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

October 14, 2020

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

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 – ISSUANCE OF AMENDMENT NO. 277 TO REVISE TECHNICAL SPECIFICATION 6.8.4.G TO ALLOW A ONE-TIME DEFERRAL OF THE STEAM GENERATOR INSPECTIONS (EPID L-2020-LLA-0178)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 277 to Renewed Facility Operating License No. NPF-49 for the Millstone Power Station (Millstone), Unit No. 3, in response to your application dated August 11, 2020, as supplemented by letter dated September 4, 2020.

The amendment revises the technical specifications (TSs) by adding a note to TS 6.8.4.g, "Steam Generator (SG) Program," to permit a one-time deferral of the Millstone, Unit No. 3, steam generator A and C inspections from fall 2020 to spring 2022.

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. 50-423

Enclosures:
1. Amendment No. 277 to NFP-49
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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DOMINION ENERGY NUCLEAR CONNECTICUT, INC., ET AL

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 277
Renewed License No. NPF-49

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

   A. The application for amendment by Dominion Energy Nuclear Connecticut, Inc. (DENC, the licensee) dated August 11, 2020, as supplemented by letter dated September 4, 2020, 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.

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. NPF-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No. 277 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DENC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

Digitally signed by James G. Danna
Date: 2020.10.14 08:52:43 -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: October 14, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 277

MILLSTONE POWER STATION, UNIT NO. 3

RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

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
4	4

Replace the following page of the Appendix A Technical Specifications 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
6-17c	6-17c

(2) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No. 277 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DENC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) DENC shall not take any action that would cause Dominion Energy, Inc. or its parent companies to void, cancel, or diminish DENC's Commitment to have sufficient funds available to fund an extended plant shutdown as represented in the application for approval of the transfer of the licenses for MPS Unit No. 3.

(4) Immediately after the transfer of interests in MPS Unit No. 3 to DNC*, the amount in the decommissioning trust fund for MPS Unit No. 3 must, with respect to the interest in MPS Unit No. 3, that DNC* would then hold, be at a level no less than the formula amount under 10 CFR 50.75.

(5) The decommissioning trust agreement for MPS Unit No. 3 at the time the transfer of the unit to DNC* is effected and thereafter is subject to the following:

(a) The decommissioning trust agreement must be in a form acceptable to the NRC.

(b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Dominion Energy, Inc. or its affiliates or subsidiaries, successors, or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(c) The decommissioning trust agreement for MPS Unit No. 3 must provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee until the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.

(d) The decommissioning trust agreement must provide that the agreement cannot be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

* On May 12, 2017, the name "Dominion Nuclear Connecticut, Inc." changed to "Dominion Energy Nuclear Connecticut, Inc."

PROCEDURES AND PROGRAMS (Continued)

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. Portions of the tube below 15.2 inches below the top of the tubesheet are excluded from this requirement. 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 each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections)*. In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the

* As approved by License Amendment No. 277, inspection of the Millstone Unit 3 SGs A and C may be deferred, on a one-time basis, from fall 2020 (Refueling Outage 20) to spring 2022 (Refueling Outage 21).

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 277

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOMINION ENERGY NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated August 11, 2020 (Reference 1), as supplemented by letter dated September 4, 2020 (Reference 2), Dominion Energy Nuclear Connecticut, Inc. (the licensee) requested changes to the Millstone Power Station, Unit No. 3 (Millstone 3), Technical Specifications (TSs). The proposed changes would allow a one-time deferral of the steam generator (SG) tube inspections required by TS 6.8.4.g, "Steam Generator (SG) Program," for Millstone Unit 3 SGs A and C from fall 2020 to spring 2022. The proposed changes were submitted in response to recommendations (e.g., social distancing) provided by the Centers for Disease Control and Prevention in an attempt to limit the spread of the Coronavirus Disease 2019 (COVID-19).

The supplemental letter dated September 4, 2020, 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, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 8, 2020 (85 FR 55506).

2.0 REGULATORY EVALUATION

2.1 Description of System

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. For the purposes of this safety evaluation, SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis.

2.2 Regulatory Requirements and Guidance

Fundamental regulatory requirements with respect to SG tube integrity are established in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The general design criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provide regulatory requirements that state that the RCPB shall have "an extremely low probability of abnormal leakage … and of gross rupture" (GDC 14, "Reactor coolant pressure boundary"), "shall be

designed with sufficient margin" (GDC 15, "Reactor coolant system design," and GDC 31, "Fracture prevention of reactor coolant pressure boundary"), shall be of "the highest quality standards practical" (GDC 30, "Quality of reactor coolant pressure boundary"), and shall be designed to permit "periodic inspection and testing … to assess … structural and leaktight integrity" (GDC 32, "Inspection of reactor coolant pressure boundary").

Section 182 of the Atomic Energy Act of 1954, as amended, requires nuclear power plant operating licenses to include TSs. The NRC regulatory requirements related to the content of TSs are in 10 CFR 50.36, "Technical specifications." Given the importance of SG tube integrity, all current pressurized-water reactor (PWR) licenses have TSs governing the surveillance of SG tubes. These TSs require that an SG program be established and implemented to ensure that SG tube integrity is maintained. Programs established by licensees, including the SG program, are listed in the administrative controls section of the TSs. According to 10 CFR 50.36(c)(5), "Administrative controls," administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The Millstone 3 SG administrative controls are provided in TSs 6.8.4.g and 6.9.1.7.

For Millstone 3, SG tube integrity is maintained by meeting the performance criteria specified in TS 6.8.4.g.b for structural and leakage integrity, consistent with the plant design and licensing basis. TS 6.8.4.g.a requires that a condition monitoring assessment be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met. TS 6.8.4.g.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 criteria, as specified in TS Section 6.8.4.g.c, are that tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness shall be plugged, unless the tubes are permitted to remain in service through the application of the alternate tube plugging criteria provided in TS 6.8.4.g.c.1.

Millstone 3 TS 3.4.6.2.c includes a limit on operational 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 SG surveillance required by the TSs, this operational leakage limit provides added assurance of timely plant shutdown before SG tube integrity is impaired, consistent with the design and licensing bases.

As part of its plant's licensing basis, the applicant for a PWR license is 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 offsite 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," 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 3 would be changed because of the proposed amendment and, thus, no radiological consequences of any accident analyses are proposed to be changed. The proposed changes would maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes at Millstone 3.

3.0 TECHNICAL EVALUATION

3.1 Background

3.1.1 Steam Generator Design

Millstone 3 has four Westinghouse Model F recirculating SGs. Each SG has 5,626 thermally-treated Alloy 600 (Alloy 600TT) tubes with a nominal outside diameter of 0.688 inches and a nominal wall thickness of 0.040 inches. The tubes are arranged in a square pitch pattern and are hydraulically expanded at each end for the full depth of the tubesheet. The tubes are supported by stainless steel tube support plates (TSPs) with quatrefoil-shaped holes and three sets of anti-vibration bars (AVBs) on the U-bend section of the tubing. There is a flow distribution baffle plate (FDB) between the tubesheet and the first TSP. The U-bend region of the tubes installed in rows 1-10 was thermally treated after bending to reduce stress.

3.1.2 Operating Experience

The last two SG inspections at Millstone 3 were in spring 2019 (end of cycle 19 (EOC19)) and fall 2017 (EOC18). More information regarding the SG inspections is available in the spring 2019 and fall 2017 SG tube inspection reports (Reference 3 and Reference 4, respectively). The most recent eddy current inspections of SGs A and C were in fall 2017.

The following existing degradation mechanisms have been detected in the Millstone 3 SGs: Anti-vibration bar (AVB) wear, TSP and FDB wear, foreign object (FO) wear, wear from a legacy sludge lancing process that is no longer used, and primary water stress corrosion cracking (PWSCC) at the tube ends. On December 6, 2012, the NRC approved permanent alternate repair criteria for the stress corrosion cracking (SCC) found near the tube ends of the Millstone 3 SGs. The amendment revised TS 6.8.4.g to exclude portions of the SG tubes more than 15.2 inches below the top of the tubesheet (TTS) from the normal inspection and plugging criteria (Reference 5). The excluded portion of the tubes is called the H* (“H-star”) depth, and the amendment is called an H* amendment.

The four Millstone 3 SGs were placed in service in 1986 and have a total of 189 plugged tubes (51 in SG A, 25 in SG B, 22 in SG C, and 91 in SG D). Table 1 shows the causes of tube plugging for all SGs. As a result of the fall 2017 inspections of SGs A and C, one tube in SG A was plugged due to AVB wear and one tube in SG A was plugged due to FO wear. No tubes were plugged in SGs B or D as a result of the spring 2019 inspections.

Table 1: Millstone 3 SG Causes of Tube Plugging

Cause	Number of Tubes Plugged
AVB Wear	77
TSP and FDB Wear	9
FO Wear	59
Tube-end SCC	23
Preservice	10
Other*	11
TOTAL	189

* Expansion transition position, tube restriction, or eddy current test inspection issue

In addition to the active degradation mechanisms, the degradation assessment for Millstone 3 includes the following potential degradation mechanisms for Alloy 600TT SG tube material: axial outside diameter stress corrosion cracking (ODSCC) at TSPs, axial ODSCC at dents/dings, and circumferential ODSCC at the TTS. The SG inspection strategy for Millstone 3 includes inspections for these potential degradation mechanisms with specialized eddy current probes. For its operational assessment, the licensee considered axial ODSCC at TSPs bounding for axial SCC mechanisms and circumferential ODSCC at the TTS bounding for circumferential mechanisms. The choice of these bounding mechanisms was based on the effect of the cracks on structural and leakage integrity and on the probability of detecting the cracks with the eddy current examination techniques used.

The Alloy 600TT fleet has some tubes with potentially higher residual stress that are more susceptible to SCC. During refueling outage 15, Millstone 3 screened the bobbin coil eddy current data for Rows 11 through 59 with two methods to identify potentially high-stress tubes. The screening identified a total of 159 tubes in the four SGs with potentially high residual stress (67 in SG A, 30 in SG B, 39 in SG C, and 23 in SG D). These tubes were classified “Tier 1” because they contained the eddy current offset in both straight legs. An additional 1,243 tubes were identified as “Tier 2” tubes based on having an eddy current offset in only one of the straight legs (Reference 6).

The inspection strategy at Millstone 3 for the Tier 1 high-stress tubes includes full-length bobbin coil examinations and array probe examinations. In SGs A and C, full-length array probe examinations of the Tier 1 tubes were performed on the hot leg in in 2017. In SGs B and D, full-length array probe examinations of the Tier 1 tubes were performed on both the hot leg and cold leg in 2019.

Secondary-side activities were conducted in all four Millstone 3 SGs in the EOC18 (2017) refueling outage and included sludge lancing, post-lancing visual examination of the TTS annulus and no-tube lane, foreign object search and retrieval (FOSAR) at the TTS, visual examination of accessible locations having eddy current possible loose part (PLP) indications, and removal of retrievable FOs associated with PLP indications. In addition, a visual examination of key steam drum components, the upper tube bundle and AVB supports, feedring internal surface, and upper TSP was performed in SG D.

Secondary-side activities were also conducted in all four Millstone 3 SGs in the EOC19 (2019) refueling outage and included sludge lancing, post-lancing visual examination of the TTS annulus and no-tube lane, FOSAR at the TTS, visual examination of accessible locations having eddy current PLP indications, and removal of retrievable FOs associated with PLP indications. In addition, a visual examination of key steam drum components, the upper tube bundle and AVB supports, feedring internal surface, and upper TSP was performed in SG C. There have been no known foreign material introductions into either the primary or secondary systems since the 2019 refueling.

The licensee stated in the proposed license amendment request (LAR) that deposit loading has been aggressively managed, including performing chemical cleanings (deposit minimization treatment) in 2014 and 2016, and no adverse trends in deposit loading currently exist in any Millstone 3 SG. Also, there have been no chemistry excursions in the operating interval since 2019 at Millstone 3. In addition, based on the Nitrogen-16 main steam radiation monitor and main condenser air ejector, as supplemented by periodic leakage calculations (72 hours under normal operation), the licensee reported that no primary-to-secondary leakage has been detected in any Millstone 3 SG.

3.2. Proposed Technical Specification Changes

3.2.1 Current Technical Specification Requirements

The SG program in TS 6.8.4.g provides the SG tube inspection requirements for Millstone 3. TS 6.8.4.g.d.2 requires, in part, that, "After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections)." TS 6.8.4.g.d.2 defines the SG tube inspection requirements for the first, second, third, and subsequent inspection periods following SG installation. Specifically, TS 6.8.4.g.d.2.c states, "During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods." Millstone 3 is currently in the fourth inspection period.

3.2.2 Description of Proposed Technical Specification Changes

The LAR, as supplemented, proposes to add to TS 6.8.4.g.d.2 a footnote to indicate that the SG inspections scheduled for SGs A and C in the fall 2020 refueling outage may be deferred on a one-time basis to the spring 2022 refueling outage. Specifically, TS 6.8.4.g.d.2 would be changed by adding an asterisk at the end of its first sentence and a corresponding footnote stating: "* As approved by License Amendment No. 277, inspection of the Millstone Unit 3 SGs A and C may be deferred, on a one-time basis, from fall 2020 (Refueling Outage 20) to spring 2022 (Refueling Outage 21)."

3.3 Staff Evaluation of Proposed Technical Specification Changes

The NRC staff evaluation of the proposed TS changes for a one-time deferral of SG inspections was performed within the context of the COVID-19 public health emergency and the potential impacts of COVID-19 to plant personnel safety. Therefore, this safety evaluation should not be considered precedent setting for future routine plant amendments or generic industry licensing actions related to SG inspection intervals.

The NRC staff evaluation of the proposed one-time deferral focused on its potential to affect SG tube integrity, since maintaining SG tube integrity ensures that the plant will meet its SG program-related TSs, thereby protecting the public health and safety. In particular, the NRC staff evaluation assessed whether the LAR demonstrates that the structural integrity performance criterion (SIPC) and accident-induced leakage performance criterion (AILPC) will be met for Millstone 3 Cycle 21, which ends in spring 2022. These criteria are defined in TS 6.8.4.g.b. The licensee has stated its intention to request a measurement uncertainty recapture power uprate (Reference 7), but this would not be implemented before Cycle 22 and, therefore, does not affect the request for a one-time SG inspection deferral.

The Millstone 3 operating experience has shown tube degradation from wear at various support structures, wear from FOs, and wear from sludge lancing equipment no longer used. These mechanisms were evaluated as existing mechanisms in the operational assessment (OA). The OA includes axial ODSCC at TSPs, axial ODSCC in dents and dings, and circumferential ODSCC at the TTS, as potential mechanisms. These mechanisms have been observed at other units with Alloy 600TT tubes. The OA was performed using a deterministic analysis for the existing mechanisms and a probabilistic analysis for the potential mechanisms.

The deterministic OA method uses the worst-case single-tube analysis from the Electric Power Research Institute (EPRI) Integrity Assessment Guidelines (Reference 8) to provide a

conservative estimate of the projected EOC condition, considering all uncertainties at 0.95 probability and 50 percent confidence. The applicable uncertainties are for burst relation, material strength, and nondestructive examination (NDE) flaw sizing. The single tube methods are referred to as "worst-case degraded tube" methods, as the most severely flawed tube is selected for evaluation. The worst-case degraded tube OA methods involve determining the most limiting flaw at the beginning of cycle (BOC) and applying conservative flaw growth over the intended inspection interval to arrive at the EOC flaw condition to determine if the SIPC and AILPC will be met at EOC21. In contrast, the probabilistic analysis approach, also referred to as "full bundle analysis," uses probabilistic models with distributions of flaw size and growth to determine the probability of burst and leakage. The projected EOC results are evaluated against the SIPC and AILPC acceptance criteria.

3.3.1 Evaluation of Existing Tube Degradation Mechanisms

Wear at Anti-Vibration Bars

Wear at AVBs has occurred in all four of the Millstone 3 SGs, and a total of 77 tubes have been plugged due to AVB wear indications. In the most recent inspections of SGs A and C (EOC18) and SGs B and D (EOC19), there were indications of wear in each SG, and one tube was plugged as a result (in SG A). In SGs A and C, the deepest wear indications returned to service were 35 and 31 percent through-wall (TW), respectively, as sized with a bobbin probe using an EPRI-qualified examination technique.

The licensee's OA for AVB wear was performed using the worst-case single tube method, which is described in Section 3.3 of this safety evaluation. Based on NDE sizing uncertainty, the licensee adjusted the largest (35 percent TW) indication returned to service in EOC18 to a value of 42.9 percent TW for use as the BOC19 depth for the assessment. To project the maximum TW depth at EOC21, the licensee applied a growth rate of 2.2 percent TW/effective full power year (EFPY). In response to an NRC staff request for additional information (RAI), the licensee clarified in its September 4, 2020, letter that this was the highest rate observed in any of the four SGs over the past eight cycles and was observed in the inspection of SGs A and C at EOC18. The other observed rates over that period varied from 0.7 to 1.8 percent TW per EFPY. The calculated growth resulted in a maximum projected wear depth of 52.1 percent TW at EOC21. The length of the flaw was assumed to be 2.5 inches, which is conservative with respect to the width of the AVBs. Since the maximum projected EOC depth is less than the structural limit of 61.2 percent TW for Millstone 3, the licensee concluded that the SIPC and AILPC would be met for AVB wear until the proposed next inspection of SGs A and C at EOC21.

Wear at Support Plates (Tube Support Plates and the Flow Distribution Baffle)

In total, nine tubes have been plugged in the Millstone 3 SGs due to wear at support plates. The two new indications detected during the most recent inspection of SGs A and C at EOC18 in 2017 were in SG A at the first and fifth TSPs on the hot leg. The measured wear depths for the two new indications were 11 and 12 percent TW, while the deepest indication returned to service at that time had a measured depth of 27 percent TW. Since the 95th percentile probability of detection (POD) for this mechanism is about 15 percent TW, 27 percent was used as the upper bound BOC flaw depth rather than the depth of an undetected flaw (i.e., 15 percent based on the POD). The licensee applied measurement uncertainty and a growth rate of 5.5 percent TW per EFPY as a conservative value from the previous condition monitoring/operational assessment report, and the resulting upper bound depth at EOC21 was 64.3 percent TW. Since this is less than the calculated structural limit of 68.9 percent TW, the

licensee concluded the SIPC and AILPC would be met for TSP/FDB wear until the proposed next inspection of SGs A and C at EOC21.

NRC Staff Evaluation for Wear at Anti-Vibration Bars, Tube Support Plates, and the Flow Distribution Baffle

The NRC staff finds the licensee's evaluation of tube wear at AVBs, TSPs, and the FDB to be acceptable. Wear at these locations in the Millstone 3 SGs has been effectively managed for many cycles without challenging SG tube integrity. 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 EOC21. The NRC staff finds the determination of BOC flaw depth and growth rates acceptable because the determination is based on industry guidelines and conservative "worst case tube" deterministic assumptions, and is in line with rates determined for other plants. For example, the licensee applied the highest 95th percentile growth rates for AVB wear from the past eight operating cycles to the largest wear flaw in service adjusted to a greater percent TW to account for NDE uncertainty. Projections of EOC21 TW depths meet the SIPC with margin. 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 the 3 x nominal full power steady state primary to secondary pressure differential (3xNOPD). Based on the above, the NRC staff concludes that both SIPC and AILPC are satisfied.

Foreign Object Wear

In addition to wear at support structures, Millstone 3 has also experienced SG tube wear from FOs. In total, 59 tubes have been plugged due to FO wear in the four SGs, most recently in EOC18 when one tube was plugged in SG A. Secondary-side inspections were performed most recently in SGs A and C in EOC19, including sludge lancing, post-lancing FOSAR of the TTS annulus and no-tube lane, and visual investigation of accessible locations with eddy current signals potentially related to FOs. The largest of the flaws returned to service at the most recent eddy current inspection of SGs A and C was 37 percent TW, and it met the SIPC at that time. With one exception, no FOs remain near any of the worn areas, including the largest flaw returned to service; thus, there is no mechanism that could cause further degradation. One FO is known to be wedged between a tube in SG C and the blowdown pipe on the cold leg. The degradation was detected in EOC8 and the tube was plugged at that time. No wear has been detected on that tube or surrounding tubes since EOC8. The licensee concludes that FO wear, although difficult to predict, will not challenge tube integrity through EOC21, given the ability to detect FOs and FO wear, lack of additional degradation at FO wear locations, the low mass of the objects found, the success of FOSAR activities in removing objects, and improved foreign material exclusion.

The NRC staff finds the licensee's analysis of FO wear reasonable based on its use of the information known about past FO wear in the Millstone 3 SGs and the existing FOs. The NRC staff notes that tubes with FO wear no longer have an object present that would cause additional degradation. The one exception where an FO is wedged between a plugged tube and the blowdown pipe on the cold leg has not experienced any degradation in the surrounding tubes since EOC8, and these tubes were inspected with an array probe at EOC18. The NRC staff also acknowledges that predicting future FO and loose parts generation is not possible, since past fleetwide operating experience has shown that new loose parts generation, transport to the SG tube bundle, and interactions with the tubes cannot be reliably predicted. However, plants can reduce the probability of 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 3, have demonstrated the ability to conservatively manage loose parts once they are detected by eddy current examinations or by secondary-side FOSAR inspections. If unanticipated aggressive tube wear from new loose parts should occur in a Millstone 3 SG, operating experience has shown that a primary-to-secondary leak would probably 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 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated November 1, 2018 (effective January 1, 2019) (Reference 9), to confirm the licensee's conservative decisionmaking.

3.3.2 Evaluation of Potential Tube Degradation Mechanisms

In addition to existing tube degradation mechanisms, the licensee considered the following potential degradation mechanisms (none of which has been identified in the Millstone 3 SGs):

- axial ODSCC at TSPs
- axial ODSCC at dents/dings
- circumferential ODSCC at the hot-leg TTS

The licensee provided the rationale for assessing these SCC mechanisms. Axial ODSCC at TSPs was considered bounding for axial SCC mechanisms because axial cracks at these locations tend to be longer than at other locations and, therefore, can have a greater effect on SG tube integrity. In addition, detection of axial ODSCC at TSPs in tubes not identified as high-stress tubes depends on bobbin probes, which have a lower POD for this mechanism than the specialized probes. Circumferential ODSCC at the TTS was considered bounding for circumferential mechanisms because of the lower POD compared to circumferential PWSCC; therefore, the potentially larger undetected flaws have a greater effect on SG tube integrity. Axial ODSCC at dents and dings was evaluated based on recent operating experience at another plant with Alloy 600TT SG tubes.

As noted in Section 3.1.2 of this safety evaluation, the NRC staff previously approved an H* amendment for Millstone 3 (Reference 5), which concluded that potential tube degradation beyond the H* depth in the tubesheet does not affect SG tube integrity. Therefore, the licensee's evaluation of potential mechanisms and this safety evaluation do not consider potential tube cracking between the Millstone 3 H* distance and the tube end.

Both PWSCC and ODSCC have occurred at various locations at different plants in the Alloy 600TT fleet. In general, plants operating at higher temperatures are more prone to SCC compared to plants operating at lower temperatures. Similarly, in general, hotter portions of the tubing are more susceptible to SCC than colder portions of the tubing. SCC in SG tubing is also accelerated by higher residual stress in the tubing. The Alloy 600TT fleet is known to have some tubes with higher residual stress that are more susceptible to cracking. Millstone 3 has identified 67 such tubes in SG A and 39 such tubes in SG C. In EOC18, along with full-length bobbin probe inspection of all tubes, the high-stress tubes in SGs A and C were examined with an array probe for the entire hot-leg straight length.

Although Millstone 3 has not detected these three potential forms of SCC, periodic eddy current examinations with specialized probes are performed to detect such cracking. These specialized probes are used for examination of 100 percent of the tubes in the hot-leg tubesheet region and

can detect circumferential and axial ODSCC and PWSCC. The specialized probes are also used to examine the entire hot-leg straight section of the known high-stress tubes and can detect axial ODSCC. (The full-length bobbin probe examinations can detect axial ODSCC, but with a lower POD than the specialized probes.) Specialized probes are also used in voltage-based sampling of dents and dings in the hot leg and cold leg for axial ODSCC. The licensee considered the examination methods for each mechanism in the assessments described below.

Axial Outside Diameter Stress Corrosion Cracking at Tube Support Plates

The licensee assessed this mechanism with a full-bundle probabilistic analysis following the methodology in Reference 8. The simulation generates Monte Carlo projections of detected and undetected flaws for multiple operating cycles, considering inspection POD, new flaw initiation, and flaw growth, to calculate burst and leakage probabilities. This analysis addresses potentially unknown high-stress tubes by using a POD from bobbin probe examination, which is used to inspect the TSP intersections of all tubes rather than using the specialized (array) probe POD, which has been used to inspect TSP intersections only in the known high-stress tubes. A site-specific POD curve was developed according to industry guidelines in Reference 10 using eddy current noise measurements from the TSP area at the Surry Power Station (Surry), Unit No. 2, SGs, which the licensee considered more limiting than the noise from Millstone 3.

While the NRC staff did not review Reference 10, the resulting POD and growth rate curves developed by the licensee appear to be reasonable and conservative with a 95th percentile value of approximately 73 percent TW. The Weibull initiation function was benchmarked to assume detection of axial ODSCC in EOC18, despite no actual detections in the Millstone 3 SGs. Initial crack lengths were sampled from the distribution of the indicated lengths of cracks detected at plants with ODSCC at TSPs. Since a site-specific crack growth rate does not exist, the simulation used the EPRI default typical crack length and depth growth rate distributions from Reference 8, adjusted for the Millstone 3 hot-leg temperature. Using these inputs, the resulting distributions of EOC21 worst-case degraded tube burst pressures and accident-induced leakage met the SIPC and AILPC with margin.

Axial Outside Diameter Stress Corrosion Cracking at Dents/Dings

To assess this mechanism, the licensee used the same fully probabilistic modeling methodology that it had used for axial ODSCC at TSPs. However, because half of the dents and dings are examined during each inspection, the licensee treated the tube bundle as two sub-populations in the assessment. The probability of burst and leakage assessment was individually computed for each sub-population and later numerically combined to give the probabilities for the full bundle. For SGs A and C, one sub-population was inspected most recently at EOC16 and the other at EOC18, both using specialized rotating probes (+Point™) with an EPRI-qualified examination technique. A site-specific POD was developed according to industry guidelines in Reference 10 using eddy current noise measurements from the Millstone 3 SGs. The resulting POD curve developed by the licensee appeared to be reasonable and conservative with a 95th percentile value of approximately 83 percent TW. The Weibull initiation function was benchmarked to assume detection of axial ODSCC in the sub-population inspected at EOC18, and the same initiation function was applied to both sub-populations.

For the crack length distribution, the licensee used the Millstone 3 measured length distribution of dents and dings. No length growth was applied based on the assumption that the length would be limited by the dent/ding residual stress and, therefore, to the length of the dent/ding. A depth growth rate distribution was applied using temperature-adjusted rates determined from a

plant where this mechanism was detected. Using these inputs, the resulting distributions of EOC21 worst-case degraded tube burst pressures and accident-induced leakage for each sub-population and for the full bundle met the SIPC and AILPC with margin.

Circumferential Outside Diameter Stress Corrosion Cracking at the Top of the Tubesheet

The licensee assessed this potential mechanism with a fully probabilistic modeling methodology used for circumferential cracking. The methodology is similar to that used for the axial ODSCC methods. Inspection for this mechanism at Millstone 3 is performed with array probes in 100 percent of the tubes on the hot leg and a sample of tubes on the cold leg using an EPRI-qualified examination technique. The most recent inspection for this mechanism in SGs A and C was at EOC18. A site-specific POD for the array probe technique was developed according to industry guidelines in Reference 10 using eddy current noise measurements from the Millstone 3 SGs. The Weibull function for crack initiation was benchmarked to assume at least one detection of this mechanism at EOC18.

The model assumed an initial crack length distribution based on reported eddy current measurements for this mechanism in Alloy 600TT tubes at other plants. The assessment for Millstone 3 treated the crack length distribution as the initial length of the cracks. Crack length and depth growth were applied using the temperature-adjusted typical growth rate distributions in Reference 8 based on operating experience with a more susceptible tube material. Using these inputs, the resulting distributions of EOC21 worst-case degraded tube burst pressures and accident-induced leakage met the SIPC and AILPC with margin.

NRC Staff Evaluation for Potential Mechanisms

The NRC staff reviewed the licensee's probabilistic evaluations of potential cracking mechanisms. These degradation mechanisms have not been detected in the Millstone 3 SGs but have been detected in other SGs with Alloy 600 TT tube material. The NRC staff finds the licensee's choice of potential SCC mechanisms acceptable for the following reasons. Axial cracks at TSPs can be bounding due to potential length and since the bobbin probe is relied on for detection in non-high-stress tubes. Axial ODSCC at TSPs and circumferential ODSCC at the TTS are the two mechanisms responsible for the most cracks in the Alloy 600 TT fleet. Analysis of cracking at dents/dings is appropriate in response to recent operating experience at another Alloy 600 TT plant.

For axial cracking at TSPs, the licensee assumed a bobbin probe POD even though all known high-stress tubes are inspected with array probes at hot-leg TSP intersections. In response to an NRC staff RAI, the licensee clarified that the POD curve was different than for Surry, Unit No. 2, SG B because it relied on the most limiting Surry, Unit No. 2, noise, which is conservative for Millstone 3. The licensee's model for this mechanism was also benchmarked to cause axial ODSCC cracking detection at TSPs at EOC18 when none was detected. For cracking at dents/dings, the licensee appropriately divided the tubes into two sub-populations based on different inspection dates with the +Point™ probe. In a similar manner to TSPs, the licensee's model was benchmarked to cause ODSCC detection at a dent/ding at EOC18 within the second half of the population when no cracking was detected. Likewise, the ODSCC model at the TTS was benchmarked to cause crack detection at EOC18 when no cracks were detected.

The NRC staff finds the probabilistic evaluation assumptions to be conservative for Millstone 3. Although no cracks have been detected, the analysis for each potential cracking mechanism was benchmarked to assume detection of cracks in the last inspection of SGs A and C,

bounding crack lengths were assumed, and an appropriate growth rate was applied. The calculated probability of burst for each potential mechanism considered satisfies the SIPC margin requirements until EOC21. The AILPC is also satisfied until EOC21 because the analyses predicted no leakage.

Based on the evaluation discussed in Sections 3.3.1 and 3.3.2 of this safety evaluation, the NRC staff concludes that there is reasonable assurance that both the tube structural integrity and leakage integrity performance criteria will be met for all Millstone 3 SGs A and C tubes for existing degradation mechanisms (wear) and for potential degradation mechanisms (cracking) until EOC21.

3.4 Cycle 21 Mitigation Strategy

Millstone 3 TS 4.4.6.2.1.e requires "Verification that primary to secondary LEAKAGE is ≤ 150 gallons per day through any one Steam Generator…." The current primary-to-secondary leakage administrative limit at Millstone 3 is 75 gallons per day (gpd) in any one SG for 1 hour to initiate plant shutdown. In Section 3.2, "Mitigation Strategy," of Attachment 1 to the LAR, the licensee describes how leakage is measured and the primary-to-secondary leakage values that initiate additional monitoring. Section 3.2 also includes actions taken in response to the loose parts monitoring system. The NRC staff finds the mitigation strategy acceptable since the primary-to-secondary leakage limits imposed to initiate plant shutdown are less than the TS shutdown criterion of 150 gpd.

3.5 Technical Evaluation Conclusion

Based on the information submitted, the NRC staff finds that the licensee has demonstrated that there is reasonable assurance that the structural and leakage integrity of the Millstone 3 SGs A and C tubes will be maintained until the proposed next SG tube inspections during Refueling Outage 21 in spring 2022. Therefore, the NRC staff concludes that the licensee may change TS 6.8.4.g.d.2 as proposed because, as changed, the TS continues to provide administrative controls necessary to assure operation of the facility in a safe manner as required by 10 CFR 50.36(c)(5).

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination under the procedures in 10 CFR 50.91 that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

An operational assessment (OA) has been performed that concludes MPS3 [Millstone Power Station, Unit No. 3] SGs A and C will continue to meet structural and leakage integrity performance criteria with margin throughout the operating period preceding the next inspection in spring 2022. In addition, the proposed change does not implement physical changes to any plant structure, system or component; hence, no new failure modes are introduced. Therefore, the probability of an accident previously evaluated is not significantly increased. The proposed change does not involve a significant increase in the consequences of any previously evaluated accident. The proposed change does not involve an increase to the Technical Specification allowable iodine limit for primary coolant activity nor does it impact any of the underlying assumptions of the steam generator tube rupture (SGTR) accident analysis. As such, the consequences of a SGTR event are not affected by the change.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change does not alter the design function or operation of the MPS3 SGs A and C or the ability of the SGs to perform their design function. The SG tubes continue to meet the SG Program performance criteria. No plant physical changes are being implemented that would result in plant operation in a configuration outside the plant safety analyses or design basis. The proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Furthermore, MPS3 SGs A and C will continue to meet its specific structural and leakage integrity performance criteria throughout the operating period preceding the next inspection in spring 2022. Finally, no new effects on existing equipment are created, nor are any new malfunctions introduced.

Therefore, based on the above evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Extending the MPS3 inspection schedule for SGs A and C does not involve changes to any limit on accident consequences specified in the MPS3 licensing bases or applicable regulations, does not modify how accidents are mitigated, and does not involve a change in a methodology.

> A forward-focused OA of MPS3 SGs A and C was performed that demonstrates there is reasonable assurance that the structural integrity and accident-induced leakage performance criteria will remain satisfied in SGs A and C throughout the period preceding the spring 2022 refueling outage inspection for a total operating duration of three cycles between primary side inspections. The OA also identified projected margin to the structural integrity and accident-induced leakage performance criteria prior to the spring 2022 refueling outage for each evaluated degradation mechanism.
>
> Therefore, operation of the facility in accordance with the proposed change will not involve a significant reduction in a margin of safety.

The NRC staff reviewed the licensee's no significant hazards consideration analysis. Based on this review and on the NRC staff's safety evaluation of the underlying LAR as discussed above, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff makes a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment on September 11, 2020. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 or changes surveillance requirements. 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 that the amendment involves no significant hazards consideration, and there has been no public comment on this finding (September 8, 2020; 85 FR 55506). 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.

7.0 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.

8.0 REFERENCES

1. Letter from M. D. Sartain, Dominion Energy Nuclear Connecticut, Inc., to U.S. NRC, "Millstone Power Station Unit 3 Proposed License Amendment Request for a One-Time Extension of the Millstone Unit 3 Steam Generator Inspections," August 11, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20224A457).

2. Letter from G. T. Bischof, Dominion Energy Nuclear Connecticut, Inc., to U.S. NRC, "Millstone Power Station Unit 3 Response to Request for Additional Information Regarding License Amendment Request for a One-Time Deferral of the Millstone Unit 3 Steam Generator Inspections," September 4, 2020 (ADAMS Accession No. ML20252A191).

3. Letter from J. R. Daugherty, Dominion Energy Nuclear Connecticut, Inc., to U.S. NRC, "Millstone Power Station Unit 3 End of Cycle 19 Steam Generator Tube Inspection Report," September 19, 2019 (ADAMS Accession No. ML19275D254).

4. Letter from J. R. Daugherty, Dominion Energy Nuclear Connecticut, Inc., to U.S. NRC, "Millstone Power Station Unit 3 End of Cycle 18 Steam Generator Tube Inspection Report," April 19, 2018 (ADAMS Accession No. ML18114A105).

5. Letter from J. Kim, U.S. NRC, to D. A. Heacock, Dominion Nuclear, "Millstone Power Station Unit No. 3 – Issuance of Amendment Re: Revise Technical Specification 6.8.4.g, 'Steam Generator (SG) Program,' and Technical Specifications 6.9.1.7, 'Steam Generator Tube Inspection Report,' for a Permanent Alternate Repair Criteria (TAC No. ME8455)," December 6, 2012 (ADAMS Accession No. ML12299A498).

6. NUREG-2188, "U.S. Operating Experience with Thermally Treated Alloy 600 Steam Generator Tubes Through December 2013," page 3-82, February 2016 (ADAMS Accession No. ML16061A159).

7. U.S. NRC, "Summary of June 29, 2020, Pre-Submittal Teleconference with Dominion Energy Nuclear Connecticut, Inc. Re: Proposed License Amendment Request Related to a Measurement Uncertainty Recapture Power Uprate (EPID L-2020-LRM-0050)," August 5, 2020 (ADAMS Accession No. ML20210M423).

8. EPRI Report 3002007571, "Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines, Revision 4," June 2016 (non-publicly available).

9. U.S. NRC, Inspection Manual, Inspection Manual Chapter 0327, "Steam Generator Tube Primary-to-Secondary Leakage," November 1, 2018 (effective January 1, 2019) (ADAMS Accession No. ML18093B067).

10. EPRI Report 3002010334, "Model Assisted Probability of Detection Using R (MAPOD-R), Version 2.1," September 2017 (non-publicly available).

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Date: October 14, 2020

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 – ISSUANCE OF AMENDMENT NO. 277 TO REVISE TECHNICAL SPECIFICATION 6.8.4.G TO ALLOW A ONE-TIME DEFERRAL OF THE STEAM GENERATOR INSPECTIONS (EPID L-2020-LLA-0178) DATED OCTOBER 14, 2020

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