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August 31, 2015

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

Serial No. 15-027A
NLOS/WDC R0
Docket No. 50-336
License No. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
SUPPLEMENT TO LICENSE AMENDMENT REQUEST TO REVISE
TS 6.19, CONTAINMENT LEAKAGE TESTING PROGRAM

By letter dated March 2, 2015, Dominion Nuclear Connecticut, Inc. (DNC) submitted a license amendment request (LAR) to revise Technical Specification (TS) 6.19, Containment Leakage Rate Testing Program, for Millstone Power Station Unit 2 (MPS2). In that letter DNC proposed to: 1) revise the definition of P_a in TS 6.19, and 2) revise the acceptance criteria for leakage rate testing of containment air lock door seals to substitute the use of the makeup flow method in lieu of the pressure decay method currently used at MPS2. This supplement provides additional information and modifies the March 2, 2015 proposed definition of P_a in TS 6.19. This supplement makes no change to the March 2, 2015 proposed revision to the acceptance criteria for leakage rate testing of containment air lock door seals.

Since MPS2 received an operating license in 1975, TS 3.6.1.2 and 3.6.1.3 equated P_a to the MPS2 containment design pressure of 54 psig. This is not consistent with the 10 CFR 50 Appendix J, Option B definition of P_a which states: *P_a (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis loss-of-coolant accident as specified in the Technical Specifications.* In the March 2, 2015 LAR, DNC requested a change to TS 6.19 to define P_a as the containment design pressure consistent with MPS2 TS 3.6.1.2 and 3.6.1.3.

Subsequent to the March 2, 2015 LAR, DNC identified a more appropriate set of TS changes to align the MPS2 TSs ~~P_a value~~ with the 10 CFR 50 Appendix J, Option B definition of P_a. DNC discussed this approach with the NRC staff in a teleconference on May 7, 2015, and proposed to submit a supplement to the LAR. This supplement modifies the March 2, 2015 LAR to align the MPS2 TSs ~~P_a value~~ with that contained in 10 CFR 50 Appendix J, Option B. Attachment 1 provides the description and assessment of the proposed change. Attachment 2 provides the marked-up TS pages to reflect the proposed TS changes. Attachment 3 provides marked-up pages to reflect the proposed change to the TS bases for information only and will be implemented in accordance with the TS bases control program.

This supplement requires a revision to the significant hazards consideration contained in the March 2, 2015 LAR. The revision to the significant hazards consideration is

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contained in Attachment 1. This proposed amendment as supplemented does not involve a significant hazards consideration pursuant to the provisions of 10 CFR 50.92.

This supplement to the March 2, 2015 LAR has been reviewed and approved by the Facility Safety Review Committee.

DNC requests approval of the proposed amendment by March 2, 2016. DNC will implement the revised TS within 60 days of NRC approval of the proposed amendment.

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

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

Sincerely,

Mark D. Sartain
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President – Nuclear Engineering 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 31st day of August, 2015.

My Commission Expires: 12/31/16.

Notary Public

CRAIG D SLY
Notary Public
Commonwealth of Virginia
Reg. # 7518653
My Commission Expires December 31, 2016

Attachments:
1. Discussion of Technical Specification Change
2. Marked-up Technical Specification Pages
3. Marked-up Technical Specification Bases Page – For Information Only

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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Attachment 1

Discussion of Technical Specification Change

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

1.0 Discussion of Proposed Technical Specification Change as Supplemented

By letter dated March 2, 2015, Dominion Nuclear Connecticut, Inc. (DNC) submitted a license amendment request (LAR) to revise Technical Specification (TS) 6.19, Containment Leakage Rate Testing Program, for Millstone Power Station Unit 2 (MPS2). In that letter DNC proposed to: 1) revise the definition of P_a in TS 6.19, and 2) revise the acceptance criteria for leakage rate testing of containment air lock door seals to substitute the use of the makeup flow method in lieu of the pressure decay method currently used at MPS2.

Since MPS2 received an operating license in 1975, TSs 3.6.1.2 and 3.6.1.3 equated P_a to the MPS2 containment design pressure of 54 psig. This is not consistent with the 10 CFR 50 Appendix J, Option B definition of P_a which states:

P_a (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis loss-of-coolant accident as specified in the Technical Specifications.

In the March 2, 2015 LAR, DNC requested a change to TS 6.19 to define P_a as the containment design pressure consistent with the value of P_a specified in MPS2 TS 3.6.1.2 and 3.6.1.3.

Subsequent to the March 2, 2015 LAR, DNC identified a more appropriate set of TS changes to align the MPS2 TSs with the 10 CFR 50 Appendix J, Option B definition of P_a. Specifically, DNC proposes to delete the containment design pressure value of 54 psig from TSs 3.6.1.2.a and 3.6.1.3.b and add the numerical value of Pa to TS 6.19. DNC discussed this approach with the NRC staff in a teleconference on May 7, 2015 and proposed to submit a supplement to the LAR. This supplement modifies the March 2, 2015 LAR to align the MPS2 TSs with that contained in 10 CFR 50 Appendix J, Option B. This supplement makes no change to the March 2, 2015 proposed revision to the acceptance criteria for leakage rate testing of containment air lock door seals.

The following two new TS changes are being proposed in this supplement to the March 2, 2015 LAR:

1) DNC proposes to revise TS 3.6.1.2.a on containment leakage rates (Note: Deleted text is struck-through):

   a. An overall integrated leakage rate of $<L_a$, 0.50 percent by weight of the containment air per 24 hours at P_a~~, 54 psig~~.

2) DNC proposes to revise TS 3.6.1.3.b on the containment air lock (Note: Deleted text is struck-through):

   b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a ~~(54 psig)~~.

The following change is a revision to the proposed TS change in the March 2, 2015 LAR.

DNC proposes to add to the definition of P_a in TS 6.19 a specific numerical value for P_a and identify that leakage rate testing will be performed at a value that bounds the containment design pressure. Therefore, the proposed change to the second paragraph of TS 6.19 would be revised as follows (Note: added text is italicized and bold):

> The peak calculated primary Containment internal pressure for the design basis loss of coolant accident is P_a. ***P_a is 53 psig. Containment leakage rate testing will be performed at the containment design pressure of 54 psig or higher.***

The proposed change to TS 6.19.b (air lock testing acceptance criteria) contained in the March 2, 2015 LAR, remains unchanged:

Markups of the proposed changes to TSs 3.6.1.2, 3.6.1.3 and 6.19 are provided in Attachment 2.

The peak calculated containment pressure for the Loss of Coolant Accident (LOCA) is 52.5 psig. The proposed value to be added to TS 6.19 is the LOCA calculated containment pressure rounded up to the next integer value which is 53 psig.

The peak calculated containment pressure for the Main Steam Line Break (MSLB) accident is 53.8 psig. The TS 3.6.1.2 maximum allowable primary containment leakage rate, L_a (0.50% of the primary containment air weight per 24 hours), is used in the MPS2 Final Safety Analysis Report (FSAR), Chapter 14, for the radiological dose calculations of both the LOCA and the MSLB. To ensure the use of the maximum allowable primary containment leakage rate, L_a, for both the LOCA and MSLB FSAR radiological dose calculations is conservative, containment leak rate testing will continue to be performed at the containment design pressure of 54 psig or higher.

2.0 Revised No Significant Hazards Consideration

The NRC has provided standards for determining whether a significant hazards consideration (SHC) exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no SHC if operation of the facility in accordance with a proposed 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. DNC has evaluated whether or not an SHC is involved with the proposed amendment. A discussion of these standards as they relate to this amendment request is provided below.

This proposed license amendment, as supplemented, would revise the definition of P_a contained in TSs 3.6.1.2 and 3.6.1.3 to be consistent with the P_a definition contained in 10 CFR 50 Appendix J, Option B. The proposed amendment also revises the method of surveillance for leakage rate testing of the containment air lock door seals as described in the March 2, 2015 LAR.

Criterion 1
Will operation of the facility in accordance with the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The design basis accident remains unchanged for the postulated events described in the MPS2 FSAR. Since the initial conditions and assumptions included in the safety analyses are unchanged, the consequences of the postulated events remain unchanged. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment also revises the method of surveillance for leakage rate testing of the containment air lock door seals. The makeup flow method will continue to provide assurance that the containment leakage rate is within the limits assumed in the radiological consequences analysis of the design basis accident, therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2
Will operation of the facility in accordance with this proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed amendment does not change the way the plant is operated and does not involve a physical alteration of the plant. No new or different types of equipment will be installed and there are no physical modifications to existing equipment associated with the proposed amendment. Similarly, the proposed amendment would not physically change any plant systems, structures, or components involved in the mitigation of any postulated accidents. Thus, no new initiators or precursors of a new or different kind of accident are created. Furthermore, the proposed amendment does not create the possibility of a new failure mode associated with any equipment or personnel failures. Therefore, the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3
Will operation of the facility in accordance with this proposed amendment involve a significant reduction in the margin of safety?

Response: No.

The proposed amendment does not represent any physical change to plant systems, structures, or components, or to procedures established for plant operation. The proposed amendment does not affect the inputs or assumptions of any of the design basis analyses and current design limits will continue to be met. Since the proposed amendment does not affect the assumptions or consequences of any accident previously analyzed, there is no significant reduction in the margin of safety.

Conclusion
Based on the above, DNC concludes that the proposed amendment does not represent a significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

3.0 Conclusion

Based on the considerations discussed above, there is reasonable assurance that (1) the health and safety of the public will not be endangered by the demonstration that MPS2 continues to meet applicable design criteria and safety analysis acceptance criteria, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

Attachment 2

Marked-up Technical Specification Pages

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of < L_a, 0.50 percent by weight of the containment air per 24 hours at P_a~~, 54 psig~~.

b. A combined leakage rate of < 0.60 L_a for all penetrations and valves subject to Type B and C tests when pressurized to P_a.

c. A combined leakage rate of < 0.014 L_a for all penetrations that are secondary containment bypass leakage paths when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.014 L_a, restore the leakage rate(s) to within the limit(s) prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated in accordance with the Containment Leakage Rate Testing Program.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 The containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a ~~(54 psig)~~.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: :

NOTE
Entry and exit through the containment air lock door is permitted to perform repairs on the affected air lock components.

a. With one containment air lock door inoperable:

1. Verify the OPERABLE air lock door is closed within 1 hour and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days.

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. Entry into an OPERATIONAL MODE or other specified condition under the provisions of Specification 3.0.4 shall not be made if the inner air lock door is inoperable.

b. With only the containment air lock interlock mechanism inoperable, verify an OPERABLE air lock door is closed within 1 hour and lock an OPERABLE air lock door closed within 24 hours. Verify an OPERABLE air lock door is locked closed at least once per 31 days there after. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (Entry into and exit from containment is permissible under the control of a dedicated individual).

c. With the containment air lock inoperable, except as specified in ACTION a. or ACTION b. above, immediately initiate action to evaluate overall containment leakage rate per Specification 3.6.1.2 and verify an air lock door is closed within 1 hour. Restore the air lock to OPERABLE status within 24 hours. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

6.19 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the primary containment as required by 10CFR50.54(o) and 10CFR50, 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 June 10, 1995 Type A test shall be performed no later than June 10, 2010.

The peak calculated primary Containment internal pressure for the design basis loss of coolant accident is P_a.

The maximum allowable primary containment leakage rate, L_a, at P_a, is 0.5% of primary containment air weight per day.

P_a is 53 psig. Containment leakage rate testing will be performed at the containment design pressure of 54 psig or higher.

Leakage rate acceptance criteria are:

a. Primary containment overall leakage rate acceptance criterion is < 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 < 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

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

   2. For each door, ~~pressure decay is ≤ 0.1 psig~~ leakage rate is $\le 0.01\ L_a$ when pressurized to ≥ 25 psig ~~for at least 15 minutes~~.

The provisions of SR 4.0.2 do not apply for test frequencies specified in the Primary Containment Leakage Rate Testing Program.

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

6.20 RADIOACTIVE EFFLUENT CONTROLS PROGRAM

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the REMODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the REMODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10CFR 20.1001-20.2402;

Attachment 3

Marked-up Technical Specification Bases Page

(For Information Only)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR 50.67 during accident conditions.

Primary CONTAINMENT INTEGRITY is required in MODES 1 through 4. This requires an OPERABLE containment automatic isolation valve system. In MODES 1, 2, and 3 this is satisfied by the automatic containment isolation signals generated by low pressurizer pressure and high containment pressure. In MODE 4 the automatic containment isolation signals generated by low pressurizer pressure and high containment pressure are not required to be OPERABLE. Automatic actuation of the containment isolation system in MODE 4 is not required because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating engineered safety features components. Since the manual actuation (trip pushbuttons) portion of the containment isolation system is required to be OPERABLE in MODE 4, the plant operators can use the manual pushbuttons to rapidly position all automatic containment isolation valves to the required accident position. Therefore, the containment isolation trip pushbuttons satisfy the requirement for an OPERABLE containment automatic isolation valve system in MODE 4.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses ~~at the peak accident pressure of P_a~~. As an added conservatism, the measured overall integrated leakage rate is further limited to < ~~0.75~~ L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

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The surveillance testing for measuring leakage rates is in accordance with the Containment Leakage Rate Testing Program.

The Millstone Unit No. 2 FSAR contains a list of the containment penetrations that have been identified as secondary containment bypass leakage paths.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and leak rate given in Specifications 3.6.1.1 and

Insert for TS Bases 3/4.6.1.2

P_a is the peak calculated primary containment internal pressure for the design basis loss of coolant accident. The peak calculated primary containment internal pressure for the design basis main steam line break accident is greater than P_a. Since the radiological dose consequence analysis of both these accidents assume containment leakage at the technical specification allowed leakage rate, containment leakage testing will be performed at a value greater than or equal to the containment design pressure.