VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

August 11, 2022

10 CFR 50
10 CFR 51
10 CFR 54

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Serial No.: 22-200
NRA/DEA: R0
Docket Nos.: 50-338/339
License Nos.: NPF-4/7

VIRGINIA ELECTRIC AND POWER COMPANY
NORTH ANNA POWER STATION (NAPS) UNITS 1 AND 2
SUBSEQUENT LICENSE RENEWAL APPLICATION (SLRA)
SECOND 10 CFR 54.21(b) ANNUAL AMENDMENT

By letter dated August 24, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20246G703), Virginia Electric and Power Company (Dominion Energy Virginia or Dominion) submitted an application for the subsequent license renewal of Renewed Facility Operating License Nos. NPF-4 and NPF-7 for North Anna Power Station (NAPS) Units 1 and 2, respectively.

10 CFR 54.21(b) requires Dominion to report changes to the current licensing basis (CLB) that materially affect the contents of the subsequent license renewal application (SLRA), including the UFSAR supplement. These changes are required to be submitted each year and at least 3 months prior to the scheduled completion of the SLRA review by the NRC.

The first annual review results were provided by letter dated August 5, 2021 (ADAMS Accession No. ML21217A187). The second annual review covers the period from July 1, 2021, through July 1, 2022. Dominion has completed the second annual review and concluded one change to the CLB has been implemented that materially affects the contents of the NAPS SLRA. This letter provides an amendment to various sections of the NAPS SLRA to address this change.

A description of the change and the associated SLRA sections are provided in Enclosure 1. Mark-ups of the affected SLRA pages are provided in Enclosure 2. To aid the staff in assessing changes, Enclosure 2 shows new text as underlined and deleted text as lined through.

If you have any questions or require additional information regarding this submittal, please contact Mr. Paul Aitken at (804) 273-2818.

Sincerely,

James E. Holloway
Vice President - Nuclear Engineering and Fleet Support

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

Kathryn Hill Barret
Notary Public
Commonwealth of Virginia
Reg. No. 7905256
My Commission Expires January 31, 2024

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by James E. Holloway, who is Vice President - Nuclear Engineering and Fleet Support of Virginia Electric and Power Company. He has affirmed before me that he is duly authorized to execute and file the foregoing document on behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 11th day of August, 2022.

My Commission Expires: January 31, 2024

Kathy H. Barret
Notary Public

Commitments made in this letter: None

Enclosures:

1. Current Licensing Basis Change that Impacts the SLRA – Second 10 CFR 54.21(b) Annual Amendment
2. SLRA Mark-ups – Second 10 CFR 54.21(b) Annual Amendment

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Enclosure 1

CURRENT LICENSING BASIS CHANGE THAT IMPACTS THE SLRA
SECOND 10 CFR 54.21(b) ANNUAL AMENDMENT

Virginia Electric and Power Company
(Dominion Energy Virginia or Dominion)
North Anna Power Station Units 1 and 2

CURRENT LICENSING BASIS CHANGE THAT IMPACTS THE SLRA SECOND 10 CFR 54.21(b) ANNUAL AMENDMENT

By letter dated August 24, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20246G703), Virginia Electric and Power Company (Dominion Energy Virginia or Dominion) submitted an application for the subsequent license renewal of Renewed Facility Operating License Nos. NPF-4 and NPF-7 for North Anna Power Station (NAPS) Units 1 and 2, respectively.

10 CFR 54.21(b) requires Dominion to report changes to the current licensing basis (CLB) that materially affect the contents of the subsequent license renewal application (SLRA), including the UFSAR supplement. These changes are required to be submitted each year and at least 3 months prior to the scheduled completion of the LRA review by the NRC.

The first annual review results were provided by letter dated August 5, 2021 (ADAMS Accession No. ML21217A187). The second annual review covers the period from July 1, 2021, through July 1, 2022. Dominion has completed the second annual review and concluded the following change to the CLB materially affects the contents of the NAPS SLRA and requires the SLRA to be supplemented:

Current Licensing Basis Change

By letter dated September 9, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21252A514), as supplemented by letter dated December 16, 2021 (ADAMS Accession No. ML21350A408), Dominion submitted a request for an amendment to the Technical Specifications (TSs) for NAPS Units 1 and 2. As requested by Dominion, the Nuclear Regulatory Commission (NRC) approved the proposed changes to the TS for NAPS Units 1 and 2 to adopt Technical Specification Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections," on March 22, 2022 (ADAMS Accession No. ML22068A071). As part of adopting TSTF-577, the SG tube inspection frequency in NAPS Technical Specification (TS) 5.5.8, "Steam Generator (SG) Program," was revised from at least every 72 effective full power months (EFPM) to at least every 96 EFPM. This allows for the SG primary-side inspections to be performed commensurate with the approved TS inspection frequency of at least 96 EFPM for the SG tubes.

NUREG-2191, Section XI.M19, Steam Generators, Element 3 specifies that SG primary-side inspections will be performed at least every 72 EFPM or every third refueling outage, whichever results in more frequent inspections.

Therefore, as a result of the CLB change described above, this submittal amends the NAPS SLRA Sections below to indicate the Steam Generators program exception to the NUREG-2191, Section XI.M19, inspection frequency requirement of at least every 72 EFPM.

- SRP Table 3.1.1 – Items 3.1.1-025, 3.1.1-069, 3.1.1-070, 3.1.1-071, 3.1.1-072, 3.1.1-074, 3.1.1-076, 3.1.1-077, 3.1.1-111, 3.1.1-125, 3.1.1-127 are revised to

indicate exceptions now apply to the NUREG-2191 Aging Management Program (AMP).

- Table 3.1.2-4 is revised to indicate exceptions now apply to the NUREG-2191 AMP.
- Table B2-1 is revised to indicate the Steam Generators program has exceptions to NUREG-2191.
- Section B2.1.10 is revised to update the SG inspection frequency to at least every 96 effective full power months and include the exception to NUREG-2191, Section XI.M19.
- Section A1.10 is revised to update the SG inspection frequency to at least every 96 effective full power months.

Enclosure 2

SLRA MARK-UPS
SECOND 10 CFR 54.21(b) ANNUAL AMENDMENT

Virginia Electric and Power Company
(Dominion Energy Virginia or Dominion)
North Anna Power Station Units 1 and 2

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report

Item Number	Component	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1-019	Stainless steel reactor vessel bottom-mounted instrument guide tubes (external to reactor vessel) exposed to reactor coolant	Cracking due to SCC	Plant-specific aging management program	Yes (SRP-SLR Section 3.1.2.2.6.1)	Consistent with NUREG-2191. Cracking of stainless steel reactor vessel bottom-mounted instrument guide tubes (external to reactor vessel) exposed to reactor coolant is managed by the ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) program and the Water Chemistry (B2.1.2) program. See further evaluation in Section 3.1.2.2.6.1.
3.1.1-020	Cast austenitic stainless steel Class 1 piping, piping components exposed to reactor coolant	Cracking due to SCC	AMP XI.M2, Water Chemistry and plant-specific aging management program	Yes (SRP-SLR Section 3.1.2.2.6.2)	Consistent with NUREG-2191. Cracking of cast austenitic stainless steel Class 1 piping, piping components exposed to reactor coolant is managed by the ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) program and the Water Chemistry (B2.1.2) program. See further evaluation in Section 3.1.2.2.6.2.
3.1.1-021	Steel and stainless steel isolation condenser components exposed to reactor coolant	Cracking due to cyclic loading	AMP XI.M1, ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	Yes (SRP-SLR Section 3.1.2.2.7)	Not applicable - BWR only.
3.1.1-022	Steel steam generator feedwater impingement plate and support exposed to secondary feedwater	Loss of material due to erosion	Plant-specific aging management program	Yes (SRP-SLR Section 3.1.2.2.8)	Not applicable. NAPS has no in-scope steel steam generator feedwater impingement plate and support exposed to secondary feedwater in the Reactor Vessel, Internals, and Reactor Coolant System. The associated NUREG-2191 aging items are not used.
3.1.1-025	Steel (with nickel alloy cladding) or nickel alloy steam generator primary side components: divider plate and tube-to-tube sheet welds exposed to reactor coolant	Cracking due to primary water SCC	AMP XI.M2, Water Chemistry, and AMP XI.M19, Steam Generators. In addition, a plant-specific program is to be evaluated.	Yes (SRP-SLR Sections 3.1.2.2.11.1 and 3.1.2.2.11.2)	Consistent with NUREG-2191 with exceptions. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation. A plant-specific program is not needed for the divider plate or for the tube-to-tubesheet weld. See further evaluation in Section 3.1.2.2.11.1 and 3.1.2.2.11.2.

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report

Item Number	Component	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1-068	Nickel alloy steam generator tubes exposed to secondary feedwater or steam	Changes in dimension (denting) due to corrosion of carbon steel tube support plate	AMP XI.M19, Steam Generators, and AMP XI.M2, Water Chemistry	No	Not applicable. NAPS has no carbon steel tube support plates in the Reactor Vessel, Internals, and Reactor Coolant System. The associated NUREG-2191 aging items are not used.
3.1.1-069	Nickel alloy steam generator tubes and sleeves exposed to secondary feedwater or steam	Cracking due to outer diameter SCC, intergranular attack	AMP XI.M19, Steam Generators, and AMP XI.M2, Water Chemistry	No	Consistent with NUREG-2191 with exceptions. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.
3.1.1-070	Nickel alloy steam generator tubes, repair sleeves, and tube plugs exposed to reactor coolant	Cracking due to primary water SCC	AMP XI.M19, Steam Generators, and AMP XI.M2, Water Chemistry	No	Consistent with NUREG-2191 with exceptions. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.
3.1.1-071	Steel, chrome plated steel, stainless steel, nickel alloy steam generator U-bend supports including anti-vibration bars exposed to secondary feedwater or steam	Cracking due to SCC or other mechanism(s); loss of material due general (steel only), pitting, crevice corrosion	AMP XI.M19, Steam Generators, and AMP XI.M2, Water Chemistry	No	Consistent with NUREG-2191 with exceptions, and with a different program for some component. The ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) program is used instead of the Steam Generators (B2.1.10) program to manage cracking and loss of material for the feedwater nozzle thermal sleeve. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.
3.1.1-072	Steel steam generator tube support plate, tube bundle wrapper, supports and mounting hardware exposed to secondary feedwater or steam	Loss of material due to general, pitting, crevice corrosion, erosion, ligament cracking due to corrosion	AMP XI.M19, Steam Generators, and AMP XI.M2, Water Chemistry (corrosion based aging effects and mechanisms only)	No	Consistent with NUREG-2191 with exceptions. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report

Item Number	Component	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1-073	Nickel alloy steam generator tubes and sleeves exposed to phosphate chemistry in secondary feedwater or steam	Loss of material due to wastage, pitting corrosion	AMP XI.M19, Steam Generators, and AMP XI.M2, Water Chemistry	No	Not applicable. NAPS has no in-scope nickel alloy steam generator tubes and sleeves exposed to phosphate chemistry in secondary feedwater or steam in the Reactor Vessel, Internals, and Reactor Coolant System. The associated NUREG-2191 aging items are not used.
3.1.1-074	Steel steam generator upper assembly and separators including feedwater inlet ring and support exposed to secondary feedwater or steam	Wall thinning due to flow-accelerated corrosion	AMP XI.M19, Steam Generators, and AMP XI.M2, Water Chemistry	No	Consistent with NUREG-2191 with exceptions. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.
3.1.1-075	Steel steam generator tube support lattice bars exposed to secondary feedwater or steam	Wall thinning due to flow-accelerated corrosion, general corrosion	AMP XI.M19, Steam Generators, and AMP XI.M2, Water Chemistry	No	Not applicable. NAPS has no in-scope steel steam generator tube support lattice bars exposed to secondary feedwater or steam in the Reactor Vessel, Internals, and Reactor Coolant System. The associated NUREG-2191 aging items are not used.
3.1.1-076	Steel, chrome plated steel, stainless steel, nickel alloy steam generator U-bend supports including anti-vibration bars exposed to secondary feedwater or steam	Loss of material due to wear, fretting	AMP XI.M19, Steam Generators	No	Consistent with NUREG-2191 with exceptions, and with a different program for some component. The ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) program is used instead of the Steam Generators (B2.1.10) program to manage loss of material for the feedwater nozzle thermal sleeve. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.
3.1.1-077	Nickel alloy steam generator tubes and sleeves exposed to secondary feedwater or steam	Loss of material due to wear, fretting	AMP XI.M19, Steam Generators	No	Consistent with NUREG-2191 with exceptions, and with a TLAA evaluation included. Wear of steam generator tubes at tube support plates is a TLAA, evaluated in Section 4.7.8, Steam Generator Tube Wear Evaluation. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report

Item Number	Component	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1-111	Nickel alloy steam generator tubes exposed to secondary feedwater or steam	Reduction of heat transfer due to fouling	AMP XI.M2, Water Chemistry, and AMP XI.M19, Steam Generators	No	Consistent with NUREG-2191 with exceptions. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.
3.1.1-113	Steel reactor vessel external attachments exposed to indoor, uncontrolled air	Loss of material due to general, pitting, crevice corrosion, wear	AMP XI.M1, ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	No	Not applicable - BWR only.
3.1.1-114	Reactor coolant system components defined as ASME Code, Section XI Code Class components (ASME Code Class 1 reactor coolant pressure boundary components or core support structure components, or ASME Code Class 2 or 3 components - including ASME defined appurtenances, component supports, and associated pressure boundary welds, or components subject to plant-specific equivalent classifications for these ASME code classes)	Cracking due to SCC, IGSCC (stainless steel, nickel alloy components only), cyclic loading; loss of material due to general corrosion (steel only), pitting corrosion, crevice corrosion, wear	AMP XI.M1, ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD, and AMP XI.M2, Water Chemistry (water chemistry-related or corrosion-related aging effect mechanisms only)	No	Not applicable. Cracking and loss of material of reactor coolant system components defined as ASME Code, Section XI Code Class components (ASME Code Class 1 reactor coolant pressure boundary components or core support structure components, or ASME Code Class 2 or 3 components - including ASME defined appurtenances, component supports, and associated pressure boundary welds, or components subject to plant-specific equivalent classifications for these ASME code classes) is addressed by rows 3.1.1-020, 3.1.1-033, 3.1.1-035, 3.1.1-036, 3.1.1-037, 3.1.1-039, 3.1.1-042, 3.1.1-045, 3.1.1-088, and 3.1.1-116. The associated NUREG-2191 aging items are not used.
3.1.1-115	Stainless steel piping, piping components exposed to concrete	None	None	Yes (SRP-SLR Section 3.1.2.2.15)	Not applicable. NAPS has no in-scope stainless steel piping, piping components exposed to concrete in the Reactor Vessel, Internals, and Reactor Coolant System. The associated NUREG-2191 aging items are not used.

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report

Item Number	Component	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1-120	Stainless steel core plate rim holddown bolts exposed to reactor coolant and neutron flux	Loss of preload due to thermal or irradiation-enhanced stress relaxation	AMP XI.M9, BWR Vessel Internals, and TLAA SRP-SLR 4.7 Other Plant-Specific TLAAs [if an analysis is performed as part of the aging management basis and conforms to the definition of a TLAA in 10 CFR 54.3(a)]	Yes (SRP-SLR Section 3.1.2.2.14)	Not applicable - BWR only.
3.1.1-121	Stainless steel jet pump assembly holddown beam bolts exposed to reactor coolant and neutron flux	Loss of preload due to thermal or irradiation-enhanced stress relaxation	AMP XI.M9, BWR Vessel Internals	No	Not applicable - BWR only.
3.1.1-124	Steel piping, piping components exposed to air-indoor uncontrolled, air-outdoor, condensation	Loss of material due to general, pitting, crevice corrosion	AMP XI.M36, External Surfaces Monitoring of Mechanical Components	No	Consistent with NUREG-2191.
3.1.1-125	Nickel alloy steam generator tubes at support plate locations exposed to secondary feedwater or steam	Cracking due to flow-induced vibration, high-cycle fatigue	AMP XI.M19, Steam Generators	No	Consistent with NUREG-2191 with exceptions. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation.

Table 3.1.1 Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL-SLR Report

Item Number	Component	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1-127	Steel (with stainless steel or nickel alloy cladding) steam generator heads and tubesheets exposed to reactor coolant	Loss of material due to boric acid corrosion	AMP XI.M2, Water Chemistry, and AMP XI.M19, Steam Generators	No	Consistent with NUREG-2191 with exceptions and a different aging management program is credited for some components. The ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) program will manage loss of material of the steel with stainless steel cladding steam generator primary inlet and outlet nozzles, and the stainless steel primary inlet and outlet nozzle safe ends, exposed to reactor coolant, instead of the Water Chemistry (B2.1.2) and Steam Generators (B2.1.10) programs. Exceptions apply to the NUREG-2191 recommendations for Steam Generators (B2.1.10) program implementation
3.1.1-128	Stainless steel, nickel alloy nozzles safe ends and welds: high pressure core spray; low pressure core spray; recirculating water, low pressure coolant injection or RHR injection mode exposed to reactor coolant	Cracking due to SCC, IGSCC	AMP XI.M7, BWR Stress Corrosion Cracking, and AMP XI.M2, Water Chemistry	No	Not applicable - BWR only.
3.1.1-129	Steel and stainless steel piping, piping components exposed to reactor coolant: welded connections between the re-routed control rod drive return line and the inlet piping system that delivers return line flow to the reactor pressure vessel exposed to reactor coolant	Cracking due to cyclic loading	AMP XI.M1, ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	No	Not applicable - BWR only.

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System - Steam Generator - Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	Table 1 Item	Notes
Anti-vibration bar	SS	Stainless steel	(E) Treated water >60°C (>140°F)	Cracking	Steam Generators (B2.1.10)	IV.D1.RP-384	3.1.1-071	~~A~~B
					Water Chemistry (B2.1.2)	IV.D1.RP-384	3.1.1-071	A
				Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-226	3.1.1-071	~~A~~B
					Water Chemistry (B2.1.2)	IV.D1.RP-226	3.1.1-071	A
					Steam Generators (B2.1.10)	IV.D1.RP-225	3.1.1-076	~~A~~B
Channel head	PB	Steel with stainless steel cladding	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Reactor coolant	Cracking	Steam Generators (B2.1.10)	IV.D1.RP-232	3.1.1-033	E, 2
					Water Chemistry (B2.1.2)	IV.D1.RP-232	3.1.1-033	C
				Cumulative fatigue damage	TLAA	IV.D1.R-221	3.1.1-008	A
				Loss of material	Steam Generators (B2.1.10)	IV.D1.R-436	3.1.1-127	~~A~~B
					Water Chemistry (B2.1.2)	IV.D1.R-436	3.1.1-127	A
Channel head divider plate	FD	Nickel alloy	(E) Reactor coolant	Cracking	Steam Generators (B2.1.10)	IV.D1.RP-367	3.1.1-025	~~A~~B
					Water Chemistry (B2.1.2)	IV.D1.RP-367	3.1.1-025	A
				Cumulative fatigue damage	TLAA	IV.D1.R-221	3.1.1-008	C
Feedwater distribution ring and J-nozzles	FD	Steel	(E) Treated water	Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-161	3.1.1-072	~~C~~D
					Water Chemistry (B2.1.2)	IV.D1.RP-161	3.1.1-072	C
					Steam Generators (B2.1.10)	IV.D1.RP-225	3.1.1-076	~~C~~D
			(I) Treated water	Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-161	3.1.1-072	~~C~~D
					Water Chemistry (B2.1.2)	IV.D1.RP-161	3.1.1-072	C
					Steam Generators (B2.1.10)	IV.D1.RP-225	3.1.1-076	~~C~~D
				Wall thinning	Steam Generators (B2.1.10)	IV.D1.RP-49	3.1.1-074	~~A~~B
					Water Chemistry (B2.1.2)	IV.D1.RP-49	3.1.1-074	A

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System - Steam Generator - Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	Table 1 Item	Notes
Feedwater nozzle	PB	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Treated water	Cumulative fatigue damage	TLAA	IV.D1.R-33	3.1.1-005	A
				Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-368	3.1.1-012	C
					Water Chemistry (B2.1.2)	IV.D1.RP-368	3.1.1-012	C
				Wall thinning	Flow-Accelerated Corrosion (B2.1.8)	IV.D1.R-37	3.1.1-061	A
Feedwater nozzle thermal sleeve	LTC	Stainless steel	(I) Treated water >60°C (>140°F)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-384	3.1.1-071	E, 5
					Water Chemistry (B2.1.2)	IV.D1.RP-384	3.1.1-071	C
				Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-226	3.1.1-071	E, 5
					Water Chemistry (B2.1.2)	IV.D1.RP-226	3.1.1-071	C
					ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-225	3.1.1-076	E, 5
Moisture separator assembly	FD	Steel	(E) Treated water	Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-161	3.1.1-072	~~G~~D
					Water Chemistry (B2.1.2)	IV.D1.RP-161	3.1.1-072	C
				Wall thinning	Steam Generators (B2.1.10)	IV.D1.RP-49	3.1.1-074	~~A~~B
					Water Chemistry (B2.1.2)	IV.D1.RP-49	3.1.1-074	A
			(I) Treated water	Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-161	3.1.1-072	~~G~~D
					Water Chemistry (B2.1.2)	IV.D1.RP-161	3.1.1-072	C
				Wall thinning	Steam Generators (B2.1.10)	IV.D1.RP-49	3.1.1-074	~~A~~B
					Water Chemistry (B2.1.2)	IV.D1.RP-49	3.1.1-074	A

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System - Steam Generator - Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	Table 1 Item	Notes
Primary inlet and outlet nozzle	PB	Steel with stainless steel cladding	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Reactor coolant	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-232	3.1.1-033	A
					Water Chemistry (B2.1.2)	IV.D1.RP-232	3.1.1-033	A
				Cumulative fatigue damage	TLAA	IV.D1.R-221	3.1.1-008	A
				Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.R-436	3.1.1-127	E, 1
					Water Chemistry (B2.1.2)	IV.D1.R-436	3.1.1-127	A
Primary inlet and outlet nozzle safe end	PB	Stainless steel	(E) Air – indoor uncontrolled	Cracking	One-Time Inspection (B2.1.20)	V.A.EP-103b	3.2.1-007	A
				Loss of material	One-Time Inspection (B2.1.20)	IV.C2.R-452a	3.1.1-136	C
			(I) Reactor coolant	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-232	3.1.1-033	A
					Water Chemistry (B2.1.2)	IV.D1.RP-232	3.1.1-033	A
				Cumulative fatigue damage	TLAA	IV.D1.R-221	3.1.1-008	A
				Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.R-436	3.1.1-127	E, 1
					Water Chemistry (B2.1.2)	IV.D1.R-436	3.1.1-127	A
Primary inlet and outlet nozzle weld	PB	Nickel alloy	(E) Air – indoor uncontrolled	Loss of material	One-Time Inspection (B2.1.20)	IV.C2.R-452a	3.1.1-136	C
			(I) Reactor coolant	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.C2.RP-159	3.1.1-045	C
					Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (B2.1.5)	IV.C2.RP-159	3.1.1-045	C
					Water Chemistry (B2.1.2)	IV.C2.RP-159	3.1.1-045	C
				Loss of material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	A

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System - Steam Generator - Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	Table 1 Item	Notes
Primary manway	PB	Steel with stainless steel cladding	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Reactor coolant	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-232	3.1.1-033	A
					Water Chemistry (B2.1.2)	IV.D1.RP-232	3.1.1-033	A
				Cumulative fatigue damage	TLAA	IV.D1.R-221	3.1.1-008	A
				Loss of material	Steam Generators (B2.1.10)	IV.D1.R-436	3.1.1-127	~~A~~B
					Water Chemistry (B2.1.2)	IV.D1.R-436	3.1.1-127	A
Primary manway cover	PB	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
Primary manway cover bolting	PB	Steel	(E) Air – indoor uncontrolled	Cumulative fatigue damage	TLAA	IV.C2.R-18	3.1.1-005	A
				Loss of material	Bolting Integrity (B2.1.9)	IV.D1.RP-166	3.1.1-064	A
				Loss of preload	Bolting Integrity (B2.1.9)	IV.D1.RP-46	3.1.1-067	A
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
Primary manway cover insert	PB	Nickel alloy	(I) Reactor coolant	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.C2.RP-159	3.1.1-045	C
					Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (B2.1.5)	IV.C2.RP-159	3.1.1-045	C
					Water Chemistry (B2.1.2)	IV.C2.RP-159	3.1.1-045	C
				Loss of material	Water Chemistry (B2.1.2)	IV.C2.RP-23	3.1.1-088	A

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System - Steam Generator - Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	Table 1 Item	Notes
Secondary closure cover	PB	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Treated water	Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-368	3.1.1-012	C
					Water Chemistry (B2.1.2)	IV.D1.RP-368	3.1.1-012	C
Secondary closure cover bolting	PB	Steel	(E) Air – indoor uncontrolled	Cumulative fatigue damage	TLAA	IV.C2.R-18	3.1.1-005	A
				Loss of material	Bolting Integrity (B2.1.9)	IV.D1.RP-166	3.1.1-064	A
				Loss of preload	Bolting Integrity (B2.1.9)	IV.D1.RP-46	3.1.1-067	A
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
Secondary manway (includes pad)	PB	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Steam	Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-368	3.1.1-012	A
					Water Chemistry (B2.1.2)	IV.D1.RP-368	3.1.1-012	A
					ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.R-31	3.1.1-044	A
Secondary side shell (lower shell, upper shell, transition cone, closure weld, girth weld)	PB	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Treated water	Cumulative fatigue damage	TLAA	IV.D1.R-33	3.1.1-005	A
				Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-368	3.1.1-012	A
					One-Time Inspection (B2.1.20)	IV.D1.RP-368	3.1.1-012	E, 4
					Water Chemistry (B2.1.2)	IV.D1.RP-368	3.1.1-012	A

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System - Steam Generator - Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	Table 1 Item	Notes
Secondary side shell (penetrations)	PB	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Treated water	Cumulative fatigue damage	TLAA	IV.D1.R-33	3.1.1-005	A
				Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-368	3.1.1-012	A
					Water Chemistry (B2.1.2)	IV.D1.RP-368	3.1.1-012	A
Secondary side shell (upper head)	PB	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Steam	Cumulative fatigue damage	TLAA	IV.D1.R-33	3.1.1-005	A
				Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-368	3.1.1-012	A
					Water Chemistry (B2.1.2)	IV.D1.RP-368	3.1.1-012	A
Stay rod and spacer	SS	Steel	(E) Treated water	Cracking	Steam Generators (B2.1.10)	IV.D1.RP-384	3.1.1-071	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.RP-384	3.1.1-071	A
				Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-226	3.1.1-071	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.RP-226	3.1.1-071	A
					Steam Generators (B2.1.10)	IV.D1.RP-225	3.1.1-076	~~AB~~
Steam flow limiter	RF	Nickel alloy	(E) Steam	Cracking	Steam Generators (B2.1.10)	IV.D1.RP-384	3.1.1-071	~~GD~~
					Water Chemistry (B2.1.2)	IV.D1.RP-384	3.1.1-071	C
				Cumulative fatigue damage	TLAA	IV.D1.R-46	3.1.1-002	C
				Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-226	3.1.1-071	~~GD~~
					Water Chemistry (B2.1.2)	IV.D1.RP-226	3.1.1-071	C

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System - Steam Generator - Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	Table 1 Item	Notes
Steam outlet nozzle	PB	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
			(I) Steam	Cumulative fatigue damage	TLAA	IV.D1.R-33	3.1.1-005	A
				Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.D1.RP-368	3.1.1-012	A
					Water Chemistry (B2.1.2)	IV.D1.RP-368	3.1.1-012	A
				Wall thinning	Flow-Accelerated Corrosion (B2.1.8)	IV.D1.R-37	3.1.1-061	A
Support pad	SS	Steel	(E) Air – indoor uncontrolled	Loss of material	External Surfaces Monitoring of Mechanical Components (B2.1.23)	IV.C2.R-431	3.1.1-124	C
			(E) Air with borated water leakage	Loss of material	Boric Acid Corrosion (B2.1.4)	IV.D1.R-17	3.1.1-049	A
Tube	HT;PB	Nickel alloy	(I) Reactor coolant	Cracking	Steam Generators (B2.1.10)	IV.D1.R-44	3.1.1-070	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.R-44	3.1.1-070	A
				Cumulative fatigue damage	TLAA	IV.D1.R-46	3.1.1-002	A
			(E) Treated water >60°C (>140°F)	Cracking	Steam Generators (B2.1.10)	IV.D1.R-47	3.1.1-069	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.R-47	3.1.1-069	A
					Steam Generators (B2.1.10)	IV.D1.R-437	3.1.1-125	~~AB~~
				Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-233	3.1.1-077	~~AB~~
					TLAA	IV.D1.RP-233	3.1.1-077	E, 3
				Reduction of heat transfer	Steam Generators (B2.1.10)	IV.D1.R-407	3.1.1-111	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.R-407	3.1.1-111	A
Tube bundle wrapper	FD;SS	Steel	(E) Treated water	Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-161	3.1.1-072	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.RP-161	3.1.1-072	A
Tube plug	PB	Nickel alloy	(E) Reactor coolant	Cracking	Steam Generators (B2.1.10)	IV.D1.R-40	3.1.1-070	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.R-40	3.1.1-070	A
				Cumulative fatigue damage	TLAA	IV.D1.R-46	3.1.1-002	C

Table 3.1.2-4 Reactor Vessel, Internals, and Reactor Coolant System - Steam Generator - Aging Management Evaluation

Subcomponent	Intended Function(s)	Material	Environment	Aging Effect Requiring Management	Aging Management Programs	NUREG-2191 Item	Table 1 Item	Notes
Tube support plate	FD;SS	Stainless steel	(E) Treated water >60°C (>140°F)	Cracking	Steam Generators (B2.1.10)	IV.D1.RP-384	3.1.1-071	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.RP-384	3.1.1-071	A
				Cumulative fatigue damage	TLAA	IV.C2.R-18	3.1.1-005	C
				Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-226	3.1.1-071	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.RP-226	3.1.1-071	A
					Steam Generators (B2.1.10)	IV.D1.RP-225	3.1.1-076	~~AB~~
Tubesheet	PB	Steel with nickel alloy cladding	(I) Reactor coolant	Cumulative fatigue damage	TLAA	IV.D1.R-221	3.1.1-008	A
				Loss of material	Steam Generators (B2.1.10)	IV.D1.R-436	3.1.1-127	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.R-436	3.1.1-127	A
			(E) Treated water	Loss of material	Steam Generators (B2.1.10)	IV.D1.RP-161	3.1.1-072	~~GD~~
					Water Chemistry (B2.1.2)	IV.D1.RP-161	3.1.1-072	C
Tube-to-tubesheet weld	SS	Nickel alloy	(E) Reactor coolant	Cracking	Steam Generators (B2.1.10)	IV.D1.RP-385	3.1.1-025	~~AB~~
					Water Chemistry (B2.1.2)	IV.D1.RP-385	3.1.1-025	A
				Cumulative fatigue damage	TLAA	IV.D1.R-221	3.1.1-008	A

Table 3.1.2-4 Plant-Specific Notes:

1. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) program is used instead of the Steam Generators (B2.1.10) program to manage loss of material due to boric acid corrosion for the primary inlet and outlet nozzle and safe end.
2. The Steam Generators (B2.1.10) program is used instead of the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) program to manage cracking due to stress corrosion cracking for the channel head stainless steel cladding.
3. Wear of steam generator tubes at the tube support plates is a plant-specific TLAA, evaluated in Steam Generator Tube Wear Evaluation (4.7.8).
4. The One-Time Inspection (B2.1.20) program, using magnetic particle test intended to cover essentially 100 percent of the total weld length of each weld on each steam generator, will verify the effectiveness of the Water Chemistry (B2.1.2) program to manage loss of material for the upper shell-to-transition cone girth weld and the new transition cone closure weld.
5. The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1) program is used instead of the Steam Generators (B2.1.10) program to manage cracking and loss of material for the feedwater nozzle thermal sleeve.

Tables 3.1.2-1 through 3.1.2-4 Industry Standard Notes:

A. Consistent with NUREG-2191 item for component, material, environment, and aging effect. AMP is consistent with NUREG-2191 AMP.

B. Consistent with NUREG-2191 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-2191 AMP.

C. Component is different, but consistent with NUREG-2191 item for material, environment, and aging effect. AMP is consistent with NUREG-2191 AMP.

D. Component is different, but consistent with NUREG-2191 item for material, environment, and aging effect. AMP takes some exceptions to the NUREG-2191 AMP.

E. Consistent with NUREG-2191 item for material, environment, and aging effect, but a different AMP is credited or NUREG-2191 identifies a plant-specific AMP.

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

G. Environment not in NUREG-2191 for this component and material.

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

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

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

A1.10 STEAM GENERATORS

The *Steam Generators* program is an existing condition monitoring program that manages the aging effects of cracking, loss of material (e.g., wall thinning), and reduction of heat transfer for the steam generators. The scope of the program includes primary-side components (e.g., U-tubes [tubes], plugs, channel head divider plate, channel head, tubesheet, etc), and secondary-side components that are contained within the steam generator. The program uses volumetric inspections for the tubes, and visual inspections for the other primary-side and secondary-side components. The visual inspections of primary-side components listed above are performed in accordance with the Degradation Assessment (DA) that is prepared as each steam generator is scheduled for examination.

Provisions in the Steam Generators program address reporting criteria, inspection scope and frequency, assessments, plugging criteria, and water chemistry monitoring to maintain consistency with established requirements. NEI 97-06, Revision 3, "Steam Generator Program Guidelines," and associated EPRI guidelines, provide a generic industry program to implement Technical Specifications.

~~As stated in the steam generator DA, tubing and primary side inspections are typically performed every third refueling outage for each steam generator, thus satisfying the guidance for visual inspections to be performed at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections.~~ By letter dated September 9, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21252A514), as supplemented by letter dated December 16, 2021 (ADAMS Accession No. ML21350A408), Dominion submitted a request for an amendment to the Technical Specifications (TSs) for NAPS Units 1 and 2. As requested by Dominion, the Nuclear Regulatory Commission (NRC) approved the proposed changes to the TS for NAPS Units 1 and 2 to adopt Technical Specification Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections," on March 22, 2022 (ADAMS Accession No. ML22068A071). The change to the TS increased the SG tube inspection frequency from at least every 72 effective full power months (EFPM) to at least every 96 EFPM. This allows for the SG primary-side inspections to be performed commensurate with the approved TS inspection frequency of at least every 96 EFPM for the SG tubes. The DA includes a review of applicable industry operating experience (OE) and plant-specific OE which has occurred since the previous DA was performed. The DA review determines the existence of any unaddressed mechanism that could adversely affect steam generator primary-side or secondary-side integrity, as well as the effects of any chemistry excursions or transients that could affect existing degradation mechanisms.

The Steam Generators program includes preventive measures to mitigate aging related to corrosion phenomena, and through foreign material exclusion as a means to inhibit tube

degradation due to wear. Identification of deposits on the secondary side of the steam generator, and the subsequent removal of sludge deposits help avoid tube degradation.

The Technical Specifications require condition monitoring and operational assessments to be performed to ensure tube integrity will be maintained until the next inspection. The operational assessments are performed after steam generator inspections have been completed to verify structural and leakage integrity.

A1.11 OPEN-CYCLE COOLING WATER SYSTEM

The *Open Cycle Cooling Water System* program is an existing preventive, mitigative, condition monitoring, and performance monitoring program that manages cracking, flow blockage, loss of material, and reduction of heat transfer, for the piping, piping components, and heat exchangers identified in the responses to NRC Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The program is comprised of the aging management aspects of the Virginia Electric and Power Company response to NRC GL 89-13 and includes: (a) surveillance and control to reduce the incidence of flow blockage problems as a result of biofouling, (b) tests to verify heat transfer of safety-related heat exchangers, (c) routine inspection and maintenance so that loss of material, corrosion, erosion, cracking, fouling, and biofouling cannot degrade the performance of systems serviced by the open-cycle cooling water system. This program includes enhancements to the guidance in NRC GL 89-13 that address operating experience (OE) to provide reasonable assurance that aging effects are adequately managed.

System and component testing, visual inspections, nondestructive examination (i.e., ultrasonic testing and eddy current testing), and chemical injection are conducted to ensure that identified aging effects are managed such that system and component intended functions and integrity are maintained. Periodic heat transfer testing, visual inspection and cleaning of safety-related heat exchangers with a heat transfer intended function is performed in accordance with the Virginia Electric and Power Company commitments to GL 89-13 to verify heat transfer capabilities.

The *Internal Coatings/Linings For In-Scope Piping, Piping Components, Heat Exchangers, and Tanks* program (A1.28) will manage the aging effects of internal surface coatings.

A1.12 CLOSED TREATED WATER SYSTEMS

The *Closed Treated Water Systems* program is an existing program that manages cracking, loss of material, and reduction of heat transfer for components exposed to a closed treated water environment.

This is a mitigation program that also includes a condition monitoring program to verify the effectiveness of the mitigation activities. The program consists of: (a) water treatment, including the use of corrosion inhibitors, to modify the chemical composition of the water such that the effects of

B2 Aging Management Programs

Table B2-1 lists the aging management programs described in this appendix and identifies the programs consistency with NUREG-2191. As discussed in Section B1.4, both plant specific and industry operating experience has been reviewed and considered as it relates to both new and existing aging management programs.

Table B2-1
NAPS Program Consistency with NUREG-2191 Program

NUREG-2191 Program	Appendix B Reference	Existing or New	Program has NUREG-2191 Enhancements	Program has Exceptions to NUREG-2191
ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	B2.1.1	Existing	X	
Water Chemistry (Primary and Secondary)	B2.1.2	Existing		
Reactor Head Closure Stud Bolting (addressed by ISI program)	B2.1.3	Existing		X
Boric Acid Corrosion	B2.1.4	Existing		
Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-induced Corrosion in Reactor Coolant Pressure Boundary Components	B2.1.5	Existing		
Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	B2.1.6	Existing		
PWR Vessel Internals	B2.1.7	Existing	X	
Flow-Accelerated Corrosion	B2.1.8	Existing		
Bolting Integrity	B2.1.9	Existing	X	
Steam Generators	B2.1.10	Existing		X
Open-Cycle Cooling Water System	B2.1.11	Existing		X
Closed Treated Water Systems	B2.1.12	Existing	X	

B2.1.10 Steam Generators

Program Description

The *Steam Generators* program does not include primary-side sleeves since these components are not used in the steam generators. The program uses volumetric inspections for the tubes, and visual inspections for the other primary-side and secondary-side components. The visual inspections of primary-side components are performed in accordance with the Degradation Assessment (DA) that is prepared as each steam generator is scheduled for examination.

The *Steam Generators* program utilizes industry endorsed guidance regarding tube inspections, evaluation and repair, and leakage monitoring techniques to ensure tube integrity of the steam generators. Aging is managed through assessment of potential degradation mechanisms, inspections, tube integrity assessments, plugging and repairs, primary-to-secondary leakage monitoring, maintenance of secondary side component integrity, primary-side and secondary-side water chemistry, and foreign material exclusion. Implementing procedures specify the performance criteria for tube integrity, condition monitoring requirements, inspection scope and frequency, acceptance criteria for the plugging or repair of flawed tubes, acceptable tube repair methods, leakage monitoring requirements, and operational leakage and accident-induced leakage requirements from the Technical Specifications (TS).

Provisions in the *Steam Generators* program address reporting criteria, inspection scope and frequency, assessments, plugging criteria, and water chemistry monitoring to maintain consistency with established requirements. Those requirements appear in the following documents:

- Technical Specifications (and Technical Requirements Manual)
- Maintenance Rule (10 CFR 50.65)
- EPRI 3002007571, "Steam Generator Integrity Assessment Guidelines"
- EPRI 3002007572, "PWR Steam Generator Examination Guidelines"
- EPRI Technical Report TR1022832, "PWR Primary-to-Secondary Leak Guidelines"
- EPRI 3002007856, "Steam Generator In-Situ Pressure Test Guidelines."

The EPRI guidelines provide a generic industry program to implement the expectations from NEI 97-06, Revision 3, "Steam Generator Program Guidelines."

The original steam generators were replaced for Unit 1 in 1993 and for Unit 2 in 1995. The steam generator replacement projects involved replacing the lower section of each steam generator and refurbishing the upper section. The replacement steam generators incorporated Alloy 690 thermally-treated tubes to improve reliability and minimize aging.

The *Steam Generators* program includes plant-specific steam generator DAs that identify existing and potential degradation mechanisms and associated aging effects that could impact the integrity of the steam generators. The DA identifies qualified tube inspection techniques and defines the scope of inspections that are appropriate for the detection and characterization of those aging effects, which consist of cracking, loss of material (e.g., wall thinning), and reduction of heat transfer. ~~As stated in the DA, U-tube and primary-side inspections are normally performed every third refueling outage for each steam generator, thus satisfying the guidance for inspections to be performed at least every 72 effective full power months or every third refueling outage, whichever results in more frequent inspections.~~By letter dated September 9, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21252A514), as supplemented by letter dated December 16, 2021 (ADAMS Accession No. ML21350A408), Dominion submitted a request for an amendment to the Technical Specifications (TSs) for NAPS Units 1 and 2. As requested by Dominion, the Nuclear Regulatory Commission (NRC) approved the proposed changes to the TS for NAPS Units 1 and 2 to adopt Technical Specification Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections," on March 22, 2022 (ADAMS Accession No. ML22068A071). The change to the TS increased the steam generator (SG) tube inspection frequency from at least every 72 effective full power months (EFPM) to at least every 96 EFPM. This allows for the SG primary-side inspections to be performed commensurate with the approved TS inspection frequency of at least every 96 EFPM for the SG tubes. The DA includes a review of applicable industry operating experience (OE) and plant-specific OE which has occurred since the previous DA was performed. The DA review determines the existence of any unaddressed mechanism that could adversely affect steam generator primary-side or secondary-side integrity, as well as the effects of any chemistry excursions or transients that could affect existing degradation mechanisms. An excursion of secondary chemistry could lead to fouling of heat transfer surfaces and a reduction of heat transfer thermal performance.

The DA indicates that primary-side inspections include video/visual examinations, specifically including:

- Tube plugs
- Tube-to-tubesheet welds
- Stub runner and divider plate
- Stub runner to divider plate weld
- Stub runner to tubesheet clad weld
- Divider plate-to-channel head clad weld
- Tubesheet cladding
- Closure ring welds
- Bottom of the bowl cladding

The analysis of the steam generator tube-to-tubesheet welds and the channel head design and loading provided by EPRI Technical Report 3002002850, "Steam Generator Management Program: Investigation of Crack Initiation and Propagation in the Steam Generator Channel Head Assembly" is applicable and bounding. A plant specific aging management program is not required for the primary-side channel head. The steam generator tubesheet is clad with Alloy 82, and the Alloy 690 thermally treated tubes are joined to the tubesheet with autogenous welds. General visual inspections of the tubesheet region looking for evidence of cracking (e.g., rust stains on the tubesheet cladding) are performed as part of this program.

The *Steam Generators* program includes preventive measures to mitigate aging related to corrosion phenomena through foreign material exclusion as a means to inhibit tube degradation due to wear. Identification of deposits on the secondary side of the steam generator, and the subsequent removal of sludge deposits help avoid tube degradation. Sludge mapping occurs when the steam generator is inspected, and inspections for remaining foreign material are performed after sludge lancing is completed. Sludge lancing, steam drum inspections, and feedring inspections typically are performed at least every third refueling outage. As an additional preventive measure, the *Water Chemistry* program (B2.1.2) monitors and controls reactor water chemistry and secondary water chemistry for the steam generators consistent with EPRI 3002000505, "PWR Primary Water Chemistry Guidelines," and EPRI 3002010645, "PWR Secondary Water Chemistry Guidelines".

The TS include the following requirements which have been incorporated in the *Steam Generators* program:

- Conducting condition monitoring assessments for each refueling outage during which steam generator tubes are inspected or plugged.
- Maintaining steam generator tube integrity by meeting performance criteria for tube structural integrity, accident-induced leakage, and operational leakage.
- Installing plugs in tubes found by inservice inspection to contain flaws with a depth equal to, or exceeding, 40% of the nominal tube wall thickness.
- Performing periodic inspections of steam generator tubes. Inspection scope, methods, and interval ensure that tube integrity is maintained until the next planned inspection.
- Monitoring primary-to-secondary leakage.
- Monitoring secondary water chemistry to ensure controls are in place to inhibit steam generator tube degradation.

Non-destructive examination techniques are used to inspect tubing materials in order to identify tubes that may need to be removed from service in accordance with the TS. The *Steam Generators* program utilizes volumetric examination techniques for the tubes, and visual examinations for other primary-side and secondary-side components. The *Steam Generators* program defines specific examination techniques, and describes criteria for the qualification of personnel, and for the acquisition and analysis of data. Assessment of tube integrity and plugging criteria of flawed tubes is in accordance with the TS and the *Steam Generators* program implementing procedures. Tube plugs with indications of aging are evaluated for corrective actions in accordance with the Corrective Action Program and the *Steam Generators* program.

Condition monitoring assessments are performed to determine whether structural and accident leakage criteria have been satisfied during the previous operating cycle(s). Operational assessments are performed after inspections are completed to verify that structural and leakage integrity will be maintained for the operating interval between inspections, which is selected in accordance with the TS and EPRI Steam Generator Integrity Assessment Guidelines. Comparison of the results of the condition monitoring assessment with the predictions of the previous operational assessment provides feedback for evaluation of the adequacy of the operational assessment and additional insights that can be incorporated into the next operational assessment. The condition monitoring, and performance monitoring methods, are effective in detecting the applicable aging effects, and the frequency of monitoring is adequate to prevent significant age-related degradation.

The *Steam Generators* program is implemented as a Fleet program at Dominion. The Fleet program requirements and Fleet implementation procedures have been previously reviewed and evaluated by the NRC Staff and a determination was made that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the subsequent period of extended operation, as required by 10 CFR 54.21(a)(3) (ADAMS Accession No. ML19360A020).

NUREG-2191 Consistency

The *Steam Generators* program is an existing program that is consistent, with exception, to NUREG-2191, Section XI.M19, Steam Generators.

Exception Summary

~~None~~The following program element(s) are affected:

Parameters Monitored/Inspected (Element 3) and Detection of Aging Effects (Element 4)

1. Section XI.M19 of NUREG-2191, Steam Generators, specifies that SG divider plate assemblies, tube-to-tubesheet welds, heads (channel or lower/upper heads), and tubesheets be visually inspected at least every 72 EFPM or every third refueling outage, whichever results in more frequent inspections. The Steam Generators program takes exception to the NUREG-2191 inspection frequency requirement of at least 72 EFPM.

Justification for Exception

The Nuclear Regulatory Commission (NRC) approved Amendment Nos. 292 and 275 to the TS for NAPS Units 1 and 2 to adopt Technical Specification Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections," on March 22, 2022 (ADAMS Accession No. ML22068A071). As part of adopting TSTF-577, the SG tube inspection frequency in NAPS TS 5.5.8, "Steam Generator (SG) Program," was revised from at least every 72 EFPM to at least every 96 EFPM. This allows for the SG primary-side inspections to be performed commensurate with the approved TS inspection frequency of at least every 96 EFPM for the SG tubes.

The revision to the SG tube inspection frequency to at least every 96 EFPM was based on operating history and justified by a unit-specific operational assessment (OA). Like the SG tubes, the inspection frequency of at least every 96 EFPM for the SG primary-side inspections is also based on operating history and justified by a unit-specific OA.

During the Spring 2018 refueling outage for Unit 1 (1RF26) and the Spring 2019 refueling outage for Unit 2 (2RF26), visual inspections of the primary-side components were performed and the examination results indicated no age-related degradation of these components.

Performance of the SG tube inspections and the SG primary-side component inspections both require access via the SG primary-side manways. As such, alignment of at least every 96 EFPM for the visual inspections of SG components with the NRC approved inspection frequency of at least every 96 EFPM for the SG tubes reduces both safety and radiological risk.

Based on the above, the exception to the NUREG-2191 inspection frequency requirement of at least every 72 EFPM is justified.

Enhancements

None

Operating Experience Summary

The following examples of operating experience provide objective evidence that the *Steam Generators* program has been, and will be effective in managing the aging effects for SSCs within the scope of the program so that the intended functions will be maintained consistent with the current licensing basis during the subsequent period of extended operation.

1. In March 2009, a foreign object was found in the Unit 1 'A' steam generator during a foreign object search and retrieval examination. The object was retrieved and the tubes in the vicinity of the foreign object were visually examined. No significant tube wall loss was identified and no further evaluation was required.
2. In September 2009, Regulatory Issue Summary 2009-04 (RIS 2009-04), "Steam Generator Tube Inspection Requirements," provided guidance on the implementation of the steam generator inspection requirements. The Steam Generator Inspections AMA (UFSAR Section 18.2.18) was reviewed and found to be consistent with the interpretations provided in RIS 2009-04. No further action was needed.
3. In March 2010, foreign material was found in the Unit 2 'C' steam generator during the post sludge-lancing top-of-tubesheet inspection. The material was found to be weld slag, and was retrieved. Inspection of surrounding tubes noted no damage or marks on tubes. No further action was required.
4. In March 2010, a small metallic piece of material was found in the Unit 2 'A' steam generator during the post sludge-lancing top-of-tubesheet inspection. There were no indications of wear on the surrounding tubes. The piece of foreign material was removed. No further action was needed.

5. In April 2010, foreign material was noted in Unit 2 'B' steam generator during the post sludge-lancing top-of-tubesheet inspection. Material was noted as wire-like, and it appeared the ends were embedded in a portion of the remaining sludge. The wire was sized less than 0.020 inch in diameter by less than 0.5 inch in length. The tubes adjacent to the wire, and ten surrounding tubes, were +Point examined and no tube degradation was identified. The exposed portion of the wire broke free during a retrieval attempt. An evaluation was provided in the Condition Monitoring and Operational Assessment document. Due to the small diameter, length, and minimal mass, the remaining portion of the wire was determined to not pose a threat to tube integrity.
6. In September 2011, foreign material was noted protruding from the hot leg end of a tube in the Unit 1 'A' steam generator. An additional piece of foreign material was located in the hot leg bowl. An apparent cause evaluation stated that the presence of the foreign material was a breakdown of the Foreign Material Exclusion boundary for the reactor coolant system. The surface of the steam generator tubesheet did not show any evidence of impacts on the cladding or tube ends. No tube wear was observed in the tube containing the foreign material. Therefore, there was no evidence that the foreign material caused any damage to the steam generator or reactor coolant system. The foreign material was removed.
7. In September 2013, ultrasonic testing (UT) examination results for Unit 1 'A' steam generator indicated wear in the vicinity of a J-tube nozzle. Additional UT examinations and visual inspections for the internal surfaces of the feedrings for all three steam generators resulted in repairs being planned for several J-tube nozzles during the following outage. An Engineering evaluation concluded that the affected J-tube nozzles were fully capable of performing their design basis function, and would maintain structural integrity for an additional 18-month cycle until the repairs could be performed.
8. In September 2013, during post sludge-lancing visual inspection of Unit 1 'A' steam generator, a loose part was identified at the top of the tubesheet. The part appeared to be weld slag, and was removed. Eddy current testing for the two tubes that were in contact with the loose part determined there was no wall loss on either tube contacted by the loose part. Further eddy current examinations for the row of surrounding tubes did not identify any degradation. No further action was required.
9. In March 2016, a foreign object was identified by eddy current testing in a Unit 2 'C' steam generator cold leg tube at the third tube support plate. Foreign object wear with a maximum wall loss of 33%, believed to be from the same foreign material, was identified on an adjacent cold leg tube. The location was inaccessible for foreign object retrieval. The two tubes were plugged at the cold leg and the hot leg, and were stabilized at the cold leg.

10. In December 2016, as part of oversight review activities, a review of procedures credited by initial license renewal AMAs was conducted to confirm the following:
    - Procedures credited for license renewal were identified
    - Procedures were consistent with the licensing basis and bases documents
    - Procedures contained a reference to conduct an aging management review prior to revising
    - Procedures credited for license renewal were identified by an appropriate program indicator and contained a reference to a license renewal document

    Procedure changes were completed as necessary to ensure the above items were satisfied.
11. In May 2017, an assessment was performed to determine the progress and substance of license commitment closure and readiness for the IP 71003 NRC Phase II inspection to be conducted for Units 1 and 2 during November and December of 2017. The conclusion was reached that no areas for improvement or enhancements were identified for the Steam Generator Inspections AMA (UFSAR Section 18.2.18).
12. In October 2017, NSAL-12-1, "Steam Generator Channel Head Degradation", was issued to describe degradation of the Steam Generator channel head cladding in a Westinghouse-designed steam generator. Recommended action from NSAL-12-1 was to perform a visual inspection to identify potential breaches in the cladding. No additional action was necessary since steam generator bowl scans of each channel head are performed during primary-side inspections.
13. In March 2018, foreign material was found on the internal surface of the feedring of Unit 1 'B' steam generator. The piece of foreign material was removed, and was determined to be a backing ring used for initial welding and installation of the feedring. No damage was identified within the feedring. An inspection of the entire feedring was conducted and other backing rings and other materials were verified intact. Inspections were also performed within the feedrings for the 'A' and 'C' Steam Generators. Backing rings and other components were intact.
14. In April 2019, an effectiveness review was performed on the Steam Generator Inspections AMA (UFSAR Section 18.2.18). The AMA was evaluated against the performance criteria identified in NEI 14-12, "Aging Management Program Effectiveness." No gaps were identified by the effectiveness review.

The above examples of operating experience provide objective evidence that the *Steam Generators* program includes activities to perform volumetric and visual inspections to identify cracking, loss of material and reduction of heat transfer for primary-side components and secondary side components contained within the steam generator that are within the scope of subsequent license renewal, and to initiate corrective actions. Occurrences identified under the *Steam Generators* program are evaluated to ensure there is no significant impact to the safe operation of the plant and corrective actions will be taken to prevent recurrence. Guidance or corrective actions for additional inspections, re-evaluation, repairs, or replacements is provided for locations where aging effects are found. The program is informed and enhanced when necessary through the systematic and ongoing review of both plant-specific and industry operating experience. There is reasonable assurance that the continued implementation of the *Steam Generators* program will effectively manage aging prior to a loss of intended function.

Conclusion

The continued implementation of the *Steam Generators* program provides reasonable assurance that aging effects will be managed such that the components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis during the subsequent period of extended operation.