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Dominion®

April 25, 2013

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No. 13-225
NSSL/MLC R0
Docket No. 50-423
License No. NPF-49

**DOMINION NUCLEAR CONNECTICUT, INC.**
**MILLSTONE POWER STATION UNIT 3**
**LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 6.8.4.F FOR PEAK CALCULATED CONTAINMENT INTERNAL PRESSURE**

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) requests an amendment to Operating License NPF-49 for Millstone Power Station Unit 3 (MPS3). The proposed license amendment would revise the peak calculated containment internal pressure for the design basis loss of coolant accident described in Technical Specification (TS) 6.8.4.f, "Containment Leakage Rate Testing Program." The peak calculated containment internal pressure, $P_a$, would increase from 41.4 psig to 41.9 psig.

Attachment 1 provides the description and assessment of the proposed change. As discussed in this attachment, the proposed amendment does not involve a significant hazards consideration pursuant to the provisions of 10 CFR 50.92. Attachment 2 contains the marked-up TS page to reflect the proposed change. The proposed change has been reviewed and approved by the Facility Safety Review Committee.

DNC requests approval of the proposed license amendment by April 25, 2014. Once approved, the license amendment will be implemented within 60 days.

In accordance with 10 CFR 50.91(b), a copy of this license amendment request is being provided to the State of Connecticut.

If you have any questions regarding this submittal, please contact Ms. Wanda Craft at (804) 273-4687.

Sincerely,

Eugene S. Grecheck
Vice President – Nuclear Engineering and Development

VICKI L. HULL
Notary Public
Commonwealth of Virginia
140542
My Commission Expires May 31, 2014

COMMONWEALTH OF VIRGINIA )
)
COUNTY OF HENRICO )

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Eugene S. Grecheck, who is Vice President – Nuclear Engineering and Development of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 25TH day of April, 2013.

My Commission Expires: 5-31-14.

Vicki L. Hull
Notary Public

A001
NRR

Commitments made in this letter: None

Attachments:

1. Evaluation of Proposed Change to Revise Technical Specification 6.8.4.f for Peak Calculated Containment Internal Pressure

2. Marked-up Technical Specifications Page

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# ATTACHMENT 1

## EVALUATION OF PROPOSED CHANGE TO REVISE TECHNICAL SPECIFICATION 6.8.4.F FOR PEAK CALCULATED CONTAINMENT INTERNAL PRESSURE

**DOMINION NUCLEAR CONNECTICUT, INC.**
**MILLSTONE POWER STATION UNIT 3**

# EVALUATION OF PROPOSED CHANGE TO REVISE TECHNICAL SPECIFICATION 6.8.4.F FOR PEAK CALCULATED CONTAINMENT INTERNAL PRESSURE

## 1.0 DESCRIPTION

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) requests an amendment to Operating License NPF-49 for Millstone Power Station Unit 3 (MPS3). The proposed license amendment would revise the peak calculated containment internal pressure for the design basis loss of coolant accident (LOCA) described in Technical Specification (TS) 6.8.4.f, "Containment Leakage Rate Testing Program." The peak calculated containment internal pressure, $P_a$, would increase from 41.4 pounds per square inch gauge (psig) to 41.9 psig.

The increase in $P_a$ for MPS3 is due to an increase in the calculated mass and energy (M&E) released into containment during the blowdown phase of the design basis LOCA event. DNC has reanalyzed MPS3's Final Safety Analysis Report (FSAR) Chapter 6 containment analyses with corrected large break LOCA M&E data and is requesting NRC review and approval to change the TS 6.8.4.f value for $P_a$ from 41.4 psig to 41.9 psig. The large break LOCA containment pressure analysis uses NRC-approved methods already described in the MPS3 FSAR, uses mass and energy inputs from an analysis that uses NRC-approved methods, meets the containment design pressure limit of 45 psig, and satisfies the Environmental Qualification and Containment Leakage Rate Testing Programs.

## 2.0 PROPOSED CHANGE

The proposed change to TS 6.8.4.f, "Containment Leakage Rate Testing Program," is shown below.

TS 6.8.4.f currently states:

> The peak calculated containment internal pressure for the design basis loss of coolant accident, $P_a$, is 41.4 psig.

The proposed change would revise TS 6.8.4.f, by replacing the $P_a$ value of 41.4 psig with a value of 41.9 psig.

The revised TS 6.8.4.f would read as follows:

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

## 3.0 BACKGROUND

Four errors have been identified in the MPS3 FSAR Chapter 6 analyses for large break LOCA M&E releases. The M&E releases are calculated by Westinghouse and input to the MPS3 FSAR Chapter 6 containment response analyses that are performed by DNC. Three of the errors were identified in Westinghouse Nuclear Safety Advisory Letter (NSAL)-11-5, "Westinghouse LOCA Mass and Energy Release Calculation Issues," dated July 25, 2011. The fourth error was independent of NSAL-11-5 and specific to MPS3 (see Item 1 below). DNC has reanalyzed the FSAR Chapter 6 containment response analyses with corrected large break LOCA M&E data and is requesting NRC review and approval to change the TS 6.8.4.f value for $P_a$ from 41.4 psig to 41.9 psig.

Specifically, the four errors applicable to the MPS3 LOCA M&E analysis are:

1. Steam generator (SG) pressure was incorrectly input as 948 pounds per square inch absolute (psia) rather than the correct value of 984 psia. This error under predicted the initial stored energy in the four SGs. This error was specific to the MPS3 analysis of record and was discovered independent of the issues identified in NSAL-11-5.

2. The reactor vessel modeling did not include the appropriate reactor vessel metal mass available from the component drawings. The discrepancy results in an inaccurate reactor vessel metal mass that affects the amount of reactor vessel stored energy initially available in the M&E model. This error was identified in NSAL-11-5.

3. The reactor vessel modeling did not include the appropriate reactor vessel metal mass in the reactor vessel barrel/baffle downcomer region. Differences were identified in the calculated metal mass and surface area input values between upflow and downflow barrel/baffle configurations, with more significant differences noted in plants that were converted to an upflow barrel/baffle configuration. Increases in the barrel/baffle metal mass impact the initial energy stored within the reactor vessel. MPS3 is an upflow plant. This error was identified in NSAL-11-5.

4. The large break LOCA M&E release analysis was initialized at a non-conservative (low) 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 pressure at the exit of the SG outlet nozzle was incorrectly used as the SG secondary side pressure, as opposed to the correct higher tube bundle pressure. The initial SG energy is under estimated; therefore, the correction results in an increase in the calculated large break LOCA M&E release. This error was identified in NSAL-11-5.

The Westinghouse errors only affected large break LOCA M&E releases. 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 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).

## 4.0 TECHNICAL ANALYSIS

Using the revised large break LOCA M&E data, DNC reanalyzed the FSAR Chapter 6 containment pressure and temperature response calculations using the NRC-approved GOTHIC containment analysis methodology documented in topical report DOM-NAF-3-0.0-P-A, Revision 0, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment," dated September 2006. This methodology is described in MPS3 FSAR Chapter 6.

The peak calculated containment internal pressure following a large break LOCA is obtained for the double-ended hot leg guillotine break. Table A compares the new analysis results to the analysis of record. In the new analysis, correction of the large break LOCA M&E errors produced an increase in containment peak pressure of 0.44 psig and a reduction of 0.1 seconds in the peak pressure time. Consistent with the analysis of record, the containment peak pressure occurs near the end of the initial RCS blowdown. The magnitude of the peak pressure is independent of the emergency core cooling and containment heat removal systems, because these systems actuate after the peak pressure occurs. The large break LOCA containment peak pressure is less than the containment design pressure of 45 psig; however, the rounded result of 41.9 psig is an increase compared to the value for $P_a$ of 41.4 psig currently reported in TS 6.8.4.f.

**Table A**
**LOCA Peak Calculated Containment Internal Pressure Results**

| | Analysis of Record | Revised Analysis |
|---|---|---|
| LOCA Containment Peak Pressure | 56.09 psia | 56.53 psia |
| Time of Peak Pressure | 21.2 seconds | 21.1 seconds |
| Peak Pressure for TS 6.8.4.f $P_a$* | 41.4 psig | 41.9 psig |

* Determined by rounding the peak calculated containment internal pressure up to the nearest 0.1 psig.

Containment Leakage Review

The total containment leakage, ($L_a$), for MPS3 consists of both filtered and bypass leakage. Per TS 6.8.4.f, the maximum allowable containment leakage rate $L_a$, at $P_a$, is 0.3 percent by weight of the containment air per 24 hours. Until the supplementary leak collection and release system (SLCRS) drawdown is effective at 2 minutes post-LOCA, 100% of the containment leak rate is assumed to bypass the secondary containment and release unfiltered at ground level directly from containment. After SLCRS drawdown at 2 minutes, the bypass leak rate, defined per TS 6.8.4.f, is 0.06 of $L_a$ or 0.018 percent by weight per day; the remaining containment leakage (0.3 - 0.018) is filtered and released through SLCRS. 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 (CR) and technical support center (TSC) dose calculations. This assumption of a reduction in the containment leak rate by 50 percent after one hour for the CR and TSC habitability analyses was used in calculations supporting Amendment No. 59 (ADAMS No. ML011790140), which eliminated the post-LOCA negative containment pressure requirement. This assumption was also referenced in the description of calculations provided as supplemental information supporting Amendment No. 211 (ADAMS No. ML023290568 and ML022470399), which changed the licensing basis for the post-accident operation of the SLCRS. The assumption of a 50 percent reduction in containment leakage after one hour is based on the fact that the MPS3 post-LOCA containment pressure is rapidly reduced compared to typical pressurized water reactors because the MPS3 containment was originally designed to be operated at sub-atmospheric pressure. The initial containment design pressure for MPS3 was for a range of 8.9 psia - 12 psia. In Amendment No. 59, the limiting condition for operation for TS 3.6.1.4 for containment initial pressure was changed from a range of 8.9 psia - 12 psia to the current range of 10.6 psia - 14.0 psia.

The long-term LOCA containment response analysis demonstrates that the containment pressure meets the FSAR requirement for the radiological analysis of a 50 percent reduction in containment leakage after one hour. Resolution of the errors in the Westinghouse-generated LOCA M&E release analysis does not modify the intrinsic characteristic of the MPS3 containment. The containment was originally designed as a negative pressure containment that allows a rapid pressure reduction following a design basis LOCA event as compared to typical pressurized water reactor containments.

The LOCA offsite radiological dose consequence analyses assume containment leakage rates based on percent by weight of the containment air. Therefore, the increase in the peak calculated containment internal pressure does not impact the offsite, CR and TSC radiological consequences of the LOCA accident analysis, as described in the MPS3 FSAR Section 15.6.5.4.

10 CFR 50 Appendix J Program Review

The containment leakage rate "Type A" test is performed in accordance with the requirements of 10 CFR 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 MPS3 Type A test was last completed on November 7, 2011. The containment pressure during the test was measured at 42.5 psig, which exceeds the peak calculated containment internal pressure of 41.9 psig that was calculated following resolution of the errors in the Westinghouse-generated large break LOCA M&E release analyses. The containment leakage rate during the test was calculated to correspond to 0.0531 weight percent per day, which is less than the maximum containment leakage rate of 0.30 weight percent of the containment air per 24 hours specified in TS 6.8.4.f and used for the offsite dose calculations.

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, as referenced in FSAR Sections 6.2.1 and 3.8.1.

Equipment Environmental Qualification Review

The change in $P_a$ does not affect environmentally qualified equipment within containment. This equipment is qualified for the containment design pressure of 45 psig. Therefore, an increase in peak calculated containment internal pressure to 41.9 psig does not affect the environmental qualification of equipment within containment.

The containment temperatures, using the corrected large break LOCA M&E releases, remain within the bounding containment temperature profile used to qualify equipment. Therefore, the post-accident operating time of the environmentally qualified equipment is unaffected.

## 5.0 REGULATORY SAFETY ANALYSIS

### 5.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

General Design Criterion 4, "Environmental and dynamic effects design bases," states that structures, systems and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including LOCAs.

General Design Criterion 16, "Containment design," states that 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.

General Design Criterion 19, "Control room," states 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 a safe condition under accident conditions, including LOCAs, and that adequate radiation protection shall be provided.

General Design Criterion 38, "Containment heat removal," states 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 maintain them at acceptable low levels.

These general design criteria continue to be met with the change in peak calculated containment internal pressure. The environmental qualification of equipment within containment is not affected by the change in peak calculated containment internal pressure following a LOCA. The change in peak calculated containment internal pressure will be reflected in future 10 CFR 50 Appendix J, Type A containment leakage rate testing, so containment integrity is not impacted by the change. The change in peak calculated containment internal pressure does not impact the maximum allowable containment leakage rate and therefore does not impact control room operator dose. The peak calculated containment internal pressure remains below the containment design pressure.

Based on the considerations above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will continue to be conducted in accordance with the MPS3 licensing basis, and

(3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, DNC has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements or criteria.

5.2 No Significant Hazards Consideration

DNC is proposing a license amendment to MPS3 TS 6.8.4.f, "Containment Leakage Rate Testing Program." The proposed amendment would increase the calculated peak containment internal pressure for the design basis loss of coolant accident, $P_a$, from 41.4 psig to 41.9 psig.

DNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No.

The proposed change to $P_a$ does not alter the assumed initiators to any analyzed event. The probability of an accident previously evaluated will not be significantly increased by this proposed change.

The change in $P_a$ will not affect radiological dose consequence analyses. MPS3 radiological dose consequence analyses assume a certain containment atmosphere leak rate based on the maximum allowable containment leakage rate, which is not affected by the change in peak calculated containment internal pressure. The Appendix J containment leakage rate testing program will continue to ensure that containment leakage remains within the leakage assumed in the offsite dose consequence analyses. The consequences of an accident previously evaluated will not be significantly increased by this proposed change.

Therefore, operation of the facility in accordance with the proposed change to $P_a$ will 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 provides a higher $P_a$ than currently described in TS 6.8.4.f. This change is a result of an increase in the M&E release input for the LOCA containment response analysis. The peak calculated containment pressure remains below the containment design pressure of 45 psig. This change does not involve any alteration in the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, operation of the facility in accordance with the proposed change to TS 6.8.4.f would not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The calculated peak containment internal pressure remains below the containment design pressure of 45 psig. Since the MPS3 radiological consequence analyses are based on the maximum allowable containment leakage rate, which is not being revised, the change in the calculated peak containment internal pressure does not represent a significant change in the margin of safety.

Therefore, operation of the facility in accordance with the proposed change to TS 6.8.4.f does not involve a significant reduction in the margin of safety.

## 6.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment would change a requirement with respect to installed facility components located within the restricted area of the plant, as defined in 10 CFR 20. However, as detailed below, the proposed amendment does not involve 1) a significant hazards consideration, 2) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or 3) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51 .22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

1. The amendment involves no significant hazards consideration.

   As demonstrated in Section 5.2 above, "No Significant Hazards Consideration," the proposed change does not involve any significant hazards consideration.

2. There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

   The proposed change will revise TS 6.8.4.f, "Containment Leakage Rate Testing Program." The proposed change does not result in an increase in power level, and does not increase the production nor alter the flow path or method of disposal of radioactive waste or byproducts; thus, there will be no change in the amounts of radiological effluents released offsite.

   Based on the above evaluation, the proposed change will not result in a significant change in the types or significant increase in the amounts of any effluent released offsite.

3. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will revise TS 6.8.4.f, "Containment Leakage Rate Testing Program." The proposed change will not result in any changes to the configuration of the facility. The proposed change will not cause a change in the level of controls or methodology used for the processing of radioactive effluents or handling of solid radioactive waste, nor will the proposed amendment result in any change in the normal radiation levels in the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

## 7.0 PRECEDENCE

This request is similar to the license amendment authorized by the NRC on January 19, 2012, for the Palisades Nuclear Plant (TAC No. ME6875, ADAMS Accession Numbers ML113220370 and ML120600415).

# Attachment 2

## Marked-Up Technical Specifications Page

**DOMINION NUCLEAR CONNECTICUT, INC.**
**MILLSTONE POWER STATION UNIT 3**

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.~~4~~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.