December 22, 2016

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

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REVISION TO EMERGENCY CORE COOLING SYSTEM TECHNICAL SPECIFICATIONS AND FINAL SAFETY ANALYSIS REPORT CHAPTER 14 TO REMOVE CHARGING PUMP FLOW (CAC NO. MF7297)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 331 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated January 25, 2016, as supplemented on June 27 and October 12, 2016.

The amendment revises the MPS2 technical specifications (TSs) to remove the requirement for the charging pumps to be operable in TS 3.5.2, "Emergency Core Cooling Systems, ECCS Subsystems – $T_{avg} \geq 300$ °F," by eliminating surveillance requirement 4.5.2.e from the TSs. The proposed change also revises the MPS2 final safety analysis report relative to the long-term analysis of the inadvertent opening of a pressurized water reactor pressurizer pressure relief valve event and clarifies the existing discussion regarding the application of single failure criteria.

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

Sincerely,

Richard V. Guzman, Sr. Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 331 to DPR-65
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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

DOMINION NUCLEAR CONNECTICUT, INC.

DOCKET NO. 50-336

MILLSTONE POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 331
Renewed License No. DPR-65

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

   A. The application for amendment by Dominion Nuclear Connecticut, Inc. (the licensee) dated January 25, 2016, as supplemented by letters dated June 27 and October 12, 2016, 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. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 331, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Stephen S. Koenick, Acting Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-65

Date of Issuance: December 22, 2016

MILLSTONE POWER STATION, UNIT NO. 2

ATTACHMENT TO LICENSE AMENDMENT NO. 331

RENEWED FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

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

Remove	Insert
3	3

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 marginal lines indicating the areas of change.

Remove	Insert
3/4 5-4	3/4 5-4

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

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

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

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

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

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

(1) <u>Maximum Power Level</u>

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

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 331 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

NOTE

Not required to be met for system vent flow paths opened under administrative control.

a. At the frequency specified in the Surveillance Frequency Control Program by verifying each Emergency Core Cooling System manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying that the following valves are in the indicated position with power to the valve operator removed:

Valve Number	Valve Function	Valve Position
2-SI-306	Shutdown Cooling Flow Control	Open*
2-SI-659	SRAS Recirc.	Open**
2-SI-660	SRAS Recirc.	Open**

* Pinned and locked at preset throttle open position.

** To be closed prior to recirculation following LOCA.

c. By verifying the developed head of each high pressure safety injection pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

d. By verifying the developed head of each low pressure safety injection pump at the flow test point is greater than or equal to the required developed head when tested pursuant to Specification 4.0.5.

e. Deleted

f. At the frequency specified in the Surveillance Frequency Control Program by verifying each Emergency Core Cooling System automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.

g. At the frequency specified in the Surveillance Frequency Control Program by verifying each high pressure safety injection pump and low pressure safety injection pump starts automatically on an actual or simulated actuation signal.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 331

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.

MILLSTONE POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By application dated January 25, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16029A168), as supplemented by letters dated June 27, 2016 and October 12, 2016 (ADAMS Accession No. ML16182A037 and ML16291A508, respectively), Dominion Nuclear Connecticut, Inc. (DNC, the licensee) requested changes to the Millstone Power Station, Unit 2 (MPS2) technical specification (TS) Section 3.5.2, "Emergency Core Cooling Systems, ECCS Subsystems – $T_{avg} \geq 300$ °F," to remove the requirement for the charging pumps to be operable in TS 3.5.2 by eliminating TS Surveillance Requirement (SR) 4.5.2.e. The proposed change would also revise the MPS2 final safety analysis report (FSAR), relative to the long-term analysis in Section 14.6.1,"Inadvertant Opening of a Pressurized Water Reactor Pressurizer Pressure Relief Valve [IOPPRV]," and would clarify the existing discussion regarding the application of single failure criteria. DNC submitted this license amendment request (LAR) to comply with paragraph 2 of Confirmatory Order EA-13-188 (ADAMS Accession No. ML15236A207) which stated:

> By no later than February 15, 2016, DNC will submit a license amendment request to the NRC addressing the use of charging pumps in the analysis of the inadvertent opening of PORVs [power operated relief valves]. If DNC does not submit a license amendment request by February 15, 2016, the Millstone Unit 2 design and licensing basis for the operation of charging pumps to mitigate the inadvertent opening of PORVs that was in place prior to Amendment No. 283 (dated September 9, 2004) will be reinstated by this Confirmatory Order, and DNC will take all actions necessary to conform Millstone Unit 2 to the reinstated design and licensing basis.

The supplemental letters dated June 27 and October 12, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on May 24, 2016, (81 FR 32804).

2.0 REGULATORY EVALUATION

2.1 Background

In 2009, DNC made changes to the MPS2 FSAR under Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59 that removed the use of charging pumps credited in the analysis for the event documented in FSAR 14.6.1. By letter dated April 29, 2015 (ADAMS Accession No. ML15119A028), the NRC identified apparent violations involving the changes made to FSAR 14.6.1 and issued Confirmatory Order EA-13-188 on August 26, 2015 (ADAMS Accession No. ML15236A207), requiring DNC to submit a license amendment request (LAR) addressing the use of charging pumps in the FSAR 14.6.1 event.

2.2 Proposed Changes

The licensee proposed to revise the TS 3.5.2, "Emergency Core Cooling Systems, ECCS Subsystems – $T_{avg} \geq 300$ °F," to remove the requirement for the charging pumps to be operable by eliminating TS Surveillance Requirement (SR) 4.5.2.e, which states that:

> 4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE: ...
>
> e. By verifying the delivered flow of each charging pump at the required discharge pressure is greater than or equal to the required flow when tested pursuant to Specification 4.0.5.

In support of the proposed TS changes, DNC provided the results of its reanalysis of FSAR Chapter 14.6.1, to show that the charging system is not required to mitigate this event. The IOPPRV event is the only FSAR Chapter 14 analysis that credits flow from the charging pumps for event mitigation.

2.3 Applicable Regulatory Requirements

The NRC used the following regulatory requirements and guidance documents in evaluating the licensee's amendment request:

General Design Criteria (GDC) establish the necessary design, fabrication, construction, testing, and performance requirements for structures, system, and components (SSCs) important to safety; that is, SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. MPS2 was designed and constructed based on the 1967 draft Atomic Energy Commission (AEC) GDC. As noted in MPS2 FSAR Section 1.A, the design bases of MPS2 are compared against the GDCs in 10 CFR Part 50, Appendix A, as amended through February 20, 1971. GDC 10 and 15 from 10 CFR 50 Appendix A are applicable to the TS change request:

- GDC 10 requires that the Reactor Coolant System (RCS) be designed with appropriate margin to assure that Specified Acceptable Fuel Design Limits (SAFDLs) are not exceeded during normal operations, including Anticipated Operational Occurrences (AOOs).

- GDC 15 requires that the RCS and associated auxiliary systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operations, including AOOs.

Appendix K to Part 50, provides the requirements for acceptable features of the ECCS Evaluation Models. Specifically, Appendix K.I.3 states that the heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the loss-of-coolant accident LOCA.

The regulation 10 CFR 50.36(c)(2)(i) states:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation [LCO] of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

Section 50.36(c)(2)(ii) requires that a TS LCO of a nuclear reactor must be established for each item meeting one or more of the following criteria:

> (A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> (B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident [DBA] or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (C) Criterion 3. A structure, system, or component [SSC] that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (D) Criterion 4. [An SSC] which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Section 50.36(c)(3), requires that TSs include surveillance requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCO will be met.

As required by Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.36(a)(1), each applicant for an operating license shall include in its application proposed technical specifications in accordance with the requirements of 10 CFR 50.36. Further, per 50.36(a)(1), a summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications.

Per 10 CFR 50.36(b), each license authorizing operation of a utilization facility will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34. The analyses submitted under 10 CFR 50.34 include the Preliminary safety analysis report (PSAR), submitted under 50.34(a) as part of the application for a construction permit, and the FSAR submitted under 50.34(b) as part of the application for an operating license. The FSAR shall include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole. Last, also per 10 CFR 50.36, the Commission may include such additional technical specifications as the Commission finds appropriate.

Paragraph 50.59(c)(1) of 10 CFR states that a licensee can make changes in the facility or procedures as described in the UFSAR and conduct tests and experiments not described in the UFSAR without obtaining a license amendment pursuant to 10 CFR 50.90 if none of the criteria in 10 CFR 50.92(c)(2) are met. Paragraph 50.59(c)(3) of 10 CFR states that the UFSAR is considered to include FSAR changes resulting from evaluations performed pursuant to 10 CFR 50.59 and analyses performed pursuant to 10 CFR 50.90 since submittal of the last UFSAR pursuant to 10 CFR 50.71.

The Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, which was published in the *Federal Register* on July 22, 1993 (58 FR 39132), presents the policy of the NRC with respect to the scope and purpose of TSs as required by 10 CFR 50.36 and establishes the guidance for determining which operating restrictions should be included in the TSs. The policy states that each limiting condition for operation, Action, and surveillance requirement (SR) should have supporting Bases which should at a minimum address questions specified in the Policy Statement and cite references to appropriate licensing documentation (e.g., FSAR, Topical Report) to support the Bases.

As provided in 10 CFR 50.90, whenever a holder of an operating license desires to amend the license, application for an amendment must be filed with the Commission fully describing the changes desired, and following as far as applicable, the form prescribed for original applications.

In determining whether an amendment to a license will be issued to the applicant, 10 CFR 50.92(a) states that the Commission will be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Considerations common to many licenses and permits that guide the Commission's determination that a license will be issued are provided in 10 CFR § 50.40. The findings that the Commission must make to issue an operating license are given in 10 CFR § 50.57(a). Per 10 CFR 50.36(c), TS will include items in, among other things, the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls.

Generic Letter (GL) 89-10, "Safety-Related (1) Motor-Operated Valve Testing and Surveillance," recommended that each nuclear power plant establish a program to demonstrate that safety-related motor operated valves (MOVs) are capable of performing their design basis functions. Program features include analysis of worst case system demands on MOV operation, MOV setup to meet demands, and demonstration via dynamic testing that the MOV will perform its safety-related function.

GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor Operated Valves," superseded GL 89-10 and its supplements with regard to MOV periodic verification and requested that plants establish a program or ensure the effectiveness of the current program by periodically verifying that MOVs continue to be capable of performing their safety-related function. The PORV block valves are within the scope of DNC MOV program.

The NRC Standard Review Plan (SRP) for the Review of Safety Analysis Reports (SARs) for Nuclear Power Plants, NUREG-0800, provided guidance for NRC staff review in Section 15.6.1, "Inadvertent Opening of a PWR [pressurized-water reactor] Pressurizer Pressure Relief Valve or a BWR [boiling water reactor] Pressure Relief Valve," (ADAMS Accession No. ML070820094). The NRC SRP listed specific acceptance criteria derived from applicable General Design Criteria and other NRC regulations and a method acceptable to the staff to demonstrate compliance with those acceptance criteria. The acceptance criteria for event escalation states:

> An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant.

3.0 TECHNICAL EVALUATION

The NRC staff's review of the IOPPRV reanalysis for supporting the proposed change to TS 3.5.2 by deleting SR 4.5.2.e and removing credit for charging pumps from the IOPPRV event in Chapter 14 of the FSAR covered the sequence of events, the analytical model used for analyses, the values of parameters used in the analytical model, and the results of the transient reanalysis.

3.1 IOPPRV analysis

The initiating event for the FSAR 14.6.1 IOPPRV event is an electrical or mechanical failure of one or more safety or relief valves. MPS2 has two PORVs and two pressurizer safety valves (PSVs). This event causes a decrease in reactor coolant inventory and RCS pressure that results in a degradation of thermal margin that could lead to a departure from nucleate boiling (DNB) in the short-term. Although the reactor shutdown protects against fuel clad damage, it does not end the RCS depressurization. In the long-term, the RCS inventory must be maintained so the reactor core remains covered, SAFDLs are not exceeded, and no fuel clad damage occurs.

The IOPPRV event causes depressurization of the RCS and the 110% of the RCS design pressure is not challenged; thus, it meets the requirements of GDC 15 (corresponding to 1967 draft GDC 33) related to the reactor coolant pressure boundary limits.

The IOPPRV short-term analysis as currently presented in FSAR 14.6.1 is not being revised since it does not credit the charging system. The short-term IOPPRV previously approved by the NRC confirmed that the DNB ratio and fuel melt limits were adequately met, meeting the requirements of GDC 10 related to the fuel integrity.

3.1.1 Analytical Methods and Input Parameters

The licensee provided an IOPPRV long-term reanalysis using AREVA's S-RELAP-5 small-break loss-of-coolant accident (SBLOCA) methodology which was approved by the staff in letter dated September 1, 2015.[1]

The licensee provided the S-RELAP-5 input parameters and assumptions in Tables 1 and 2 in its response to request for additional information (RAI)-5 dated June 27, 2016 (ADAMS Accession No. ML16182A037). The licensee stated that the IOPPRV long-term analysis employed several key assumptions, summarized below, to ensure a conservative calculation.

The initiation of the high pressure safety injection (HPSI) system was delayed for 10 seconds following safety injection actuation signal (SIAS) activation. No HPSI flow was delivered until system pressure fell below the HPSI pump shutoff head. This occurred more than 85 seconds after the HPSI system was available. Minimum HPSI flow rates with two HPSI pumps were used to include the effect of pump degradation. The NRC staff finds that the assumptions are conservative since they minimize the HPSI flow and the RCS inventory during the transient, resulting in a lower margin to the core uncovery and SAFDLs conditions.

The heat generation rate in the S-RELAP-5 reactor core model was determined from reactor kinetics equations with actinide and decay heat as prescribed by 10 CFR 50, Appendix K, "ECCS Evaluation Models." The NRC staff finds that this assumption is conservative, since it maximizes the decay heat level and causes more mass to release through the PSV, resulting in a minimum RCS inventory and a lower margin to the core uncovery SAFDLs conditions.

The reactor SCRAM is delayed based on the low pressurizer pressure thermal margin/low pressure (TM/LP) (floor or minimum) reactor trip and includes Technical Requirements Manual (Section 3/4.3, "Instrumentation") maximum delays for Reactor Protection System (RPS) circuitry and Control Element Assembly (CEA) coil delay. The NRC staff finds that the assumptions for delay of a reactor trip are conservative, since the assumptions results in more energy generated in the core before the reactor trip, causing more mass to release through the

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[1] NRC Letter, "'Final Safety Evaluation by the Office of Nuclear Reactor Regulation for Topical Report EMF-2328(P)(A), Revision 0, Supplement 1, Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based' (TAC No. ME8227)" (ADAMS Accession No. ML15210A252)

PSV, and resulting in a minimum RCS inventory and a lower margin to the core uncovery and SAFDLs conditions.

As stated by the licensee in Attachment 4 in its application, "Millstone Power Station Unit 2 Proposed License Amendment Request, Small Break Loss of Coolant Accident Reanalysis," dated September 1, 2015 (ADAMS Accession No. ML15253A205), the licensee performed an RCP trip sensitivity for both cold and hot leg breaks with delayed time following loss of subcooling margin to demonstrate compliance with 10 CFR 50.46(b)[2]. The results of the delayed RCP trip sensitivity demonstrated that there is at least 2 minutes for operators to trip all four RCPs after subcooling margin is lost in the cold leg pump suction in order to meet the requirements of 10 CFR 50.46(b). The NRC staff finds that this assumption is adequate since it is consistent with an NRC-approved realistic SBLOCA analysis (NRC License Amendment No. 329 for MPS2, ADAMS Accession No. ML16249A001).

The initiating event assumes a flow area for one open PSV (0.01767 ft2) which bounds the combined flow area of 2 PORVs (0.01730 ft^2, RAI-4 response in ADAMS Accession No. ML16182A037). The NRC staff finds the assumption of a large-break flow area is conservative, resulting in more mass to release through the applicable PSV or PORVs and a lower margin to the core uncovery and SAFDLs conditions.

3.1.2 Results of the Licensee's Reanalysis

As stated in its LAR, charging flow and HPSI flow were credited for mitigating the long-term portion of the IOPPRV event for MPS2 in DNCs previous analysis of the IOPPRV event[3]. This long-term analysis demonstrated that the capacity of the charging and HPSI pumps was sufficient to compensate for the loss of primary coolant mass through the IOPPRV such that there was no core uncovery. Therefore, with no core uncovery, there would be no long-term challenge to the DNBR and fuel centerline melt specified acceptable fuel design limits. The long-term analysis was later revised using the NRC-approved S-RELAP-5 small break LOCA methodology (ADAMS Accession No. ML15210A252). In this long-term analysis, the charging pumps are not credited for accident mitigation. In the postulated IOPPRV event, the RCS pressure decreases rapidly and reactor trip occurs on TM/LP. An SIAS occurs on low pressurizer pressure and the analysis assumes that flow from only the HPSI pumps starts. The pressurizer water level begins to increase and reaches 100% pressurizer level within 3 to 4 minutes after the HPSI pumps start. Once the pressurizer is filled, the failed open valve is assumed to pass water and/or a two-phase mixture for about 35 minutes and remain failed open. After the RCPs trip, the pressurizer level decreases and eventually reaches a level of approximately 80% to 85%.

The results of the IOPPRV reanalysis demonstrates that with flow from two HPSI pumps and no flow from the charging system, the core remains adequately covered with a two-phase liquid/vapor mixture to prevent exceeding the SAFDLs. Since the IOPPRV reanalysis shows no challenge to the fuel clad integrity, the NRC staff determined that the reanalysis meets the

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[2] Section 4.3, "RCP Trip Sensitivity Study," of ANP-3315NP, Revision 0, Millstone Unit 2 M5 Upgrade, Small Break LOCA Analysis Licensing Report, April 2015 (ADAMS Accession No. ML15253A206).
[3] EMF-87-161, Revision 0, "Millstone Unit 2 Plant Transient Analysis Report: Analysis of Chapter 15 Events," dated September 19, 1988, as supplemented on October 28, 1988.

requirements of GDC 10. The inadvertent opening of two PORVs was previously evaluated as part of the IOPPRV event for MPS2 as discussed in FSAR Section 14.6.1. The IOPPRV reanalysis identified that inadvertent opening of a PSV is the limiting event rather than an inadvertent opening of two PORVs. As discussed in section 3.1.1 of this SE, the flow area for one open PSV bounds the combined flow area of two pressurizer PORVs. Thus, the NRC staff finds that the inadvertent opening of a PSV as the limiting event remains consistent with the previous analysis with respect to core uncovery, fuel clad temperature, SAFDL, and the event non-escalation criteria.

Since the pressurizer fills during the long-term reanalysis, the failed valve will pass water and/or a two-phase liquid/vapor mixture. The MPS2 PSVs and PORVs are not qualified to relieve water and must be assumed to fail in the open position after water relief. Since the initiating event is a mechanical failure of a PSV, passing water through the failed PSV has no additional impact on the operability of the valve (it will not close even if it does not pass water). However, the reanalysis shows that while the pressurizer is filled, RCS pressure is much less than the valve's opening setpoint, so a non-failed valve (PORV or PSV) would not open during the analyzed IOPPRV transient. As such, there would be no further challenge to this RCS fission product barrier filling the pressurizer beyond that caused by the initiating IOPPRV event. Therefore, the NRC staff finds that (1) the reanalysis provides reasonable assurance that the IOPPRV event would not generate a more serious plant condition without other faults occurring independently, and (2) there is no additional challenge to the RCS boundary in addition to the initiating event, thereby satisfying the SRP 15.6.1 acceptance criteria regarding event escalation limitation.

3.1.3 Block Valves to Isolate the PORVs in Water or Water- Steam Conditions

As noted in the existing MPS2 FSAR 14.6.1, the IOPPRV event is initiated by the inadvertent opening of one or more PORVs or PSVs due to an electrical or mechanical failure. DNC indicated in its response to RAI-6 (ADAMS Accession No. ML16182A037) that if the initial event involves the inadvertent opening of the PORV (IOPORV), MPS2 Emergency Operating Procedure (EOP)-2525, "Standard Post-Trip Actions," directs the operator to close the associated PORV block valve(s). The IOPORV event, an AOO, is a depressurization event. The steam releases from the open PORVs which results in a decrease in the RCS pressure. If operators do not take appropriate actions to terminate the RCS depressurization by either closing the PORV or its block valve, the safety injection (SI) system will be actuated when the RCS pressure decreases to the low pressurizer RPS signal. Injection of the HPSI pump flow following the SI actuation signal could fill the pressurizer and lead the PORV and the associated block valve to discharge water or steam-water mixtures. Similar to the above PSV analysis, after the PORV and its block valve discharge water or steam-water mixture, the valves are assumed to fail open, unless they are qualified for water or steam-water mixture releases. As a result, the initiating AOO event (i.e., IOPORV event) could escalate to an accident (i.e., an un-isolable SBLOCA). This result would not meet the acceptance criterion for the analysis of the AOOs. For the IOPORV event, the SRP 15.6.1 guidance states that in meeting the acceptance criterion, the event must not generate a more serious plant condition without other faults occurring independently.

Like the above PSV analysis, passing water through the initially failed PORV, has no impact on operability of the valve. However, unlike the above PSV analysis, there is an initially operable block valve. At the NRC staff's request, DNC provided plant data showing that the PORV block valve(s) will close, on demand, in water or a steam-water mixture condition. The NRC staff's review of the plant data is discussed in the following section.

NRC Staff Evaluation of the Licensee's Plant Data for PORV Block Valves

NRC GL 89-10 recommended that each nuclear power plant establish a program to demonstrate that safety-related MOVs are capable of performing their design basis functions. Program features include analysis of worst case system demands on MOV operation, MOV setup to meet demands, and demonstration via dynamic testing that the MOV will perform its safety-related function. GL 96-05 superseded GL 89-10 and requested plants to establish a program or ensure the effectiveness of the current program by periodically verifying that MOVs continue to be capable of performing their safety-related function. The PORV block valves are within the scope of the licensee's MOV program.

During the implementation phase of GL 89-10, the industry realized that there was a population of MOVs that cannot be dynamically tested in situ due to various hardships such as system configuration, as low as reasonably achievable (ALARA) concerns, inaccessibility, testing could cause system or component damage, and excessive personnel hazards. In response, the Electric Power Research Institute (EPRI) initiated efforts to develop a computational methodology to be used in demonstrating the design basis capability of MOVs when valve specific test data is not available. The EPRI Performance Prediction Methodology (PPM) includes computer models, software, and hand calculation models to predict individual valve performance. The PPM model is based on extensive testing of several model valves under various conditions and evaluating the test results with the predicted computational value. The final product was the EPRI MOV PPM Program. The key elements of the PPM program are:

1) System Flow Model to predict the differential pressure and fluid pressure for pumped flow and blowdown system configurations
2) Gate Valve Model to predict the thrust required to operate gate valves and potential damage at sliding surfaces
3) Globe Valve Model to predict the thrust required to operate globe valves
4) Butterfly Valve Model to predict the torque required to operate butterfly valves

The results of the EPRI PPM efforts were captured in topical report TR-103237 "EPRI MOV Performance Prediction Program" and submitted by the Nuclear Energy Institute (NEI) on February 22, 1994 (Revision 0) and November 3, 1995 (Revision 1) to the Nuclear Regulatory Commission (NRC) for evaluation. NRC staff completed the safety evaluation on March 15, 1996 (ADAMS Accession No. ML15142A761) and accepted the PPM methodology described in the topical report, with certain conditions and limitations.

The PORV block valves at MPS2 are a normally open Velan 2 ½ inch flexible wedge gate valve with a Limitorque SMB-00 actuator. Due to system design and other hardships, the PORV block valves are difficult to dynamically test at projected design pressure and flow. The licensee elected to use the PPM method to validate the MPS2 PORV block valves to be operationally ready to perform their safety function.

The EPRI PPM gate valve computer model predicts the thrust required to operate gate valves throughout their stroke up to initial wedging under specified fluid conditions and differential pressure. The model uses theoretical equations that address fluid loading on the disk as well as the detailed mechanical interaction between the stem, disk, guides and seat. Valve internal information (dimensions and materials) are required to develop the model as well as piping configuration and fluid conditions for the system flow model. The use of the PPM gate valve model assumes that the valve is in good condition. However, the staff notes that it is necessary for model users to ensure that an adequate internal valve preventive maintenance program is established for the thrust or torque requirements predicted by the model to remain valid. End users are also cautioned that aging conditions or valve degradation can influence valve performance, which may or may not be accelerated by nonstandard orientations. The NRC staff approved the gate valve PPM model with certain conditions and limitations as noted in its safety evaluation March 15, 1996 (ADAMS Accession No. ML15142A761).

DNC responded to the NRC staff's follow-up RAI (RAI-1) in its supplement dated October 12, 2016 (ADAMS Accession No. ML16291A508) to demonstrate that the MPS2 pressurizer PORV block valves can be credited for closure under conditions predicted from the analysis of an IOPORV by performing a PPM calculation for the PORV block valve MOVs. The calculation was completed using EPRI MOV PPM software program version 3.5. This version has been approved by the NRC staff in a letter dated April 2, 2015 (ADAMS Accession No. ML15075A012). The NRC staff reviewed the calculation results, parameters selected, model valve, current MOV settings, and the projected flow and pressure conditions and finds the analysis provides reasonable assurance that the PORV block valves will close on demand. This conclusion is based on the PPM gate valve model prediction.

To ensure that the PPM calculation for PORV block valves remains valid, MPS2 has a preventive maintenance program that maintains valve internals in good condition. In addition, the PORV block valve MOVs are periodically verified via diagnostic testing to ensure the MOVs have positive margin and are operationally ready. Diagnostic data is also used to monitor actuator performance and degradation such as stem nut wear, stroke time, torque switch repeatability, and spring pack operation. The NRC staff concludes that the PORV block valve MOVs will close under conditions predicted from the analysis of an IOPORV.

In addition, based on the review discussed above, the NRC staff finds that the plant data provides reasonable assurance that the block valves could be closed in water or a water-steam mixture condition, and avoid escalation of the IOPORV event (an AOO) to an un-isolable SBLOCA (an accident), thereby satisfying SRP 15.6.1 acceptance criteria regarding event escalation limitation.

3.1.4 Long Term IOPPRV Piping Structure Consideration

The PSVs and PORVs are connected to nozzles on the top of the pressurizer vessel. As discussed in FSAR Section 4.3.5, the discharge from the PSVs and PORVs is piped to a quench tank where it is cooled and condensed by water in the tank. Because the current FSAR clearly states that steam is discharged from the pressurizer safety and relief valves, the NRC staff considered that the hydrodynamic analysis may address short-term steam discharge only. Therefore, the NRC staff noted that the current IOPPRV structural analysis may not support the discharge of water or a two-phase steam/water mixture associated with the new long-term IOPPRV event and requested the licensee to address this concern.

In its response to RAI-2 (ADAMS Accession No. ML16182A037), DNC provided a justification to support that the current analysis is bounding. DNC's response stated that the current hydrodynamic analysis for PSV and PORV piping considers multiple valves discharging at the same time. The IOPPRV transient is bounded by the current analysis condition since IOPPRV considers one stuck open PSV or two open PORVs. DNC also stated that the current hydrodynamic analysis has higher operating pressure and larger momentum change (due to rapid valve opening).

The staff acknowledged that the structural analysis applied force (F= ma, where 'a' is acceleration, and 'm' is mass) is the derivative of momentum (Momentum = mv, where v is velocity). At the initial rapid opening, the choked steam flow velocity changed from initial zero to the speed of sound at a 25-millisecond opening time for PSV and 160-millisecond opening time for PORV. At the time at which the pressurizer fills with a saturated two-phase mixture, the choked two-phase flow velocity is at the speed of sound of the corresponding pressure. The associated momentum change for the two-phase flow or water occurs relatively slower compared to the initial valve opening stage. Therefore, the largest applied loads to the associated piping and supports occur at the initial opening of a PSV or a PORV due to the rapid fluid momentum change. On the basis that the largest loads occur at the initial stage, the NRC staff concludes that the current structural analysis adequately address the new long-term IOPPRV event.

Based on its review discussed in above, the NRC staff finds that: (1) DNC's re-analysis of the long-term IOPPRV event appropriately used the NRC-approved AREVA S-RELAP-5 methodology; (2) the assumptions used in the reanalysis correctly reflected the proposed removal of the charging system from the TSs; and (3) the results of the reanalysis satisfy the GDC-10 requirements related to the fuel integrity, GDC-15 requirements related to the reactor coolant pressure boundary design conditions, and SRP 15.6.1 acceptance criteria regarding event escalation limitation. Therefore, the NRC staff concludes that the reanalysis of the IOPPRV event is acceptable for supporting the proposed TS change that would remove SR 4.5.2.e for charging pump flow verification.

3.1.5 NRC Staff Evaluation of the Proposed TS Change per 10 CFR 50.36(c)(2)(ii)

The proposed change to TS 3.5.2 is to remove the requirement to verify charging pump flow by deleting SR 4.5.2.e, which states that:

> 4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE: ...
>
> e. By verifying the delivered flow of each charging pump at the required discharge pressure is greater than or equal to the required flow when tested pursuant to Specification 4.0.5.

As described in 10 CFR 50.36(c)(4), the purpose of a SR is to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. Logically, if there is no LCO, then there is no need for a SR to assure that the non-existent LCO will be met, although the SR could still fulfill the other elements in 10 CFR 50.36(c)(4). Applying that logic to the current LAR, which proposes to delete a SR on the charging system, the staff considered whether a technical specification LCO must be established for the charging system based on the four criteria of 10 CFR 50.36(c)(2)(ii). The applicability of the each of the four criteria to the charging system is discussed as follows:

> (A) Criterion 1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion1 is not applicable since the charging system does not detect or indicate degradation of the RCS pressure boundary.

> (B) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (C) Criterion 3. A SSC that is part of the primary success path and which functions or actuates to mitigate a DBA or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criteria 2 and 3 are not applicable to the charging system, since all analyses supporting UFSAR Chapter 14 events do not credit the charging pumps for event mitigation. FSAR events that typically credit flow from the charging pumps for mitigation are LOCA and Main Steam Line Break (MSLB). For MPS2, both FSAR 14.6.5 LOCA and FSAR 14.1.5 MSLB have been analyzed without the charging system. With the proposed re-analysis of the long-term portion of the IOPPRV without charging, MPS2 does not credit the charging systems for mitigation in any of the FSAR Chapter 14 events. Although some of the FSAR events still use the charging system when it has no impact on the results or makes the consequences worse, use of the charging system is not credited for event mitigation. These included FSAR 14.2.7 Loss of Normal Feedwater Flow, FSAR 14.4.6 Decrease in Boron Concentration, and 14.6.3 Radiological Consequences of Steam Generator Tube Failures.

> (D) Criterion 4. A SSC which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Criterion 4 does not apply to the charging system, as the licensee's risk evaluation of the ECCS function of the charging system indicates that it is not significant to public health and safety. In the NRC safety evaluation of License Amendment No. 283 (ADAMS Accession No. ML042240103), the NRC acknowledged that subsequent to the 2002 LAR submittal, the probabilistic risk assessment model was revised to show that the charging pumps are no longer risk significant.

Accordingly, no technical specification LCO must be established for the charging system, and there is no need for a surveillance requirement that otherwise would be used to verify that the LCO would be met. Therefore, the NRC staff concludes that the proposed change to TS 3.5.2 is acceptable.

3.1.6 FSAR Changes to Remove Credit for Charging from Chapter 14 and Clarify Single Failure Criteria Application

DNC proposed changes to FSAR Chapter 14, as published prior to the 2009 change, to remove credit for charging pumps from the IOPPRV event in FSAR Section 14.6.1.6 and FSAR Table 14.0.9-1 that were inappropriately incorporated under 10 CFR 50.59. As discussed above, the NRC staff concludes that the proposed changes to licensing basis associated with the removal of charging pump flow credit from the IOPPRV event (in FSAR section 14.6.1 and Table 14.0.9-1) are acceptable.

DNC also proposed to modify FSAR Section 14.0.11, "Plant Licensing Basis and Single Failure Criteria," to clarify the application of single failure criteria (SFC) for the RPS, engineered safety features (ESF), and onsite and offsite power systems considered in the plant safety analysis. Specifically, the change clarifies that the postulation of an additional single failure is applied to design basis accident scenarios or limiting fault event scenarios, but is not applied to all FSAR Chapter 14 events. The Chapter 14 events that result in a reactor trip assume that the reactor trip will occur considering a single failure in the RPS. Thus, in terms of the IOPPRV event, which is a moderate frequency, or Condition II, event, the proposed change clarifies that an additional single failure does not apply[4].

Additionally, DNC proposed to update the initiating event description of the IOPPRV event in FSAR section 14.6.1.1 to identify that the maximum capacity of a single PSV is greater than the maximum capacity of two PORVs. As discussed in Section 3.1.1 in this SE, the initiating event for the FSAR 14.6.1 IOPPRV assumes a flow area for one open PSV which bounds the combined flow area of two PORVs. The NRC staff determined that the assumption of a large-break flow area is conservative, resulting in more mass to release through the applicable PSV or PORVs and a lower margin to the core uncovery and SAFDLs conditions.

The NRC staff reviewed the proposed changes to FSAR Section 14.0.11 and Section 14.6.1.1 to verify consistency with the analyses. Based on the above evaluation, the staff finds that the information provided in the proposed FSAR changes reflect the revised analyses. Therefore, the staff concludes that the proposed FSAR changes are acceptable.

In addition to the changes to the FSAR, the licensee provided a mark-up of the associated technical bases pages for supporting the proposed TS changes. As described in TS Section 6.23, the licensee maintains a TS Bases Control Program. TS Section 6.23.c states that the Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR. Accordingly, the NRC expects to the licensee to update its TS Bases

[4] The IOPPRV event with the postulation of a single failure (other than within the RPS) would be categorized as an accident that would be bounded by the results of the small break LOCA presented in FSAR Section 14.6.5.

in a manner consistent with the information provided in the LAR dated January 25, 2016, as supplemented by letters dated June 27 and October 12, 2016.

3.3 NRC Staff Conclusion

Based on its review discussed in the above sections, the NRC staff finds that: (1) the DNC's re-analysis of the long-term IOPPRV event appropriately used the NRC-approved AREVA S-RELAP-5 methodology; (2) the assumptions used in the reanalysis were adequate and correctly reflected the proposed removal of the charging system from TS; and (3) the results of the reanalysis satisfy GDC-10 requirements for fuel integrity, GDC-15 requirements that the reactor coolant pressure boundary design conditions not be exceeded, and SRP 15.6.1 acceptance criteria regarding event escalation limitation. Therefore, the NRC staff concludes that the reanalysis of the IOPPRV event is acceptable for supporting the proposed TS change that would remove SR 4.5.2.e for charging pump flow verification and the proposed changes to FSAR Chapter 14 to remove credit for charging pumps from the IOPPRV event. The NRC staff also finds that with the acceptable IOPPRV reanalysis, there were no FSAR Chapter 14 events that credited the charging pumps for event mitigation. The NRC staff concludes that the charging pumps do not meet any of the criteria for LCOs as specified in 10 CFR 50.36(c)(2)(ii). Therefore, the proposed change to TS 3.5.2 "Emergency Core Cooling System, ECCS Subsystems – Tavg > 300 °F" to remove the requirement for charging pumps to be operable and eliminate TS SR 4.5.2.e is acceptable for MPS2.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified on November 3, 2016, of the proposed issuance of the amendment. The State official responded with no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes the SRs. The NRC staff has determined that no significant change in the types or significant increase in the amounts 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 (81 FR 32804). 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: S. Sun
M. Farnan

Date: December 22, 2016

December 22, 2016

Mr. David A. Heacock
President and Chief Nuclear Officer
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SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: REVISION TO EMERGENCY CORE COOLING SYSTEM TECHNICAL SPECIFICATIONS AND FINAL SAFETY ANALYSIS REPORT CHAPTER 14 TO REMOVE CHARGING PUMP FLOW (CAC NO. MF7297)

Dear Mr. Heacock:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 331 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated January 25, 2016, as supplemented on June 27 and October 12, 2016.

The amendment revises the MPS2 technical specifications (TSs) to remove the requirement for the charging pumps to be operable in TS 3.5.2, "Emergency Core Cooling Systems, ECCS Subsystems – $T_{avg} \geq 300$ °F," by eliminating surveillance requirement 4.5.2.e from the TSs. The proposed change also revises the MPS2 final safety analysis report relative to the long-term analysis of the inadvertent opening of a pressurized water reactor pressurizer pressure relief valve event and clarifies the existing discussion regarding the application of single failure criteria.

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

Sincerely,

/RA/
Richard V. Guzman, Sr. Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 331 to DPR-65
2. Safety Evaluation

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Accession No.: ML16308A485 * See memo dated July 22, 2016

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