UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 9, 2021

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Dominion Energy Nuclear Connecticut, Inc.
Millstone Power Station
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 – ISSUANCE OF AMENDMENT NO. 343 RE: REVISION TO TECHNICAL SPECIFICATIONS FOR STEAM GENERATOR INSPECTION FREQUENCY (EPID L-2020-LLA-0227)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 343 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (Millstone 2), in response to your application dated October 8, 2020, as supplemented by letters dated December 8, 2020, and April 15, 2021.

The amendment revises the Millstone 2 Technical Specification (TS) Section 6.26, "Steam Generator (SG) Program," and the SG tube inspection reporting requirements in TS Section 6.9.1.9, "Steam Generator Tube Inspection Report," and makes several editorial changes to the Millstone 2 TSs.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 343 to DPR-65
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

MILLSTONE POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 343
Renewed License No. DPR-65

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Dominion Energy Nuclear Connecticut, Inc. (the licensee) dated October 8, 2020, as supplemented by letters dated December 8, 2020, and April 15, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 343 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

Digitally signed by James G. Danna
Date: 2021.09.09 10:59:06 -04'00'

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: September 9, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 343

MILLSTONE POWER STATION, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
6-20	6-20
6-20a	6-20a
---	6-20b
6-30	6-30
6-31	6-31
6-31a	6-31a

Connecticut, in accordance with the procedures and limitations set forth in this renewed operating license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 2700 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 343 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.9 A report shall be submitted within 180 days after initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.26, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

1. The nondestructive examination techniques utilized;

2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

4. The number of tubes plugged during the inspection outage.

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, input, and results;

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG; and

f. The results of any SG secondary side inspections.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator, Region I, and one copy to the NRC Resident Inspector within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Deleted

b. Deleted

c. Deleted

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Deleted

f. Deleted

g. RCS Overpressure Mitigation, Specification 3.4.9.3.

h. Deleted

i. Tendon Surveillance Report, Specification 6.25

j. Deleted

k. Accident Monitoring Instrumentation, Specification 3.3.3.8.

l. Radiation Monitoring Instrumentation, Specification 3.3.3.1.

m. Deleted

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

6.12.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of a Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

6.26 STEAM GENERATOR (SG) PROGRAM

An SG Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following:

a. Provisions for condition monitoring assessments: Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Provisions for performance criteria for SG tube integrity: SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

    1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including STARTUP, operation in the power range, HOT STANDBY, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

    2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gpd per SG.

    3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational LEAKAGE."

6.26 STEAM GENERATOR (SG) PROGRAM (Continued)

c. Provisions for SG tube plugging criteria: Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections: Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1., d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

   1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

   2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

   3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 343

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-65

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated October 8, 2020 (Reference 1), and as supplemented by letters dated December 8, 2020 (Reference 2) and April 15, 2021 (Reference 3), Dominion Energy Nuclear Connecticut, Inc. (the licensee), requested changes to the technical specifications (TS) (Reference 4) for Millstone Power Station, Unit 2 (Millstone 2). The proposed changes would alter the steam generator (SG) tube inspection requirements in TS Section 6.26, "Steam Generator (SG) Program," and the SG tube inspection reporting requirements in TS Section 6.9.1.9, "Steam Generator Tube Inspection Report," and would make several editorial changes to the Millstone 2 TSs. The licensee requested that the changes be approved as a license amendment in accordance with Section 50.90, "Application for amendment of license, construction permit, or early site permit," of Title 10 of the *Code of Federal Regulations* (10 CFR), "Energy."

The supplemental letters dated December 8, 2020, and April 15, 2021, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 26, 2021 (86 FR 7115).

2.0 REGULATORY EVALUATION

2.1 System Description

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. SG tube integrity means that the tubes can perform this safety function in accordance with the plant design and licensing basis.

2.2 Regulatory Requirements and Guidance

Fundamental regulatory requirements with respect to the integrity of the SG tubing are established in 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities.” Specifically, General Design Criterion (GDC) 14, “Reactor coolant pressure boundary” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, states that the RCPB shall be “designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.” GDC 15, “Reactor coolant system design” states that “the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.” GDC 30, “Quality of reactor coolant pressure boundary,” states in part that “components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical.” GDC 31, “Fracture prevention of reactor coolant pressure boundary” states, in part, that the RCPB “shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.” GDC 32, “Inspection of reactor coolant pressure boundary” states, in part, that RCPB components shall be “…designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity.”

For plants that were issued construction permits before the effective date of 10 CFR Part 50, Appendix A, the plant-specific principal design criteria in the plant design basis established similar fundamental regulatory requirements pertaining to the integrity of the SG tubing. Millstone 2 received a construction permit prior to May 21, 1971, which is the date the GDC in Appendix A of 10 CFR Part 50 became effective. The Millstone 2 Updated Final Safety Analysis Report (UFSAR) addresses the general design criteria of Appendix A to 10 CFR Part 50 in Section 1.A, “AEC [Atomic Energy Commission] General Design Criteria for Nuclear Power Plants,” of the UFSAR (Reference 5).

Section 50.55a to 10 CFR specifies that components which are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), except as provided in 10 CFR 50.55a(c)(2), (3), and (4). Section 50.55a further requires that throughout the service life of pressurized-water reactor (PWR) facilities like Millstone 2, ASME Code Class 1 components must meet the Section XI requirements of the ASME Code to the extent practical, except for design and access provisions, and pre-service examination requirements. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. The Section XI requirements pertaining to in-service inspection of SG tubing are augmented by additional requirements in the TS.

Section 182(a) of the Atomic Energy Act requires nuclear power plant operating licenses to include TS as part of any license. In 10 CFR 50.36, “Technical specifications,” the U.S. Nuclear Regulatory Commission (NRC) regulatory requirements related to the content of the TSs are established. The TSs for all current PWR licenses require that an SG program be established and implemented to ensure that SG tube integrity is maintained.

2.3 Steam Generator Tube Integrity Requirements in the Millstone 2 Technical Specifications

At Millstone, programs established by the licensee to operate the facility in a safe manner, including the SG Program, are listed in the administrative controls section of the TS. For Millstone 2, the requirements for performing SG tube inspections and repair are in TS Section 6.26, while the requirements for reporting the SG tube inspections and repair are in TS Section 6.9.1.9.

For Millstone 2, SG tube integrity is maintained by meeting the performance criteria specified in TS Section 6.26.b for structural and leakage integrity, consistent with the plant design and licensing basis. TS Section 6.26.a requires that a condition monitoring (CM) assessment be performed during each outage in which the SG tubes are inspected to confirm that the performance criteria are being met. TS Section 6.26.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube and that may satisfy the applicable tube plugging criteria. The applicable tube plugging criterion specified in TS Section 6.26.c is that tubes found during in-service inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged.

Millstone 2 operational LEAKAGE performance criterion is specified in Limiting Condition for Operation 3.4.6.2, “Reactor Coolant System Operational LEAKAGE,” and includes a limit on primary-to-secondary leakage beyond which the plant must be promptly shut down. Should a flaw exceeding the tube plugging limit not be detected during the periodic tube surveillance required by the plant TS, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired, consistent with the design and licensing bases.

As part of the plant’s licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture and a steam line break. These analyses consider primary-to-secondary leakage that may occur during these events and must show that the radiological consequences do not exceed the applicable limits of 10 CFR 50.67, “Accident source term,” or 10 CFR 100.11, “Determination of exclusion area, low population zone, and population center distance,” for offsite doses; GDC 19, “Control room,” of 10 CFR Part 50, Appendix A, for control room operator doses (or some fraction thereof, as appropriate to the accident); or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for Millstone 2 are being changed as part of the proposed amendment; thus, no radiological consequences of any accident analysis are being changed. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes.

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 Steam Generator Design

Millstone 2 has two replacement SGs manufactured by Babcock and Wilcox that were installed in 1992. Each SG contains 8,523[1] thermally treated Alloy 690 tubes, which have a nominal outside diameter of 0.75 inches and a nominal wall thickness of 0.0445 inches. The tubes were hydraulically expanded into the tubesheet, which is 21.06 inches thick and triangular pitch. The straight portion of the tubes are supported by 7, Type 410 stainless steel lattice grids, and the U-bend portion of the tubes are supported by 12, Type 410 stainless steel fan bar assemblies.

3.1.2 Operating Experience

Both SGs in Millstone 2 were last inspected in spring 2017 (Refueling Outage 24 (2R24)). Full-length bobbin probe examinations of 100 percent of the in-service tubes in both SGs were performed during 2R24. Additional information regarding the SG inspections at Millstone 2 is available in the spring 2017 SG Tube Inspection Report (Reference 6).

Since the SGs were placed in service in 1992, a total of 32 tubes have been plugged (19 in SG 1 and 13 in SG 2). Section 3.10 in Attachment 1 of Reference 1 identifies that one tube in SG 1 was plugged preservice and the remaining tubes in both SGs were plugged due to foreign objects (FOs). The licensee stated that seven tubes were plugged due to FO wear; five of which contained minor FO wear and were plugged due to a qualified sizing technique not being available at the time. The remaining tubes were plugged because the FOs were determined to be irretrievable at the time.

Millstone 2 has the following existing degradation mechanisms: mechanical wear at fan bar supports and wear from FOs. Inspections during 2R24 reported a total of 20 wear indications from all mechanisms in the two Millstone 2 SGs. Table 1 summarizes the wear indications by mechanism reported during 2R24, the latest inspection.

Table 1: Wear Indications Reported in Millstone 2 SGs (2R24, Spring 2017)

Wear Mechanism	Total Number of Indications in Each SG		
	SG 1	SG 2	Total
Fan Bar	2	2	4
FOs (New and Historical)	1	15	16

Millstone 2 has not reported any indications of corrosion degradation, such as stress corrosion cracking (SCC).

Reference 6 summarizes the primary side visual inspections performed during 2R24, which included visual inspection of the channel heads of both SGs and all previously installed tube plugs. The licensee reported that the channel head inspections revealed no degradation of the

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[1] SG 1 has 8,522 tubes because one hot-leg tubesheet hole was plugged during construction and the associated cold-leg tubesheet hole was not drilled.

divider plates, divider plate retaining bars/welds, primary closure rings/welds, and cladding; and that there was no degradation, leakage, or misplacement of the previously installed tube plugs.

Reference 6 summarizes the secondary-side activities and inspections performed during 2R24, which included sludge lancing in both SGs followed by a post sludge lancing visual inspection of the top-of-tubesheet annulus and no-tube lane, and foreign object search and retrieval (FOSAR). In addition, visual inspections of accessible locations with eddy current indications potentially related to FOs; and visual inspections of the steam drums, including moisture separators, drain systems, and interior surfaces were performed during 2R24. The licensee reported that no degradation was detected during the secondary-side visual inspections. In Reference 3, the licensee clarified that orange discoloration was observed on the skimmer vanes, lower cylinder assembly sidewalls, swirl vanes, and baseplates of the secondary moisture separators during 2R24. However, the licensee stated there was an absence of magnetite in these regions and that has not changed appreciably over multiple operating cycles. The licensee also stated that neither perforations nor visible material loss were evident in these regions.

Sixteen (1 in SG 1 and 15 in SG 2) total indications of FO wear on fourteen tubes were identified during 2R24. Only four (1 in SG 1 and 3 in SG 2) of the sixteen FO wear indications were newly identified during 2R24. Based on review of previous bobbin probe data, the new FO wear indication in SG 1 (Row 82, Column 143) was present since 1997 but 2R24 was the first time a +Point™ probe was used in the location of the FO wear indication. No FOs were identified at the newly identified FO wear indications in either SG and no tubes were plugged in either SG due to FOs during 2R24.

The licensee stated that no primary-to-secondary leakage was reported during the previous operating period.

3.2. Proposed TS Changes

3.2.1 Description of Current TS Requirements

Section 6.26 of the Millstone 2 TS provides the SG tube inspection requirements for Millstone 2. Section 6.26.d.2 requires, in part, that, "After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months [EFPM] or at least every third refueling outage (whichever results in more frequent inspections)."

The Millstone 2 TS Sections 6.26.d.2.a), b), c), and d) define the SG tube inspection requirements for the first, second, third, and fourth and subsequent inspection periods following SG installation. Millstone 2 is currently in the second inspection period. Specifically, Millstone 2 TS Section 6.26.d.2.b) states, "During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period."

Section 6.26.c of the Millstone 2 TS requires the next inspection of each affected and potentially affected SG after crack indications are found not to exceed 24 EFPM or one refueling outage (whichever results in more frequent inspections).

Section 6.9.1.9 of the Millstone 2 TS requires that a report be submitted within 180 days after the initial entry into hot shutdown (MODE 4) following completion of an inspection of the SGs

performed in accordance with TS Section 6.26. In addition, TS Section 6.9.1.9 includes the information that shall be included in the report.

3.2.2 Description of Proposed TS Changes

The license amendment request (LAR) proposes to revise Millstone 2 TS 6.26.d.2 from requiring SG inspections every 72 EFPM, or at least every third refueling outage (whichever results in more frequent inspections), to every 96 EFPM. The change would remove the requirement to base inspection frequency on the more restrictive metric between either the EFPM or refueling outage and use just the EFPM metric. Because Millstone 2 operates on an 18-month operating cycle, this change would result in SG inspections being required at least every fifth refueling outage.

The LAR also proposes to eliminate Millstone 2 TS Sections 6.26.d.2.a), b), c), and d), which currently define the first, second, third, and fourth and subsequent inspection periods. As previously stated, TS Section 6.26.d.2 would be revised with a 96 EFPM inspection period that would be applicable for all future inspection periods. By using a common frequency of 96 EFPM for both the required SG inspections and the requirement to inspect 100 percent of the tubes, SG inspections will occur less frequently, but all tubes in the SG will be required to be inspected more frequently.

The LAR proposes to revise Millstone 2 TS Section 6.26.d.3 from requiring the next inspection after crack indications are found not to exceed 24 EFPM or one refueling outage (whichever results in more frequent inspections), to requiring that each affected and potentially affected SG be inspected at the next refueling outage after crack indications are found.

The LAR also proposes to revise Millstone 2 TS Section 6.9.1.9 by adding requirements to report (1) nondestructive examination techniques utilized for tubes with increased degradation susceptibility; (2) a description of the CM results, including the margin to tube integrity performance criteria and comparison with the margin predicted to exist by the previous forward-looking integrity assessment (operational assessment (OA)); (3) a summary of the tube integrity conditions predicted to exist by the OA at the next inspection, including the analysis methodology, inputs and results; and (4) the results of any secondary-side inspections performed. In addition, a requirement that only the total number of tube wear indications at support structures less than 20 percent through-wall (TW) needs to be reported is proposed to be added.

Additionally, TS Sections 6.26 and 6.9.1.9 would be revised by other editorial and punctuation changes.

3.3 <u>Staff Evaluation of Proposed TS Changes</u>

The NRC staff evaluation of the proposed TS changes focused on the potential for these changes to affect SG tube integrity, since maintaining SG tube integrity is a current TS requirement that plays a key role in protecting the public's health and safety. In particular, the evaluation assessed whether the technical justification in References 1 and 2 demonstrate that the structural integrity performance criterion (SIPC) and accident-induced leakage performance

criterion (AILPC) will continue to be met with the revised inspection intervals proposed in Reference 1. These tube integrity criteria are defined in TS Section 6.26.b.

3.3.1 Evaluation of Existing Tube Degradation Mechanisms

The Millstone 2 inspections have detected tube degradation from mechanical wear at fan bar supports and wear from FOs, and the OA provided by the licensee evaluates these as existing mechanisms using arithmetic deterministic analyses. These analyses use the worst-case single-tube analysis method of the Electric Power Research Institute (EPRI) Steam Generator Integrity Assessment Guidelines (IAG) (Reference 7) to provide a conservative estimate of the projected end-of-cycle (EOC) condition considering all uncertainties at a probability of 0.95 and at 50 percent confidence. The uncertainties used in the assessment are for the burst equation, the material strength, and the nondestructive examination flaw sizing technique. The single tube methods are referred to as “worst-case degraded tube” methods because the most severely flawed tube is selected for evaluation. The worst-case degraded tube OA methods involve selecting the most severely flawed tube at the beginning-of-cycle (BOC) and applying conservative flaw growth over the intended inspection interval, to arrive at a predicted EOC flaw size and then determine if the SIPC and AILPC will be met at the EOC.

Tube Wear at Fan Bars

The licensee’s OA for fan bar wear was performed using the arithmetic approach described in the EPRI IAG. This approach adjusts the deepest fan bar wear flaw returned to service to account for eddy current sizing uncertainty, applies a 95^{th} percentile growth rate, and then compares the projected flaw size at the next EOC inspection to the acceptable structural limits.

The 2R24 inspections for fan bar wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes. There were four fan bar wear indications detected during 2R24 that ranged in depth from 12 to 19 percent TW. The fan bar wear indications were sized using an EPRI qualified examination technique. No tubes were plugged during 2R24 due to fan bar wear. The largest fan bar wear indication allowed to remain in service in 2R24 was 19 percent TW in SG 1. Given the small number of fan bar wear indications, the licensee trended the wear progression over multiple inspections. The licensee assumed a fan bar wear rate of 3.0 percent TW per effective full power years (EFPY), even though the highest average growth rate between the two Millstone 2 SGs was determined to be 0.74 percent TW/EFPY. The fan bar wear rate of 3.0 percent TW/EFPY was applied and the wear depth at 2R29 (after five operating cycles) was projected to be 48 percent TW, which is less than the structural limit of 50.2 percent TW for fan bar wear with a bounding length of 3.20 inches. The structural limit is calculated assuming a bounding wear length, applying a three times normal operating pressure differential (3xNOPD), and including material property and burst pressure equation uncertainties. For flaws of this type, for pressure loading only, satisfying the SIPC demonstrates that the AILPC will also be satisfied since the limiting accident induced pressure differentials are much less than 3xNOPD. Therefore, the licensee concluded the SIPC and AILPC would be met for fan bar wear until the next inspection in 2R29 (fall 2024).

Foreign Object Wear

In addition to wear at support structures, Millstone 2 has also experienced tube wear from FOs that have been transported into the SGs. The 2R24 inspections for FO wear consisted of full-length bobbin probe examinations of 100 percent of the in-service tubes, 50 percent +Point™ probe examinations of a six tube deep periphery at the top-of-tubesheet (TTS) in both

the hot- and cold-legs, bounding +Point™ probe examinations of indications of potential FOs, secondary-side inspection of TTS annulus and bundle periphery, and FOSAR. Sixteen total indications of FO wear on 14 tubes were identified during 2R24. Four of the 16 FO wear indications were newly identified during 2R24. No FOs were identified at the newly identified FO wear indications in either SG and no tubes were plugged in either SG due to FOs during 2R24. All eddy current possible loose parts (PLPs) identified during 2R24 were dispositioned as either the FO was removed; no FO identified; or the PLP, which has been present for multiple cycles without causing detectable wear, will continue to be monitored. New FOs identified during 2R24, not associated with wear indications, that remain in the SGs were determined not to present risk to tube degradation. During 2R24, historical FO wear indications where FOs were previously removed were resized. The licensee stated that the historical FO wear indications exhibited no growth considering the sizing uncertainty. In addition, locations with known parts remaining in the SGs were reexamined to confirm the part is still present and that there is no change in tube wear.

The NRC staff also acknowledges that predicting future FO and loose part generation is not possible since past fleet-wide operating experience has shown that new loose part generation, transport to the SG tube bundle, and interactions with the tubes cannot be reliably predicted. However, plants can reduce the probability of FOs and loose parts by maintaining robust foreign material exclusion programs and applying lessons learned from previous industry operating experience with loose parts. Plants in general, including Millstone 2, have demonstrated the ability to conservatively manage loose parts once they are detected by eddy current examinations or by secondary-side FOSAR. If unanticipated aggressive tube wear from new FOs or loose parts should occur in a Millstone 2 SG, operating experience has shown that a primary-to-secondary leak is more likely to occur, rather than a loss of tube integrity. In the event of a primary-to-secondary leak, the NRC staff will interact with the licensee in accordance with established procedures in Inspection Manual Chapter (IMC) 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (Reference 8), to confirm the licensee's conservative decision making.

Evaluation Summary of Existing Tube Degradation Mechanisms

The NRC staff finds the licensee's evaluation of tube wear at fan bars to be acceptable. Wear at these locations in the SGs has been effectively managed for many cycles without challenging tube integrity. No tubes have been plugged due to fan bar wear. Wear at support structures is readily detected with standard eddy current examination techniques and wear sizing errors are considered in the projection of existing flaws until 2R29. Due to the small number of fan bar wear indications at Millstone 2, the licensee trended the wear progression over multiple inspections and assumed a fan bar wear rate greater than the highest average growth rate between the two Millstone 2 SGs. The NRC staff found the determination of BOC flaw depth and growth rates acceptable because they are based on industry guidelines and conservative assumptions. For flaws of this type, for pressure loading only, satisfying the SIPC demonstrates that the AILPC will also be satisfied since the limiting accident induced pressure differentials are much less than 3xNOPD. Therefore, the NRC staff concludes there is reasonable assurance that the SIPC and AILPC will be met for all tubes until the next inspection in 2R29 (fall 2024). The NRC staff finds the evaluation of existing FO wear acceptable because it considered FO wear without a part present, eddy current PLPs without wear, known FOs remaining in the SGs, and FOs that may enter the SGs during the upcoming operating cycle. The licensee determined that FOs remaining in the SGs would not cause significant wear through the next SG

inspections and that tubes that no longer have a FO present will not incur additional wear and have been demonstrated to meet tube integrity.

3.3.2 Evaluation of Potential Tube Degradation Mechanisms

The OA performed following 2R24 considered lattice support wear, tube-to-tube wear, and thinning as potential degradation mechanisms.

Even though lattice support wear has not been detected at Millstone 2, it is considered a potential degradation mechanism because it has been detected at other replacement Babcock and Wilcox SGs. The licensee stated that lattice support wear has not been detected at Millstone 2 for almost 30 years of operation, therefore, it is “unlikely to initiate and rapidly progress to an unacceptable depth during the interval between inspections.”

Tube-to-tube wear has not been detected at Millstone 2; however, it is considered a potential degradation mechanism because it has been detected at other operating SGs. Causes of tube-to-tube wear at other operating SGs include less than the nominal gap between tubes and fluid elastic instability. The licensee stated that “neither of these conditions are known to exist” in the Millstone 2 SGs.

Tube thinning adjacent to support structures has not been detected at Millstone 2 and the licensee stated that it has a low likelihood of initiation and progression due to secondary water chemistry controls. Specifically, Millstone 2 does not use a phosphate chemistry, sulfate limits are very low, and crevice pH is typically not acidic.

The OA performed following 2R24 did not include any SCC mechanisms as potential mechanisms, as OA projections are only performed for existing degradation mechanisms, in accordance with the guidance provided in Reference 7. The NRC staff notes that while the licensee did not assume any SCC mechanisms as potential in the OA of the Alloy 690TT SG tubing, the licensee is required by their TS to perform a degradation assessment prior to each SG inspection and to perform inspections with inspection methods that are capable of detecting flaws of any type that may be present along the length of the tube. This TS requirement ensures that each SG inspection will look for all types of degradation that may be present, whether they are existing or potential.

The NRC staff finds the evaluation of potential degradation mechanisms acceptable because it considered potential degradation mechanisms that have occurred at other operating SGs that may be applicable to the Millstone 2 SGs. In addition, Millstone 2 has not reported any SCC mechanisms, and to date, the NRC staff is unaware of any corrosion degradation in operating SGs with Alloy 690TT tubing, and the Millstone 2 TS requires each SG inspection to look for all types of degradation that may be present. Therefore, the NRC staff concludes there is reasonable assurance that the SIPC and AILPC will be met for all tubes until the next inspection in 2R29 (fall 2024).

3.3.3 Evaluation of Steam Generator Inspection Reporting Requirements

Licensees are required to submit a SG tube inspection report to the NRC in accordance with their TS, typically within 180 days of the completion of the SG inspection. The NRC staff reviews each SG tube inspection report to ensure that the report includes the information required by the licensee’s TS, ensure that the inspections performed appear to be capable of detecting potential SG tube degradation, ensure that SG tube integrity is being effectively

managed, and determine whether the inspection results appear to be consistent with the operating experience at similarly designed and operated units.

The NRC staff has reviewed the proposed changes to the SG tube inspection reporting requirements described in Section 3.2.2 of this safety evaluation and determined that they are acceptable because they will provide additional detailed information to allow the staff to better understand the overall condition of the SGs. For example, new reporting requirement “f.” would be added to require the results of any SG secondary side inspections to be reported. Revising reporting requirement “c.2.” (renumbered from existing reporting requirement “d.”), to require that only the total number of tube wear indications at support structures less than 20-percent TW be reported, will allow the NRC staff to focus on the most significant wear indications and more efficiently trend tube wear indications at support structures if a large number of wear indications are reported. U.S. PWR operating experience has shown that individual tube reporting of support structure wear indications less than 20-percent TW is not needed to maintain SG tube integrity or for NRC staff to evaluate wear rate trends.

3.4 Technical Evaluation Conclusion

The NRC staff concludes that the information submitted by the licensee provides reasonable assurance that the Millstone 2 SG tubes will continue to maintain structural and leakage integrity. The NRC staff finds that the licensee has demonstrated the acceptability of the proposed changes to the Millstone 2 SG Program and inspection reporting requirements with respect to meeting the requirements of GDC 14, 15, 30, 31, and 32 of Appendix A to 10 CFR Part 50, 10 CFR 50.36, and 10 CFR 50.55a. Therefore, the NRC staff finds it acceptable for the licensee to incorporate the proposed changes into Millstone 2 TS Sections 6.26 and 6.9.1.9.

4.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Connecticut State official was notified on August 10, 2021, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 or changes SRs. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding, which was published in the *Federal Register* on January 26, 2021 (86 FR 7115), that the amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Proposed License Amendment Request to Revise the Millstone Power Station, Unit 2, Technical Specifications for Steam Generator Inspection Frequency, dated October 8, 2020 (ADAMS Accession No. ML20282A594).

2. Supplement to Proposed License Amendment Request to Revise the Millstone Power Station, Unit 2, Technical Specifications for Steam Generator Frequency, dated December 8, 2020 (ADAMS Accession No. ML20343A259).

3. Response to Request for Additional Information for Proposed License Amendment Request to Revise the Millstone, Unit 2, Technical Specification for Steam Generator Inspection Frequency, dated April 15, 2021 (ADAMS Accession No. ML21105A482).

4. Millstone Power Station, Unit 2, Current Facility Operating License DPR-65, Technical Specifications, Revised 03/01/2021 (ADAMS Accession No. ML052720294).

5. Millstone Power Station, Unit 2, Revision 38, to Updated Final Safety Analysis Report, Chapter 1: Introduction and Summary, dated June 22, 2020 (ADAMS Accession No. ML20209A362.

6. Millstone, Unit 2, End of Cycle 24 Steam Generator Tube Inspection Report, dated September 18, 2017 (ADAMS Accession No. ML17269A030).

7. Electric Power Research Institute, Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines, Revision 4, dated June 2016 (ADAMS Accession No. ML16208A272).

8. Inspection Manual Chapter 0327 Steam Generator Tube Primary-to-Secondary Leakage, dated January 1, 2019 (ADAMS Accession No. ML18093B067).

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Date of Issuance: September 9, 2021

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 – ISSUANCE OF AMENDMENT NO. 343 RE: REVISION TO TECHNICAL SPECIFICATIONS FOR STEAM GENERATOR INSPECTION FREQUENCY (EPID L-2020-LLA-0227) DATED SEPTEMBER 9, 2021

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