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

April 8, 2014

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT REGARDING CALCULATED CONTAINMENT INTERNAL PRESSURE (TAC NO. MF1731)

Dear Mr. Heacock:

The Commission has issued the enclosed Amendment No. 259 to Renewed Facility Operating License No. NPF-49 for the Millstone Power Station, Unit No. 3. This amendment is in response to your application dated April 25, 2013, as supplemented by letters dated September 19, and December 11, 2013.

The amendment revises the Technical Specification Section 6.8.4.f, "Containment Leakage Rate Testing Program," to change the peak calculated containment internal pressure for the design basis loss-of-coolant accident. The Amendment approves an increase in the value of the peak calculated containment internal pressure, P_a, from 41.4 pounds per square inch gage (psig) to 41.9 psig.

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

Sincerely,

Mohan C. Thadani

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures: 1. Amendment No. 259 to NPF-49
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

DOMINION NUCLEAR CONNECTICUT, INC., ET AL.

DOCKET NO. 50-423

MILLSTONE POWER STATION, UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 259
Renewed License No. NPF-49

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

    A. The application for amendment by Dominion Nuclear Connecticut, Inc. et. al., dated April 25, 2013, as supplemented by letters dated September 19, and December 11, 2013 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, as revised through Amendment No. 259, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into the license. DNC 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

Benjamin G. Beasley

Benjamin G. Beasley, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: April 8, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 259

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 marginal lines indicating the areas of change.

Remove	Insert
4	4

Replace the following page of the Appendix A Technical Specification, with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
6-17	6-17

(2) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No. 259 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DNC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) DNC shall not take any action that would cause Dominion Resources, Inc. (DRI) or its parent companies to void, cancel, or diminish DNC=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 Resources, 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 decommissiong trust agreement for MPS Unit No. 3 must provide that no disbursement 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 agreements must provide that the agreement can not be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

PROCEDURES AND PROGRAMS (Continued)

2) Pre-planned operating procedures and backup instrumentation to be used if one or more monitoring instruments become inoperable, and

3) Administrative procedures for returning inoperable instruments to OPERABLE status as soon as practicable.

f. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions*. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Performance Based Option of 10 CFR Part 50 Appendix J": The first Type A test performed after the January 6, 1998 Type A test shall be performed no later than January 6, 2013.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 41.9 psig.

The maximum allowable containment leakage rate L_a, at P_a, shall be 0.3 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.06\ L_a$ for all penetrations that are Secondary Containment bypass leakage paths, and $< 0.75\ L_a$ for Type A tests;

2) Air lock testing acceptance criteria are:

   a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   b. For each door, seal leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

   The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

   The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

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* An exemption to Appendix J, Option A, paragraph III.D.2(b)(ii), of 10 CFR Part 50, as approved by the NRC on December 6, 1985.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 259

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 3

DOCKET NO. 50-423

1.0 INTRODUCTION

By letter dated April 25, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13120A158), Dominion Nuclear Connecticut, Inc. (DNC or the licensee) proposed a change to Technical Specifications (TSs) for Millstone Power Station, Unit No. 3 (MPS3). The proposed change would revise the TS Section 6.8.4.f, "Containment Leakage Rate Testing Program" to increase the value of the peak calculated containment internal pressure for the design basis loss-of-coolant accident, P_a, from 41.4 pounds per square inch gage (psig) to 41.9 psig. According to the licensee, this increase is needed to address an increase in the calculated mass and energy (M&E) release during the blowdown phase of the design basis Loss-of-Coolant Accident (LOCA).

The licensee provided responses to two sets of requests for additional information (RAIs) from the U. S. Nuclear Regulatory Commission (NRC) staff related to this license amendment request (LAR). By a letter dated September 19, 2013, the licensee provided supplemental information in response to the NRC staff's first RAI (ADAMS Accession No. ML13275A240). By a letter dated December 11, 2013, the licensee provided supplemental information in response to the NRC staff's second RAI (ADAMS Accession No. ML13353A296).

The supplemental letter of September 19 and December 11, 2013 provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on June 25, 2013 (78 FR 38081).

2.0 REGULATORY EVALUATION

The regulations at Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 Appendix A, General Design Criteria (GDC) 16 and 50 address the capability of the containment to withstand the containment pressure resulting from a postulated design basis LOCA.

Criterion 16, Containment Design, requires that the reactor containment and associated systems shall be provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 19, Control Room, specifies that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in safe shutdown under accident conditions, including LOCAs, and that adequate radiation protection shall be provided.

Criterion 38, Containment Heat Removal, specifies that a system to remove heat from the reactor containment shall be provided that rapidly reduces, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintains them at acceptably low levels.

Criterion 50, Containment Design Basis, specifies that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartment can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a design basis LOCA.

The regulations at 10 CFR Part 50, Appendix J Option B define calculated peak containment internal pressure as the calculated peak containment internal pressure related to the design basis LOCA as specified in the TS and specify the requirements for containment leakage rate testing. The requirements of MPS3 TS 6.8.4.f, "Containment Leakage Rate Testing" provide more detailed containment leakage rate testing requirements.

As additional background, the NRC staff has previously issued license amendments to a number of reactor units to implement increased P_a values when revised containment analyses were performed for reasons such as correcting the calculation input errors in power uprates. For instance, the licensee cited Palisades Nuclear Plant as precedent for a similar license amendment approved by the NRC on January 19, 2012 (ADAMS Accession Nos. ML113220370 and ML120600415).

The regulations at 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," require, in part, that licensees establish programs to qualify electric equipment important to safety located in a harsh environment.

Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," describes a method acceptable to the NRC staff for complying with 10 CFR 50.49 with regard to qualification of electric equipment important to safety for service in nuclear power plants to ensure that the equipment can perform its safety function during and after a design basis accident.

3.0 TECHNICAL EVALUATION

Westinghouse provided the licensee with the mass and energy release data, which is used as an input to the LOCA containment response analysis of record (AOR), and is the basis for the current value of P_a. The licensee used these mass and energy release values to calculate P_a using the DOM-NAF-3-0.0-P-A GOTHIC (GOTHIC) containment analysis methodology previously approved by the NRC. This methodology is described in MPS3's Final Safety Analysis Report (FSAR) Chapter 6. The AOR and the revised analysis performed in support of this LAR utilize the same version of GOTHIC.

In its April 25, 2013, letter the licensee stated that it had identified four errors in the MPS3 FSAR Chapter 6 analyses for large break LOCA M&E releases. The M&E releases were calculated by Westinghouse and input to the MPS3 FSAR Chapter 6 containment response analyses that were performed by DNC. A total of six issues were identified in Westinghouse Nuclear Safety Advisory Letter (NSAL)-11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues," dated July 25, 2011, out of which three were determined to affect large break LOCA containment M&E responses. The fourth error was independent of NSAL-11-5 and specific to MPS3 (see Item 1 below).

The licensee stated that the four errors applicable to the MPS3 LOCA M&E analysis are:

1. Lower steam generator (SG) primary side pressure was used in the AOR (948 pounds per square inch absolute (psia)), rather than the correct value of 984 psia. This error under predicts the initial stored energy in the four MPS3 SGs. This error was specific to the MPS3 analysis of record and was discovered independent of the issues identified in NSAL-11-5.

2. Reactor vessel model in the AOR did not include the appropriate reactor vessel metal mass available, which affects the amount of reactor vessel stored energy initially available in the M&E model.

3. Reactor vessel model did not include the appropriate reactor vessel metal mass in the reactor vessel barrel/baffle downcomer region, this impacts the initial energy stored within the reactor vessel. MPS3 is an upflow plant.

4. The break flow was initialized at a non-conservative SG secondary pressure condition. This input value determines the initial SG secondary side temperature and pressure used in the large break LOCA M&E release calculations.

The licensee stated that the Westinghouse errors only affected large break LOCA M&E releases and that steam line break and small break LOCA M&E releases are unaffected. Westinghouse reanalyzed the large break LOCA M&E releases with the errors corrected and no other design input changes. The large break LOCA M&E analysis methods that were applied are consistent with those referenced in MPS3 FSAR Section 6.2.1.3 (see below).

- WCAP-8264-P-A, Revision 1, "Topical Report: Westinghouse Mass and Energy Release Data for Containment Design," August 1975.

- WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," May 1983 (Proprietary).

The NRC staff reviewed the licensee's response in a letter dated September 19, 2013, to the Staff's RAI. The current TS Section 3.6.1.4 value for containment initial pressure is a range of 10.6 psia - 14.0 psia. The staff concurs with the licensee's approach to use an initial containment pressure of 14.2 psia as it would result in a conservative value for peak containment pressure. Therefore, with respect to containment safety analyses, there is no need to change the limiting safety system setting contained in TS Section 3.6.1.4 "Containment Pressure."

Through the licensee letter dated September 19, 2013, the staff was able to get confirmation from the licensee that there were no additional generic errors that affect the MPS3 large break LOCA containment M&E release analysis, including the EPITOME computer modeling error discovered by Westinghouse. The proposed value of P_a, which includes the corrected mass and energy release data and the most adverse analytical initial containment pressure, is 41.9 psig. The revised calculated P_a remains below the containment design pressure of 45 psig, as referenced in FSAR Sections 6.2.1 and 3.8.1. Thus, the increase in peak calculated containment internal pressure does not affect systems and components in containment because these are designed for a containment design pressure limit of 45 psig. Therefore the proposed change in the P_a value from 41.4 psig to 41.9 psig is acceptable.

The maximum allowable containment leakage rate (L_a), at P_a, for MPS3 is 0.3 percent by weight of the containment air per 24 hours. During the first 2 minutes post-LOCA, 100 percent of the containment leak rate is assumed to bypass the secondary containment and release unfiltered at ground. After the Supplementary Leak Collection and Release System (SLCRS) drawdown is effective at 2 minutes, the bypass leakage rate is 0.06 of L_a (or 0.018 percent by weight per day) as defined in TS 6.8.4.f. The remaining containment leakage is filtered and releases through SLCRS. The LAR further states that the containment leak rate, L_a, is reduced from 0.3 to 0.15 percent by weight at 24 hours for offsite dose calculations, and at one hour for control room and technical support center dose calculations. The licensee stated that the long-term LOCA containment response AOR demonstrates that containment pressure meets the radiological analysis requirement in the FSAR for a 50 percent reduction in containment leakage after one hour.

As stated in the licensee's April 25, 2013, letter, the containment leakage rate "Type A" test is performed in accordance with the requirements of 10 CFR Part 50, Appendix J to demonstrate that leakage of systems and components penetrating the primary containment do not exceed the allowable leakage rates specified in MPS3 TS 6.8.4.f. Specifically, the Type A test verifies that the measured containment leakage rate at P_a does not exceed the maximum allowable leakage rate, L_a, which is used to calculate the dose consequences following a postulated LOCA.

The most recent Type A test at MPS3 was completed on November 7, 2011. The containment pressure during the test was measured at 42.5 psig, which exceeds the proposed peak calculated containment internal pressure of 41.9 psig. The containment leakage rate during the

test was calculated to correspond to 0.0531 weight percent per day, which is significantly less than the maximum containment leakage rate of 0.30 weight percent of the containment air per 24 hours as specified in TS 6.8.4.f and used in the offsite dose calculations.

The licensee also stated in the September 19, 2013, letter that Appendix J Type B and C test procedures do not require revision upon approval of this proposed LAR. ANSI 56.8-1994 Section 3.3.2 requires that Type B and C testing be performed at a pressure not less than P_a (except for airlock door seals, which may have a lower pressure specified) and not more than 1.1 times P_a when a higher differential pressure results in increased sealing. MPS3 site procedures for Type B and C testing require that testing be performed within a range of pressures that, with the revised P_a, will continue to be within the range of pressures required by ANSI 56.8-1994. Therefore, the Type B and C test procedures will not require revision upon approval of the license amendment request.

The LOCA offsite, control room, and technical support center dose calculations are based on the allowable leakage rate (i.e., 0.3 percent by weight of containment air) in the TSs. Based on the significant margin between the tested and allowed leakage rates exhibited by the most recent Type A test completed on November 7, 2011, the staff concludes that the proposed change to increase the peak calculated containment pressure, P_a, from 41.4 psig to 41.9 psig will have no adverse impact on the dose calculations and the plant's compliance with GDC-19.

The NRC staff concludes that the proposed change meets the requirements of 10 CFR Part 50 Appendix A, (1) GDC 16, because the licensee showed that the containment design conditions important to safety are not exceeded during a postulated Design-Basis Accident (DBA); (2) GDC 19, because the licensee showed that control room dose analysis is unaffected by the proposed change; (3) GDC 38, because the licensee showed that the containment heat removal system would reduce the containment pressure and temperature rapidly following a DBA and would maintain them at acceptable levels; and (4) GDC 50, because the licensee showed that the containment heat removal system is designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a design basis LOCA. Therefore, the proposed increase in the P_a, from 41.4 psig to 41.9 psig is acceptable.

The licensee stated that the calculated peak containment internal pressure, P_a, would increase from 41.4 psig to 41.9 psig. This increase in P_a is due to an increase in the calculated M&E released into containment during the blowdown phase of the design basis LOCA event. The licensee stated that it has reanalyzed MPS3's FSAR Chapter 6 containment analyses with corrected large break LOCA M&E data. The licensee also stated that the large break LOCA containment pressure analysis uses NRC-approved methods already described in the MPS3 FSAR. The licensee stated that the calculated peak containment internal pressure will remain below the design limit of 45 psig.

In the LAR, Section 4, the licensee provided a review of the environmental qualification (EQ) of equipment in the containment addressing the effect of the increase in M&E on EQ. The licensee determined that the increase in P_a to 41.9 psig does not adversely affect environmentally qualified equipment within the containment because this equipment is qualified

for the containment design pressure of 45 psig. The licensee also determined that the containment temperatures, using the corrected large break LOCA M&E releases, remain within the bounding containment temperature profile used to qualify the equipment and concluded that the post-accident operating time of the environmentally qualified equipment remains unaffected.

By letter dated December 11, 2013, the licensee responded to the NRC staff's request for additional information regarding the EQ reanalysis to show equipment qualification inside containment meets the requirements of 10 CFR 50.49. The licensee stated that the EQ bounding temperature and pressure profiles taken from the MPS3 Environmental Specification are used for the EQ of plant equipment. The licensee also stated that the most limiting LOCA for peak containment pressure and temperature is a double-ended guillotine break of the hot leg. The licensee provided Figures 1 and 2, which compared the EQ bounding temperature and pressure profiles and containment temperature and pressure profiles (EQ overlays) from the double-ended hot leg break analysis. The NRC staff reviewed the figures and concluded that the EQ bounding temperature and pressure profiles envelope the containment temperature and pressure profiles from the double-ended hot leg break analysis respectively.

The licensee further stated in the letter dated December 11, 2013, that the temperature and pressure margins in the MPS3 EQ program are applied in accordance with 10 CFR 50.49 and Regulatory Guide 1.89. Also, the licensee stated that these margins are in addition to the EQ bounding temperature and pressure profiles for equipment qualified according to 10 CFR 50.49. Since the EQ bounding temperature and pressure profiles exceed the expected peak containment temperature and pressure from the double-ended hot leg break analysis, and the margins are applied in addition to the EQ bounding temperature and pressure, the NRC staff concluded that there will be no adverse impact on the EQ of electric equipment for the revised temperature and pressure inside containment due to the proposed amendment request.

Additionally, the licensee stated that the proposed change in calculated peak containment internal pressure for the design basis LOCA does not result in any adverse impact on any area of the plant outside containment, and as a result, environmental conditions outside containment remain bounded by existing calculations. Based on this information, the NRC staff concludes that there will be no adverse impact on the qualifications of electrical equipment for both temperature and pressure outside containment.

Based on the technical evaluation provided above, the NRC staff concludes that the EQ bounding profiles for temperature and pressure are bounded by the EQ test profiles under harsh environmental conditions. The NRC staff concludes that the revised LOCA containment pressure and temperature profiles remain enveloped by the current MPS3 EQ profiles and therefore the electric equipment inside and outside containment remains environmentally qualified in accordance with 10 CFR 50.49.

Therefore, the NRC staff concludes that the licensee's application for the amendment to increase peak calculated containment internal pressure $P_{a,}$ due to reanalyzed design basis LOCA is acceptable, because it continues to satisfy GDC 16, GDC 19, GDC 38, and GDC 50, and the potential impact on the environmental qualifications of electrical equipment under the newly created environmental conditions of increased mass and energy release is not significant, because the equipment continues to remain qualified in accordance with the requirements of 10 CFR 50.49.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified 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 a facility component located within the restricted area as defined in 10 CFR Part 20. 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 such finding as published in the *Federal Register* on June 25, 2013 (78 FR 38081). 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.

Principal Contributors: R. Torres
S. Som

Date: April 8, 2014

April 8, 2014

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT REGARDING CALCULATED CONTAINMENT INTERNAL PRESSURE (TAC NO. MF1731)

Dear Mr. Heacock:

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

Sincerely,
/ra/
Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423
Enclosures: 1. Amendment No. 259 to NPF-49
2. Safety Evaluation
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ADAMS Accession NO.: ML14073A055 ***Via Memo ML14007A663** ****Via Memo ML13329A444**

OFFICE	NRR/LPLI-1/PM	NRR/LPLI-1/LA	OGC	NRR//EEEB/BC	NRR/SCVB/BC	NRR/LPLI-1/BC
NAME	MThadani	KGoldstein	JWachutka	JZimmerman*	RDenning**	BBeasley
DATE	03/25/2014	03/21/2014	03/31/2014	01/27/2014	02/10/2014	04/08/2014

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