Constellation Energy
Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, New York 13093

December 6, 2004
NMP1L 1892

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

SUBJECT: Nine Mile Point Units 1 and 2
Docket Nos. 50-220 and 50-410
Facility Operating License Nos. DPR-63 and NPF-69

License Renewal Application – Submittal of Supplemental Information Resulting from the NRC Audit of Aging Management Reviews (TAC Nos. MC0691 and MC0692)

Gentlemen:

By letter dated May 26, 2004, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted an application to renew the operating licenses for Nine Mile Point Units 1 and 2.

As a result of NRC audits of the aging management reviews, supplemental information in support of the License Renewal Application is being submitted as Attachments 1 and 2. This letter contains no new regulatory commitments.

If you have any questions about this submittal, please contact Peter Mazzaferro, NMPNS License Renewal Project Manager, at (315) 349-1019.

Very truly yours,

James A. Spina
Vice President Nine Mile Point

JAS/DEV/jm

A107

STATE OF NEW YORK :
: TO WIT:
COUNTY OF OSWEGO :

I, James A. Spina, being duly sworn, state that I am Vice President Nine Mile Point, and that I am duly authorized to execute and file this supplemental information on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this submittal are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 6th day of December, 2004.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

12/6/04
Date

SANDRA A. OSWALD
Notary Public, State of New York
No. 01OS6032276
Qualified in Oswego County
Commission Expires 10/25/05

Attachments:
1. Supplemental Information in Support of the License Renewal Application Resulting from the NRC Audit of Aging Management Reviews
2. Re-Typed Text and Tables for License Renewal Application Section 3.1, Aging Management of Reactor Vessel, Internals, and Reactor Coolant Systems

cc: Mr. S. J. Collins, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR
Mr. N. B. Le, License Renewal Project Manager, NRR
Mr. J. P. Spath, NYSERDA

ATTACHMENT 1

Nine Mile Point Nuclear Station

Supplemental Information in Support of the License Renewal Application

Resulting from the NRC Audit of Aging Management Reviews

This supplemental information consists of revisions to License Renewal Application (LRA) Section 3.1, "Aging Management of Reactor Vessel, Internals, and Reactor Coolant Systems," for both Nine Mile Point Unit 1 (NMP1) and Nine Mile Point Unit 2 (NMP2). The revisions to the text and tables of LRA Section 3.1 incorporate changes resulting from resolution of the NRC aging management review audit items listed in Table 1 below. The re-typed Section 3.1 text and tables, provided in Attachment 2, replace the existing corresponding LRA text and tables in their entirety. The changes are highlighted by shading.

Table 1 – NRC Aging Management Review Audit Items Addressed by the Revisions to LRA Section 3.1

AMR Audit Item No.	Revised LRA Table(s)
1	Various
3	3.1.2.A-1
4	3.1.2.A-1
7	3.1.1.A, 3.1.2.A-3
8	3.1.1.A, 3.1.2.A-4
39	Various
67	3.1.2.A-1
71	3.1.2.A-3
76	3.1.2.B-1
77	3.1.2.B-1
78	3.1.2.B-1
79	3.1.2.B-1
80	3.1.2.B-1
81	3.1.2.B-1
83	3.1.2.B-1
87	3.1.2.B-2
88	3.1.2.B-2
91	3.1.2.B-2
93	3.1.2.B-2
94	3.1.1.B, 3.1.2.B-3
96	3.1.2.B-3, 3.1.2.B-5
100	3.1.2.B-4
211	3.1.1.A
212	3.1.1.A
213	3.1.1.A, 3.1.1.B
214	3.1.1.A, 3.1.1.B
215	3.1.1.A, 3.1.1.B
216	3.1.1.A, 3.1.1.B
217	3.1.1.A, 3.1.1.B
222	3.1.1.B
223	3.1.1.B
224	3.1.1.A, 3.1.1.B
225	3.1.1.B, 3.1.2.B-1
227	3.1.1.B, 3.1.2.B-1
228	3.1.1.A, 3.1.2.B-1

ATTACHMENT 2

Nine Mile Point Nuclear Station

Re-Typed Text and Tables for License Renewal Application Section 3.1,

Aging Management of Reactor Vessel, Internals, and Reactor Coolant Systems

3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEMS

3.1.1 INTRODUCTION

This section provides the results of the aging management review for those components identified in Section 2.3.1, Reactor Vessel, Internals, and Reactor Coolant Systems (RCS), as being subject to aging management review. The systems, or portions of systems, which are addressed in this section, are described in the indicated sections.

NMP1

- NMP1 Reactor Pressure Vessel (2.3.1.A.1)
- NMP1 Reactor Pressure Vessel Internals (2.3.1.A.2)
- NMP1 Reactor Pressure Vessel Instrumentation System (2.3.1.A.3)
- NMP1 Reactor Recirculation System (2.3.1.A.4)
- NMP1 Control Rod Drive System (2.3.1.A.5)

NMP2

- NMP2 Reactor Pressure Vessel (2.3.1.B.1)
- NMP2 Reactor Pressure Vessel Internals (2.3.1.B.2)
- NMP2 Reactor Pressure Vessel Instrumentation System (2.3.1.B.3)
- NMP2 Reactor Recirculation System (2.3.1.B.4)
- NMP2 Control Rod Drive System (2.3.1.B.5)

Tables 3.1.1.A, NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801, and 3.1.1.B, NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801, provide the summary of the programs evaluated in NUREG-1801 for the RCS component groups that are relied on for license renewal.

These tables use the format described in Section 3.0 above. Note that these tables only include results for those component groups that are applicable to a BWR.

3.1.2 RESULTS

The following tables summarize the results of the aging management review for systems in the RCS group.

NMP1

- Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System - NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation
- Table 3.1.2.A-2 Reactor Vessel, Internals, and Reactor Coolant System - NMP1 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation
- Table 3.1.2.A-3 Reactor Vessel, Internals, and Reactor Coolant System - NMP1 Reactor Pressure Vessel Instrumentation System – Summary of Aging Management Evaluation
- Table 3.1.2.A-4 Reactor Vessel, Internals, and Reactor Coolant System - NMP1 Reactor Recirculation System – Summary of Aging Management Evaluation
- Table 3.1.2.A-5 Reactor Vessel, Internals, and Reactor Coolant System - NMP1 Control Rod Drive System – Summary of Aging Management Evaluation

NMP2

- Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System - NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation
- Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System - NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation
- Table 3.1.2.B-3 Reactor Vessel, Internals, and Reactor Coolant System - NMP2 Reactor Pressure Vessel Instrumentation System – Summary of Aging Management Evaluation

- Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System - NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation
- Table 3.1.2.B-5 Reactor Vessel, Internals, and Reactor Coolant System - NMP2 Control Rod Drive System – Summary of Aging Management Evaluation

The materials from which specific components are fabricated, the environments to which components are exposed, the aging effects requiring management, and the aging management programs used to manage these aging effects are provided for each of the above systems in the following subsections of Section 3.1.2.A, NMP1 Materials, Environments, Aging Effects Requiring Management and Aging Management Programs and Section 3.1.2.B, NMP2 Materials, Environments, Aging Effects Requiring Management and Aging Management Programs:

NMP1

- Section 3.1.2.A.1, NMP1 Reactor Pressure Vessel
- Section 3.1.2.A.2, NMP1 Reactor Pressure Vessel Internals
- Section 3.1.2.A.3, NMP1 Reactor Pressure Vessel Instrumentation System
- Section 3.1.2.A.4, NMP1 Reactor Recirculation System
- Section 3.1.2.A.5, NMP1 Control Rod Drive System

NMP2

- Section 3.1.2.B.1, NMP2 Reactor Pressure Vessel
- Section 3.1.2.B.2, NMP2 Reactor Pressure Vessel Internals
- Section 3.1.2.B.3, NMP2 Reactor Pressure Vessel Instrumentation System
- Section 3.1.2.B.4, NMP2 Reactor Recirculation System
- Section 3.1.2.B.5, NMP2 Control Rod Drive System

3.1.2.A NMP1 MATERIALS, ENVIRONMENTS, AGING EFFECTS REQUIRING MANAGEMENT AND AGING MANAGEMENT PROGRAMS

3.1.2.A.1 NMP1 REACTOR PRESSURE VESSEL

Materials

The materials of construction for the NMP1 Reactor Pressure Vessel components are:

- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)
- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)
- Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)
- Nickel Based Alloys
- Wrought Austenitic Stainless Steel

Environments

The NMP1 Reactor Pressure Vessel components are exposed to the following environments:

- Air, Moisture or Wetting, temperature <140°F
- Air, Moisture or Wetting, Temperature ≥212°F
- Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F, Leaking Fluid
- Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel
- Treated Water or Steam, High temperature, Neutron Fluence $\geq 1\times10^{17}$n/cm^2. - BWR Reactor Pressure Vessel
- Treated Water or Steam, temperature ≥ 482°F, Low Flow

Aging Effects Requiring Management

The following aging effects, associated with the NMP1 Reactor Pressure Vessel, require management:

- Cracking
- Cumulative Fatigue Damage
- Loss of Fracture Toughness
- Loss of Material

Aging Management Programs

The following aging management programs manage the aging effects for the NMP1 Reactor Pressure Vessel components:

- ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program
- BWR Feedwater Nozzle Program
- BWR Penetrations Program
- BWR Stress Corrosion Cracking Program
- BWR Vessel ID Attachment Welds Program
- BWR Vessel Internals Program
- Fatigue Monitoring Program
- One-Time Inspection Program
- Reactor Head Closure Studs Program
- Reactor Vessel Surveillance Program
- Water Chemistry Control Program

3.1.2.A.2 NMP1 REACTOR PRESSURE VESSEL INTERNALS

Materials

The materials of construction for the NMP1 Reactor Pressure Vessel Internals components are:

- Cast Austenitic Stainless Steel
- Nickel Based Alloys
- Wrought Austenitic Stainless Steel

Environments

The NMP1 Reactor Pressure Vessel Internals components are exposed to the following environments:

- Treated Water or Steam, temperature ≥482°F
- Treated Water or Steam, High temperature, Neutron Fluence < 5×10^{20} n/cm^2. – BWR Reactor Vessel Internals
- Treated Water or Steam, High temperature, Neutron Fluence ≥5×10^{20} n/cm^2. – BWR Reactor Vessel Internals

Aging Effects Requiring Management

The following aging effects, associated with the NMP1 Reactor Pressure Vessel Internals, requires management:

- Cracking
- Cumulative Fatigue Damage

Aging Management Programs

The following aging management programs manage the aging effects for the NMP1 Reactor Pressure Vessel Internals components:

- BWR Vessel Internals Program
- Fatigue Monitoring Program

- Water Chemistry Control Program

3.1.2.A.3 NMP1 REACTOR PRESSURE VESSEL INSTRUMENTATION SYSTEM

Materials

The materials of construction for the NMP1 Reactor Pressure Vessel Instrumentation System components are:

- Any (this applies to NSR piping, fittings, and equipment)
- Carbon or Low Alloy Steel(Yield Strength < 100 Ksi)
- Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)
- Wrought Austenitic Stainless Steel

Environments

The NMP1 Reactor Pressure Vessel Instrumentation System components are exposed to the following environments:

- Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F
- Treated Water, temperature < 140°F, Low Flow
- Treated Water, temperature ≥140°F, but < 212°F, Low Flow
- Treated Water or Steam, temperature ≥482°F, Low Flow

Aging Effects Requiring Management

The following aging effects, associated with the NMP1 Reactor Pressure Vessel Instrumentation System, require management:

- Cracking
- Cumulative Fatigue Damage
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the NMP1 Reactor Pressure Vessel Instrumentation System components:

- ASME Section XI, Subsections IWB, IWC, & IWD, Inservice Inspection Program
- Fatigue Monitoring Program
- One-Time Inspection Program
- Systems Walkdown Program
- Water Chemistry Program

3.1.2.A.4 NMP1 REACTOR RECIRCULATION SYSTEM

Materials

The materials of construction for the NMP1 Reactor Recirculation System components are:

- Any (this applies to NSR piping, fittings, and equipment)
- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)
- Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)
- Cast Austenitic Stainless Steel
- Wrought Austenitic Stainless Steel

Environments

The NMP1 Reactor Recirculation System components are exposed to the following environments:

- Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F
- Treated Water, temperature < 140°F, Low Flow
- Treated Water or Steam, temperature > 482°F

- Treated Water or Steam, temperature > 482°F, Low Flow

Aging Effects Requiring Management

The following aging effects, associated with the NMP1 Reactor Recirculation System, require management:

- Cracking
- Cumulative Fatigue Damage
- Loss of Fracture Toughness
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the NMP1 Reactor Recirculation System components:

- ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program
- BWR Stress Corrosion Cracking Program
- Fatigue Monitoring Program
- One-Time Inspection Program
- Water Chemistry Control Program

3.1.2.A.5 NMP1 CONTROL ROD DRIVE SYSTEM

Materials

The materials of construction for the NMP1 Control Rod Drive System components are:

- Any (this applies to NSR piping, fittings, and equipment)
- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)

- Cast Austenitic Stainless Steel
- Copper Alloys (Zinc > 15%) and Aluminum Bronze
- Wrought Austenitic Stainless Steel

Environments

The NMP1 Control Rod Drive System components are exposed to the following environments:

- Dried Air or Gas
- Treated Water, temperature ≥140°F, but < 212°F
- Treated Water, temperature ≥140°F, but < 212°F, Low Flow

Aging Effects Requiring Management

The following aging effects, associated with the NMP1 Control Rod Drive System, require management:

- Cracking
- Cumulative Fatigue Damage
- Loss of Material

Aging Management Programs

The following aging management programs manage the aging effects for the NMP1 Control Rod Drive System components:

- ASME Section XI, Subsections IWB, IWC, & IWD, Inservice Inspection Program
- One-Time Inspection Program
- Selective Leaching of Materials Program
- Systems Walkdown Program
- Water Chemistry Program

3.1.2.B NMP2 MATERIALS, ENVIRONMENTS, AGING EFFECTS REQUIRING MANAGEMENT AND AGING MANAGEMENT PROGRAMS

3.1.2.B.1 NMP2 REACTOR PRESSURE VESSEL

Materials

The materials of construction for the NMP2 Reactor Pressure Vessel components are:

- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)
- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)
- Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)
- Nickel Based Alloys
- Wrought Austenitic Stainless Steel

Environments

The NMP2 Reactor Pressure Vessel components are exposed to the following environments:

- Air With Thermal Fatigue
- Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F, Leaking Fluid
- Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel
- Treated Water or Steam, High temperature, Neutron Fluence $\geq 1x10^{17} n/cm^2$. – BWR Reactor Pressure Vessel

Aging Effects Requiring Management

The following aging effects, associated with the NMP2 Reactor Pressure Vessel, require management:

- Cracking

- Cumulative Fatigue Damage
- Loss of Fracture Toughness
- Loss of Material

Aging Management Programs

The following aging management programs manage the aging effects for the NMP2 Reactor Pressure Vessel components:

- ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program
- BWR Penetrations Program
- BWR Stress Corrosion Cracking Program
- BWR Vessel ID Attachment Welds Program
- Fatigue Monitoring Program
- Reactor Head Closure Studs Program
- Reactor Vessel Surveillance Program
- Systems Walkdown Program
- Water Chemistry Control Program

3.1.2.B.2 NMP2 REACTOR PRESSURE VESSEL INTERNALS

Materials

The materials of construction for the NMP2 Reactor Pressure Vessel Internals components are:

- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)
- Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)
- Cast Austenitic Stainless Steel
- Nickel Based Alloys

- Wrought Austenitic Stainless Steel

Environment

The NMP2 Reactor Pressure Vessel Internals components are exposed to the following environment:

- Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel
- Treated Water or Steam, High temperature, Neutron Fluence $\geq 1x10^{17} n/cm^2$. – BWR Reactor Pressure Vessel
- Treated Water or Steam, High temperature, Neutron Fluence $< 5x10^{20} n/cm^2$. – BWR Reactor Vessel Internals
- Treated Water or Steam, High temperature, Neutron Fluence $\geq 5x10^{20} n/cm^2$. – BWR Reactor Vessel Internals
- Treated Water or Steam, temperature ≥482°F

Aging Effect Requiring Management

The following aging effect, associated with the NMP2 Reactor Pressure Vessel Internals, requires management:

- Cracking
- Cumulative Fatigue Damage

Aging Management Programs

The following aging management programs manage the aging effect for the NMP2 Reactor Pressure Vessel Internals components:

- BWR Vessel Internals Program
- Fatigue Monitoring Program
- Water Chemistry Control Program

3.1.2.B.3 NMP2 REACTOR PRESSURE VESSEL INSTRUMENTATION SYSTEM

Materials

The materials of construction for the NMP2 Reactor Pressure Vessel Instrumentation System components are:

- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)
- Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)
- Martensitic, Precipitation Hardenable, and Superferritic Stainless Steels
- Nickel Based Alloys
- Wrought Austenitic Stainless Steel

Environments

The NMP2 Reactor Pressure Vessel Instrumentation System components are exposed to the following environments:

- Air
- Air, Moisture or Wetting, temperature ≥140°F
- Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F
- Treated Water, temperature < 140°F, Low Flow
- Treated Water or Steam, temperature ≥482°F, Low Flow

Aging Effects Requiring Management

The following aging effects, associated with the NMP2 Reactor Pressure Vessel Instrumentation System, require management:

- Cracking
- Cumulative Fatigue Damage
- Loss of Material

- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the NMP2 Reactor Pressure Vessel Instrumentation System components:

- ASME Section XI, Subsections IWB, IWC, & IWD, Inservice Inspection Program
- Fatigue Monitoring Program
- One-Time Inspection Program
- Systems Walkdown Program
- Water Chemistry Program

3.1.2.B.4 NMP2 REACTOR RECIRCULATION SYSTEM

Materials

The materials of construction for the NMP2 Reactor Recirculation System components are:

- Any (this applies to NSR piping, fittings, and equipment)
- Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)
- Cast Austenitic Stainless Steel
- Nickel Based Alloys
- Wrought Austenitic Stainless Steel

Environments

The NMP2 Reactor Recirculation System components are exposed to the following environments:

- Air
- Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F

- Hydraulic Fluid
- Treated Water, temperature < 140°F, Low Flow
- Treated Water, temperature ≥140°F, but < 212°F, Low Flow
- Treated Water or Steam, temperature ≥482°F
- Treated Water or Steam, temperature ≥482°F, Low Flow

Aging Effects Requiring Management

The following aging effects, associated with the NMP2 Reactor Recirculation System, require management:

- Cracking
- Cumulative Fatigue Damage
- Loss of Fracture Toughness
- Loss of Material
- Loss of Preload

Aging Management Programs

The following aging management programs manage the aging effects for the NMP2 Reactor Recirculation System components:

- ASME Section XI, Subsections IWB, IWC, & IWD, Inservice Inspection Program
- BWR Stress Corrosion Cracking Program
- Fatigue Monitoring Program
- One-Time Inspection Program
- Systems Walkdown Program
- Water Chemistry Program

3.1.2.B.5 NMP2 CONTROL ROD DRIVE SYSTEM

Materials

The materials of construction for the NMP2 Control Rod Drive System components are:

- Any (this applies to NSR piping, fittings, and equipment)
- Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)
- Cast Austenitic Stainless Steel
- Copper Alloys (Zinc ≤15%)
- Wrought Austenitic Stainless Steel

Environments

The NMP2 Control Rod Drive System components are exposed to the following environments:

- Air
- Dried Air or Gas
- Treated Water, temperature < 140°F, Low Flow
- Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow

Aging Effects Requiring Management

The following aging effects, associated with the NMP2 Control Rod Drive System, require management:

- Cracking
- Cumulative Fatigue Damage
- Loss of Material

Aging Management Programs

The following aging management programs manage the aging effects for the NMP2 Control Rod Drive System components:

- ASME Section XI, Subsections IWB, IWC, & IWD, Inservice Inspection Program
- Fatigue Monitoring Program
- One-Time Inspection Program
- Water Chemistry Program

3.1.3 TIME-LIMITED AGING ANALYSES

The Time-Limited Aging Analyses (TLAAs) identified below are associated with the RCS components. The section of the LRA that contains the TLAA review results is indicated in parenthesis.

- Reactor Vessel Neutron Embrittlement (Section 4.2)
- Metal Fatigue Analysis (Section 4.3)
- NMP2 Core Plate Holdown Bolts (Section 4.7.3)
- NMP1 Reactor Vessel Weld Flaw Evaluation (Section 4.7.4)

3.1.4 CONCLUSIONS

The RCS components that are subject to aging management review have been identified in accordance with the requirements of 10 CFR 54.4. The aging management programs selected to manage aging effects for the RCS components are identified in the summary tables and Section 3.1.2. A description of these aging management programs is provided in Appendix B, along with the demonstration that the identified aging effects will be managed for the period of extended operation. Therefore, based on the demonstrations provided in Appendix B, the effects of aging associated with the RCS components will be adequately managed so that there is reasonable assurance that the intended function(s) will be maintained consistent with the current licensing basis during the period of extended operation.

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-01	Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Consistent with NUREG-1801. The TLAA is further evaluated in Section 4.3. Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Condensing pots • Control Rod Drive (CRD) Assemblies • Control Rod Drive Return Line nozzle thermal sleeves • Core Shroud support plates, rings, and welds • Core Shroud head bolts and collars • Core Spray nozzles, Emergency Condenser Steam outlet nozzles and Reactor Recirculation nozzles • Feedwater nozzle thermal sleeves • Flow elements • Instrumentation Penetrations • Orifices in the NMP1 Shutdown Cooling System (see Table 3.3.2.A-20) (continued on next page)

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-01 (cont'd)	Reactor coolant pressure boundary components (cont'd)	Cumulative fatigue damage (cont'd)	TLAA, evaluated in accordance with 10 CFR 54.21(c) (cont'd)	Yes, TLAA (cont'd)	Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Steam Dryers • Main Steam, Core Differential Pressure, Core Spray, Emergency Condenser Steam Feedwater, Reactor Recirculation, and Safety Valve nozzle safe ends • Temperature Equalizing Columns • Top Head Enclosure with cladding • Top Head Nozzles • Vessel Drain Penetrations • Vessel Welds
3.1.1.A-02	PWR only				
3.1.1.A-03	Isolation Condenser	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection; water chemistry	Yes, plant specific	NMP1 is consistent with NUREG-1801 with the exception that eddy current testing of the tubes cannot be performed due to all welded fabrication of the condenser, i.e., there is no access to the condenser tubes. Continuous radioactivity monitoring of the condenser vent is provided in the Control Room. Temperature monitoring is conducted by a Preventive Maintenance Program (B2.1.32) procedure.

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-04	Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm2 (E>1MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10CFR50 and RG 1.99	Yes, TLAA	The only RCS components with this environment and aging effect are the Reactor Vessel beltline shell and welds which are addressed in row 3.1.1.A-05. The TLAA is further evaluated in Section 4.2.
3.1.1.A-05	Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.19). Further evaluation is documented in Section 4.2 and B2.1.19 (Reactor Vessel Surveillance Program).
3.1.1.A-06	PWR only				

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-07	Small-bore reactor coolant system and connected systems piping	Crack initiation and growth due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), and thermal and mechanical loading	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/ inspected and detection of aging effects are to be further evaluated	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.1 and B2.1.2). Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Accumulators • Condensing Pots • CRD System filters • Reactor Vessel Instrumentation Valves • Small bore valves • Temperature Equalizing Columns A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted. This will be performed as part of a one-time inspection that will be conducted to verify that service-induced weld cracking is not occurring in the small-bore piping. Additionally, for small bore piping and fittings in the NMP1 CRD System that are not part of the Inservice Inspection Testing Program, NMP1 only credits the Water Chemistry and One-Time Inspection Programs.

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-08	Jet pump sensing line and reactor vessel flange leak detection line	Crack initiation and growth due to SCC, IGSCC, or cyclic loading	Plant specific	Yes, plant specific	NMP1 does not have jet pumps; therefore, there are no jet pump sensing lines. The vessel flange leak detection lines for NMP1 are not within the scope of license renewal.
3.1.1.A-09	Isolation Condenser	Crack initiation and growth due to SCC or cyclic loading	Inservice inspection; water chemistry	Yes, plant specific	NMP1 is consistent with NUREG-1801 with the exception that eddy current testing of the tubes cannot be performed due to all welded fabrication of the condenser, i.e., there is no access to the condenser tubes. Continuous radioactivity monitoring of the condenser vent is provided in the Control Room. Temperature monitoring is conducted by a Preventive Maintenance Program (B2.1.32) procedure. Further evaluation is documented in Appendix B2.1.2 (Water Chemistry Control Program) and B2.1.32 (Preventive Maintenance Program).
3.1.1.A-10	PWR only				
3.1.1.A-11	PWR only				
3.1.1.A-12	PWR only				
3.1.1.A-13	PWR only				
3.1.1.A-14	PWR only				
3.1.1.A-15	PWR only				

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-16	PWR only				
3.1.1.A-17	PWR only				
3.1.1.A-18	PWR only				
3.1.1.A-19	PWR only				
3.1.1.A-20	PWR only				
3.1.1.A-21	PWR only				
3.1.1.A-22	Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No	Consistent with NUREG-1801: NMP1 credits the Fatigue Monitoring Program (Section B3.2) for Closure Head Studs and Nuts that have an aging effect of cumulative fatigue damage, and the Reactor Head Closure Studs Program (Section B2.1.3) for Closure Head Studs and Nuts that have an aging effect/mechanism of loss of material due to general corrosion.
3.1.1.A-23	CASS pump casing and valve body pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.1).
3.1.1.A-24	CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of Cast Austenitic Stainless Steel	No	Not applicable because this component does not exist at NMP1.

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-25	BWR piping and fittings; steam generator components	Wall thinning due to flow accelerated corrosion	Flow accelerated corrosion	No	Consistent with NUREG-1801 for valves with this aging effect/mechanism that are part of the Reactor Coolant Pressure Boundary (Note: NUREG-1801 Volume 2 Item IV.C1.3-a, which applies to this row number, addresses valves). Additionally, NMP1 Main Steam flow elements, which are part of the NMP1 Reactor Coolant Pressure Boundary, are consistent with, but not addressed in, NUREG-1801. NUREG-1801 items that identify FAC as a mechanism are IV.C1.1-a (main steam piping and fittings), IV.C1.1-c (feedwater piping and fittings), and IV.C1.3-a (carbon steel valves). Items IV.C1.1-a and IV.C1.1-c are addressed in Steam and Power Conversion Systems Tables 3.4.2.A-2 and 3.4.2.A-4. Not applicable for steam generator components because NMPNS does not have steam generators.

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-26	Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high- pressure and high-temperature systems	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	Consistent with NUREG-1801, with the following exceptions: • NMP1 credits the ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program (ISI Program) in lieu of the Bolting Integrity Program to manage the aging effect of loss of material and loss of preload in a high-pressure and high-temperature environment. As noted in Appendix B2.1.1, the ISI Program manages aging of pressure-retaining components including bolting. • Not applicable for the pressurizer bolting because this component does not exist at NMP1.
3.1.1.A-27	Feedwater and control rod drive (CRD) return line nozzles	Crack initiation and growth due to cyclic loading	Feedwater nozzle; CRD return line nozzle	No	Consistent with NUREG-1801 with the following exception: NMP1 credits the ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program (ISI Program) for management of crack initiation and growth due to cyclic loading for the control rod drive return line nozzles. All requirements of NUREG-1801 Program XI.M6 (Control Rod Drive Return Line Program) are implemented under the NMP1 ISI Program Plan.
3.1.1.A-28	Vessel shell attachment welds	Crack initiation and growth due to SCC and/or IGSCC	BWR vessel ID attachment welds; water chemistry	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.2).

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-29	Nozzle safe ends, recirculation pump casing, connected systems piping and fittings, body and bonnet of valves	Crack initiation and growth due to SCC and/or IGSCC	BWR stress corrosion cracking; water chemistry	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.6 and B2.1.2). Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Core Differential Pressure nozzle safe end • Emergency Condenser Steam nozzle safe ends • Flow elements • Reactor Recirculation nozzle safe ends • Safety Valve nozzle safe ends
3.1.1.A-30	Penetrations	Crack initiation and growth due to SCC, IGSCC, and/or cyclic loading	BWR bottom head penetrations; water chemistry	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.2).

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-31	Core shroud and core plate, support structure, top guide, core spray lines and spargers, jet pump assemblies, control rod drive housing, and nuclear instrumentation guide tubes	Crack initiation and growth due to SCC, IGSCC, and/or IASCC	BWR vessel internals; water chemistry	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.2). Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Control Rod Guide tubes • Core Shroud clamps, spacers, support rings, and tie rods • Core Shroud head bolts and collars • Steam Dryers
3.1.1.A-32	Core shroud and core plate access hole cover (welded and mechanical covers)	Crack initiation and growth due to SCC, IGSCC, and/or IASCC	ASME Section XI inservice inspection; water chemistry	No	The core shroud and supporting components that have this aging effect/mechanism are evaluated in row 3.1.1.A-31 since NMP1 credits the BWR Vessel Internals Program (Section B2.1.8) and Water Chemistry Control Program (Section B2.1.2) for managing the aging effects for these components. Not applicable for the core plate access hole cover since this component does not exist at NMP1. Additionally, the NMP1 Feedwater Sparger thermal sleeves are consistent with, but not addressed in, NUREG-1801.

Table 3.1.1.A NMP1 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.A-33	Jet pump assembly castings and orificed fuel support	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal aging and neutron irradiation embrittlement	No	This item is not applicable for the jet pump components since NMP1 does not have jet pumps. Aging management of the orificed fuel supports is conducted in accordance with BWRVIP-47 of the BWR Vessel Internals Program, XI.M9.
3.1.1.A-34	Unclad top head and nozzles	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection; water chemistry	No	Not applicable because NMP1 has a cladded top head enclosure and nozzles. The NMP1 top head and nozzles are evaluated in row 3.1.1.A-01.
3.1.1.A-35	PWR only				
3.1.1.A-36	PWR only				
3.1.1.A-37	PWR only				
3.1.1.A-38	PWR only				
3.1.1.A-39	PWR only				
3.1.1.A-40	PWR only				
3.1.1.A-41	PWR only				
3.1.1.A-42	PWR only				
3.1.1.A-43	PWR only				
3.1.1.A-44	PWR only				
3.1.1.A-45	PWR only				
3.1.1.A-46	PWR only				
3.1.1.A-47	PWR only				
3.1.1.A-48	PWR only				

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-01	Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Consistent with NUREG-1801. The TLAA is further evaluated in Section 4.3. Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Core Spray, Drain, Jet Pump Instrumentation, Reactor Recirculation, Residual Heat Removal , spray nozzles, Top Head, and vent nozzles • Core Spray, CRD Return Line, Feedwater, Main Steam, Jet Pump Instrumentation, Residual Heat Removal, and Reactor Recirculation nozzle safe ends • Core Spray, CRD Return Line, Feedwater, Residual Heat Removal, and Reactor Recirculation nozzle thermal sleeves • CRD housings • Drain line penetrations • Head bolts • Instrumentation penetrations • Leak detection lines (continued on next page)

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-01 (cont'd)	Reactor coolant pressure boundary components (cont'd)	Cumulative fatigue damage (cont'd)	TLAA, evaluated in accordance with 10 CFR 54.21(c) (cont'd)	Yes, TLAA (cont'd)	The following components are consistent with, but not addressed in, NUREG-1801: • Main Steam flow elements, condensing chambers, and restriction orifices (see Table 3.4.2.B-3) • Steam Dryers • Stub tube welds • Top Head Enclosure without cladding • Vessel welds
3.1.1.B-02	PWR only				
3.1.1.B-03	Isolation Condenser	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection; water chemistry	Yes, plant specific	Not applicable because this component does not exist at NMP2.
3.1.1.B-04	Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm2 (E>1MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10CFR50 and RG 1.99	Yes, TLAA	The evaluation of the RHR/LPCI nozzle (NUREG-1801 item IV.A1.3-c) will be completed prior to the period of extended operation.
3.1.1.B-05	Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.19). Further evaluation is documented in Section 4.2 and Section B2.1.19 (Reactor Vessel Surveillance Program).
3.1.1.B-06	PWR only				

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-07	Small-bore reactor coolant system and connected systems piping	Crack initiation and growth due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), and thermal and mechanical loading	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/ inspected and detection of aging effects are to be further evaluated	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.1 and B2.1.2). Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Accumulators • Condensing chambers • Control Rod Hydraulic Control Units • Flow elements in the Reactor Water Cleanup System (see Table 3.3.2.B-24) • Restriction orifices • Valves A plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping is to be conducted. This will be performed as part of a one-time inspection that will be conducted to verify that service-induced weld cracking is not occurring in the small-bore piping. Additionally, for small bore piping and fittings in the NMP2 Reactor Vessel Instrumentation, Reactor Recirculation, and CRD Systems that are not part of the Inservice Inspection Testing Program, NMP2 only credits the Water Chemistry and One-Time Inspection Programs.

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-08	Jet pump sensing line and reactor vessel flange leak detection line	Crack initiation and growth due to SCC, IGSCC, or cyclic loading	Plant specific	Yes, plant specific	For NMP2, the jet pump sensing lines are not within scope of license renewal. The vessel flange leak detection lines are comprised of both stainless steel and carbon steel components. The design internal environment for these lines is air. They contain water only during refueling operations and undergo an ASME pressure test following refueling. These lines are managed by the ASME Section XI, One-Time Inspection and System Walkdown programs. The carbon steel line sections are managed for loss of material instead of cracking.
3.1.1.B-09	Isolation Condenser	Crack initiation and growth due to SCC or cyclic loading	Inservice inspection; water chemistry	Yes, plant specific	Not applicable because this component does not exist at NMP2.
3.1.1.B-10	PWR only				
3.1.1.B-11	PWR only				
3.1.1.B-12	PWR only				
3.1.1.B-13	PWR only				
3.1.1.B-14	PWR only				
3.1.1.B-15	PWR only				
3.1.1.B-16	PWR only				
3.1.1.B-17	PWR only				
3.1.1.B-18	PWR only				
3.1.1.B-19	PWR only				

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-20	PWR only				
3.1.1.B-21	PWR only				
3.1.1.B-22	Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No	Consistent with NUREG-1801. NMP2 credits the Fatigue Monitoring Program (Section B3.2) for Closure Head Studs and Nuts that have an aging effect of cumulative fatigue damage, and the Reactor Head Closure Studs Program (Section B2.1.3) for Closure Head Studs and Nuts that have an aging effect/mechanism of loss of material due to general corrosion and SCC.
3.1.1.B-23	CASS pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.1).
3.1.1.B-24	CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of Cast Austenitic Stainless Steel	No	Not applicable because this component does not exist at NMP2.

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-25	BWR piping and fittings; steam generator components	Wall thinning due to flow accelerated corrosion	Flow accelerated corrosion	No	Consistent with NUREG-1801 for valves with this aging effect/mechanism that are part of the Reactor Coolant Pressure Boundary (Note: NUREG-1801 Volume 2 Item IV.C1.3-a, which applies to this row number, addresses valves). Additionally, NMP2 Main Steam flow elements, which are part of the NMP2 Reactor Coolant Pressure Boundary, are consistent with, but not addressed in, NUREG-1801. NUREG-1801 items that identify FAC as a mechanism are IV.C1.1-a (main steam piping and fittings), IV.C1.1-c (feedwater piping and fittings), and IV.C1.3-a (carbon steel valves). Items IV.C1.1-a and IV.C1.1-c are addressed in Steam and Power Conversion Systems Tables 3.4.2.B-3 and 3.4.2.B-4. Not applicable for steam generator components because NMPNS does not have steam generators.

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-26	Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high- pressure and high-temperature systems	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No	Consistent with NUREG-1801, with the following exceptions: • NMP2 credits the ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program (ISI Program) in lieu of the Bolting Integrity Program to manage the aging effect of loss of material and loss of preload in a high-pressure and high-temperature environment. As noted in Appendix B2.1.1, the ISI Program manages aging of pressure-retaining components including bolting. • Not applicable for the pressurizer bolting because this component does not exist at NMP2.

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-27	Feedwater and control rod drive (CRD) return line nozzles	Crack initiation and growth due to cyclic loading	Feedwater nozzle; CRD return line nozzle	No	NMP2 takes exception to NUREG-1801. NMP2 credits the ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program (ISI Program) for management of crack initiation and growth due to cyclic loading for the feedwater and control rod drive return line nozzles. The feedwater nozzles employ the improved interference fit (triple-sleeve) sparger design that was generically approved by the NRC, as documented in NUREG-0619. The improved thermal sleeve design is considered to have greatly reduced susceptibility to the type of cracking addressed in NUREG-0619; therefore, the augmented inspections required by NUREG-1801 Program XI.M5, Feedwater Nozzle, are not required. The NMP2 Control Rod Drive Return Line Nozzle has been cut and capped and is not used, so it is also less susceptible to cracking. Therefore, NMP2 does not credit program XI.M6, "BWR Control Rod Drive Return Line Nozzle."

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-28	Vessel shell attachment welds	Crack initiation and growth due to SCC and/or IGSCC	BWR vessel ID attachment welds; water chemistry	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.2). Additionally, the NMP2 Stub tube welds are consistent with, but not addressed in, NUREG-1801. For Reactor Vessel nickel based alloy weld overlays, NMP2 credits the ISI program (Appendix B2.1.1) in lieu of the BWR Vessel ID Attachment Welds Program. These welds are currently part of the ISI program, and this program adequately manages the aging effects for these components.
3.1.1.B-29	Nozzle safe ends, recirculation pump casing, connected systems piping and fittings, body and bonnet of valves	Crack initiation and growth due to SCC and/or IGSCC	BWR stress corrosion cracking; water chemistry	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.6 and B2.1.2). Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Feedwater, Jet Pump, Instrumentation, and Reactor Recirculation nozzle safe ends • Feedwater nozzle safe end inserts • Main Steam condensing chambers and restriction orifices (see Table 3.4.2.B-3)
3.1.1.B-30	Penetrations	Crack initiation and growth due to SCC, IGSCC, and/or cyclic loading	BWR bottom head penetrations; water chemistry	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.2).

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-31	Core shroud and core plate, support structure, top guide, core spray lines and spargers, jet pump assemblies, control rod drive housing, and nuclear instrumentation guide tubes	Crack initiation and growth due to SCC, IGSCC, and/or IASCC	BWR vessel internals; water chemistry	No	Consistent with NUREG-1801 with exceptions (see Appendix B2.1.2). Additionally, the following components are consistent with, but not addressed in, NUREG-1801: • Access hole covers • Clamps and keepers • Core spray line brackets • Differential Pressure Liquid Control lines • Flanges • Peripheral Fuel supports • Head bolts • Steam Dryers • Core Spray and Feedwater nozzle thermal sleeves and extensions • CRD Return Line and Residual Heat Removal nozzle thermal sleeves
3.1.1.B-32	Core shroud and core plate access hole cover (welded and mechanical covers)	Crack initiation and growth due to SCC, IGSCC, and/or IASCC	ASME Section XI inservice inspection; water chemistry	No	The core shroud, access hole cover, and supporting components that have this aging effect/mechanism are evaluated in row 3.1.1.B-31 since NMP2 credits the BWR Vessel Internals Program (Appendix B2.1.8) and Water Chemistry Control Program (Appendix B2.1.2) for managing the aging effects for these components.

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-33	Jet pump assembly castings and orificed fuel support	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement	Thermal aging and neutron irradiation embrittlement	No	The jet pumps and this aging effect are managed by BWRVIP-41 of the BWR Vessel Internals Program, XI.M9. Aging management of the orificed fuel supports is conducted in accordance with BWRVIP-47 of the BWR Vessel Internals Program, XI.M9.
3.1.1.B-34	Unclad top head and nozzles	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection; water chemistry	No	Consistent with BWRVIP-74-A, Table 3-1, "BWR RPV Aging Mechanism Assessment Summary," NMP2 is managing the only aging effect, cumulative fatigue damage, identified for the RPV nozzles with the Fatigue Monitoring Program. NMP2 is also managing the unclad top head for this aging effect even though it is not identified for the top head in that table. Additionally, consistent with NUREG-1801, the unclad top head and RPV nozzles are also being managed for loss of material by the ASME Inservice Inspection and Water Chemistry Programs; therefore, credit will be taken for implementation of the NUREG-1801 guidance.

Table 3.1.1.B NMP2 Summary of Aging Management Programs for the Reactor Vessel, Internals, and Reactor Coolant Systems Evaluated in Chapter IV of NUREG-1801

Item Number	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended	Discussion
3.1.1.B-35	PWR only				
3.1.1.B-36	PWR only				
3.1.1.B-37	PWR only				
3.1.1.B-38	PWR only				
3.1.1.B-39	PWR only				
3.1.1.B-40	PWR only				
3.1.1.B-41	PWR only				
3.1.1.B-42	PWR only				
3.1.1.B-43	PWR only				
3.1.1.B-44	PWR only				
3.1.1.B-45	PWR only				
3.1.1.B-46	PWR only				
3.1.1.B-47	PWR only				

Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Bottom Head	PB SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.6-a	3.1.1.A-01	A
Nozzles	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-a	3.1.1.A-01	C, 1
						IV.A1.3-d	3.1.1.A-01	A
				Cracking	BWR Feedwater Nozzle Program	IV.A1.3-b	3.1.1.A-27	A
					BWR CRD Return Line Nozzle Program	IV.A1.3-c	3.1.1.A-27	E
Nozzle Safe Ends	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-a	3.1.1.A-01	C, 2

Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Nozzle Safe Ends (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.A1.4-a	3.1.1.A-29	B D, 3
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.A-01	A C, 4
Penetrations: • Core Differential Pressure • CRD Stub Tube • Flux Monitor • Instrumentation • Vessel Drain	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, temperature ≥482°F, Low Flow	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-a	3.1.1.A-01	C, 54
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program Water Chemistry Control Program			M
		Nickel Based Alloys; Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature – BWR Reactor Pressure Vessel	Cracking	BWR Penetrations Program Water Chemistry Control Program	IV.A1.5-a	3.1.1.A-30	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.5-b	3.1.1.A-01	A

Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Penetrations (cont'd)	PB (cont'd)	Nickel Based Alloys; Wrought Austenitic Stainless Steel (cont'd)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel (cont'd)	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.A1.4-a	3.1.1.A-29	D, 5
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.A-01	C, 5
						IV.A1.5-b	3.1.1.A-01	A
Support Skirt and Attachment Welds	SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Air, Moisture or Wetting, temperature <140°F	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program			G
			Air, Moisture or Wetting, Temperature ≥212°F	Cumulative Fatigue Damage	Fatigue Monitoring Program		3.1.1.A-01	G
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program			G

Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Thermal Sleeves	PB	Nickel Based Alloys	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.4-b	3.1.1.A-01	C, 6
				Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program Water Chemistry Control Program	IV.B1.1-e	3.1.1.A-32	D, 6
	PB TS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	Water Chemistry Control Program			Q, 68
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-d	3.1.1.A-01	C, 68
	TS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.3-b	3.1.1.A-01	A
				Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.3-a	3.1.1.A-31	B
Top Head	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.1-b	3.1.1.A-01	C, 8

Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Top Head (Closure Studs and Nuts)	PB	Carbon or Low Alloy Steel (Yield Strength ≥ 100 Ksi)	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥ 212°F, Leaking Fluid	Cracking	Reactor Head Closure Studs Program	IV.A1.1-c	3.1.1.A-22	B
				Loss of Material	Reactor Head Closure Studs Program			L
				Cumulative Fatigue Damage	Fatigue Monitoring Program			H
Top Head (Flanges)	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.1-b	3.1.1.A-01	A
Top Head (Nozzles)	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-d	3.1.1.A-01	C, 10

Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, temperature ≥482°F, Low Flow	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.A-01	A
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program Water Chemistry Control Program			H
Vessel Shell (Flange)	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-a	3.1.1.A-01	A
Vessel Shells • Beltline • Lower Shell • Upper Nozzle Shell • Upper RPV Shell	PB SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-a	3.1.1.A-01	A
			Treated Water or Steam, High temperature, Neutron Fluence ≥$1x10^{17}$n/cm^2. - BWR Reactor Pressure Vessel	Loss of Fracture Toughness	Reactor Vessel Surveillance Program	IV.A1.2-d	3.1.1.A-05	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-b	3.1.1.A-01	A

Table 3.1.2.A-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Vessel Shell Welds (including attachment welds)	PB SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High temperature, Neutron Fluence $\geq 1x10^{17} n/cm^2$. - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-b	3.1.1.A-01	A
				Loss of Fracture Toughness	Reactor Vessel Surveillance Program	IV.A1.2-d	3.1.1.A-05	B
		Nickel Based Alloys	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Vessel ID Attachment Welds Program Water Chemistry Control Program	IV.A1.2-e	3.1.1.A-28	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.A-01	C, 11
	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence $\geq 1x10^{17} n/cm^2$. - BWR Reactor Pressure Vessel	Cracking	BWR Vessel ID Attachment Welds Program Water Chemistry Control Program	IV.A1.2-e	3.1.1.A-28	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.A-01	C, 11

Table 3.1.2.A-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
CRD Assemblies (includes drive mechanism and housing)	PB SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.5-b	3.1.1.A-01	C, 12
				Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.5-c	3.1.1.A-31	B
Control Rod Guide Tubes	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.5-c	3.1.1.A-31	D, 13
Core Plate and Bolts	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-b	3.1.1.A-31	B

Table 3.1.2.A-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Core Shroud	DF SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence ≥5x10^{20} n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-a	3.1.1.A-31	B
Core Shroud Head Bolts and Collars	SFS	Nickel Based Alloys	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-f	3.1.1.A-31	D, 14
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.4-b	3.1.1.A-01	C, 14
		Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-b	3.1.1.A-31	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.1-c	3.1.1.A-01	C, 55

Table 3.1.2.A-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Core Shroud Support Structures • Clamps • Core Plate Spacers • Support Plates • Support Rings • Support Welds • Tie Rod Assemblies	SFS	Nickel Based Alloys	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-f	3.1.1.A-31	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.4-b	3.1.1.A-01	C, 16
			Treated Water or Steam, High temperature, Neutron Fluence $\geq 5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-f	3.1.1.A-31	D, 15
		Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-b	3.1.1.A-31	D, 56
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.1-c	3.1.1.A-31	C, 56

Table 3.1.2.A-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Core Shroud Support Structures (cont'd)	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence $\geq 5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-a	3.1.1.A-31	D, 15, 57
Core Spray Lines and Spargers	DF, SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.3-a	3.1.1.A-31	B
In-core Instrumentation Dry Tubes and Guide Tubes	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence $\geq 5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internal	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.6-a	3.1.1.A-31	B

**Table 3.1.2.A-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation**

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Orificed Fuel Support	DF SFS	Cast Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Loss of Fracture Toughness	BWR Vessel Internals Program	IV.B1.5-a	3.1.1.A-33	E
Steam Dryer	NSR Functional Support	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.1-c	3.1.1.A-01	C, 17
				Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-a	3.1.1.A-31	D, 17
Top Guides	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence ≥$5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.2-a	3.1.1.A-31	B

Table 3.1.2.A-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Closure Bolting	PB	Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-g	3.1.1.A-01	A
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.3-e	3.1.1.A-26	E
				Loss of Preload	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.3-f	3.1.1.A-26	E
Condensing Pots	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F, Low Flow	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.1-h	3.1.1.A-01	C, 18
				Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 18
NSR piping, fittings, and equipment	PFASRE	Any	Treated Water, temperature < 140°F, Low Flow	Cracking; Loss of Material	Water Chemistry Control Program			J

Table 3.1.2.A-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings	PB	Wrought Austenitic Stainless Steel	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	B
Temperature Equalizing Columns	PB	Wrought Austenitic Stainless Steel	Air	None	None			None
Valves	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature < 140°F, Low Flow	Loss of Material	One-Time Inspection Program Water Chemistry Control Program			H
		Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None

Table 3.1.2.A-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel (cont'd)	Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 19
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.A-01	A

Table 3.1.2.A-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Closure Bolting	PB	Carbon or Low Alloy Steel (Yield Strength ≥ 100 Ksi)	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.2-f	3.1.1.A-01	A
						IV.C1.3-g	3.1.1.A-01	A
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.2-d	3.1.1.A-26	E
						IV.C1.3-e	3.1.1.A-26	E
				Loss of Preload	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.2-e	3.1.1.A-26	E
						IV.C1.3-f	3.1.1.A-26	E
Flow Elements	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.C1.1-f	3.1.1.A-29	D, 20
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.1-h	3.1.1.A-01	C, 20
NSR piping, fittings, and equipment	PFASRE	Any	Treated Water, temperature < 140°F, Low Flow Treated Water or Steam, temperature ≥482°F	Cracking; Loss of Material	Water Chemistry Control Program			J

Table 3.1.2.A-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
NSR piping, fittings, and equipment (cont'd)	PFASRE (cont'd)	Any (cont'd)	Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking; Loss of Material	Water Chemistry Control Program			J
Piping and Fittings	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.C1.1-f	3.1.1.A-29	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.1-h	3.1.1.A-01	A
				Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	B

Table 3.1.2.A-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Pumps	PB	Cast Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.2-a	3.1.1.A-01	A
				Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.C1.2-b	3.1.1.A-29	B
				Loss of Fracture Toughness	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.2-c	3.1.1.A-23	B
Pump Seal Flanges	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.2-a	3.1.1.A-01	A
				Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program Water Chemistry Control Program			Q

Table 3.1.2.A-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature < 140°F, Low Flow	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			H, 21
					One-Time Inspection Program Water Chemistry Control Program			H
		Cast Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.C1.3-c	3.1.1.A-29	B

**Table 3.1.2.A-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Recirculation System – Summary of Aging Management Evaluation**

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Cast Austenitic Stainless Steel (cont'd)	Treated Water or Steam, temperature ≥482°F (cont'd)	Cracking (cont'd)	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 49
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.A-01	A
				Loss of Fracture Toughness	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.3-b	3.1.1.A-23	B
		Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 49

**Table 3.1.2.A-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Reactor Recirculation System – Summary of Aging Management Evaluation**

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel (cont'd)	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.A-01	A
			Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 49
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.A-01	A

Table 3.1.2.A-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Accumulators	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			P
		Wrought Austenitic Stainless Steel	Dried Air or Gas	None	None			None
			Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 50

**Table 3.1.2.A-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Control Rod Drive System – Summary of Aging Management Evaluation**

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Filters	PB	Cast Austenitic Stainless Steel	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 23
					One-Time Inspection Program Water Chemistry Control Program			Q
NSR piping, fittings, and equipment	PFASRE	Any	Treated Water, temperature ≥140°F, but < 212°F Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking Loss of Material	Systems Walkdown Program Water Chemistry Control Program			J

Table 3.1.2.A-5 Reactor Vessel, Internals, and Reactor Coolant System NMP1 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			H
		Wrought Austenitic Stainless Steel	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	B
					One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	E,24

Table 3.1.2.A-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			H
		Cast Austenitic Stainless Steel	Treated Water, temperature ≥140°F, but < 212°F	Cracking	One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	E, 24
			Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 49
		Copper Alloys (Zinc > 15%) and Aluminum Bronze	Dried Air or Gas	None	None			None

Table 3.1.2.A-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Copper Alloys (Zinc > 15%) and Aluminum Bronze (cont'd)	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program Selective Leaching of Materials Program Water Chemistry Control Program			M
		Wrought Austenitic Stainless Steel	Dried Air or Gas	None	None			None
			Treated Water, temperature ≥140°F, but < 212°F	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 49
					One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	E, 24

Table 3.1.2.A-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP1 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel (cont'd)	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.A-07	D, 49

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Bottom Head	PB SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.6-a	3.1.1.B-01	A
Nozzles	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-a	3.1.1.B-01	A
						IV.A1.3-d	3.1.1.B-01	A
								C, 25
		Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-d	3.1.1.B-01	C, 26
				Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.A1.3-b	3.1.1.B-27	E
						IV.A1.3-c	3.1.1.B-27	E
		Carbon or Low Alloy Steel (Yield Strength ≤ 100 Ksi)	Treated Water or Steam, High temperature, Neutron Fluence $\geq 1 \times 10^{17} n/cm^2$ - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV-A1.3-d	3.1.1.B-01	C
				Loss of Fracture Toughness	TLAA	IV.A1.3-e	3.1.1.B-04	A

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Nozzle Safe Ends	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-a	3.1.1.B-01	C, 27
						IV.A1.3-d	3.1.1.B-01	C, 28
		Nickel Based Alloys	Treated Water or Steam, High Temperature – BWR Reactor Pressure Vessel	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.A1.4-a	3.1.1.B-29	D, 29
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.B-01	C, 29
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature – BWR Reactor Pressure Vessel	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.A1.4-a	3.1.1.B-29	B, 30
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.B-01	C, 30

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Nozzle Thermal Sleeves	PB	Nickel Based Alloys	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.A1.4-a	3.1.1.B-31	E, 31
					Water Chemistry Control Program	IV.A1.4-a	3.1.1.B-31	E, Q, 58
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.B-01	C, 31, 58
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.A1.4-a	3.1.1.B-31	D, 59
					Water Chemistry Control Program	IV.A1.4-a	3.1.1.B-31	E, Q, 60
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.B-01	C, 59, 60

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Penetrations: • Core Differential Pressure and Liquid Control • CRD Stub Tubes • Drain Lines • In-core Instruments • Instrumentation	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-d	3.1.1.B-01	C, 5, 7
		Nickel Based Alloys	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Penetrations Program Water Chemistry Control Program	IV.A1.5-a	3.1.1.B-30	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.5-b	3.1.1.B-01	A
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Penetrations Program Water Chemistry Control Program	IV.A1.5-a	3.1.1.B-30	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.5-b	3.1.1.B-01	A
Support Skirt	SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Air With Thermal Fatigue	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.7-a	3.1.1.B-01	A
				Loss of Material	Systems Walkdown Program			H

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Top Head and Nozzles (unclad)	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.1-b	3.1.1.B-01	C, 8, 33
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program Water Chemistry Control Program	IV.A1.1-a	3.1.1.B-34	B
Top Head (Closure Studs and Nuts)	PB	Carbon or Low Alloy Steel (Yield Strength ≥ 100 Ksi)	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥ 212°F, Leaking Fluid	Cumulative Fatigue Damage	Fatigue Monitoring Program		3.1.1.B-01	H
				Loss of Material	Reactor Head Closure Studs Program			H
				Cracking	Reactor Head Closure Studs Program	IV.A.1.1-c	3.11.A-22	A
Top Head (Flanges)	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.1-b	3.1.1.B-01	A

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Top Head (Leak Detection Lines)	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Air	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program		3.1.1.B-08	G
			Air	Loss of Material	Systems Walkdown Program	VII.I1-b	3.1.1.B-08	H, 45
		Wrought Austenitic Stainless Steel	Air	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program		3.1.1.B-08	G
Vessel Shells (Flange)	PB SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-a	3.1.1.B-01	A

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Vessel Shells • Lower Intermediate Shell • Lower Shell • Upper Intermediate Shell • Upper Shell	PB SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-a	3.1.1.B-01	A
			Treated Water or Steam, High temperature, Neutron Fluence $\geq 1x10^{17} n/cm^2$. – BWR Reactor Pressure Vessel	Loss of Fracture Toughness	Reactor Vessel Surveillance Program	IV.A1.2-d	3.1.1.B-05	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-b	3.1.1.B-01	A
Vessel Welds (including attachment welds)	PB	Nickel Based Alloys	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Vessel ID Attachment Welds Program Water Chemistry Control Program	IV.A1.2-e	3.1.1.B-28	D, 34
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.B-01	C, 34

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Vessel Welds (including attachment welds) (cont'd)	PB SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, High temperature, Neutron Fluence $\geq 1x10^{17} n/cm^2$. – BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-b	3.1.1.B-01	A
				Loss of Fracture Toughness	Reactor Vessel Surveillance Program	IV.A1.2-d	3.1.1.B-05	B
		Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (Clad with Stainless Steel)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-b	3.1.1.B-01	A
	SFS	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.2-b	3.1.1.B-01	C, 11
		Nickel Based Alloys	Treated Water or Steam, High Temperature - BWR Reactor Pressure- Vessel	Cracking	BWR Vessel ID Attachment Welds Program Water Chemistry Control Program	IV.A1.2-e	3.1.1.B-28	B

Table 3.1.2.B-1 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Vessel Welds (including attachment welds) (cont'd)	SFS (cont'd)	Nickel Based Alloys (cont'd)	Treated Water or Steam, High Temperature - BWR Reactor Pressure- Vessel (cont'd)	Cracking (cont'd)	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program Water Chemistry Control Program	IV.A1.2-e	3.1.1.B-28	E
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.B-01	C, 11
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Vessel ID Attachment Welds Program Water Chemistry Control Program	IV.A1.2-e	3.1.1.B-28	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.4-b	3.1.1.B-01	C, 11

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Access Hole Covers	PB	Nickel Based Alloys	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-f	3.1.1.B-31	D, 35
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.2-a	3.1.1.B-31	D, 35
CRD Assemblies (includes drive mechanism and housing)	PB SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.5-b	3.1.1.B-01	C, 36
				Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.5-c	3.1.1.B-31	B

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Control Rod Guide Tubes	PB SFS	Cast Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	None	None			None
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.5-c	3.1.1.B-31	B
Core Plate, Bolts, and Supports	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-b	3.1.1.B-31	B

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Core Shroud	DF SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence $\geq 5 \times 10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-a	3.1.1.B-31	B
Core Shroud Head Bolts	SFS	Nickel Based Alloys	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program			F
				Cracking	BWR Vessel Internals Program Water Chemistry Control Program			F
		Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-b	3.1.1.B-31	D, 37
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.1-c	3.1.1.B-01	C, 37

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Core Shroud Support Structures • Bolts • Brackets • Cap Screws • Clamps • Keepers • Restraints • Supports	SFS	Carbon or Low Alloy Steel (Yield Strength ≥ 100 Ksi)	Treated Water or Steam, High temperature, Neutron Fluence $< 5x10^{20}$ n/cm². – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program			J
		Nickel Based Alloys	Treated Water or Steam, High temperature, Neutron Fluence $\geq 1x10^{17}$n/cm². – BWR Reactor Pressure Vessel	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-f	3.1.1.B-31	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program			F
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence $< 5x10^{20}$ n/cm². – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-b	3.1.1.B-31	B
						IV.B1.3-a	3.1.1.B-31	D, 38

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Core Spray Lines and Spargers	DF PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.3-a	3.1.1.B-31	B
	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.3-a	3.1.1.B-31	B
Differential Pressure Liquid Control Line	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.3-a	3.1.1.B-31	D, 39
Flanges	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-a	3.1.1.B-31	D, 40

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
In-core Housings	PB SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.6-a	3.1.1.B-31	B
In-core Instrumentation Dry Tubes	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.6-a	3.1.1.B-31	B

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Jet Pump Assemblies	DF	Cast Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Loss of Fracture Toughness	BWR Vessel Internals Program	IV.B1.4-c	3.1.1.B-33	E, 61
		Nickel Based Alloys	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.4-a	3.1.1.B-31	B, 62
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.4-a	3.1.1.B-31	B, 63
	SFS	Cast Austenitic Stainless Steel	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.4-b	3.1.1.B-01	A, 61

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Jet Pump Assemblies (cont'd)	SFS (cont'd)	Nickel Based Alloys	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.4-a	3.1.1.B-31	B, 64
		Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.4-a	3.1.1.B-31	B, 65
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.4-b	3.1.1.B-01	A, 65
			Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.4-a	3.1.1.B-31	B, 66

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Jet Pump Assemblies (cont'd)	TS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.4-a	3.1.1.B-31	B, 67
LPCI Couplings	DF PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-g	3.1.1.B-31	B
Orificed Fuel Supports	DF SFS	Cast Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence < $5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Loss of Fracture Toughness	BWR Vessel Internals Program	IV.B1.5-a	3.1.1.B-33	E
Peripheral Fuel Supports	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.5-c	3.1.1.B-31	D, 41

Table 3.1.2.B-2 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Internals – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Power Range Detector Assemblies	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, High Temperature, Neutron Fluence < $5x10^{20}$ n/cm.^{2}- BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.6-a	3.1.1.B-31	B
Spray Nozzles	DF PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, High Temperature - BWR Reactor Pressure Vessel	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.A1.3-d	3.1.1.B-01	C, 42
Steam Dryer	NFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.1-a	3.1.1.B-31	D, 17
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.B1.1-c	3.1.1.B-01	C, 17
Top Guide and Supports	SFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, High temperature, Neutron Fluence ≥$5x10^{20}$ n/cm^2. – BWR Reactor Vessel Internals	Cracking	BWR Vessel Internals Program Water Chemistry Control Program	IV.B1.2-a	3.1.1.B-31	B

Table 3.1.2.B-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Closure Bolting	PB	Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥ 212°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-g	3.1.1.B-01	A
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.3-e	3.1.1.B-26	E
				Loss of Preload	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.3-f	3.1.1.B-26	E
		Martensitic, Precipitation Hardenable, and Superferritic Stainless Steels	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥ 212°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-g	3.1.1.B-01	A
				Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program			H, 43
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.3-e	3.1.1.B-26	E
				Loss of Preload	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.3-f	3.1.1.B-26	H

Table 3.1.2.B-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Condensing Chambers	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	D, 44
Piping and Fittings	NFS	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	B
	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Air, Moisture or Wetting, temperature ≥140°F	Loss of Material	Systems Walkdown Program			H, 45

Table 3.1.2.B-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings (cont'd)	PB (cont'd)	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (cont'd)	Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	B
			Treated Water or Steam, temperature ≥482°F, Low Flow (cont'd)	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			H
		Nickel Based Alloys	Air, Moisture or Wetting, temperature ≥140°F	Loss of Material	Systems Walkdown Program			N, 46
			Treated Water, temperature < 140°F, Low Flow	None	None			None

Table 3.1.2.B-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings (cont'd)	PB (cont'd)	Nickel Based Alloys (cont'd)	Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			F
		Wrought Austenitic Stainless Steel	Air, Moisture or Wetting, temperature ≥140°F	Cracking	Systems Walkdown Program			G, 47
				Loss of Material	Systems Walkdown Program			L, 47
			Treated Water, temperature < 140°F, Low Flow	None	None			None

Table 3.1.2.B-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel (cont'd)	Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	B
Radiation Collars	RS	Wrought Austenitic Stainless Steel	Air	None	None			None
Restriction Orifices	PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	D, 48
Vacuum Breakers	PB	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None

Table 3.1.2.B-3 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Pressure Vessel Instrumentation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves	PB	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None
			Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	E, 24
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.B-01	A

**Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation**

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Closure Bolting	PB	Carbon or Low Alloy Steel (Yield Strength ≥100 Ksi)	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.2-f	3.1.1.B-01	A
						IV.C1.3-g	3.1.1.B-01	A
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.2-d	3.1.1.B-26	E
						IV.C1.3-e	3.1.1.B-26	E
				Loss of Preload	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.2-e	3.1.1.A-26	E
						IV.C1.3-f	3.1.1.A-26	E
		Wrought Austenitic Stainless Steel	Closure Bolting for Non-Borated Water Systems with operating temperatures ≥212°F	Cumulative Fatigue Damage	Fatigue Monitoring Program			F
				Loss of Preload	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.2-e	3.1.1.A-26	H

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
NSR piping, fittings, and equipment	PFASRE	Any	Treated Water, temperature < 140°F, Low Flow Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking; Loss of Material	Water Chemistry Control Program			J
			Treated Water or Steam, temperature ≥482°F Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking; Loss of Material	Water Chemistry Control Program			J

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings	PB	Nickel Based Alloys	Treated Water, temperature < 140°F, Low Flow	None	None			None
			Treated Water, temperature ≥140°F, but < 212°F, Low Flow	None	None			None
			Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			F

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel	Hydraulic Fluid	None	None			None
			Treated Water, temperature < 140°F, Low Flow	None	None			None
			Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	B
			Treated Water or Steam, temperature ≥482°F	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.C1.1-f	3.1.1.B-29	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.1-h	3.1.1.B-01	A

**Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation**

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel (cont'd)	Treated Water or Steam, temperature ≥482°F (cont'd)	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	B
			Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	B
Pumps	PB	Cast Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.2-a	3.1.1.B-01	A
				Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.C1.2-b	3.1.1.B-29	B

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Pumps (cont'd)	PB (cont'd)	Cast Austenitic Stainless Steel (cont'd)	Treated Water or Steam, temperature ≥482°F (cont'd)	Loss of Fracture Toughness	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.2-c	3.1.1.B-23	B
		Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.2-a	B.1.1.B-01	A
				Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program Water Chemistry Control Program			F
Radiation Collars	RS	Wrought Austenitic Stainless Steel	Air	None	None			None
Restriction Orifices	FR PB	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Restriction Orifices (cont'd)	FR PB (cont'd)	Wrought Austenitic Stainless Steel (cont'd)	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	D, 48
			Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	D, 48
Seal Coolers	HT PB	Wrought Austenitic Stainless Steel	Treated Water or Steam, temperature ≥482°F	Cracking	One-Time Inspection Program Water Chemistry Control Program			Q
Valves	PB	Cast Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Cast Austenitic Stainless Steel (cont'd)	Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	E, 24
			Treated Water or Steam, temperature ≥482°F	Cracking	BWR Stress Corrosion Cracking Water Chemistry Control Program	IV.C1.3-c	3.1.1.B-29	B
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.B-01	A
			Treated Water or Steam, temperature ≥482°F	Loss of Fracture Toughness	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program	IV.C1.3-b	3.1.1.B-23	B
		Wrought Austenitic Stainless Steel	Hydraulic Fluid	None	None			None
			Treated Water, temperature < 140°F, Low Flow	None	None			None
			Treated Water, temperature ≥140°F, but < 212°F, Low Flow	Cracking	One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	E, 24

Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel (cont'd)	Treated Water or Steam, temperature ≥482°F	Cracking	BWR Stress Corrosion Cracking Program Water Chemistry Control Program	IV.C1.3-c	3.1.1.B-29	B
					One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	E, 24
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.B-01	A
			Treated Water or Steam, temperature ≥482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	D, 49

**Table 3.1.2.B-4 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Reactor Recirculation System – Summary of Aging Management Evaluation**

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel (cont'd)	Treated Water or Steam, temperature ≥482°F, Low Flow (cont'd)	Cracking (cont'd)	One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	E, 24
				Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.B-01	A

Table 3.1.2.B-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Control Rod Drive System -- Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Accumulators	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Dried Air or Gas	None	None			None
			Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow	Cracking	<u>ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program</u> <u>One-Time Inspection Program</u> <u>Water Chemistry Control Program</u>	IV.C1.1-i	<u>3.1.1.B-07</u>	<u>D</u>, <u>50</u>
				Loss of Material	<u>ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program</u> <u>One-Time Inspection Program</u> <u>Water Chemistry Control Program</u>			<u>Q</u>

Table 3.1.2.B-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
CRD Hydraulic Control Units	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	D, 51
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			Q
Filters	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature < 140°F, Low Flow	Loss of Material	One-Time Inspection Program Water Chemistry Control Program			Q

Table 3.1.2.B-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Filters (cont'd)	PB (cont'd)	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None
Flow Elements	PB	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None
Flow Indicators	PB	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None
Flow Orifices	FR PB	Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None
NSR piping, fittings, and equipment	PFASRE	Any	Treated Water, temperature < 140°F, Low Flow Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow	Cracking; Loss of Material	Water Chemistry Control Program			J

Table 3.1.2.B-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature < 140°F, Low Flow	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			H, 22
					One-Time Inspection Program Water Chemistry Control Program			H, 24
			Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow	Cracking	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	B

Table 3.1.2.B-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Piping and Fittings (cont'd)	PB (cont'd)	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (cont'd)	Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow (cont'd)	Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			H, 22
					One-Time Inspection Program Water Chemistry Control Program			H, 24
		Copper Alloys (Zinc ≤15%)	Air	None	None			None
		Wrought Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None
			Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow	Cracking	One-Time Inspection Program Water Chemistry Control Program	IV.C1.1-i	3.1.1.B-07	E, 24

Table 3.1.2.B-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Pumps	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature < 140°F, Low Flow	Loss of Material	One-Time Inspection Program Water Chemistry Control Program			H
Rupture Discs	PB	Wrought Austenitic Stainless Steel	Dried Air or Gas	None	None			None
Valves	PB	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi)	Treated Water, temperature < 140°F, Low Flow	Loss of Material	One-Time Inspection Program Water Chemistry Control Program			H, 24
			Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow	Cumulative Fatigue Damage	Fatigue Monitoring Program	IV.C1.3-d	3.1.1.B-01	A
				Loss of Material	ASME Section XI Inservice Inspection (Subsections IWB, IWC, IWD) Program One-Time Inspection Program Water Chemistry Control Program			H, 22

Table 3.1.2.B-5 Reactor Vessel, Internals, and Reactor Coolant System
NMP2 Control Rod Drive System – Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves (cont'd)	PB (cont'd)	Carbon or Low Alloy Steel (Yield Strength < 100 Ksi) (cont'd)	Treated Water or Steam, temperature ≥212°F, but < 482°F, Low Flow (cont'd)	Loss of Material (cont'd)	One-Time Inspection Program Water Chemistry Control Program			H, 24
		Cast Austenitic Stainless Steel	Treated Water, temperature < 140°F, Low Flow	None	None			None
		Wrought Austenitic Stainless Steel	Dried Air or Gas	None	None			None
			Treated Water, temperature < 140°F, Low Flow	None	None			None

Notes for Tables 3.1.2.A-1 through 3.1.2.B-5:

A. Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

B. Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

C. Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

D. Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

E. Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program is credited.

F. Material not in NUREG-1801 for this component.

G. Environment not in NUREG-1801 for this component and material.

H. Aging effect not in NUREG-1801 for this component, material, and environment combination.

I. Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

J. Neither the component nor the material and environment combination is evaluated in NUREG-1801.

K. Material and environment not in NUREG-1801 for this component and aging effect.

L. Aging effect and environment not in NUREG-1801 for this component and material.

M. Aging effect and material not in NUREG-1801 for this component and environment.

N. Aging effect, material, and environment not in NUREG-1801 for this component.

P. Component and aging effect not in NUREG-1801 for this material and environment.

Q. Component not in NUREG-1801 for this material, environment, and aging effect.

(Note "O" was not used to avoid confusion with the number zero)

Plant Specific Notes:

1. Core Spray nozzles, Emergency Condenser Steam outlet nozzles, and Reactor Recirculation nozzles are not identified in NUREG-1801 for this GALL row number.
2. Feedwater nozzle safe ends and Steam nozzle safe ends are not identified in NUREG-1801 for this GALL row number.
3. Core Differential Pressure nozzle safe ends, Emergency Condenser Steam nozzle safe ends, Reactor Recirculation nozzle safe ends, and Safety Valve nozzle safe ends are not identified in NUREG-1801 for this GALL row number.
4. Core Differential Pressure nozzle safe ends, Core Spray nozzle safe ends, Emergency Condenser Steam nozzle safe ends, Safety Valve nozzle safe ends, and Reactor Recirculation nozzle safe ends are not identified in NUREG-1801 for this GALL row number.
5. Instrumentation Penetrations are not identified in NUREG-1801 for this GALL row number.
6. Feedwater Sparger thermal sleeves and are not identified in NUREG-1801 for this GALL row number.
7. Drain line penetrations are not identified in NUREG-1801 for this GALL row number.
8. The NMP1 Reactor Vessel Top Head (with cladding) and the NMP2 Reactor Vessel Top Head (without cladding) are not identified in NUREG-1801 for this GALL row number.
9. Reactor Vessel flange leak detection lines are not identified in NUREG-1801 for this GALL row number.

10. Top Head nozzles are not identified in NUREG-1801 for this GALL row number.
11. Reactor Vessel attachment welds are not identified in NUREG-1801 for this GALL row number.
12. Control Rod Drive assemblies are not identified in NUREG-1801 for this GALL row number.
13. Control Rod Guide Tubes are not identified in NUREG-1801 for this GALL row number.
14. Core Shroud head bolts are not identified in NUREG-1801 for this GALL row number.
15. Core Shroud spacers and tie rods are not identified in NUREG-1801 for this GALL row number.
16. Core Shroud support plates and welds are not identified in NUREG-1801 for this GALL row number.
17. Steam Dryers are not identified in NUREG-1801 for this GALL row number.
18. Condensing Pots and Temperature Equalizing Columns are not identified in NUREG-1801 for this GALL row number.
19. Reactor Vessel Instrumentation Valves are not identified in NUREG-1801 for this GALL row number.
20. Flow Elements are not identified in NUREG-1801 for this GALL row number.
21. Valve VLV-32-424 is part of the reactor coolant pressure boundary and is included in the ISI program.
22. This row applies to small bore valves and piping that are included in the Inservice Inspection Testing program.
23. These filters are part of the reactor coolant pressure boundary and are not identified in NUREG-1801 for this GALL row number.
24. This row applies to small bore valves and piping that are not included in the Inservice Inspection Testing program.
25. Core Spray nozzles, Drain nozzles and Residual Heat Removal nozzles are not identified in NUREG-1801 for this GALL row number.

26. Jet Pump Instrumentation nozzles and Reactor Recirculation nozzles are not identified in NUREG-1801 for this GALL row number.

27. Core Spray nozzle safe end extensions and Main Steam nozzle safe ends are not identified in NUREG-1801 for this GALL row number.

28. CRD Return Line nozzle safe ends, Residual Heat Removal nozzle safe end extensions, and Feedwater nozzle safe ends are not identified in NUREG-1801 for this GALL row number.

29. Core Spray nozzle safe ends, Feedwater nozzle safe end inserts, and Residual Heat Removal nozzle safe ends are not identified in NUREG-1801 for this GALL row number.

30. Reactor Recirculation nozzle safe ends and Jet Pump Instrumentation nozzle safe ends are not identified in NUREG-1801 for this GALL row number.

31. Core Spray nozzle thermal sleeve extensions are not identified in NUREG-1801 for this GALL row number.

32. Core Spray nozzle thermal sleeves, CRD Return Line nozzle thermal sleeves, Feedwater nozzle thermal sleeves, Residual Heat Removal nozzle thermal sleeves, and Reactor Recirculation nozzle thermal sleeves are not identified in NUREG-1801 for this GALL row number.

33. Top Head nozzles, spray nozzles, and vent nozzles are not identified in NUREG-1801 for this GALL row number.

34. Stub Tube Welds are not identified in NUREG-1801 for this GALL row number.

35. Access hole covers are not identified in NUREG-1801 for this GALL row number.

36. Control Rod Drive housings are not identified in NUREG-1801 for this GALL row number.

37. Head bolts are not identified in NUREG-1801 for this GALL row number.

38. Head clamps, keepers, and core spray line brackets are not identified in NUREG-1801 for this GALL row number.

39. Differential Pressure Liquid Control lines are not identified in NUREG-1801 for this GALL row number.

40. Flanges are not identified in NUREG-1801 for this GALL row number.
41. Peripheral fuel supports are not identified in NUREG-1801 for this GALL row number.
42. Head Cooling spray nozzles are not identified in NUREG-1801 for this GALL row number.
43. This row applies to bolting that has an aging effect/mechanism of cracking/stress corrosion cracking which is not addressed in NUREG-1801 Volume 2 Items IV.C1.2-d, IV.C1.2-e, IV.C1.2-f, IV.C1.3-e, IV.C1.3-f, or IV.C1.3-g.
44. Condensing chambers are not identified in NUREG-1801 for this GALL row number.
45. This row applies to the external surfaces of carbon steel components.
46. This row applies to the external surfaces of nickel based alloy components.
47. This row applies to the external surfaces of stainless steel components.
48. Restriction orifices are not identified in NUREG-1801 for this GALL row number.
49. Valves are not identified in NUREG-1801 for this GALL row number.
50. Accumulators are not identified in NUREG-1801 for this GALL row number.
51. Control Rod Hydraulic Control Units are not identified in NUREG-1801 for this GALL row number.
52. Feedwater Nozzle safe ends are not identified in NUREG-1801 for this GALL row number.
53. Feedwater Nozzle thermal sleeves are not identified in NUREG-1801 for this GALL row number.
54. Vessel Drain Penetrations are not identified in NUREG-1801 for this GALL row number.
55. Core Shroud collars are not identified in NUREG-1801 for this GALL row number.
56. Core Shroud support rings are not identified in NUREG-1801 for this GALL row number.

57. Core Shroud clamps are not identified in NUREG-1801 for this GALL row number

58. Feedwater nozzle thermal sleeve extensions are not identified in NUREG-1801 for this GALL row number.

59. Core Spray nozzle thermal sleeves and Residual Heat Removal thermal sleeves are not identified in NUREG-1801 for this GALL row number.

60. Feedwater nozzle thermal sleeves and CRD Return Line nozzle thermal sleeves are not identified in NUREG-1801 for this GALL row number.

61. This line applies to the following Jet Pump subcomponents: diffuser collars, inlet mixers, and jet pump nozzles.

62. This line applies to the following Jet Pump subcomponents: diffuser adaptors.

63. This line applies to the following Jet Pump subcomponents: riser pipes and diffuser shells.

64. This line applies to the following Jet Pump subcomponents: riser restrainer brackets and restrainer wedges.

65. This line applies to the following Jet Pump subcomponents: beams.

66. This line applies to the following Jet Pump subcomponents: riser braces and bolts.

67. This line applies to the following Jet Pump subcomponents: thermal sleeves.

68. This line applies to the Control Rod Drive Return Line nozzle thermal sleeves.