors)	Fire Barrier	Reinforced Concrete	Air - Indoor, Uncontrolled (External)	Loss of Material	Structures Monitoring (B.2.1.33)	VII.G.A-90	3.3.1-060	B, <u>13</u>
35177	Fire Barriers (Concrete Walls, Ceilings, Floors)	Fire Barrier	Reinforced Concrete	Air - Outdoor (External)	Cracking	Fire Protection (B.2.1.15)	VII.G.A-90	3.3.1-060	A, <u>13</u>
45177	Fire Barriers (Concrete Walls, Ceilings, Floors)	Fire Barrier	Reinforced Concrete	Air - Outdoor (External)	Cracking	Structures Monitoring (B.2.1.33)	VII.G.A-90	3.3.1-060	B, <u>13</u>
35178	Fire Barriers (Concrete Walls, Ceilings, Floors)	Fire Barrier	Reinforced Concrete	Air - Outdoor (External)	Loss of Material	Fire Protection (B.2.1.15)	VII.G.A-90	3.3.1-060	A, <u>13</u>
45178	Fire Barriers (Concrete Walls, Ceilings, Floors)	Fire Barrier	Reinforced Concrete	Air - Outdoor (External)	Loss of Material	Structures Monitoring (B.2.1.33)	VII.G.A-90	3.3.1-060	B, <u>13</u>

SLRA Table 3.3.2-9, Fire Detection and Protection System Summary of Aging Management Evaluation, page 3.3-215 is revised as shown below.

Table 3.3.2-9 Fire Detection and Protection System Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
35192	Fire Barriers (Masonry Walls)	Fire Barrier	Concrete Block	Air - Indoor, Uncontrolled (External)	Cracking	Fire Protection (B.2.1.15)	VII.G.A-626	3.3.1-179	A, <u>13</u>
45192	Fire Barriers (Masonry Walls)	Fire Barrier	Concrete Block	Air - Indoor, Uncontrolled (External)	Cracking	Masonry Walls (B.2.1.32)	VII.G.A-626	3.3.1-179	B, <u>13</u>
35193	Fire Barriers (Masonry Walls)	Fire Barrier	Concrete Block	Air - Indoor, Uncontrolled (External)	Loss of Material	Fire Protection (B.2.1.15)	VII.G.A-626	3.3.1-179	A, <u>13</u>
45193	Fire Barriers (Masonry Walls)	Fire Barrier	Concrete Block	Air - Indoor, Uncontrolled (External)	Loss of Material	Masonry Walls (B.2.1.32)	VII.G.A-626	3.3.1-179	B, <u>13</u>

SLRA Table 3.3.2-9, Fire Detection and Protection System Summary of Aging Management Evaluation, page 3.3-216 is revised as shown below.

Table 3.3.2-9 Fire Detection and Protection System Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
36404	Fire Barriers (Penetration Seals)	Fire Barrier	Polymer	Air - Indoor, Uncontrolled (External)	Loss of Material Due to Peeling, Delamination, Wear	Fire Protection (B.2.1.15)	VII.I.A-797a	3.3.1-263	E, 1
39292	Fire Barriers (Penetration Seals)	Fire Barrier	Polymer	Air - Indoor, Uncontrolled (External)	Separation	Fire Protection (B.2.1.15)	VII.G.A-805	3.3-1, 267	H, 11
39298	Fire Barriers (Penetration Seals)	Fire Barrier	Polymer	Air - Indoor, Uncontrolled (External)	None	None	VII.G.A-805	3.3-1, 267	H, 12
35451	Fire Barriers (Penetration Seals)	Fire Barrier	Silicate Fireproofing	Air - Indoor, Uncontrolled (External)	Change in Material Properties	Fire Protection (B.2.1.15)	VII.G.A-807	3.3.1-269	A

SLRA Table 3.3.2-9, Fire Detection and Protection System Summary of Aging Management Evaluation, page 3.3-233, Plant Specific Notes 11, 12, and 13 are added as shown below. This SLRA markup includes changes submitted in Supplement 1, dated July 30, 2026, which added Plant Specific Note 10 via Change #7 and is duplicated here as an aid to the reviewer.

Table 3.3.2-9 Fire Detection and Protection System Summary of Aging Management Evaluation

Plant Specific Notes:

10. The aging effects of the buried fiberglass tank exposed to soil will be managed by monitoring the interstitial space between the primary and secondary tank walls for leakage. External visual inspections will not be performed.
11. Flamemastic (Polymer) Fire Barriers (Penetration Seals) are susceptible to separation based on plant operating experience. This aging effect will be managed by the Fire Protection (B.2.1.15) program.
12. Flamemastic (Polymer) Fire Barriers (Penetration Seals) are not susceptible to change in material properties due to gamma irradiation exposure being below the limit defined in EPRI 3002013084 Section 6.3.3.2.
13. Grout, used to repair masonry and concrete walls, is managed under Fire Barriers (Masonry Walls) and Fire Barriers (Concrete Walls, Ceilings, Floors).

Change #5 – Clarify Testing Requirements for Sprinklers

Affected SLRA Sections: Appendix A, Section A.2.1.16 and Appendix B, Section B.2.1.16

SLRA Page Numbers: A-21 and B-91

Description of Change:

The Fire Water System program description in Appendix A, Section A.2.1.16 and Appendix B, Section B.2.1.16 is being revised to clarify that testing of sprinklers will be performed per NFPA Section 5.3.1 on the ten-year frequency described and remove discussion of a station specific evaluation resulting in no further testing.

Appendix B, Section B.2.1.16 is also being revised to include the station basis for not performing five-year sprinkler testing per footnote (e) of NUREG-2191 Table XI.M27-1.

These changes are in response to a Fire Water System program Breakout Question (i.e., TRP 27-6).

Accordingly, SLRA Appendix A, Section A.2.1.16 and Appendix B, Section B.2.1.16 are revised.

SLRA Appendix A, Section A.2.1.16, Fire Water System, page A-21, fourth paragraph is revised as shown below.

A.2.1.16 Fire Water System

Testing or replacement of sprinklers that have been in place for 50 years, 60 years, 70 years, and 80 years is performed using the guidance of NFPA 25, 2011 Edition, Section 5.3.1 if a station-specific evaluation cannot determine the water is non-corrosive to sprinklers.

SLRA Appendix B, Section B.2.1.16, Fire Water System, page B-91, Program Description discussion, fifth paragraph is revised as shown below.

B.2.1.16 Fire Water System

Program Description

The Fire Water System program includes replacement or testing of a representative sample of sprinklers after they reach 50 years, 60 years, 70 years, and 80 years of service in accordance with NFPA 25, 2011 Edition, Section 5.3.1. ~~if a plant-specific evaluation cannot determine the water is non-corrosive to sprinklers.~~ The water source for the fire water system is a freshwater body, which is not a chemically reactive water source. Additionally, the external environment of the fire water system does not meet the definitions of a harsh environment described within NFPA 25, 2011 Edition, Section A.5.3.1.1.2. Previously performed sprinkler testing on a test sample population of 28 sprinklers identified a single removed sprinkler as slightly corroded during visual inspection. However, there was no impact due to the corrosion since the sprinkler passed its functional testing following the visual inspection. The previous sprinkler testing also identified a single sprinkler as abnormal due to not flowing water at the test water pressure despite the release mechanism functioning. The condition of the sprinkler, which was found to be abnormal, was documented in the test report as being dusty and did not reference any aging effects associated with corrosion. The station evaluated the test failure as age-related but did not attribute the failure to a specific aging effect. All the sprinklers from the same manufacturer in the zone of the sprinkler with the abnormal test result were replaced. As the water supply is not chemically reactive and there has not been any plant-specific evidence of corrosion impacts to system functionality, the five-year testing frequency per NFPA 25, 2011 Edition, Section 5.3.1.1.2 described under Footnote (e) of Table XI.M27-1 is not required.

Change #6 – Add AMR Items Associated with the Reduction in Fracture Toughness Aging Effect and Revise SLRA Further Evaluation Sections Related to Irradiation of Concrete and Steel Structures and Components

Affected SLRA Sections: Sections 3.5.2.1.1, 3.5.2.1.2, 3.5.2.2.1.2, 3.5.2.2.2.6, 3.5.2.2.2.8, and Tables 3.5.1, 3.5.2-1, and 3.5.2-2

SLRA Page Numbers: 3.5-2, 3.5-3, 3.5-11, 3.5-40, 3.5-41, 3.5-45, 3.5-46, 3.5-47, 3.5-77, 3.5-96, 3.5-98, 3.5-117, 3.5-119, 3.5-120, 3.5-127, 3.5-129, and 3.5-133

Description of Change:

SLRA Section 3.5.2.1.1 is being revised to include the full description of the aging effect for reduction in fracture toughness to align with NUREG-2191.

SLRA Section 3.5.2.1.2 is being revised to include reduction in fracture toughness, i.e., cracking, and/or loss of intended function as an aging effect requiring management.

SLRA Section 3.5.2.2.1.2 is being revised to provide additional information regarding primary containment ventilation.

SLRA Section 3.5.2.2.2.6 is being revised to update the maximum fluence level in the reactor cavity area based on a vendor corrected fluence report and to provide additional information regarding the reactor pedestal steel liner inspection criteria.

SLRA Section 3.5.2.2.2.8 is being revised to clarify the design of the reactor shield wall, include evaluation of the reactor shield wall steel columns (27WF177 columns), and discuss the energy groups used in the evaluation of radiation exposure damage. SLRA Tables 3.5.1, 3.5.2-1, and 3.5.2-2 are also being revised to include new AMR items associated with the reduction in fracture toughness aging effect for components evaluated in SLRA Section 3.5.2.2.2.8.

These changes are in response to Breakout Questions (i.e., TRP 76-5, 76-6, 76-7, 76-8, 76-9, 76-11, 76-14 and 76-17).

Accordingly, SLRA Sections 3.5.2.1.1, 3.5.2.1.2, 3.5.2.2.1.2, 3.5.2.2.2.6, 3.5.2.2.2.8 and Tables 3.5.1, 3.5.2-1, and 3.5.2-2 are revised.

SLRA Section 3.5.2.1.1, Components Supports Commodity, page 3.5-2 is revised as shown below.

3.5.2.1.1 Component Supports Commodity

Aging Effect Requiring Management

The following aging effects associated with the Component Supports Commodity components require management:

- Cracking
- Loss of Material
- Loss of Mechanical Function
- Loss of Preload
- Reduction in Concrete Anchor Capacity
- Reduction in Fracture Toughness, i.e., Cracking, and/or Loss of Intended Function

SLRA Section 3.5.2.1.2, Primary Containment, page 3.5-3 is revised as shown below.

3.5.2.1.2 Primary Containment

Aging Effect Requiring Management

The following aging effects associated with the Primary Containment components require management:

- Cracking
- Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling)
- Cracking, Loss of Material
- Cumulative Fatigue Damage
- Increase in Porosity and Permeability, Cracking, Loss of Material (Spalling, Scaling)
- Loss of Coating or Lining Integrity
- Loss of Leak tightness
- Loss of Material
- Loss of Mechanical Function
- Loss of Preload
- Loss of Sealing
- Reduction in Fracture Toughness, i.e., Cracking, and/or Loss of Intended Function
- Reduction of Strength; Loss of Mechanical Properties

SLRA Section 3.5.2.2.1.2, Reduction of Strength and Modulus Due to Elevated Temperature, page 3.5-11, response, first paragraph is revised as shown below.

3.5.2.2.1.2 Reduction of Strength and Modulus Due to Elevated Temperature

Item Number 3.5.1-003 is not applicable to the NMP1 Mark I steel containment. This Item Number is for concrete containment components. However, the temperature criteria in this section applies to all concrete within primary containment. NMP1 UFSAR Section VI E.1.2 limits the bulk air temperature inside the drywell during normal plant operation to 150 degrees F. The drywell is cooled during normal power operation by a closed loop ventilation system designed to hold the average temperature to 150 degrees F or lower. The primary containment ventilation system is described in UFSAR Section VI E.1.0 and in SLRA Section 2.3.3.4 Containment Systems. UFSAR Figure VI-23 illustrates the primary containment ventilation system and the airflow paths through primary containment. Plant operating experience has not identified elevated general area and local area temperatures as a concern for concrete structural components. Therefore, concrete structural components located inside the drywell are not subject to general area temperatures greater than 150 degrees F, or local area concrete temperature greater than 200 degrees F.

SLRA Section 3.5.2.2.2.6, Reduction of Strength and Mechanical Properties of Concrete Due to Irradiation, pages 3.5-40 and 3.5-41, response, is revised as shown below.

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

The calculated maximum fluence level is 1.323×10^{18} neutrons/cm² (fluence cutoff energy $E > 0.1$ MeV) at the concrete surface adjacent to the reactor shield wall interior plate for the 67 EFY projected for the SPO. By comparison, the calculated neutron fluence level for the reactor pedestal is 9.18×10^{15} neutrons/cm². This evaluation makes use of the NRC approved BWR fluence calculation methodology (ADAMS Accession No. ML032760696), which utilizes a three-dimensional discrete ordinates transport model to estimate peak neutron fluence regarding potential concrete damage. The analysis is consistent with RG 1.190, and the calculations have been benchmarked to plant-specific BWR data. The calculated maximum fluence level for NMP1 is within the NUREG-2192 threshold limit of 1×10^{19} neutrons/cm². Since the neutron fluence is below the NUREG-2192 threshold limit, potential effects from radiation induced volumetric expansion of the aggregates in the concrete are expected to be negligible.

A separate analysis was performed to evaluate a representative and bounding gamma dose for the NMP1 reactor shield wall. This analysis considers radiation sources including gamma dose from the core and secondary gamma dose. The maximum calculated gamma dose projected at 67 EFY is 3.23×10^9 rad concrete at the inner concrete region of the reactor shield wall. The largest contribution to the concrete dose is from the core, and the second largest is from the vessel activation, which is approximately 28 times lower than the dose from the core. By comparison, the maximum gamma dose computed at the surface of supporting reactor pedestal is 1.40×10^8 rad concrete. For NMP1, the peak gamma dose value at 67 EFY at the inner concrete region of the reactor shield wall at the RPV beltline does not surpass the NUREG-2192 threshold limit of 1×10^{10} rad concrete.

The results of both irradiation evaluations support the conclusion that the integrity of the reactor shield wall concrete is assured, and that no additional aging management beyond periodic inspections in accordance with the current Structures Monitoring (B.2.1.33) program is necessary to address the potential for reduction of strength, loss of mechanical properties, and cracking due to irradiation. The reactor pedestal concrete and pedestal steel liner inspections will be performed and evaluated using the second tier criteria in Chapter 5 of ACI 349.3R-02. As demonstrated above, the projected neutron fluence and gamma dose received by any portion of the reactor shield wall concrete remain below the NUREG-2192 threshold limits during the SPO. Therefore, a plant-specific aging management program is not required to manage this aging effect.

SLRA Section 3.5.2.2.2.8, Combined Effects of Aging Associated with Irradiation of RV Steel Structural Support Components and Loss of Function of Other RV Structural Support Components That are Not Concrete, pages 3.5-45, 3.5-46, and 3.5-47 are revised as shown below:

3.5.2.2.2.8 Combined Effects of Aging Associated with Irradiation of RV Steel Structural Support Components and Loss of Function of Other RV Structural Support Components That are Not Concrete

The reactor shield wall, surrounding the reactor vessel, is a 2-foot-thick hollow cylinder of concrete with 5/8-inch-thick steel liner plates covering its interior and exterior surfaces. The wall includes a structurally reinforced section at its base, while the remaining concrete primarily serves a non-structural shielding function with steel columns embedded within. The reactor shield wall, which is approximately 45 feet tall, is supported on the same concrete pedestal as the RPV. The top of the reactor shield wall is covered by a channel-shaped ring girder. The ring girder is anchored by rigid bars to the drywell walls. The reactor shield wall is described in UFSAR Section XVI B.2.6 and shown on UFSAR Figure XVI-17. In general, the reactor shield wall has been analyzed as a double-walled cylinder simply supported at its ends and pressurized internally. Both the interior and exterior steel liner plates are stressed in tension due to internal pressure. Since the loading distribution between the interior and outer plates was within 5 percent, they were assumed equal. The plates are connected to steel columns by continuous 3/8-inch fillet welds on the interior plate and intermittent 5/16-inch fillet welds on the exterior plate. As described in UFSAR Section VI.B, the containment is designed for the simultaneous occurrence of the design internal pressure and temperature, plus the dead, live, and seismic loads imposed on the shell. UFSAR Section VI.B further identifies the maximum internal design temperature in the drywell as 310 degrees F. Components of the containment systems must be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless the temperatures are well above the nil-ductility temperatures (NDT). For ferritic materials of the containment, the minimum temperature is NDT plus 30 degrees Fahrenheit.

The following RPV steel support components were evaluated against the NUREG-1509 Section 4.2.4 criterion: reactor support skirt, reactor shield wall interior plate, reactor shield wall steel columns, stabilizer, and ring girders. In accordance with NUREG-2192, the plant-specific radiation exposure level at a specific location in the RPV steel structural supports can be considered low if the radiation exposure damage level is 1×10^{-4} displacements per atom (dpa) or less. The NRC approved BWR fluence calculation methodology (ADAMS Accession No. ML032760696), which utilizes a three-dimensional discrete ordinates transport model, shows that ~~there is one steel component (the reactor shield wall interior plate) that exceeds this threshold with a value of~~ 6.03×10^{-4} dpa. Note that while the neutron fluence parameters have a cut off threshold energy, the dpa calculations were performed using all energy groups. The reactor shield wall steel columns are welded to the reactor shield wall interior plate as shown on UFSAR Figure XVI-17 and are exposed to a similar radiation exposure damage level of 6.03×10^{-4} dpa. By comparison, the ring girder on top of the reactor shield wall, has the next highest exposure damage level of 3.81×10^{-5} dpa, which is below the threshold limit. Note that all other steel structural elements have lower exposure damage level than this ring girder. As a result, a NUREG-1509 evaluation was subsequently performed for the reactor shield wall interior plate and the reactor shield wall steel columns.

NUREG-1509 is a resource for addressing the potential effects of irradiation on the steel elements of the reactor shield wall for SLR. Accordingly, a further evaluation of the reactor shield wall interior liner plate and steel columns for radiation induced embrittlement is provided below. In accordance with NUREG-1509, the reduction in fracture toughness assessment of the reactor shield wall steel can be based on a transition temperature analysis. The transition temperature approach is based on the proposition that catastrophic failure by brittle fracture can be avoided by maintaining the normal operating reactor shield wall service temperature above the NDT temperature of the steel. When using the transition temperature to evaluate the reactor shield wall integrity, the NDT temperature at EOL should include the irradiation induced shift.

The reactor shield wall interior plate and steel columns are is made from ASTM A36 steel. Since there is no test data available for NMP1 RPV structural steel components, which could be used to establish the initial NDT, the conservative generic values of NUREG-1509 Tables 4.1 and 4.2 are used. The value of the NDT temperature used in this evaluation is NDT mean plus 1.3 standard deviation. Based on the carbon-manganese (hot rolled) data for ASTM A36 steel, the NDT plus 1.3 standard deviation temperature is 39 degrees F. NMP1 UFSAR design criterion 11 requires an additional 30 degrees F margin to be applied to ferritic materials of the containment. Therefore, the initial NDT temperature used in the NUREG-1509 evaluation is 69 degrees F.

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 NUREG-1509) 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 displacements per atom (dpa) that incorporated the effects of flux and fluence, irradiation temperature, and gamma heating for application to the vessel supports as shown by the upper bound line in NUREG-1509 Figure 3-1. That model included an upper bound transition temperature shift that was adjusted with zero-degree shift at a dpa of 10^{-5} . For this evaluation of the reactor shield wall liner plate and steel columns, the use of the NUREG-1509 trend curve model for NDT shift versus dpa is very conservative for the following reasons: i) there is a very little copper in the ASTM A36 materials, and ii) the ratio of low energy neutrons to fast neutrons in the reactor shield wall is much smaller than that used in the test reactor experiments.

For this further evaluation, fluence calculations were performed to confirm the attenuation effects through the RPV internals, the RPV and outward to the reactor shield wall. The peak fluence at the reactor shield wall interior surface for 67 EFPY equates to 6.03×10^{-4} dpa.

The prediction of the potential irradiation induced NDT shift using the NUREG-1509 Figure 3-1 is described below. The potential irradiation induced NDT shift is a function of the dpa fluence as shown in the figure. The dashed upper bound curve is based on the fit to the experimental test data for reactor vessel carbon steel support materials under low temperature, low flux neutron exposure conditions. For the reactor shield wall with a peak inner diameter surface fluence equating to 6.03×10^{-4} dpa, the corresponding NDT shift is 45 degrees F. Therefore, the total adjusted NDT (EOL NDT) is 114 degrees F. The minimum average bulk average temperature of the Drywell during normal operation is 115 degrees F (UFSAR Section VI E.1.2). Note that plant operating experience has shown that the average reactor shield wall temperatures are at least 15 degrees F higher

(>130 degrees F) than the minimum average bulk drywell temperature. Since the EOL NDT is less than the minimum average bulk average drywell temperature, the reactor shield wall interior plate and steel columns areis adequate through SPEO (67 EFPY).

Welding of the reactor shield wall was performed in accordance with the original design specifications and the construction records that required the use of E-7018 electrodes in accordance with the relevant AWS specifications, and do not incorporate a copper covering on the electrode. The original specification did not specifically add requirements for any additional copper or nickel and there are no chemical measurements for copper or nickel in material receipt records for the well rods used for the NMP1 reactor shield wall beyond the standard material requirements, which do not include requirements for copper or nickel. As a result, the weld materials are conservatively considered to be similar to the ASTM A36 materials, therefore the same conclusions are made regarding the potential effects of irradiation induced embrittlement for the weld materials incorporated into the reactor shield wall as were made regarding the reactor shield wall steel elements.

An evaluation of the potential effects of the irradiation on the steel elements of the reactor shield wall was performed for NMP1, using the projected fluence values for 67 EFPY and an adjusted NDT for the steel materials using the methodology in NUREG-1509. The evaluation demonstrates that there is margin between the normal operating temperature and the ductile-to-brittle fracture mode transition temperature that was adjusted for the potential effects due to irradiation. The evaluation concludes that the potential effects due to irradiation on the steel elements of the reactor shield wall materials, including the welding material, are not significant. As a result, the integrity of the reactor shield wall is assured, and no additional aging management of the reactor shield wall beyond the current Structures Monitoring (B.2.1.~~33~~34) program is necessary for aging effects due to irradiation during the SPEO at NMP1.

In accordance with NUREG-1509 Section 4.3.1, physical examination is essential to reevaluation. The most recent plant operating experience has revealed areas of light corrosion on the reactor shield wall with no sign of pitting. In general, structural steel components including the reactor stabilizers are in good condition with no noted degradation. Accordingly, there is reasonable assurance that a loss (or reduction) of fracture toughness due to irradiation embrittlement will not affect the ability of the reactor shield wall support steel to perform its component intended functions through the SPEO. Since the radiation dose of the reactor shield wall bounds other Class 1 steel structures at NMP1 and the aging effects due to irradiation are not significant for the reactor shield wall, a plant-specific program is not necessary to manage the aging effects due to irradiation of structural steel components.

Additionally, plant operating experience has not identified consequential degradation in the reactor vessel steel supports. A 2019 inspection of the reactor vessel support skirt found the skirt to be in acceptable condition, with no significant degradation identified. The concrete and embedments adjacent to the ring girder were also inspected and determined to be acceptable. While these inspections did not directly examine the ring girder, the absence of degradation in adjacent structural components provides additional confidence that the reactor vessel support skirt and ring girder remain capable of performing their intended functions. Furthermore, the Structures Monitoring (B.2.1.33) program is being enhanced to establish quantitative baseline inspection data against the

acceptance criteria provided in the enhanced program prior to the SPEO. As part of this enhancement, baseline inspections of the ring girder will be completed no later than the last refueling outage prior to the SPEO. Therefore, the integrity of the reactor vessel supports is assured, and no additional aging management of the reactor vessel supports beyond the current ASME Section XI, Subsection IWF (B.2.1.~~3031~~) program and the Structures Monitoring (B.2.1.33) program is necessary for aging effects due to irradiation during the 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 RPV support steel to perform its component intended functions through the SPEO.

SLRA Table 3.5.1, Summary of Aging Management Evaluations for the Structures and Component Supports, page 3.5-77 is revised as shown below:

Table 3.5.1 Summary of Aging Management Evaluations for the Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-066	Groups 1-5, 7, 9: concrete (accessible areas): interior and above-grade exterior	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	AMP XI.S6, "Structures Monitoring"	No	Consistent with NUREG-2191 with exceptions. The Structures Monitoring (B.2.1.33) program will be used to manage cracking, loss of bond, and loss of material (spalling, scaling) of the accessible concrete areas including curbs, above-grade exterior, foundation, interior, near reactor vessel (reactor shield wall) <u>- including inaccessible areas</u>, equipment supports and foundations, hatches/plugs exposed to air - indoor uncontrolled, and air – outdoor environments in the Primary Containment, Reactor Building, Turbine, Waste Handling, and Admin. Structures, and Yard Structures. Exceptions apply to the NUREG-2191 recommendations for the Structures Monitoring (B.2.1.33) program implementation.

SLRA Table 3.5.1, Summary of Aging Management Evaluations for the Structures and Component Supports, page 3.5-96 is revised as shown below:

Table 3.5.1 Summary of Aging Management Evaluations for the Structures and Component Supports

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.5.1-102	Reactor vessel steel structural supports, their assembled components (e.g., reactor vessel steel support skirt assembly; reactor vessel support girders/columns structure; neutron shield tank; reactor vessel support sliding feet assembly; reactor vessel seismic restraints; welds; bolted connections; support anchorage to building structure)	Reduction in fracture toughness, cracking, and/or loss of intended function (mechanical/ structural) due to irradiation induced combined mechanisms	Plant-specific aging management program, or plant-specific enhancements to selected GALL-SLR AMPs (e.g., AMP XI.S3, "ASME Section XI, Subsection IWF," and/or AMP XI.S6, "Structures Monitoring," and if/when necessary*, AMP XI.M1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD) * i.e., when there is evidence of combined aging effects associated with radiation exposure in areas of high tensile stresses	Yes	Consistent with NUREG-2191 <u>with exceptions</u> . The ASME Section XI, Subsection IWF (B.2.1.30) <u>and Structures Monitoring (B.2.1.33) programs</u> will be used to manage reduction in fracture toughness, <u>cracking, and/or loss of intended function</u> of carbon steel supports for ASME Class 1 piping and components (including the reactor vessel to reactor shield wall stabilizers <u>and the reactor vessel support skirt</u>), <u>steel components (ring girder for the reactor vessel support skirt, reactor shield wall steel columns, ring girder at the top of the reactor shield wall and reactor shield wall interior liner)</u> , exposed to air - indoor uncontrolled in Component Supports Commodity <u>and Primary Containment</u> . <u>Exceptions apply to the NUREG-2191 recommendations for the Structures Monitoring (B.2.1.33) program implementation.</u> See Section 3.5.2.2.2.8

SLRA Table 3.5.2-1, Component Supports Commodity Summary of Aging Management Evaluation, page 3.5-98 is revised as shown below:

Table 3.5.2-1 Component Supports Commodity Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
39060	Supports for ASME Class 1 piping and components: stabilizers (reactor vessel to reactor shield wall)	Structural Support	Carbon Steel	Air - Indoor, Uncontrolled	Reduction in Fracture Toughness, <u>i.e., Cracking, and/or Loss of Intended Function</u>	ASME Section XI, Subsection IWF (B.2.1.30)	III.A4.T-36	3.5.1-102	A
34624	Supports for ASME Class 1 piping and components: support members; welds; bolted connections; support anchorage to building structure	Structural Support	Carbon and Low Alloy Steel Bolting	Air - Indoor, Uncontrolled	Loss of Material	ASME Section XI, Subsection IWF (B.2.1.30)	III.B1.1.TP-226	3.5.1-081	A
34623	Supports for ASME Class 1 piping and components: support members; welds; bolted connections; support anchorage to building structure	Structural Support	Carbon and Low Alloy Steel Bolting	Air - Indoor, Uncontrolled	Loss of Preload	ASME Section XI, Subsection IWF (B.2.1.30)	III.B1.1.TP-229	3.5.1-087	A
34620	Supports for ASME Class 1 piping and components: support members; welds; bolted connections; support anchorage to building structure	Structural Support	Carbon Steel	Air - Indoor, Uncontrolled	Loss of Material	ASME Section XI, Subsection IWF (B.2.1.30)	III.B1.1.T-24	3.5.1-091	A
<u>39286</u>	<u>Supports for ASME Class 1 piping and components: support members; welds; bolted connections; support anchorage to building structure</u>	<u>Structural Support</u>	<u>Carbon Steel</u>	<u>Air - Indoor, Uncontrolled</u>	<u>Reduction in Fracture Toughness, i.e., Cracking, and/or Loss of Intended Function</u>	<u>ASME Section XI, Subsection IWF (B.2.1.30)</u>	<u>III.A4.T-36</u>	<u>3.5.1-102</u>	<u>A, 3</u>
35877	Supports for ASME Class 1 piping and components: support members; welds; bolted connections; support anchorage to building structure	Structural Support	High Strength Low Alloy Steel Bolting with Yield Strength of 150 ksi or Greater	Air - Indoor, Uncontrolled	Cracking	ASME Section XI, Subsection IWF (B.2.1.30)	III.B1.1.TP-41	3.5.1-068	A

SLRA Table 3.5.2-1, Component Supports Commodity Summary of Aging Management Evaluation, page 3.5-117, Plant Specific Note 3 is added as shown below:

Table 3.5.2-1 Component Supports Commodity Summary of Aging Management Evaluation

Plant Specific Notes:

1. The stabilizer at the top of the reactor vessel allows for movement and does not use Lubrite or similar material but instead consists of steel to steel connections.
2. Support designs allow for movements and do not use Lubrite or similar material but instead consist of steel to steel or stainless steel to stainless steel connections.
3. The component exposed to the reduction in fracture toughness, i.e., cracking, and/or loss of intended function aging effect is the reactor vessel support skirt.

SLRA Table 3.5.2-2, Primary Containment Summary of Aging Management Evaluation, pages 3.5-119 and 3.5-120 are revised as shown below:

Table 3.5.2-2 Primary Containment Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
39064	Concrete: Interior (inaccessible areas - drywell floor and reactor pedestal)	Direct Flow, Structural Support	Reinforced concrete	Air - Indoor, Uncontrolled	Reduction of Strength; Loss of Mechanical Properties	Structures Monitoring (B.2.1.33)	III.A4.T-35	3.5.1-097	B, 2
35405	Concrete: near reactor vessel (reactor shield wall)	Direct Flow, HELB/MELB Shielding, Shielding, Structural Support	Reinforced concrete	Air - Indoor, Uncontrolled	Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling)	Structures Monitoring (B.2.1.33)	III.A4.TP-26	3.5.1-066	B, <u>9</u>
35406	Concrete: near reactor vessel (reactor shield wall)	Direct Flow, HELB/MELB Shielding, Shielding, Structural Support	Reinforced concrete	Air - Indoor, Uncontrolled	Reduction of Strength; Loss of Mechanical Properties	Structures Monitoring (B.2.1.33)	III.A4.T-35	3.5.1-097	B, <u>2, 9</u>
35435	Equipment supports and foundations	Structural Support	Reinforced concrete	Air - Indoor, Uncontrolled	Cracking, Loss of Bond, and Loss of Material (Spalling, Scaling)	Structures Monitoring (B.2.1.33)	III.A4.TP-26	3.5.1-066	B

SLRA Table 3.5.2-2, Primary Containment Summary of Aging Management Evaluation, page 3.5-127 is revised as shown below:

Table 3.5.2-2 Primary Containment Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
35592	Steel Components: Ring Girder for the Reactor Vessel Support Skirt	Structural Support	Carbon Steel	Air - Indoor, Uncontrolled	Loss of Material	Structures Monitoring (B.2.1.33)	III.A4.TP-302	3.5.1-077	B
<u>39287</u>	<u>Steel Components: Ring Girder for the Reactor Vessel Support Skirt</u>	<u>Structural Support</u>	<u>Carbon Steel</u>	<u>Air - Indoor, Uncontrolled</u>	<u>Reduction in Fracture Toughness, i.e., Cracking, and/or Loss of Intended Function</u>	<u>Structures Monitoring (B.2.1.33)</u>	<u>III.A4.T-36</u>	<u>3.5.1-102</u>	<u>B</u>
39066	Steel Components: Stabilizers (truss arrangements between reactor shield wall to drywell)	Structural Support	Carbon Steel	Air - Indoor, Uncontrolled	Loss of Material	Structures Monitoring (B.2.1.33)	III.A4.TP-302	3.5.1-077	B
35589	Steel Components: structural steel	Structural Support	Carbon Steel	Air - Indoor, Uncontrolled	Loss of Material	Structures Monitoring (B.2.1.33)	III.A4.TP-302	3.5.1-077	B
<u>39288</u>	<u>Steel Components: structural steel</u>	<u>Structural Support</u>	<u>Carbon Steel</u>	<u>Air - Indoor, Uncontrolled</u>	<u>Reduction in Fracture Toughness, i.e., Cracking, and/or Loss of Intended Function</u>	<u>Structures Monitoring (B.2.1.33)</u>	<u>III.A4.T-36</u>	<u>3.5.1-102</u>	<u>B, 7</u>
35056	Steel Components: structural steel (refueling seal platform)	Structural Support, Water Retaining Boundary	Carbon Steel	Air - Indoor, Uncontrolled	Loss of Material	Structures Monitoring (B.2.1.33)	III.A4.TP-302	3.5.1-077	B, 1
36470	Steel Components: structural steel (refueling seal platform)	Structural Support, Water Retaining Boundary	Stainless Steel	Air - Indoor, Uncontrolled	Cracking, Loss of Material	Structures Monitoring (B.2.1.33)	III.B5.T-37b	3.5.1-100	D, 1

SLRA Table 3.5.2-2, Primary Containment Summary of Aging Management Evaluation, page 3.5-129 is revised as shown below:

Table 3.5.2-2 Primary Containment Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
36028	Steel Elements: liner, liner anchors, integral attachments (reactor pedestal and reactor shield wall)	Direct Flow, Structural Support	Carbon Steel	Air - Indoor, Uncontrolled	Loss of Material	Structures Monitoring (B.2.1.33)	III.A4.TP-302	3.5.1-077	B
39291	Steel Elements: liner, liner anchors, integral attachments (reactor pedestal and reactor shield wall)	Direct Flow, Structural Support	Carbon Steel	Air - Indoor, Uncontrolled	Reduction in Fracture Toughness, i.e., Cracking, and/or Loss of Intended Function	Structures Monitoring (B.2.1.33)	III.A4.T-36	3.5.1-102	B, 8
35053	Steel elements: Refueling Bellows assemblies	Expansion/Separation, Water Retaining Boundary	Stainless Steel	Air - Indoor, Uncontrolled	Cracking, Loss of Material	One-Time Inspection (B.2.1.20)	III.B5.T-37a	3.5.1-100	D, 1, 5

SLRA Table 3.5.2-2, Primary Containment Summary of Aging Management Evaluation, page 3.5-133, Plant Specific Notes 7, 8, and 9 are added as shown below:

Table 3.5.2-2 Primary Containment Summary of Aging Management Evaluation

Plant Specific Notes:

5. This component is not in low-flow or stagnant areas since the area is drained upon completion of the refueling outage and heat from the reactor pressure vessel causes rapid evaporation of any moisture remaining in the bellows. Plant-specific operating experience reviews have not identified any failures or leakage from the refueling bellows.
6. This component is constructed of 6061-T6 aluminum alloy, which is not susceptible to stress corrosion cracking.
7. The components exposed to the reduction in fracture toughness, i.e., cracking, and/or loss of intended function aging effect are the reactor shield wall steel columns and the ring girder at the top of the reactor shield wall.
8. The component exposed to the reduction in fracture toughness, i.e., cracking, and/or loss of intended function aging effect is the reactor shield wall interior liner.
9. This component addresses the aging effects of the inaccessible structural and non-structural concrete in the reactor shield wall.

Change #7 – Clarification Regarding Use of NRC Approved DORT Methodology to Perform Fluence Calculations

Affected SLRA Sections: Sections 4.2.1 and 4.8

SLRA Page Numbers: 4.2-4 and 4.8-4

Description of Change:

SLRA Sections 4.2.1 and 4.8 are being revised to clarify how fluence calculations were performed using an NRC approved Two-Dimensional Discrete Ordinates Code (DORT) methodology and provide an appropriate reference. In addition, SLRA Section 4.2.1 is being revised to state that the methodology conforms with NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

This change is in response to a Breakout Question (i.e., TRP 142.1-1).

Accordingly, SLRA Sections 4.2.1 and 4.8 are revised.

SLRA Section 4.2.1, Neutron Fluence Analyses, page 4.2-4 is revised as shown below.

4.2.1 NEUTRON FLUENCE ANALYSES

Neutron fluence is the term used to represent the cumulative number of neutrons per square centimeter (flux) that contact the RPV shell and its internal components over a given period of time. The fluence projections that quantify the number of neutrons that contact these surfaces have been used as inputs to the neutron embrittlement analyses that evaluate the loss of fracture toughness aging effect resulting from neutron fluence.

The transport calculations for Nine Mile Point Nuclear Station, Unit 1 (NMP1) were carried out in R- θ and R-Z geometry using the Two-Dimensional Discrete Ordinates Code (DORT) and the BUGLE-96 cross-section library. The DORT code is an update of the DOT code which has been in use for this type of analysis for many years. The BUGLE-96 library is a 47-energy group ENDF/B-VI based data set produced specifically for light water reactor applications (an update of the earlier SAILOR library).

The NMP1 RPV fluence evaluation for SPEO was generated using methods consistent with the guidance in NRC RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The fluence methodology for NMP1 was approved on a plant-specific basis and is documented in an NRC Safety Evaluation Report, dated October 27, 2003 (Reference 4.8.57). There are no deviations made in the development of the SLRA neutron transport model from the previously approved NRC methodology.

Analysis above and below the active fuel length was carried out using the Three-Dimensional Transport Code (TORT). The TORT code uses the same methodology as the DORT code except for the extension into the third dimension. These transport codes have been used in developing the 80-year fluence projections and associated RPV embrittlement analyses.

SLRA Section 4.8, References, page 4.8-4, Reference 4.8.57 is added as shown below.

4.8 REFERENCES

4.8.56 *NRC letter dated March 31, 1995, Nine Mile Point Nuclear Station Unit No. 1 (NMP1), Evaluation of Core Shroud Stabilizer Design (TAC NO. M91273) (ML17059A774) (Section 4.7.9)*

4.8.57 *NRC Safety Evaluation Report Related to Amendment No. 183, Nine Mile Point Nuclear Station, Unit 1, dated October 27, 2003 (ML032760696) (Section 4.2.1)*

Change #8 – Clarification on Environmentally Assisted Fatigue Consideration for Components Located in the Dry Steam Environment

Affected SLRA Section: Section 4.3.5

SLRA Page Number: 4.3-19

Description of Change:

SLRA Section 4.3.5 is being revised to provide clarification on the methodology and assumptions used in the environmentally assisted fatigue (EAF) screening methodology regarding components located in the steam environment during normal operation.

This change is in response to a Breakout Question (i.e., TRP 143.5-2).

Accordingly, SLRA Section 4.3.5 is revised.

SLRA Section 4.3.5, Environmentally Assisted Fatigue Analysis, page 4.3-19 is revised as shown below.

4.3.5 ENVIRONMENTALLY ASSISTED FATIGUE ANALYSIS

TLAA Evaluation:

Environmental Fatigue Screening

The environmental fatigue screening calculation methodology is described below.

- 1) All reactor coolant pressure boundary component locations that have documented cumulative usage factors in the NMP1 CLB were identified.
- 2) Locations were excluded if not in contact with reactor water.
 - a) Final steam quality leaving the steam dryer is greater than 99.9 percent at normal operating conditions. Therefore, components located downstream of the steam dryer are located in a dry steam environment and are not considered wetted. However, two component locations in the dry steam environment were evaluated considering environmental effects using the following conservative assumptions. The location "RPV closure region, limiting vessel location" considered environmental effects during flood-up conditions and the "emergency condenser nozzle" location considered environmental effects assuming the location is wetted with condensate during normal operation. Note per the methodology steps below, the "RPV closure region, limiting vessel location" did not "Screen In" as the 80-year screening CUF_{en} value was less than 1.0 and the "emergency condenser nozzle" location did not "Screen In" as this location was bounded by the "Feedwater Nozzle Blend Radius" (SLRA Table 4.3.1-2, component number 5) location in the same thermal zone.

Change #9 – Clarification on Core Shroud Tie-Rod X-750 Inconel Component Service Life Enhancement

Affected SLRA Sections: Section 4.7.9; Appendix A, Section A.2.1.7; Appendix A, Section A.4.7.9; Appendix A, Table A.5; Appendix B, Section B.2.1.7; and Appendix C, BWRVIP-315-A

SLRA Page Numbers: 4.7-22, 4.7-23, A-14, A-75, A-76, A-77, B-48, B-49, and C-26

Description of Change:

BWR Vessel Internals program Enhancement #2 associated with developing a justification for the service life of the core shroud tie-rod X-750 subcomponents through the subsequent period of extended operation is being revised to provide clarification on the approved NRC guidance that may be used to provide reasonable assurance of component functionality through the subsequent period of extended operation. In addition, references to NRC approvals of the original core shroud repair and subsequent core shroud repair modification are being provided in the revised enhancement. Corresponding changes are being made to Section 4.7.9, Section A.4.7.9, and Appendix C for consistency with the changes to BWR Vessel Internals program Enhancement #2.

This change is in response to a BWR Vessel Internals program Breakout Question (i.e., TRP 9-1).

Accordingly, SLRA Section 4.7.9; Appendix A, Section A.2.1.7; Appendix A, Section A.4.7.9; Appendix A, Table A.5; Appendix B, Section B.2.1.7; and Appendix C, BWRVIP-315-A are revised.

SLRA Section 4.7.9, Core Shroud Tie Rod X-750 Inconel Component Service Life, pages 4.7-22 and 4.7-23 are revised as shown below.

4.7.9 CORE SHROUD TIE ROD X-750 INCONEL COMPONENT SERVICE LIFE

TLAA Evaluation:

The NMP1 BWR Vessel Internals (B.2.1.7) program specifies that materials used for repairs or replacements meet the guidelines of BWRVIP-84, Revision 2-A, as modified by BWRVIP Letter 2011-011, "Guidance for Evaluating Repair Hardware with X-750 Components."

As a result, the NMP1 BWR Vessel Internals (B.2.1.7) program will be enhanced to develop a justification for the service life through the SPEO, using NRC approved BWRVIP guidelines consistent with BWRVIP-94, Revision 4.7. This which may include supplemental inspections, inspection frequencies frequency changes, engineering changes, and technical evaluations to validate that the core shroud tie-rod X-750 subcomponents can perform their intended function through the SPEO. Specifically, the NRC approved guidance in BWRVIP-138, Revision 2-A, Section 5 and BWRVIP-84, Revision 2-A, Section B.3.1 as modified by BWRVIP Letter 2011-011, may be used to ensure the X-750 subcomponents can perform their intended function through the SPEO. The original core shroud repair and subsequent core shroud repair modification were reviewed and found acceptable by the NRC (References 4.8.56 and 4.8.31, respectively) as an alternative to the ASME Code pursuant to 10 CFR 50.55a(a)(3)(i). Any proposed engineering changes to the core shroud tie-rod licensing basis will be evaluated by CEG in accordance with 10 CFR 50.59 to determine if prior NRC review and approval is required. This enhancement will be implemented no later than six months prior to the SPEO. Therefore, this TLAA will be managed by the NMP1 BWR Vessel Internals (B.2.1.7) program through the SPEO.

SLRA Appendix A, Section A.2.1.7, BWR Vessel Internals, page A-14, Enhancement #2 is revised as shown below.

A.2.1.7 BWR Vessel Internals

The BWR Vessel Internals aging management program will be enhanced to:

2. Develop a justification for the service life of the core shroud tie-rod X-750 subcomponents through the subsequent period of extended operation, using NRC approved BWRVIP guidelines consistent with BWRVIP-94, Revision 4.7. This which may include supplemental inspections, inspection frequencies frequency changes, engineering changes, and technical evaluations to validate that the core shroud tie-rod X-750 subcomponents can perform their intended function through the subsequent period of extended operation. Specifically, the NRC approved guidance in BWRVIP-138, Revision 2-A, Section 5 and BWRVIP-84, Revision 2-A, Section B.3.1 as modified by BWRVIP Letter 2011-011, may be used to ensure the X-750 subcomponents can perform their intended function through the subsequent period of extended operation. The original core shroud repair and subsequent core shroud repair modification were reviewed and found acceptable by the NRC (ADAMS Accession Nos. ML17059A774 and ML072600195, respectively) as an alternative to the ASME Code pursuant to 10 CFR 50.55a(a)(3)(i). Any proposed engineering changes to the core shroud tie-rod licensing basis will be evaluated by CEG in accordance with 10 CFR 50.59 to determine if prior NRC review and approval is required.

The program will be enhanced no later than six months prior to the subsequent period of extended operation.

SLRA Appendix A, Section A.4.7.9, Core Shroud Tie Rod X-750 Inconel Component Service Life, page A-75, concluding paragraph is revised as shown below.

A.4.7.9 Core Shroud Tie Rod X-750 Inconel Component Service Life

As a result, the NMP1 BWR Vessel Internals (A.2.1.7) program will be enhanced to develop a justification for a the service life through the subsequent period of extended operation, using NRC approved BWRVIP guidelines consistent with BWRVIP-94, Revision 4.7. This which may include supplemental inspections, inspection frequencies frequency changes, engineering changes, and technical evaluations to validate that the core shroud tie-rod X-750 subcomponents can perform their intended function through the subsequent period of extended operation. Specifically, the NRC approved guidance in BWRVIP-138, Revision 2-A, Section 5 and BWRVIP-84, Revision 2-A, Section B.3.1 as modified by BWRVIP Letter 2011-011, may be used to ensure the X-750 subcomponents can perform their intended function through the subsequent period of extended operation. The original core shroud repair and subsequent core shroud repair modification were reviewed and found acceptable by the NRC (ADAMS Accession Nos. ML17059A774 and ML072600195, respectively) as an alternative to the ASME Code pursuant to 10 CFR 50.55a(a)(3)(i). Any proposed engineering changes to the core shroud tie-rod licensing basis will be evaluated by CEG in accordance with 10 CFR 50.59 to determine if prior NRC review and approval is required. This enhancement will be implemented no later than six months prior to the subsequent period of extended operation. Therefore, this TLAA will be managed by the NMP1 BWR Vessel Internals (A.2.1.7) program through the subsequent period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, Commitment 7, BWR Vessel Internals, pages A-76 and A-77, Enhancement #2 is revised as shown below.

TABLE A.5 – SUBSEQUENT LICENSE RENEWAL COMMITMENT LIST				
NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
7	BWR Vessel Internals	BWR Vessel Internals is an existing program that will be enhanced to: 2. Develop a justification for the service life of the core shroud tie-rod X-750 subcomponents through the subsequent period of extended operation, <u>using NRC approved BWRVIP guidelines</u> consistent with BWRVIP-94, Revision 4.7. <u>This which may include supplemental inspections, inspection frequencies frequency changes, engineering changes,</u> and technical evaluations to validate that the core shroud tie-rod X-750 subcomponents can perform their intended function through the subsequent period of extended operation. <u>Specifically, the NRC approved guidance in BWRVIP-138, Revision 2-A, Section 5 and BWRVIP-84, Revision 2-A, Section B.3.1 as modified by BWRVIP Letter 2011-011, may be used to ensure the X-750 subcomponents can perform their intended function through the subsequent period of extended operation. The original core shroud repair and subsequent core shroud repair modification were reviewed and found acceptable by the NRC (ADAMS Accession Nos. ML17059A774 and ML072600195, respectively) as an alternative to the ASME Code pursuant to 10 CFR 50.55a(a)(3)(i). Any proposed engineering changes to the core shroud tie-rod licensing basis will be evaluated by CEG in accordance with 10 CFR 50.59 to determine if prior NRC review and approval is required.</u>	Program will be enhanced no later than six months prior to the subsequent period of extended operation.	Section A.2.1.7

SLRA Appendix B, Section B.2.1.7, BWR Vessel Internals, pages B-48 and B-49, Enhancement #2 is revised as shown below.

B.2.1.7 BWR Vessel Internals

Enhancements

2. Develop a justification for the service life of the core shroud tie-rod X-750 subcomponents through the subsequent period of extended operation, using NRC approved BWRVIP guidelines consistent with BWRVIP-94, Revision 4, This which may include supplemental inspections, inspection frequencies frequency changes, engineering changes, and technical evaluations to validate that the core shroud tie-rod X-750 subcomponents can perform their intended function through the subsequent period of extended operation. Specifically, the NRC approved guidance in BWRVIP-138, Revision 2-A, Section 5 and BWRVIP-84, Revision 2-A, Section B.3.1 as modified by BWRVIP Letter 2011-011, may be used to ensure the X-750 subcomponents can perform their intended function through the subsequent period of extended operation. The original core shroud repair and subsequent core shroud repair modification were reviewed and found acceptable by the NRC (ADAMS Accession Nos. ML17059A774 and ML072600195, respectively) as an alternative to the ASME Code pursuant to 10 CFR 50.55a(a)(3)(i). Any proposed engineering changes to the core shroud tie-rod licensing basis will be evaluated by CEG in accordance with 10 CFR 50.59 to determine if prior NRC review and approval is required. **Program Elements Affected: Scope of Program (Element 1), Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4), and Corrective Actions (Element 7)**

The program will be enhanced no later than six months prior to the subsequent period of extended operation.

SLRA Appendix C, BWRVIP-315-A, Limitation 7, page C-26 is revised as shown below.

APPENDIX C

RESPONSE TO BWRVIP LICENSE RENEWAL APPLICANT ACTION ITEMS

BWRVIP-315-A, “Reactor Internals Aging Management Evaluation for Extended Operations”	
Action Item Description	NMP1 Response
BWRVIP-315-A (Condition 5) Applicants for renewed operating licenses extending beyond 60 years must describe in the application how they meet the seven BWRVIP-defined limitations listed in BWRVIP-315-A, Section 4.5.1. <u>Limitation 7 – Core Shroud Repair Hardware Aging Management:</u> Core shroud tie rod repairs require plant-specific evaluation. Inspections should, as a minimum, meet the requirements listed in BWRVIP-76, Rev. 1-A. However, additional evaluations must be performed to address aging management associated with operation beyond the original repair hardware service life specified by the designer. See Section 4.3.13 for additional discussion. Also see Appendix B, Section B.5.	NMP1 installed a core shroud tie-rod repair during the refueling outage in 1995. Subsequent tie-rod inspections meet the requirements of BWRVIP-76, Revision 1-A and vendor repair design requirements. In response to the tie-rod cracking observed at Hatch Unit 1, NMP1 replaced all existing tie rod upper supports and tie rod nuts during the refueling outage in 2007. A new baseline exam was performed during the refueling outage in 2009. Re-inspection of all four tie-rods and their upper supports was performed during the refueling outage in 2019. No indications of cracking were found. NMP1 will continue inspecting the tie-rod hardware and upper supports per BWRVIP-76, Revision 1-A and vendor repair design requirements. <u>Additional evaluations will be performed to address the potential for IGSCC initiation beyond the original 40-year service life of the core shroud tie-rod X-750 subcomponents. The core shroud tie-rod X-750 Inconel component service life has been identified as a TLAA that is evaluated in SLRA Section 4.7.9.</u>

BWRVIP-315-A, “Reactor Internals Aging Management Evaluation for Extended Operations”	
Action Item Description	NMP1 Response
	NMP1 plant specific fluence projections were performed for the NMP1 core shroud tie-rod repair assemblies. The projected fluence at the end of the subsequent period of extended operation for the core shroud tie-rod repair hardware is below the vendor design requirement. The Core Shroud Stabilizer Assembly Preload Relaxation Analysis has been identified as a TLAA that is evaluated in SLRA Section 4.2.7. No further actions are required.

Change #10 – Revisions to Turbine Building HVAC System and Reactor Building HVAC System Aging Management Evaluations to Include Air, Outdoor (External) Environment for Galvanized Steel Ducting and Components

Affected SLRA Sections: Section 3.3.2.1.14 and Tables 3.3.2-14 and 3.3.2-21

SLRA Page Numbers: 3.3-21, 3.3-269, and 3.3-330

Description of Change:

SLRA Table 3.3.2-14, Reactor Building HVAC System – Summary of Aging Management Evaluation, is being revised to include the Air, Outdoor (External) environment for galvanized steel ducting and components. The Reactor Building HVAC License Renewal System includes portions of the stack breaching, which are located outside.

SLRA Table 3.3.2-21, Turbine Building HVAC System – Summary of Aging Management Evaluation, is being revised to include the Air, Outdoor (External) environment for galvanized steel ducting and components. The Turbine Building HVAC License Renewal system includes outdoor backdraft dampers that interface with the stack breaching.

SLRA Section 3.3.2.1.14 is being revised for consistency with the Reactor Building HVAC System aging management review changes discussed above.

This change is in response to a Breakout Question (Aux System-5).

Accordingly, SLRA Section 3.3.2.1.14 and Tables 3.3.2-14 and 3.3.2-21 are revised.

SLRA Section 3.3.2.1.14, Reactor Building HVAC System, page 3.3-21 is revised as shown below.

3.3.2.1.14 Reactor Building HVAC System

Materials

The materials of construction for the Reactor Building HVAC System components are:

- Carbon and Low Alloy Steel Bolting
- Carbon Steel
- Cast Iron
- Copper Alloy with less than 15% Zinc
- Elastomers
- Galvanized Bolting
- Galvanized Steel

Environments

The Reactor Building HVAC System components are exposed to the following environments:

- Air - Indoor, Uncontrolled
- Air - Outdoor
- Condensation
- Raw Water

Aging Effect Requiring Management

The following aging effects associated with the Reactor Building HVAC System components require management:

- Flow Blockage
- Hardening or Loss of Strength Due to Elastomeric Degradation
- Long-Term Loss of Material
- Loss of Material
- Loss of Material Due to Wear
- Loss of Preload

SLRA Table 3.3.2-14, Reactor Building HVAC System Summary of Aging Management Evaluation, page 3.3-269 is revised as shown below.

Table 3.3.2-14 Reactor Building HVAC System Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
33480	Ducting and Components	Pressure Boundary	Carbon Steel	Air - Indoor, Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VII.I.A-77	3.3.1-078	A
33481	Ducting and Components	Pressure Boundary	Carbon Steel	Condensation (Internal)	Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.24)	VII.F2.A-08	3.3.1-090	A
33363	Ducting and Components	Pressure Boundary	Galvanized Steel	Air - Indoor, Uncontrolled (External)	None	None	VII.J.AP-13	3.3.1-116	C
<u>39294</u>	<u>Ducting and Components</u>	<u>Pressure Boundary</u>	<u>Galvanized Steel</u>	<u>Air - Outdoor (External)</u>	<u>Loss of Material</u>	<u>External Surfaces Monitoring of Mechanical Components (B.2.1.23)</u>	<u>VII.I.A-77</u>	<u>3.3.1-078</u>	<u>A</u>
33364	Ducting and Components	Pressure Boundary	Galvanized Steel	Condensation (Internal)	Loss of Material	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components (B.2.1.24)	VII.F2.A-08	3.3.1-090	A

SLRA Table 3.3.2-21, Turbine Building HVAC System Summary of Aging Management Evaluation, page 3.3-330 is revised as shown below.

Table 3.3.2-21 Turbine Building HVAC System Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
33464	Ducting and Components	Pressure Boundary	Carbon Steel (with Internal Coatings)	Air - Indoor, Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VII.I.A-77	3.3.1-078	A
33462	Ducting and Components	Pressure Boundary	Carbon Steel (with Internal Coatings)	Condensation (Internal)	Loss of Coating or Lining Integrity	Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B.2.1.28)	V.D2.E-401	3.2.1-072	B
33463	Ducting and Components	Pressure Boundary	Carbon Steel (with Internal Coatings)	Condensation (Internal)	Loss of Material	Internal Coatings/Linings for In-Scope Piping, Piping Components, Heat Exchangers, and Tanks (B.2.1.28)	V.D2.E-414	3.2.1-073	B
33176	Ducting and Components	Pressure Boundary	Galvanized Steel	Air - Indoor, Uncontrolled (External)	None	None	VII.J.AP-13	3.3.1-116	C
<u>39293</u>	<u>Ducting and Components</u>	<u>Pressure Boundary</u>	<u>Galvanized Steel</u>	<u>Air – Outdoor (External)</u>	<u>Loss of Material</u>	<u>External Surfaces Monitoring of Mechanical Components (B.2.1.23)</u>	<u>VII.I.A-77</u>	<u>3.3.1-078</u>	<u>A</u>

Change #11 – Clarification on Average Temperature Approach and Feedwater Nozzle Inspections

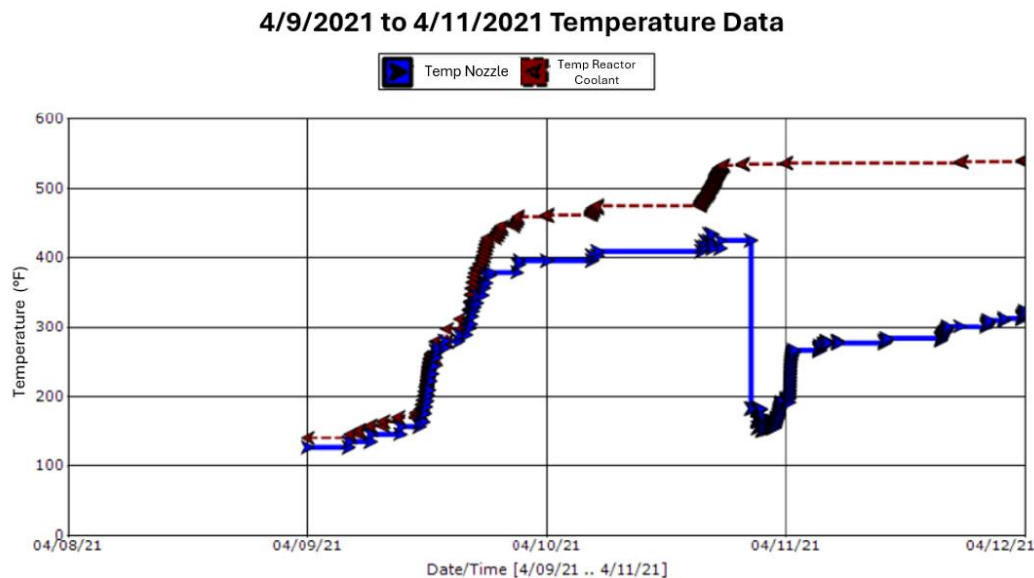
Affected SLRA Sections: SLRA Table of Contents and Sections 4.3.5 and 4.7.3

SLRA Page Numbers: 13, 4.3-21 and 4.7-9

Description of Change:

SLRA Section 4.3.5 is being revised to provide clarification on the average temperature approach. Confirmation of the conservatism of the average temperature approach is provided by a NMP1 specific demonstration that evaluated a transient for the feedwater nozzles using the average temperature approach and the modified rate approach. The transient occurred at NMP1 in April 2021, during an RPV startup from a refueling outage. As shown in the figure below, the feedwater nozzle heats up with the reactor and comes back down in temperature with increasing feedwater flow and then back up with feedwater heating. This actual (plant data) transient is a complex thermal transient. Note, the blue line in this figure shows the temperature trend for the feedwater nozzle while the brown line shows the reactor temperature.

Figure – Feedwater Nozzle RPV Startup Transient 4/9/2021 to 4/11/2021 Temperature Data



In addition, SLRA Section 4.7.3 is being revised to clarify feedwater nozzle blend radius and safe end inspection results.

These changes are in response to a Breakout Question (i.e., TRP 143.5-4).

Accordingly, the SLRA Table of Contents and Sections 4.3.5 and 4.7.3 are revised.

SLRA Table of Contents, List of Tables, page 13 is revised as shown below.

List of Tables

Table 4.2.2-5	67 EFPY EMA for NMP1 Plate Material (Heat No. P2091-2), NMP1 Plate G-307-10, BWR/2 Plate.....	4.2-14
Table 4.2.2-6	67 EFPY EMA for NMP1 Plate Material (Heat No. P2112-1), NMP1 Plate G-8-1, BWR/2 Plate.....	4.2-15
Table 4.2.2-7	67 EFPY EMA for NMP1 Plate Material (Heat No. P2106-1), NMP1 Plate G-8-2, BWR/2 Plate.....	4.2-16
Table 4.2.2-8	67 EFPY EMA for NMP1 Plate Material (Heat No. P2196-2), NMP1 Plate G-9-R1, BWR/2 Plate.....	4.2-17
Table 4.2.2-9	67 EFPY EMA for NMP1 Plate Material (Heat No. P2161-2), NMP1 Plate G-8-9, BWR/2 Plate.....	4.2-18
Table 4.2.2-10	67 EFPY EMA for NMP1 Weld Material (Heat No. 1248), NMP1 Weld 1-572, BWR/2-6 Weld	4.2-19
Table 4.2.3-1	NMP1 - 67 EFPY Adjusted Reference Temperature (ART) Values for Extended Beltline and Beltline Plates, Axial and Circumferential Welds, and Integrated Surveillance Program (ISP) Plates and Welds	4.2-22
Table 4.3.1-1	NMP1 - 80-Year Transient Occurrence Projections	4.3-5
Table 4.3.1-2	NMP1 - Limiting Locations and 80-Year Projected CUF and CUF _{en} Values	4.3-8
Table 4.3.4-1	Stress Range Reduction Factors.....	4.3-14
Table 4.3.4-2	80-Year Transient Occurrence Projections for Class 2, Class 3, and ANSI B31.1 Piping Systems Affected by Transients Other Than RCS Transients	4.3-17
<u>Table 4.3.5-1</u>	<u>NMP1 Specific F_{en} Determination Using the Average Temperature Approach and Modified Rate Approach Comparison</u>	<u>4.3-21</u>
Table 4.7.8-1	Nine Mile Point Nuclear Station, Unit 1 Reactor Building Crane Load Cycles.....	4.7-19
Table 4.7.8-2	Nine Mile Point Nuclear Station, Unit 1 Turbine Building Crane Load Cycles.....	4.7-20
Table 4.7.8-3	Nine Mile Point Nuclear Station, Unit 1 Screenhouse Building Crane Load Cycles.....	4.7-21

SLRA Section 4.3.5, Environmentally Assisted Fatigue Analysis, page 4.3-21 is revised as shown below. This SLRA markup includes changes submitted in Supplement 1, dated July 30, 2026, which clarified the environmental fatigue analysis conservatisms via Change #18 and is duplicated here as an aid to the reviewer.

4.3.5 ENVIRONMENTALLY ASSISTED FATIGUE ANALYSIS

Environmental Fatigue Analyses

Detailed EAF evaluations were performed on all "Screened In" component locations as follows.

Eighty-year fatigue usage values were calculated using ASME Code methodology (N-415.2 or NB-3222.4(e)) except that the applicable fatigue curves from NUREG/CR-6909, Revision 1 were used in addition to the applicable ASME Code fatigue curves. Projected 80-year usage values adjusted for environmental fatigue (CUF_{en}) were calculated by multiplying the projected 80-year usage values (CUF) by the environmental fatigue correction factors (F_{en}). The 80-year projected transient occurrences shown in SLRA Table 4.3.1-1 were used as inputs in the environmental fatigue analyses. ~~An average temperature value approach was used in the detailed EAF analysis.~~

Conservatism in the F_{en} determination of the NMP1 detailed EAF analysis included the use of the maximum sulfur content for carbon and low alloy steels and all materials used the minimum strain rate provided by the F_{en} equations of NUREG/CR-6909, Revision 1, Appendix A.

Conservatism was reduced in the detailed analysis by the temperatures used, which is summarized as follows.

The screening analysis assumed bounding screening F_{en} values which are based on the maximum specified transient temperature from the applicable thermal zone. In contrast, for the detailed evaluation, the calculation developed individual F_{en} values for each specified transient. The screening evaluations multiplied the 40 or 60 year usage values by conservative bounding screening F_{en} values. Since usage values are the ratio of the number of assumed occurrences divided by the number of allowable occurrences, the multiplication by a conservative F_{en} value based on the maximum operating temperature effectively assumes that all the specified transient profiles are assumed to be equal to the maximum peak temperature on the entire thermal cycle diagram. However, the ASME Section III and NUREG/CR-6909, Revision 1 methodology allows for the calculation of transient specific CUF_{en} values for each transient or load pair, based on the number of specified transient occurrences for the individual transient and an F_{en} value corresponding to the specified temperature profile for each transient. The calculated CUF_{en} values for each transient are then added together into a total CUF_{en} value for the component. In addition, the ASME Section III and NUREG/CR-6909, Revision 1 methodology allows for the pairing of transients into load pairs to calculate usage values. This methodology allows for finer granularity in calculating a total CUF_{en} value.

An average temperature value approach was used in the detailed EAF analysis. The average temperature is calculated using the transient maximum temperature and the greater of the transient specified minimum temperature or the threshold temperature for the F_{en} for each material type, as described in NUREG/CR-6909, Revision 1. The average temperature approach was used for simple and complex transients. Since multiple increasing and decreasing temperature excursions can result in multiple stress peaks and valleys, the complex transients were split into multiple simple transients in the detailed environmental fatigue analyses.

An NMP1 specific analysis was performed to demonstrate that the conservatism of the average temperature approach is comparable to, or greater than, that of the modified rate approach described in the NUREG/CR-6909, Revision 1, Section 4.4 using data from an actual complex transient associated with the feedwater nozzle safe end location (SLRA Table 4.3.1-2, component number 6), which is monitored in SI:FatiguePro™ using the stress-based fatigue monitoring methodology. The SI:FatiguePro™ stress-based fatigue monitoring methodology calculates F_{en} values based on the modified rate approach. The transient occurred at NMP1 in April 2021, during a RPV startup from a refueling outage and is considered a complex thermal transient. The NMP1 specific demonstration results showed that the evaluated transient resulted in a CUF value of 0.000003, which corresponds to a CUF_{en} value of 0.000006 using the F_{en} value of 2.01 determined by the modified rate approach and a CUF_{en} value of 0.000096 using the F_{en} value of 3.2 determined by the average temperature approach. The results summarized in SLRA Table 4.3.5-1 show that the F_{en} value calculated using the average temperature approach is greater than the F_{en} value calculated by SI:FatiguePro™ using the modified rate approach. The NMP1 specific demonstration provides additional confirmation that the average temperature approach is conservative when calculating the F_{en} and CUF_{en} values for complex thermal transients at NMP1.

Table 4.3.5-1					
NMP1 Specific F_{en} Determination Using the Average Temperature Approach and Modified Rate Approach Comparison					
<u>Transient Date</u>	<u>Transient Maximum Temperature (°F)</u>	<u>Transient Minimum Temperature (°F)</u>	<u>Transient Average Temperature (°F)</u>	<u>Calculated FatiguePro™ F_{en} Value Based on Modified Rate Approach</u>	<u>Calculated F_{en} Value Based on Average Temperature Approach</u>
<u>4/9/2021 through 4/11/2021</u>	<u>434</u>	<u>302 (Note 1)</u>	<u>368</u>	<u>2.01</u>	<u>3.2</u>

NOTES:

Note 1: Because the transient minimum temperature is below the minimum allowable temperature for carbon steel under NUREG/CR-6909, Revision 1, a value of 302 °F is used instead.

All projected 80-year usage values (CUF) and projected 80-year usage values adjusted for environmental fatigue (CUF_{en}) are documented in SLRA Table 4.3.1-2.

SLRA Section 4.7.3, Generic Letter 81-11, "Crack Growth Analysis to Demonstrate Conformance to the Intent of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", page 4.7-9, last paragraph is revised as shown below.

4.7.3 GENERIC LETTER 81-11, "CRACK GROWTH ANALYSIS TO DEMONSTRATE CONFORMANCE TO THE INTENT OF NUREG-0619, "BWR FEEDWATER NOZZLE AND CONTROL ROD DRIVE RETURN LINE NOZZLE CRACKING""

TLAA Evaluation:

Periodic examinations of the FW and CRDRL nozzles, performed under the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.1.1) program, provide confirmation that the cracking due to rapid thermal cycling fatigue and low cycle fatigue is not occurring. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.2.1.1) program is credited for managing rapid thermal cycling fatigue through the subsequent period of extended operation (SPEO).

Examinations performed under the ISI program for the FW nozzles and the CRDRL nozzle blend radius and safe-end locations in the fourth and fifth ISI intervals have discovered no reportable indications.

Change #12 – Table 4.3.1-2 Component Location Clarifications and Addition of an Operating Experience Example Related to the Inspection of FW/RWCU Mixing Region

Affected SLRA Sections: Table 4.3.1-2 and Appendix B, Section B.3.1.1

SLRA Page Numbers: 4.3-8, 4.3-9, and B-238

Description of Change:

SLRA Table 4.3.1-2 is being revised to clarify the location descriptions of component number 9 and component number 15. In addition, SLRA Appendix B, Section B.3.1.1 is being revised to add an Operating Experience example on the inspection of the FW/RWCU mixing region which supports the validity of the environmentally assisted fatigue analysis discussed in SLRA Section 4.3.5.

These changes are in response to a Breakout Question (i.e., TRP 143.5-5).

Accordingly, SLRA Table 4.3.1-2 and Appendix B, Section B.3.1.1 are revised.

SLRA Table 4.3.1-2, NMP1 – Limiting Locations and 80-Year Projected CUF and CUF_{en} Values, pages 4.3-8 and 4.3-9 are revised to clarify Note 7 and add Note 8 as shown below. This SLRA markup includes changes submitted in Supplement 1, dated July 30, 2026, which added Note 7 via Change #6 and is duplicated here as an aid to the reviewer.

Table 4.3.1-2 NMP1 – Limiting Locations and 80-Year Projected CUF and CUF _{en} Values						
No.	Location Description	Material (Note 5)	Thermal Zone	80-Year CUF _{en}	80-Year CUF	Acceptance Criterion
1	EC Nozzle Junction (EC_NOZJNC)	SS	EC Shell Side	0.027	(Note 2)	1.0
2	EC Tube-to-Tubesheet Weld (EC_TSHTWLD)	SS	EC Shell and Tube	0.037	(Note 2)	1.0
3	EC Tubing (EC_TUBE)	SS	EC Shell and Tube	0.033	(Note 2)	1.0
4	EC Tubesheet (EC_TUBESHT)	SS	EC Shell and Tube	0.030	(Note 2)	1.0
5	Feedwater Nozzle Blend Radius (FW_BR) (Note 3)	LAS	RPV Region A	0.592	(Note 2)	1.0
6	Feedwater Nozzle Safe End (FW_SE) (Note 3)	CS	FW System Near	0.231	(Note 2)	1.0
7	Core Spray Piping Point 1J (PIPE_CS) (Note 3)	SS	Core Spray Near	0.003	(Note 2)	1.0
8	Feedwater Piping Point 255, Welding Tee (PIPE_FW) (Note 3)	CS	FW System Far	0.061	(Note 2)	1.0
9	Feedwater/RWCU Mixing Region (PIPE_FWRWCU) (Note 4) (Note 8)	CS	FW/RWCU Mixing	0.842	(Note 2)	1.0
10	Recirculation Piping Point 3905, As-Welded Transition (PIPE_RECIRC) (Note 3)	SS	Recirc System	0.055	(Note 2)	1.0
11	RWCU Piping Point A20, Pipe-Valve Transition (PIPE_RWCU_A20) (Note 4)	SS	RWCU System	0.468	(Note 2)	1.0
12	RWCU Piping Point A80 (PIPE_RWCU_A80) (Note 4)	SS	RWCU System	0.422	(Note 2)	1.0
13	Basin Seal Skirt to Vessel Flange Junction (RPV_BASSKT)	CS/LAS	RPV Region A	(Note 1)	0.236	1.0

Table 4.3.1-2 NMP1 – Limiting Locations and 80-Year Projected CUF and CUF _{en} Values						
No.	Location Description	Material (Note 5)	Thermal Zone	80-Year CUF _{en}	80-Year CUF	Acceptance Criterion
14	Closure Studs (Bolts) (RPV_BOLTS)	LAS	RPV Region A	(Note 1)	0.175	1.0
15	RPV Lower Head, Element D cut D Inside (RPV_BOTTHEAD) (Note 3) (Note 7)	LAS	RPV Region C	0.263	(Note 2)	1.0
16	RPV CRD Return Nozzle (RPV_CRDRTN) (Note 4)	LAS	CRD Return Nozzle	0.751	(Note 2)	1.0
17	RPV CRD Stub Tube (RPV_CRDSTUB) (Note 4)	SS	RPV Region C	0.268	(Note 2)	1.0
18	Core Spray Nozzle Forging (302B Point 35) (RPV_CSNOZ) (Note 3)	LAS	Core Spray Near	0.002	(Note 2)	1.0
19	Core Spray Nozzle Safe End (304SS Point 20) (RPV_CSSE) (Note 3)	SS	Core Spray Near	0.002	(Note 2)	1.0
20	Recirculation Inlet Nozzle Safe End to Nozzle Weld (RPV_RECIN) (Note 3)	SS	Recirc System	0.041	(Note 2)	1.0
21	Recirculation Outlet Nozzle Blend Radius (RPV_RECOUT) (Note 3)	LAS	RPV Region B	0.045	(Note 2)	1.0
22	Torus Shell (TORUS_SHELL) (Note 4)	CS	(Note 6)	(Note 1)	0.057	1.0
23	Torus-Attached Piping XS-326 (TORUS_XS326)	CS	(Note 6)	(Note 1)	0.030	1.0
24	Torus-Attached Piping XS-337 (TORUS_XS337)	CS	(Note 6)	(Note 1)	0.090	1.0

NOTES:

Note 1: CUF_{en} is not applicable for this location because it is not exposed to the reactor coolant environment.

Note 2: The CUF value is not included in the table because it is bounded by the CUF_{en} value.

Note 3: NUREG/CR-6260 location for the older-vintage GEH plant.

Note 4: New monitored location.

Note 5: Materials listed are LAS – Low Alloy Steel, CS – Carbon Steel, and SS – Stainless Steel.

Note 6: This location does not have an applicable thermal zone and was not considered in EAF screening as this location is not exposed to the reactor coolant environment.

Note 7: This location bounds the RPV support skirt location, including its weldment to the RPV pressure boundary.

Note 8: This location is associated with the FW/RWCU mixing tees (downstream side). There are two FW/RWCU mixing tees (one in each main feedwater line).

SLRA Appendix B, Section B.3.1.1, Fatigue Monitoring, page B-238, Operating Experience Example #4 is added as shown below.

B.3.1.1 Fatigue Monitoring

Operating Experience

4. In 2023, NMP1 performed a screening and evaluation of the NEI 03-08 "Good Practice" recommendations for the assessment of thermal fatigue susceptibility of mixing tees detailed in BWRVIP-196, Revision 1, "Assessment of Mixing Tee Thermal Fatigue Susceptibility in BWR Plants." The screening evaluation recommended a one-time inspection of both feedwater/reactor water cleanup (FW/RWCU) mixing tees. As a result, two component locations were inspected by ultrasonic testing during the NMP1 spring 2025 refueling outage and all inspections were satisfactory. However, further review of the results in 2026 identified that one of the two component locations was incorrectly identified. The FW/RWCU mixing tee downstream of penetration X-4B was inspected for thermal fatigue cracking during the spring 2025 refueling outage, but the FW/RWCU mixing tee downstream of penetration X-4A was not inspected, as intended. An issue report was entered into the corrective action program to address the missed inspection.

The inspection of the correct location associated with the FW/RWCU mixing tee downstream of penetration X-4A, consistent with BWRVIP-196, Revision 1 recommendations, is currently scheduled to be performed during the upcoming NMP1 spring 2027 refueling outage. The issue report noted that the two mixing tees are identical in geometry, size, and fabrication and thus experience the same temperature and flows. Therefore, the mixing tee to be examined in the spring 2027 refueling outage is expected to have the same inspection results as the mixing tee inspected in 2025. It should be noted that both FW/RWCU mixing regions were inspected by ultrasonic testing in 1989 and no cracking indications were identified.

This operating experience provides objective evidence that deficiencies identified during the implementation of the program are entered into the corrective action program for evaluation and timely resolution. Additionally, inspection results of components monitored in the Fatigue Monitoring aging management program can be used to support the validity of fatigue and environmentally assisted fatigue analyses.

Change #13 –Turbine Building HVAC System Scoping and Screening Results
Clarification Regarding Battery Room Exhaust Ducting

Affected SLRA Section: Section 2.3.3.21

SLRA Page Number: 2.3-106

Description of Change:

SLRA Section 2.3.3.21 is being revised to state that the portions of the Turbine Building HVAC System that exhaust hydrogen from battery rooms 11 and 12 are not in scope for subsequent license renewal. The battery room exhaust ventilation is nonsafety-related and does not directly support the function of the station batteries. Additionally, the battery room exhaust ventilation is not credited to perform a function in support of the 10 CFR 54.4(a)(3) regulated events. If forced ventilation is lost, the hydrogen concentration in battery rooms 11 and 12 would remain acceptably low due to natural circulation into the Turbine Building HVAC System ductwork. Further, if the pressure boundary of the battery room exhaust ductwork were to be degraded, natural circulation out of the battery rooms would continue, and any present hydrogen would migrate into the vast air volume of the turbine building.

This change is in response to a Breakout Question (i.e., Aux System-6).

SLRA Section 2.3.3.21, Turbine Building HVAC System, page 2.3-106, Description is revised as shown below.

2.3.3.21 Turbine Building HVAC System

Description

The Turbine Building HVAC (TBV) System is a normally operating system designed to provide a continuous flow of filtered air throughout the building while maintaining a negative pressure to control the release of contaminated air and prevent out leakage. The TBV System consists of the following plant systems:

- Turbine building ventilation exhaust
- Diesel generator building ventilation

The TBV System includes air intakes, filters, electric heating units, flow control dampers, dampers, and ductwork to distribute air to various areas of the turbine building. Outside air enters via louvered, screened penthouses, passing through filters and heating coils. Exhaust air returns through a plenum to the stack for discharge and is monitored (alarm only) for radiation. The system also includes water-cooled heat exchanger cooling units in areas where equipment operating at a high temperature are located in order to dissipate the radiant heat loss from this equipment and to maintain desired temperatures for personnel comfort and equipment protection.

The diesel generator building ventilation includes two safety-related roof exhaust fans and a roll-up door for each diesel generator room. The fans and the roll-up door operate together to maintain the diesel room temperature below maximum temperatures for operation of the diesel. Thermostatically controlled electric heaters are provided to maintain the required ambient room temperature to support diesel readiness.

Smoke and heat removal capability is provided for three smoke zones in the turbine building. These systems have sufficient smoke exhaust fans, isolation dampers, and controls dedicated to smoke removal. Smoke removal for the upper elevation of the turbine building is provided by motor-operated vents installed in the turbine building roof and five sidewall vents.

The TBV System also includes safety-related blocking valves, backdraft dampers, and ductwork in the turbine building ventilation, off-gas building ventilation, waste disposal building ventilation, waste disposal building extension ventilation and radwaste solidification and storage building ventilation plant systems. These blocking valves and backdraft dampers prevent backflow from the other ventilation systems that also exhaust into the safety-related stack plenum. This prevents ground level radioactive releases when the ventilation system exhaust fans are not operating.

The TBV System includes fire dampers. These are evaluated in the Fire Detection and Protection System.

The portions of the TBV System that exhaust hydrogen from battery rooms 11 and 12 are not in scope for license renewal. The battery room exhaust ventilation is nonsafety-related and does not directly support the function of the station batteries. Additionally, the battery room exhaust ventilation is not credited to perform a function in support of the

10 CFR 54.4(a)(3) regulated events. If forced ventilation is lost, the hydrogen concentration in battery rooms 11 and 12 would remain acceptably low due to natural circulation into the TBV System ductwork. The nonsafety-related TBV System ductwork is protected from hydrogen accumulation by dampers that automatically open to the turbine building upon loss of power. If the pressure boundary of the TBV System exhaust ducting were to be degraded, natural circulation of hydrogen out of the battery rooms would continue, and any present hydrogen would migrate into the vast air volume of the turbine building.

Change #14 – Revise Fire Water System Program Description in Appendix A and Appendix B for Consistency with NUREG-2191

Affected SLRA Sections: Appendix A, Section A.2.1.16 and Appendix B, Section B.2.1.16

SLRA Page Numbers: A-21 and B-91

Description of Change:

SLRA Appendix A, Section A.2.1.16 is being revised to include the cracking aging effect in the list of managed aging effects by the program. Additionally, Appendix A, Section A.2.1.16 is being revised to include discussion of the need to perform piping wall thickness measurements when unexpected levels of degradation are identified and the need to remove material and determine the source when obstructing material is identified. These changes are being made for consistency with NUREG-2191 Table XI-01. Corresponding changes are also being made to Appendix B, Section B.2.1.16.

This change is in response to a Fire Water System Program Breakout Question (i.e., TRP 27-9).

Accordingly, Appendix A, Section A.2.1.16, and Appendix B, Section B.2.1.16 are revised.

SLRA Appendix A, Section A.2.1.16, Fire Water System, page A-21 is revised as shown below. This SLRA markup includes changes described in Change #5 which clarified the testing requirements for sprinklers and is duplicated here as an aid to the reviewer.

A.2.1.16 Fire Water System

The Fire Water System aging management program is an existing condition monitoring, performance monitoring, and preventive program that manages loss of material, cracking, and flow blockage due to fouling. The program manages these aging effects by monitoring system pressure, flushing the system fire header, flow testing the fire main, performance testing the fire pumps, full flow flushing and flow verification of the fire hydrants, flushing and flow testing sprinkler and deluge systems, and visually inspecting systems using the guidance of NFPA 25, 2011 Edition.

The program applies to water-based fire protection systems that consist of sprinklers, fittings, valves, hydrants, hose stations, standpipes, strainers, pumps, and aboveground and buried piping and components. The program manages aging of fire protection system components exposed to raw water, indoor and outdoor air, and condensation environments. Aging of the external surfaces of buried fire main piping is managed as described in the Buried and Underground Piping and Tanks (A.2.1.27) aging management program.

Inspections and tests follow station procedures that include inspection parameters for items such as lighting, distance, offset, presence of protective coatings, and cleaning processes for an adequate examination.

Testing or replacement of sprinklers that have been in place for 50 years, 60 years, 70 years, and 80 years is performed using the guidance of NFPA 25, 2011 Edition. Section 5.3.1. ~~if a station specific evaluation cannot determine the water is non-corrosive to sprinklers.~~

The water-based fire protection system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated. Piping wall thickness measurements are conducted when visual inspections detect surface irregularities indicative of unexpected levels of degradation. When the presence of sufficient organic or inorganic material sufficient to obstruct piping or sprinklers is detected, the material is removed and the source is detected and corrective actions initiated.

SLRA Appendix B, Section B.2.1.16, Fire Water System, page B-91, Program Description discussion, third paragraph is revised as shown below.

B.2.1.16 Fire Water System

Program Description

Inspections and tests follow station procedures that include inspection parameters for items such as lighting, distance, offset, presence of protective coatings, and cleaning processes for an adequate examination. Piping wall thickness measurements are conducted when visual inspections detect surface irregularities indicative of unexpected levels of degradation. When the presence of sufficient organic or inorganic material sufficient to obstruct piping or sprinklers is detected, the material is removed and the source is detected and corrective actions initiated.

Change #15 – Add Exception to One-Time Inspection Program to Specify Inspection Methods for Flow Blockage Due to Fouling

Affected SLRA Section: Appendix B, Section B.2.1.20

SLRA Page Numbers: B-118 and B-119

Description of Change:

The One-Time Inspection (B.2.1.20) program is being revised to add an exception to NUREG-2191, Table XI.M32-1 to specify the appropriate inspection methods for flow blockage due to fouling. The options are either a visual inspection, such as a VT-3 examination, or performance of a containment spray air flow test to inspect for flow blockage in the containment spray nozzles.

Accordingly, SLRA Appendix B, Section B.2.1.20 is revised.

SLRA Appendix B, Section B.2.1.20, One-Time Inspection, page B-118, is revised as shown below.

B.2.1.20 One-Time Inspection

NUREG-2191 Consistency

The One-Time Inspection aging management program will be consistent with the ten elements of aging management program XI.M32, "One-Time Inspection," specified in NUREG-2191 with the following exception~~s~~:

SLRA Appendix B, Section B.2.1.20, One-Time Inspection, page B-119, Exception #2 is added as shown below.

B.2.1.20 One-Time Inspection

Exceptions to NUREG-2191

1. The NUREG-2191 Chapter XI.M32 "Parameters Monitored or Inspected," Table XI.M32-1 does not include inspection methods for the identification of loss of material due to wear on the external surface of components. The scope of the One-Time Inspection program includes the capability to perform visual inspections for wear on the external surfaces of components. **Program Elements Affected: Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4)**

Justification for Exception

Recent fleet operating experience, as described in NRC Information Notice (IN) 2007-21, "Pipe Wear Due to Interaction of Flow-Induced Vibration and Reflective Metal Insulation," has demonstrated the potential for loss of material due to wear to occur on Class 1 small-bore piping exposed to end caps on reflective metal insulation (RMI). There is no operating experience demonstrating this aging effect at NMP1, and the One-Time Inspection aging management program will be implemented to verify the aging effect is not occurring. A VT-3 or equivalent inspection will be performed to detect loss of material due to wear on the external surfaces of Class 1 small-bore piping exposed to RMI end caps.

2. The NUREG-2191, Table XI.M32-1, "Examples of Parameters Monitored or Inspected and Aging Effect for Specific Structure or Component," does not specify inspection methods for the identification of flow blockage due to fouling. The One-Time Inspection program manages the stainless steel containment spray nozzles for the aging effect of flow blockage due to fouling. The one-time inspection will be accomplished by either a visual inspection, such as a VT-3 examination, or performance of a containment spray air flow test. Either method is acceptable to confirm that flow blockage due to fouling is not occurring in the stainless steel containment spray nozzles. **Program Elements Affected: Parameters Monitored or Inspected (Element 3), Detection of Aging Effects (Element 4)**

Justification for Exception

NUREG-2192 Section 3.2.2.2.3 identifies the One-Time Inspection Program as an acceptable program to manage loss of material due to general corrosion and flow blockage due to fouling. However, no line item in NUREG-2191 Table XI.M32-1 specifies an inspection method for flow blockage due to fouling.

A visual inspection, such as a VT-3 examination, of a representative sample of the containment spray nozzles will confirm that flow blockage due to fouling is not occurring. If the one-time inspection does not meet acceptance criteria, the

results will be evaluated in accordance with the corrective action program. For a similar aging effect, reduction of heat transfer due to fouling, Table XI.M32-1 recommends performance of a visual inspection to confirm that fouling is not occurring. As noted in Table XI.M32-1 Note (b), "Visual inspection may be used only when the inspection methodology examines the surface potentially experiencing the aging effect."

The containment spray air flow test will verify air flow through the nozzles in the containment spray system to confirm that flow blockage is not occurring. The existing station containment spray air flow test procedure is used to verify operability of the containment spray system in accordance with Technical Specification surveillance requirement 4.3.7.b following maintenance that could result in nozzle flow blockage. The surveillance requirement frequency was revised from "at least once per operating cycle" to "following maintenance that could result in nozzle blockage" in 2011 by License Amendment No. 208 (ADAMS Accession Number ML110650003). The basis for the frequency change was satisfactory performance of ten containment spray air flow tests from 1990 through 2009. There was no nozzle blockage identified in these tests. These results demonstrate that the containment spray air flow test can confirm the absence of flow blockage in the nozzles.

Change #16 – Add Enhancement to Lubricating Oil Analysis Aging Management Program

Affected SLRA Sections: Appendix A, Section A.2.1.25; Appendix A, Table A.5; and Appendix B, Section B.2.1.25

SLRA Page Numbers: A-29, A-84, and B-140

Description of Change:

An enhancement is being added to the Lubricating Oil Analysis program to provide for sampling and testing for corrosion products and moisture in the hydraulic oil of the turbine-driven reactor feedwater pump clutch actuation oil system.

Accordingly, SLRA Appendix A, Section A.2.1.25; Appendix A, Table A.5; and Appendix B, Section B.2.1.25 are revised.

SLRA Appendix A, Section A.2.1.25, Lubricating Oil Analysis, page A-29, Enhancement #4 is added as shown below. This SLRA markup includes changes submitted in Supplement 1, dated July 30, 2026, which added Enhancement #3 via Change #8 and is duplicated here as an aid to the reviewer.

A.2.1.25 Lubricating Oil Analysis

The Lubricating Oil Analysis aging management program will be enhanced to:

1. Perform periodic sampling and testing of lubricating oil for moisture and corrosion particles in the liquid poison system pump gearboxes, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the liquid poison system pump gearboxes based on vendor intervals or station experience.
2. Perform periodic sampling and testing of hydraulic oil for moisture and corrosion particles in the circulating water gates and hoists system hydraulic fluid reservoir, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the circulating water gates and hoists system hydraulic fluid reservoir based on vendor intervals or station experience.
3. Perform periodic sampling and testing of lubricating oil for moisture and corrosion particles in the diesel driven fire pump right angle gear drive, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the diesel driven fire pump right angle gear drive based on vendor intervals or station experience.
4. Perform periodic sampling and testing of hydraulic oil for moisture and corrosion particles in the turbine-driven reactor feedwater pump clutch actuation oil system, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the turbine-driven reactor feedwater pump clutch actuation oil system based on vendor intervals or station experience.

The program will be enhanced no later than six months prior to the subsequent period of extended operation.

SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, Commitment 25, Lubricating Oil Analysis, page A-84, Enhancement #4 is added as shown below. This SLRA markup includes changes submitted in Supplement 1, dated July 30, 2026, which added Enhancement #3 via Change #8 and is duplicated here as an aid to the reviewer.

Table A.5 Subsequent License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
25	Lubricating Oil Analysis	Lubricating Oil Analysis is an existing program that will be enhanced to: 1. Perform periodic sampling and testing of lubricating oil for moisture and corrosion particles in the liquid poison system pump gearboxes, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the liquid poison system pump gearboxes based on vendor intervals or station experience. 2. Perform periodic sampling and testing of hydraulic oil for moisture and corrosion particles in the circulating water gates and hoists system hydraulic fluid reservoir, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the circulating water gates and hoists system hydraulic fluid reservoir based on vendor intervals or station experience. 3. Perform periodic sampling and testing of lubricating oil for moisture and corrosion particles in the diesel driven fire pump right angle gear drive, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the diesel driven fire pump right angle gear drive based on vendor intervals or station experience. 4. <u>Perform periodic sampling and testing of hydraulic oil for moisture and corrosion particles in the turbine-driven reactor feedwater pump clutch actuation oil system, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the turbine-driven reactor feedwater pump clutch actuation oil system based on vendor intervals or station experience.</u>	Program will be enhanced no later than six months prior to the subsequent period of extended operation.	Section A.2.1.25

SLRA Appendix B, Section B.2.1.25, Lubricating Oil Analysis, page B-140, Enhancement #4 is added as shown below. This SLRA markup includes changes submitted in Supplement 1, dated July 30, 2026, which added Enhancement #3 via Change #8 and is duplicated here as an aid to the reviewer.

B.2.1.25 Lubricating Oil Analysis

Enhancements

1. Perform periodic sampling and testing of lubricating oil for moisture and corrosion particles in the liquid poison system pump gearboxes, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the liquid poison system pump gearboxes based on vendor intervals or station experience. **Program Elements Affected: Detection of Aging Effects (Element 4)**
2. Perform periodic sampling and testing of hydraulic oil for moisture and corrosion particles in the circulating water gates and hoists system hydraulic fluid reservoir, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the circulating water gates and hoists system hydraulic fluid reservoir based on vendor intervals or station experience. **Program Elements Affected: Detection of Aging Effects (Element 4)**
3. Perform periodic sampling and testing of lubricating oil for moisture and corrosion particles in the diesel driven fire pump right angle gear drive, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the diesel driven fire pump right angle gear drive based on vendor intervals or station experience. **Program Elements Affected: Detection of Aging Effects (Element 4)**
4. Perform periodic sampling and testing of hydraulic oil for moisture and corrosion particles in the turbine-driven reactor feedwater pump clutch actuation oil system, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the turbine-driven reactor feedwater pump clutch actuation oil system based on vendor intervals or station experience. **Program Elements Affected: Detection of Aging Effects (Element 4)**

Change #17 – Correct Component Types and Materials in the Control Room HVAC System

Affected SLRA Sections: Table 2.3.3-5, Section 3.3.2.1.5, and Tables 3.3.1, 3.3.2-5, and 3.4.1

SLRA Page Numbers: 2.3-53, 3.3-8, 3.3-77, 3.3-98, 3.3-185, 3.3-186, and 3.4-57

Description of Change:

SLRA Table 3.3.2-5 for the Control Room HVAC System is being revised to reflect the current material types of the installed valve bodies. For insulated valve bodies, this includes adding cast iron material type and removing the copper alloy with 15 percent zinc or more material type. For un-insulated valve bodies, this includes adding copper alloy with 15 percent zinc or more material type and the air – indoor uncontrolled environment. Corresponding changes are being made to SLRA Table 2.3.3-5; SLRA Section 3.3.2.1.5; SLRA Table 3.3.1, Item Numbers 3.3.1-072 and 3.3.1-132; and SLRA Table 3.4.1, Item Number 3.4.1-106 to reflect the component, material and aging management program changes in SLRA Table 3.3.2-5.

Accordingly, SLRA Table 2.3.3-5, Section 3.3.2.1.5, and Tables 3.3.1, 3.3.2-5, and 3.4.1 are revised.

SLRA Table 2.3.3-5, Control Room HVAC System, page 2.3-53 is revised as shown below.

Table 2.3.3-5 Control Room HVAC System
Components Subject to Aging Management Review

Component Type	Intended Function
Tanks (Chilled Water Expansion Tank)	Pressure Boundary
Tanks (Chilled Water Side Stream Demineralizer)	Leakage Boundary (Spatial)
<u>Valve Body</u>	<u>Pressure Boundary</u>
Valve Body, Insulated	Leakage Boundary (Spatial)
	Pressure Boundary

SLRA Section 3.3.2.1.5, Control Room HVAC System, page 3.3-8 is revised as shown below:

3.3.2.1.5 Control Room HVAC System

Materials

The materials of construction for the Control Room HVAC System components are:

- Aluminum Alloy
- Carbon and Low Alloy Steel Bolting
- Carbon Steel
- Cast Iron
- Copper Alloy with 15% Zinc or More
- Copper Alloy with less than 15% Zinc
- Elastomers
- Galvanized Bolting
- Galvanized Steel
- Stainless Steel
- Stainless Steel Bolting

SLRA Table 3.3.1, Summary of Aging Management Evaluations for the Auxiliary Systems, page 3.3-77 is revised as shown below.

Table 3.3.1 Summary of Aging Management Evaluations for the Auxiliary Systems

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-072	Gray cast iron, ductile iron, malleable iron, copper alloy (>15% zinc or >8% aluminum) piping, piping components, heat exchanger components exposed to treated water, closed-cycle cooling water, soil, raw water, raw water (potable), waste water	Loss of material due to selective leaching	AMP XI.M33, "Selective Leaching"	No	Consistent with NUREG-2191 with exceptions. The Selective Leaching (B.2.1.21) program will be used to manage LOM due to selective leaching of the cast iron and copper alloy with 15% ~~or~~ zinc <u>or more</u> fire hydrants, HX components, piping, pump casings, steel components, & tanks, <u>and valve bodies</u> exposed to CCCW, raw water, soil, treated water, & <u>and</u> waste water in Circulating Water System, City Water System, Compressed Air Systems, Control Room HVAC System, Emergency Diesel Generator System, Fire Detection and Protection System, Makeup Demineralizer System, Radioactive Waste System, Reactor Building Closed Loop Cooling System, Reactor Water Cleanup, Service Water System, Spent Fuel Pool Cooling and Storage System, Turbine Building Closed Loop Cooling System, Containment Spray System, Core Spray System, Control Rod Drive System, Main Generator and Auxiliary System, & <u>and</u> Screen and Pump House Building. LOM due to selective leaching of HX tubes fabricated from admiralty brass (inhibited brass) and exposed to CCCW, raw water, TW, and waste water is not an applicable aging effect requiring management consistent with NUREG-2191 Table IX.C for inhibited brass. Heat exchanger tubes fabricated from inhibited brass are identified through plant-specific notes provided in SLRA Table 3.3.2-12 for the Radioactive Waste System, Table 3.3.2-19 for the Spent Fuel Pool Cooling and Storage System, and Table 3.3.2-21 for the Turbine Building Closed Loop Cooling System.

SLRA Table 3.3.1, Summary of Aging Management Evaluations for the Auxiliary Systems, page 3.3-98 is revised as shown below.

Table 3.3.1 Summary of Aging Management Evaluations for the Auxiliary Systems

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.3.1-132	Insulated steel, copper alloy (>15% Zn or >8% Al), piping, piping components, tanks, tanks (within the scope of AMP XI.M29, "Outdoor and Large Atmospheric Metallic Storage Tanks") exposed to air, condensation	Loss of material due to general, pitting, crevice corrosion (steel only); cracking due to SCC (copper alloy (>15% Zn or >8% Al) only)	AMP XI.M36, "External Surfaces Monitoring of Mechanical Components" or AMP XI.M29, "Outdoor and Large Atmospheric Metallic Storage Tanks"	No	Consistent with NUREG-2191. The External Surfaces Monitoring of Mechanical Components (B.2.1.23) program will be used to manage loss of material of the insulated carbon steel, cast iron, ~~copper alloy with 15% zinc or more~~ , and galvanized steel ducting and components, heat exchanger components, piping, piping components, pump casings tanks, and valve bodies exposed to condensation in the Control Room HVAC System, Service Water System, and Turbine Building Closed Loop Cooling System. ~~The One-Time Inspection (B.2.1.20) program has been substituted and will be used to manage cracking of the copper alloy with 15% zinc or more insulated valve bodies exposed to condensation in the Control Room HVAC System.~~

SLRA Table 3.3.2-5, Control Room HVAC System Summary of Aging Management Evaluation, pages 3.3-185 and 3.3-186 are revised as shown below.

Table 3.3.2-5 Control Room HVAC System Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
35217	Tanks (Chilled Water Side Stream Demineralizer)	Leakage Boundary (Spatial)	Carbon Steel	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.1.12)	VII.F1.AP-202	3.3.1-045	A
39299	Valve Body	Pressure Boundary	Copper Alloy with 15% Zinc or More	Air - Indoor, Uncontrolled (External)	Cracking	One-Time Inspection (B.2.1.20)	VIII.H.S-454	3.4.1-106	E, 4
39303	Valve Body	Pressure Boundary	Copper Alloy with 15% Zinc or More	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.1.12)	VII.F1.AP-199	3.3.1-046	A
39304	Valve Body	Pressure Boundary	Copper Alloy with 15% Zinc or More	Closed Cycle Cooling Water (Internal)	Loss of Material/Selective Leaching	Selective Leaching (B.2.1.21)	VII.F1.AP-43	3.3.1-072	B
39302	Valve Body	Pressure Boundary	Copper Alloy with 15% Zinc or More	Closed Cycle Cooling Water (Internal)	None	None	VII.C2.A-473a	3.3.1-160	I, 3
36236	Valve Body, Insulated	Leakage Boundary (Spatial)	Copper Alloy with less than 15% Zinc	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.1.12)	VII.F1.AP-199	3.3.1-046	A
36234	Valve Body, Insulated	Leakage Boundary (Spatial)	Copper Alloy with less than 15% Zinc	Condensation (External)	None	None	VII.J.AP-144	3.3.1-114	A
37483	Valve Body, Insulated	Leakage Boundary (Spatial)	Stainless Steel	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.1.12)	VII.C2.A-52	3.3.1-049	A
37487	Valve Body, Insulated	Leakage Boundary (Spatial)	Stainless Steel	Condensation (External)	Cracking	One-Time Inspection (B.2.1.20)	VII.I.A-734b	3.3.1-205	B
37591	Valve Body, Insulated	Leakage Boundary (Spatial)	Stainless Steel	Condensation (External)	Loss of Material	One-Time Inspection (B.2.1.20)	VII.I.A-761b	3.3.1-232	B
39085	Valve Body, Insulated	Pressure Boundary	Carbon Steel	Closed Cycle Cooling Water (Internal)	Long-Term Loss of Material	One-Time Inspection (B.2.1.20)			H, 2
36239	Valve Body, Insulated	Pressure Boundary	Carbon Steel	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.1.12)	VII.F1.AP-202	3.3.1-045	A
36238	Valve Body, Insulated	Pressure Boundary	Carbon Steel	Condensation (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VII.I.A-405a	3.3.1-132	A

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
39305	Valve Body, Insulated	Pressure Boundary	Cast Iron	Closed Cycle Cooling Water (Internal)	Long-Term Loss of Material	One-Time Inspection (B.2.1.20)			H, 2
36243	Valve Body, Insulated	Pressure Boundary	Cast Iron	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.1.12)	VII.F1.AP-202	3.3.1-045	A
36244	Valve Body, Insulated	Pressure Boundary	Cast Iron	Closed Cycle Cooling Water (Internal)	Loss of Material/Selective Leaching	Selective Leaching (B.2.1.21)	VII.F1.AP-43	3.3.1-072	B
36564	Valve Body, Insulated	Pressure Boundary	Cast Iron	Condensation (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VII.I.A-405a	3.3.1-132	A
36243	~~Valve Body, Insulated~~	~~Pressure Boundary~~	~~Copper Alloy with 45% Zinc or More~~	~~Closed Cycle Cooling Water (Internal)~~	~~Loss of Material~~	~~Closed Treated Water Systems (B.2.1.12)~~	~~VII.F1.AP-199~~	~~3.3.1-046~~	~~A~~
36244	~~Valve Body, Insulated~~	~~Pressure Boundary~~	~~Copper Alloy with 45% Zinc or More~~	~~Closed Cycle Cooling Water (Internal)~~	~~Loss of Material/Selective Leaching~~	~~Selective Leaching (B.2.1.21)~~	~~VII.F1.AP-43~~	~~3.3.1-072~~	~~B~~
36242	~~Valve Body, Insulated~~	~~Pressure Boundary~~	~~Copper Alloy with 45% Zinc or More~~	~~Closed Cycle Cooling Water (Internal)~~	~~None~~	~~None~~	~~VII.C2.A-473a~~	~~3.3.1-160~~	~~I, 3~~
36241	~~Valve Body, Insulated~~	~~Pressure Boundary~~	~~Copper Alloy with 45% Zinc or More~~	~~Condensation (External)~~	~~Cracking~~	~~One-Time Inspection (B.2.1.20)~~	~~VII.I.A-405a~~	~~3.3.1-132~~	~~E, 4~~
36564	~~Valve Body, Insulated~~	~~Pressure Boundary~~	~~Copper Alloy with 45% Zinc or More~~	~~Condensation (External)~~	~~Loss of Material~~	~~External Surfaces Monitoring of Mechanical Components (B.2.1.23)~~	~~VII.I.A-405a~~	~~3.3.1-132~~	~~A~~
36246	Valve Body, Insulated	Pressure Boundary	Copper Alloy with less than 15% Zinc	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.1.12)	VII.F1.AP-199	3.3.1-046	A

SLRA Table 3.4.1, Summary of Aging Management Evaluations for the Steam and Power Conversion System, page 3.4-57 is revised as shown below.

Table 3.4.1 Summary of Aging Management Evaluations for the Steam and Power Conversion System

Item Number	Component	Aging Effect/Mechanism	Aging Management Programs	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	The One-Time Inspection (B.2.1.20) program has been substituted and will be used to manage cracking of the copper alloy with 15% zinc or more heat exchanger components, hoses, piping, piping components, spray nozzles and valve bodies exposed to air - indoor, uncontrolled and condensation in the Circulating Water System, City Water System, Compressed Air Systems, Containment Systems, <u>Control Room HVAC System</u>, Emergency Diesel Generator System, Fire Detection and Protection System, Makeup Demineralizer System, Reactor Building Closed Loop Cooling System, Reactor Water Cleanup, Turbine Building Closed Loop Cooling System, Control Rod Drive System, and Feedwater/High Pressure Coolant Injection System. Exceptions apply to the NUREG-2191 recommendations for One-Time Inspection (B.2.1.20) program implementation.

Change #18 – Correct the Material Type for Turbine Building Closed Loop Cooling System Piping Components

Affected SLRA Section: Table 3.3.2-20

SLRA Page Numbers: 3.3-323 and 3.3-324

Description of Change:

SLRA Table 3.3.2-20 for the Turbine Building Closed Loop Cooling System is being revised to reflect the current material type of installed piping components. The material type of copper alloy with 15 percent zinc or more is being revised to copper alloy with less than 15 percent zinc.

Accordingly, SLRA Table 3.3.2-20 is revised.

SLRA Table 3.3.2-20, Turbine Building Closed Loop Cooling System Summary of Aging Management Evaluation, pages 3.3-323 and 3.3-324 are revised as shown below.

Table 3.3.2-20 Turbine Building Closed Loop Cooling System Summary of Aging Management Evaluation

AMR ID	Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	NUREG-2192 Table 1 Item	Notes
39116	Piping, piping components	Leakage Boundary (Spatial)	Carbon Steel	Closed Cycle Cooling Water (Internal)	Long-Term Loss of Material	One-Time Inspection (B.2.1.20)			H, 1
36685	Piping, piping components	Leakage Boundary (Spatial)	Carbon Steel	Closed Cycle Cooling Water (Internal)	Loss of Material	Closed Treated Water Systems (B.2.1.12)	VII.C2.AP-202	3.3.1-045	A
37440	~~Piping, piping components~~	~~Leakage Boundary (Spatial)~~	~~Copper Alloy with 15% Zinc or More~~	~~Air - Indoor, Uncontrolled (External)~~	~~Cracking~~	~~One-Time Inspection (B.2.1.20)~~	~~VIII.H.S-454~~	~~3.4.1-106~~	~~E, 4~~
37442	~~Piping, piping components~~	~~Leakage Boundary (Spatial)~~	~~Copper Alloy with 15% Zinc or More~~	~~Closed Cycle Cooling Water (Internal)~~	~~Loss of Material~~	~~Closed Treated Water Systems (B.2.1.12)~~	~~VII.C2.AP-199~~	~~3.3.1-046~~	~~A~~
37443	~~Piping, piping components~~	~~Leakage Boundary (Spatial)~~	~~Copper Alloy with 15% Zinc or More~~	~~Closed Cycle Cooling Water (Internal)~~	~~Loss of Material/Selective Leaching~~	~~Selective Leaching (B.2.1.21)~~	~~VII.C2.AP-43~~	~~3.3.1-072~~	~~B~~
37444	~~Piping, piping components~~	~~Leakage Boundary (Spatial)~~	~~Copper Alloy with 15% Zinc or More~~	~~Closed Cycle Cooling Water (Internal)~~	~~None~~	~~None~~	~~VII.C2.A-473a~~	~~3.3.1-160~~	~~I, 2~~
39306	<u>Piping, piping components</u>	<u>Leakage Boundary (Spatial)</u>	<u>Copper Alloy with less than 15% Zinc</u>	<u>Air - Indoor, Uncontrolled (External)</u>	<u>None</u>	<u>None</u>	<u>VII.J.AP-144</u>	<u>3.3.1-114</u>	<u>A</u>
39307	<u>Piping, piping components</u>	<u>Leakage Boundary (Spatial)</u>	<u>Copper Alloy with less than 15% Zinc</u>	<u>Closed Cycle Cooling Water (Internal)</u>	<u>Loss of Material</u>	<u>Closed Treated Water Systems (B.2.1.12)</u>	<u>VII.C2.AP-199</u>	<u>3.3.1-046</u>	<u>A</u>
36690	Piping, piping components	Pressure Boundary	Carbon Steel	Air - Indoor, Uncontrolled (External)	Loss of Material	External Surfaces Monitoring of Mechanical Components (B.2.1.23)	VII.I.A-77	3.3.1-078	A
39117	Piping, piping components	Pressure Boundary	Carbon Steel	Closed Cycle Cooling Water (Internal)	Long-Term Loss of Material	One-Time Inspection (B.2.1.20)			H, 1

Enclosure B

Nine Mile Point Nuclear Station, Unit 1 Subsequent License Renewal Commitment List Update

Introduction

This enclosure identifies commitments made in this document and is an update to the Nine Mile Point Nuclear Station, Unit 1 (NMP1) Subsequent License Renewal Application (SLRA) Appendix A, Table A.5, Subsequent License Renewal Commitment List. Any other actions discussed in the submittal represent intended or planned actions. They are described to the NRC for the NRC's information and are not regulatory commitments.

Changes to the NMP1 SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, are a result of this SLRA Supplement.

These Enclosure B pages are the same as the corresponding NMP1 SLRA Appendix A, Table A.5 pages included in Enclosure A.

To facilitate understanding, relevant portions of the previously submitted Subsequent License Renewal Commitment List have been repeated in this Enclosure, with revisions indicated. Previously submitted information is shown in normal font. Additions due to this submittal are highlighted with red underlined font for inserted text, and ~~strikethroughs~~ for deleted text.

SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, Commitment 7, BWR Vessel Internals, pages A-76 and A-77, Enhancement #2 is revised as shown below.

TABLE A.5 – SUBSEQUENT LICENSE RENEWAL COMMITMENT LIST				
NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
7	BWR Vessel Internals	BWR Vessel Internals is an existing program that will be enhanced to: 2. Develop a justification for the service life of the core shroud tie-rod X-750 subcomponents through the subsequent period of extended operation, <u>using NRC approved BWRVIP guidelines</u> consistent with BWRVIP-94, Revision 4.7. <u>This which may include supplemental inspections, inspection frequencies frequency changes, engineering changes,</u> and technical evaluations to validate that the core shroud tie-rod X-750 subcomponents can perform their intended function through the subsequent period of extended operation. <u>Specifically, the NRC approved guidance in BWRVIP-138, Revision 2-A, Section 5 and BWRVIP-84, Revision 2-A, Section B.3.1 as modified by BWRVIP Letter 2011-011, may be used to ensure the X-750 subcomponents can perform their intended function through the subsequent period of extended operation. The original core shroud repair and subsequent core shroud repair modification were reviewed and found acceptable by the NRC (ADAMS Accession Nos. ML17059A774 and ML072600195, respectively) as an alternative to the ASME Code pursuant to 10 CFR 50.55a(a)(3)(i). Any proposed engineering changes to the core shroud tie-rod licensing basis will be evaluated by CEG in accordance with 10 CFR 50.59 to determine if prior NRC review and approval is required.</u>	Program will be enhanced no later than six months prior to the subsequent period of extended operation.	Section A.2.1.7

SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, Commitment 16, Fire Water System, page A-81, Enhancement #4 is revised as shown below:

A.5 Subsequent License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
16	Fire Water System	Fire Water System is an existing program that will be enhanced to: 4. Perform, on a five-year frequency, internal visual inspections of <u>a representative sample of each type of</u> ~~every other~~ in-scope suppression system. <u>The representative sample will consist of no less than 50 percent of the in-scope areas for each suppression system type. The inspections will be performed on a rotating basis such that all in-scope suppression systems will be inspected at least once during each 10-year period during the subsequent period of extended operation.</u> These inspections will include nozzle inspections when performed on deluge systems. ~~The systems not inspected in the first five-year interval will be inspected in the second five-year interval.~~ In dry pipe systems and pre-action systems, the sprinkler removed for inspection shall be from the most remote branch line from the source of water that is not equipped with the inspector's test valve. Tubercules or slime, if found, shall be tested for indications of microbiologically influenced corrosion (MIC). If the presence of foreign organic and/or inorganic material is found in any system in a building during the five-year internal inspection of piping, all systems shall be assessed. <u>This extent of condition assessment shall determine the appropriate scope and method of follow-up actions that will be performed through the corrective action program.</u> If the presence of sufficient foreign organic and/or inorganic material is found to obstruct pipe or sprinklers, an obstruction investigation will be conducted as described in Section 14.3 of NFPA 25. Alternatively, nondestructive examinations capable of ensuring flow blockage in piping and nozzles will not occur will be performed. The population of suppression systems eligible for inspection consists of the in-scope wet pipe sprinkler systems, dry-pipe systems, preaction systems, and deluge systems that are not functionally tested.	Program will be enhanced no later than six months prior to the subsequent period of extended operation. One-time inspections will be completed within the five years prior to the subsequent period of extended operation and no later than the last refueling outage prior to the subsequent period of extended operation.	Section A.2.1.16

SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, Commitment 16, Fire Water System, page A-81, Enhancement #6 is revised as shown below.

A.5 Subsequent License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
16	Fire Water System	Fire Water System is an existing program that will be enhanced to: 6. Perform a system mainline strainer inspection and cleaning after system flow exceeding that of a nominal two-inch orifice. <u>This enhancement does not apply to the private service main fire pump discharge strainers.</u>	Program will be enhanced no later than six months prior to the subsequent period of extended operation. One-time inspections will be completed within the five years prior to the subsequent period of extended operation and no later than the last refueling outage prior to the subsequent period of extended operation.	Section A.2.1.16

SLRA Appendix A, Table A.5, Subsequent License Renewal Commitment List, Commitment 25, Lubricating Oil Analysis, page A-84, Enhancement #4 is added as shown below. This SLRA markup includes changes submitted in Supplement 1, dated July 30, 2026, which added Enhancement #3 via Change #8 and is duplicated here as an aid to the reviewer.

Table A.5 Subsequent License Renewal Commitment List

NO.	PROGRAM OR TOPIC	COMMITMENT	IMPLEMENTATION SCHEDULE	SOURCE
25	Lubricating Oil Analysis	Lubricating Oil Analysis is an existing program that will be enhanced to: 1. Perform periodic sampling and testing of lubricating oil for moisture and corrosion particles in the liquid poison system pump gearboxes, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the liquid poison system pump gearboxes based on vendor intervals or station experience. 2. Perform periodic sampling and testing of hydraulic oil for moisture and corrosion particles in the circulating water gates and hoists system hydraulic fluid reservoir, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the circulating water gates and hoists system hydraulic fluid reservoir based on vendor intervals or station experience. 3. Perform periodic sampling and testing of lubricating oil for moisture and corrosion particles in the diesel driven fire pump right angle gear drive, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the diesel driven fire pump right angle gear drive based on vendor intervals or station experience. 4. <u>Perform periodic sampling and testing of hydraulic oil for moisture and corrosion particles in the turbine-driven reactor feedwater pump clutch actuation oil system, following industry or vendor recommended practices. Alternatively, test old oil after periodic oil changes for the turbine-driven reactor feedwater pump clutch actuation oil system based on vendor intervals or station experience.</u>	Program will be enhanced no later than six months prior to the subsequent period of extended operation.	Section A.2.1.25