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

April 18, 2014

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: REVISE TECHNICAL SPECIFICATION 3/4.7.11 ULTIMATE HEAT SINK (TAC NO. MF1779)

Dear Mr. Heacock:

The Commission has issued the enclosed Amendment No. 318 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2, in response to your application dated May 3, 2013, as supplemented by letters dated June 27, 2013, July 19, July 30, August 1, and October 2, 2013.

The amendment would revise Technical Specification (TS) 3/4.7.11, "Ultimate Heat Sink", to increase the current ultimate heat sink water temperature limit from 75 °F to 80 °F and change the TS Action to state, "With the ultimate heat sink water temperature greater than 80 °F, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours."

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,

James Kim

James Kim, Project Manager
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 318 to DPR-65
2. Safety Evaluation

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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. 318
Renewed License No. DPR-65

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

   A. The application for amendment by the applicant dated May 3, 2013, as supplemented by letters dated June 27, 2013, July 19, July 30, August 1, and October 2, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

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. 318, 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 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley

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

Attachment: Changes to the License
and Technical Specifications

Date of Issuance: April 18, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 318

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 pages of the Appendix A Technical Specifications, with the attached revised page. The revised page is identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 7-34	3/4 7-34

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) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 318, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3/4.7.11 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.11 The ultimate heat sink shall be OPERABLE with a water temperature of less than or equal to 80°F.

APPLICABILITY: MODES 1, 2, 3, AND 4

ACTION:

With the UHS water temperature greater than 80°F, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.11 The ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the water temperature to be within limits.

b. At least once per 6 hours by verifying the water temperature to be within limits when the water temperature exceeds 75°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 318

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 letter dated May 3, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13133A033), as supplemented by letters dated June 27, 2013 (ADAMS Accession No. ML13198A271), July 19, 2013 (ADAMS Accession No. ML13204A035), July 30, 2013 (ADAMS Accession No. ML13213A024), August 1, 2013 (ADAMS Accession No. ML13219A109), and October 2, 2013 (ADAMS Accession No. ML13281A809), Dominion Nuclear Connecticut, Inc. (the licensee) proposed changes to the Technical Specifications (TSs) for Millstone Power Station Unit 2 (MPS2).

The proposed change revises TS 3/4.7.11, "Ultimate Heat Sink" (UHS), to increase the current UHS water temperature limit from 75 °F to 80 °F and change the TS Action to state, "With the ultimate heat sink water temperature greater than 80 °F, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours."

The supplemental letters dated June 27, 2013, July 19, July 30, August 1, and October 2, 2013, 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* on August 20, 2013 (78 FR 51225).

2.0 REGULATORY EVALUATION

The regulatory requirements and the guidance upon which the staff based its review of the effects on containment analyses due to the proposed change are based on the following 10 CFR 50 Appendix A General Design Criteria (GDC):

- GDC-16 as it relates to the containment and associated systems establishing a leak-tight barrier against the uncontrolled release of radioactivity to the environment and assuring that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

- GDC-38 as it relates to the containment heat removal system safety function which shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any Loss-Of-Coolant Accident (LOCA) and to maintain them at acceptably low levels.

- GDC-50 as it relates to the containment heat removal system which shall be designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

The MPS2 Final Safety Analysis Report (FSAR) Section 1A lists the extent of MPS compliance with 10 CFR Part 50 Appendix A of which Criterion 44, Cooling Water, requires a system to transfer the combined heat from Systems Structures and Components (SSC) important to safety to an UHS under normal and accident conditions with suitable redundancy and specified electrical system availabilities and assuming a single failure. FSAR Section 1A, regarding Criterion 44, requires the RBCCW and SW systems to transfer the combined heat from SSCs under normal and accident conditions. FSAR Section1A further states, "The RBCCW and SW systems are provided with suitable redundancy in components and suitable interconnections to assure heat removal capability. The systems are designed to enable isolation of system components or headers and to detect system mal-operation. The RBCCW and SW systems are designed to operate with onsite power (assuming offsite power is not available) and with offsite power (assuming onsite power is not available). The systems are designed such that a single failure in either system will not adversely affect safe operation, accident mitigation, or safe shutdown of the plant."

FSAR Section 9.9.16, "Vital Switchgear Ventilation System," specifies the maximum allowed room temperature limits for the vital AC and DC switchgear rooms. The upper and lower 4160/6190 volt switchgear rooms, the west 480 volt switchgear room, and the east and west vital DC switchgear rooms are the vital switchgear rooms cooled by the UHS. The SW system from the UHS supplies the cooling water to the ventilation systems and refrigerant system that cool these vital switchgear rooms. The upper and lower 4160/6190 volt switchgear rooms have a room temperature limit of 122 °F. The west 480 volt switchgear room and the east and west vital DC switchgear rooms have room temperature limits of 104°F.

FSAR Section 9.9.8 states that the Engineered Safety Features Room Air Recirculation System (ESFRARS) is designed to limit the maximum ambient temperature to 145 °F except for a brief transient temperature excursion following an accident. ESFRARS is cooled by RBCCW which is cooled by SW and the UHS.

FSAR Section 6.5.2 states that each containment air recirculation cooling unit is designed for removing 80 x 10^6 Btu/hr under Main Steam Line Break (MSLB) accident or LOCA conditions prior to recirculation with air flow of 34,800 cfm and a fouling factor of 0.0005 for the RBCCW side of the coil. The containment air recirculation and cooling units are cooled by RBCCW which is cooled by SW and the UHS.

FSAR Section 6.3.2 states that the high pressure and low pressure safety injection pumps have mechanical seals. The seals are designed for operation with a pumped fluid temperature of 350°F. To permit extended operation under these conditions, a portion of the pump fluid is externally cooled by the RBCCW system and re-circulated to the seals. The containment spray pumps also have mechanical seals cooled by RBCCW. The seal coolers for these pumps are cooled by RBCCW which is cooled by SW and the UHS.

FSAR, Section 6.1.2.1 defines engineered safety features to include safety injection, containment air recirculation and cooling, reactor building closed cooling system (RBCCW), emergency electrical power (diesel generators) and SW among other safety features. The SW system supports safety injection, Emergency Diesel Generators (EDG), and RBCCW.

FSAR, Section 9.7.2, states that the SW system shall be designed with suitable redundancy that in event of a LOCA and a concurrent loss of offsite power and single active failure that the SW system can perform its safety functions.

3.0 TECHNICAL EVALUATION

The UHS consists of the Service Water (SW) system supplying cooling water to safety related loads, i.e. the Reactor Building Closed Cooling Water (RBCCW) system, diesel engine heat exchangers, and the vital AC and DC switchgear ventilation system. The SW system also supplies cooling water to non-safety related loads, i.e. Turbine Building Closed Cooling water, chilled water heat exchangers and chlorination system. The water source for the UHS is Long Island Sound which is connected to the Atlantic Ocean. The SW system has three half capacity pumps taking suction downstream of the traveling screens in the intake structure. Two SW pumps are in continuous operation with a spare pump provided. One pump supplies sufficient heat removal capability for the RBCCW heat exchangers to safely shut down the plant and for accident mitigation.

TS LCO 3.7.11, "Ultimate Heat Sink," currently requires the UHS to have a temperature of less than or equal to 75°F in order to be OPERABLE. If 75 °F is exceeded, the ACTION is:

a.) With temperature > 75 °F and ≤ 77 °F, operation may continue provided the water temperature averaged over the previous 24 hour period is verified ≤ 75 °F at least once per hour. Otherwise be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b.) With the UHS water temperature > 77 °F, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours."

The proposed amendment revises TS LCO 3.7.11, by raising the maximum temperature limit of the UHS from 75 °F, as stated above, to 80 °F. Also TS 3.7.11 ACTIONS (a) and (b) that are stated above are replaced by new TS 3.7.11 ACTION: "With the UHS water temperature greater than 80 °F, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours."

The Surveillance Requirement (SR) 4.7.11b currently requires that the UHS be determined OPERABLE: "At least once per 6 hours by verifying the water temperature to be within limits when the water temperature exceeds 70°F." The proposed amendment changes 70°F to 75°F in SR 4.7.11.

The NRC staff performed a review of Millstone Power Station Unit 2 License Amendment Request (LAR) and the licensee's design inputs, assumptions and methodology, where appropriate, for adherence to regulatory requirements and guidelines. In order to allow a 5 °F rise in UHS temperature and still maintain operability, the equipment important to safety that is supported by the UHS must remain operable. The UHS supports SSC cooled by the SW system and the RBCCW system. The UHS directly supplies the SW system which in turn cools RBCCW heat exchangers, the EDG, and supports the Vital Switchgear Ventilation Systems that cool vital AC and DC switchgear rooms.

The RBCCW system in turn cools the containment air recirculation and cooling units, containment spray pump mechanical seal coolers, high and low pressure safety injection pump mechanical seal coolers, Engineering Safety Features (ESF) room air recirculation coolers, and the Spent Fuel Pool (SFP) heat exchangers. The SW system and the RBCCW system serve other loads which are non-safety and thus are not evaluated in this safety evaluation.

3.1 Emergency Diesel Generators

The EDGs are directly cooled by SW. The diesel generators are rated at 2750 KW and must be capable of running for a minimum of 30 days after a design basis accident. For each EDG, the SW system cools three heat exchangers, i.e. an air cooler, a lube oil cooler and a jacket water cooler. The three heat exchangers are in series on the SW side. The licensee stated in their LAR that the minimum required SW flow at 80 °F inlet temperature is 637 gallons per minute (gpm) for each EDG and that a minimum of 672 gpm was determined available by calculation. However, the NRC staff noted that the SW flow rate and temperature corresponding to the heat exchange values given by the licensee did not agree with vendor data sheets for the EDG heat exchangers.

Therefore, in an NRC letter dated July 18, 2013 [Reference 6], and email dated July 23, 2013 [Reference 7], the NRC staff requested additional information and justification for the licensee's statements that there is sufficient margin in the SW flow to the EDGs to support operation with the UHS at 80 °F, and to resolve the apparent disagreement with the vendor data sheets. The licensee responded in letters dated July 19, 2013 [Reference 3] and July 30, 2013 [Reference 4], that the vendor data sheets listed heat exchanger parameters for 118 percent design load (i.e. the 30 minute rating), as explained in DNC calculation 94-DES-1111-M2 was used to determine the EDG cooling parameters. This calculation determined that 507 gpm of SW was required for continuous EDG loading (2750 KW) with a SW temperature of 77 °F as compared to 700 gpm for 118 percent load with SW at 75 °F as specified on the vendor data sheet. The licensee provided calculation 12-328 (Attachment 9 of Reference 3), which used prevalent nuclear industry software, Proto-HX, to show that 637 gpm SW would be required to remove the EDG heat load at design rating (2750 KW) and SW at 80 °F. According to Reference 5, the licensee stated that the analytical models of Proto-HX are benchmarked against the actual system by verifying results of the model match results obtained in the field. In Reference 3 the licensee stated that the predicted SW flow of 672 gpm to the EDGs was determined where 637 gpm is required when SW temperature is 80 °F. The licensee used Proto-FLO, a widely used nuclear industry software to model the SW system to determine that the predicted flow of 672 gpm would be available to each EDG. The Proto-FLO models are benchmarked to actual flows obtained during flow testing. With pump degradation inputted as part of the model, predicted flows are below actual flows which add conservatism. The licensee reduced predicted flow rate by 10 percent before comparing it to the acceptance criteria. The SW flow rate tests performed during 2R20 and 2R21 confirmed that SW flow to each EDG was meeting minimum predicted flow rates. Based on the licensee's calculations and flow rate tests, the staff finds the licensee's analysis that a SW temperature of 80 °F is satisfactory for continuous loading of the EDGs.

3.2 Vital Switchgear Ventilation Systems

The SW system supports the Vital Switchgear Ventilation System which cools the vital AC switchgear rooms containing the west 480 volt, upper and lower 4160/6190 volt switchgear rooms, and the east and west vital DC volt switchgear rooms. The 4160/6190 vital AC switchgear rooms have a maximum room temperature limit of 122 °F. The west 480 volt switchgear room has a maximum room temperature limit of 104 °F. In order for UHS temperature of 80 °F to be acceptable, the SW must be capable of keeping the room temperatures at or below the above stated temperatures.

In the LAR dated May 3, 2013 [Reference 1], the licensee stated that their SW thermal-hydraulic flow analysis with SW/UHS temperature at 80 °F gives acceptable results provided the west 480 volt load center switchgear room cooler is cleaned at 18-month intervals. In a Request for Additional Information (RAI) email dated July 26, 2013 [Reference 8], the NRC staff asked the licensee specifically whether the room temperature limits specified above and in the FSAR would not be exceeded and whether the thermal performance of the associated cooling coils as specified on the vendor data sheets would be obtained with the proposed increase in UHS temperature. In the response dated August 1, 2013 [Reference 5], the licensee stated that Proto-HX models were developed for the vital AC switchgear room cooling coils (X-181, X-182, and X-183) and a Mathcad model was developed for the vital DC switchgear room coils (X-169A/B). The models are documented in calculations which determine the minimum SW flow rates to achieve the required heat transfer. To determine predicted SW flow, Proto-FLO model of the SW system benchmarked against testing flow data determine whether predicted flow is adequate to remove the required heat. The most recent flow testing data was used to assess flow model uncertainty. Predicted flows were reduced 10 percent for the RBCCW, EDG and west 480 volt cooling coil and by 15 percent for the other vital AC and DC switchgear room cooling coils. Comparisons of predicted flow and required flow follow.

The calculations show that 90 gpm is required for the west 480 volt switchgear room coils (X-181A and X-181B combined) with 80 °F UHS temperature, while the predicted flow is 145 gpm. The calculations show that 17 gpm is required for the upper 4160/6900 volt switchgear room coil (X-183) with an 80 °F UHS temperature, while the predicted flow is 23 gpm for X-183. The calculations show that 15 gpm is required for lower 4160/6900 volt switchgear room coil (X-182) with an 80 °F UHS temperature, while the predicted flow is 28 gpm for X-182. The calculations show that 26.9 gpm is required for the east and west vital DC, while the predicted delivered flow is 30 gpm.

Based on the licensee's calculations and flow rate tests, the staff finds the licensee's analysis for a SW/UHS temperature of 80 °F is satisfactory because there are sufficient margins in flow rates for supporting the Vital Switchgear Ventilation Systems and meeting the maximum room temperature requirements specified in FSAR Section 9.9.16.

3.3 Reactor Building Component Cooling Water System

The RBCCW system has two redundant trains each with a RBCCW heat exchanger. The maximum load on a RBCCW heat exchanger is 204 MBTU/hr during a design basis event. As stated above, the RBCCW heat exchangers are directly cooled by the SW and the UHS. The RBCCW system in turn cools containment air recirculation and cooling units, containment spray pump mechanical seal coolers, high and low pressure safety injection pump mechanical seal coolers, ESF room air recirculation coolers, and the SFP heat exchangers.

In the LAR the licensee stated that the RBCCW supply temperature of 85 °F in Modes 1,2,3 will be a design requirement and will be obtained using modified operating procedures to minimize RBCCW heat loads and maximize SW flow to the RBCCW heat exchangers whenever UHS temperature exceeds 75 °F. The licensee stated that the RBCCW system will be able to perform its intended functions with an UHS temperature of 80 °F. The licensee did not sufficiently justify the proposed increase in the temperature limit of the UHS. Therefore, the NRC staff submitted RAIs in a letter dated July 18, 2013 [Reference 6] and emails dated July 23 and 26, 2013 [Reference 7 and 8] asking the licensee to provide the quantitative effects and acceptability of the increase in RBCCW cooling water on all safety related loads cooled by RBCCW. The staff asked the licensee to discuss the ability of the RBCCW system to meet the design requirements ensuring that the cooling requirements for the ESF room air recirculation coolers, containment air recirculation and cooling units, high pressure and low pressure and containment spray pump mechanical seals and SFP are achieved.

In responses dated July 19, 2013 [Reference 3], July 30, 2013 [Reference 4] and August 1, 2013 [Reference 5], the licensee explained that in performing the design basis accident (DBA) analyses, it analyzed for both maximum effect on the containment and maximum effect on the RBCCW heat exchanger outlet temperatures considering the maximum heat input to RBCCW in both the injection and recirculation modes of LOCA mitigation as well as the MSLB analysis. The licensee stated that with a UHS temperature of 80 °F, the RBCCW cooled ESF room air recirculation coolers and still maintained the ESF room maximum ambient temperature below 145 °F and was within the Equipment Environmental Qualification (EEQ) limits. Using the Dominion Gothic methodology for containment response following a LOCA and MSLB with a UHS temperature of 80 °F, the licensee determined that safety related equipment in containment will be within their EEQ limits. Analyses also showed that the increase in UHS to 80 °F was within acceptance criteria for seal performance of the mechanical seals of the safety injection and containment spray pumps because they within their EEQ limits. The staff finds the licensee's evaluation acceptable because the licensee used NRC approved methodology and the RBCCW cooled ESF room air recirculation coolers and still maintained the ESF room maximum ambient temperature within the Equipment Environmental Qualification (EEQ) limits.

For SFP cooling with the UHS at 80 °F and the limiting RBCCW temperature profile, the licensee stated that the maximum SFP temperature has been analytically determined to be below 200 °F. This is based on an SFP maximum initial temperature of 150 °F and heat up due to suspension of cooling for four hours when RBCCW is isolated to the SFP at the start of a LOCA and manually restored fours after the LOCA. The analytically determined limit of below 200 °F is less than 212 °F which is the assumed temperature during accident conditions as specified in FSAR Section 5.4.3.1.3, "Thermal Loads." Therefore, the staff finds the increase in UHS limit to 80 °F acceptable for SFP cooling.

3.4 Containment

3.4.1 Mass and Energy (M&E) Release Data

The licensee stated that the revised containment analysis incorporated corrected M&E release data for use as input to the LOCA and Main Steamline Break (MSLB) containment response analysis. The NRC staff requested in a RAI that the licensee describe the basis and reasons for the revision in the M&E release data. In response to SCVB-RAI-1 (Reference 9), the licensee stated that Westinghouse identified that the computer code CEFLASH-4A and erroneously under predicted the M&E release data for the LOCA and MSLB blowdown phase. Westinghouse provided the revised data for the reactor coolant system hot leg, pump suction and pump discharge legs double ended guillotine large breaks for the LOCA blowdown and reflood phases. Westinghouse also provided corrected M&E release data using the NRC approved CE methodology using SGN-III computer program (FSAR Section 14.8.2.1.3) for MSLB initiated at 0-, 25-, 50-, 75-, and 102-percent power levels assuming several cases of a single failure. The staff finds the licensee's evaluation acceptable because the licensee used NRC approved methodology and the corrected M&E data.

3.4.2 Containment Response Analysis

The current containment analysis was performed by the licensee using the maximum value of UHS temperature of 77 °F (Reference 11).

3.4.2.1 LOCA Pressure and Temperature Response

The licensee re-analyzed the LOCA containment response using the GOTHIC methodology approved, for use by DNC only, by NRC in Reference 10, with the corrected Westinghouse M&E data for the LOCA blowdown and reflood phases. For evaluating the LOCA post-reflood and long term phases, the licensee used NRC approved GOTHIC methodology both for determining M&E release data and the containment response (Reference 10). The licensee's evaluated cases with and without concurrent loss of offsite power while assuming a UHS temperature of 80 °F. The analysis resulted in a peak containment pressure of 52.5 psig, and a peak containment gas temperature of 279.2 °F. The calculated peak containment pressure and gas temperature are bounded by the containment design pressure 54 psig and the containment liner and structural design temperature 289 °F. The effect of the increased UHS temperature from the current 77 °F to the proposed TS limit of 80 °F increases the calculated containment peak pressure by 0.03 psi, because at the time of peak pressure, the safety related containment air recirculation and cooling system is in operation. The licensee stated calculation of the peak containment liner temperature is not required because the peak containment gas temperature for EQ is less than the containment structural and liner design temperature. The staff finds the licensee's evaluation acceptable because the licensee used NRC approved methodology and the calculated peak containment pressure and gas temperature are bounded by the containment design pressure and the containment liner and structural design temperature.

3.4.2.2 MSLB Pressure and Temperature Response

The licensee re-analyzed the MSLB containment response using the NRC approved GOTHIC methodology (Reference 10), with the corrected Westinghouse M&E data, and assuming a UHS temperature of 80 °F. The analysis resulted in a peak containment pressure of 53.8 psig and a peak containment gas temperature of 360.9 °F. The licensee stated that the effect of the increased UHS temperature from the current 77 °F to the proposed TS limit of 80 °F increases the calculated containment peak pressure by 0.1 psi, and the predicted peak containment gas temperature increases by less than 0.1 °F. Since the calculated peak drywell gas temperature is greater than the containment liner and structural design temperature, the licensee determined the liner temperature response using the methodology approved by NRC in Reference 10. The calculated peak containment liner temperature was 259.7 °F, which is less than its design temperature of 289 °F. The effect of the increased UHS temperature from the current 77 °F to the proposed TS limit of 80 °F increases the calculated containment peak liner temperature by less than 0.1 °F. The staff finds the licensee's evaluation acceptable because the licensee used NRC approved methodology and the calculated peak containment liner temperature was less than its design temperature.

3.4.2.3 Net Positive Suction Head Analysis

The NRC staff requested in a RAI that the licensee describe the impact of increasing the UHS temperature to the proposed TS limit of 80 °F on the post-accident maximum sump water temperature and the available Net Positive Suction Head (NPSH) for the Emergency Core Cooling System (ECCS) and containment spray pumps following the Sump Recirculation Actuation Signal (SRAS). The staff also requested to describe if the NPSH analysis complies with Regulatory Guide (RG) 1.1 (Safety Guide 1). In response to SCVB-RAI-2 (Reference 9), the licensee stated that the analysis was based on the NRC approved methodology in Reference 10 and a UHS temperature of 80 °F which resulted in a peak sump temperature of 233.5 °F following the SRAS. The licensee stated that the impact of the increased UHS temperature to 80 °F increased the peak value of the sump water temperature following SRAS by 1.2 °F. Following RG 1.1, assuming a sump water temperature of 212 °F at 14.7 psia pressure, the licensee conservatively calculated the minimum available NPSH without crediting the Containment Accident Pressure (CAP) for the ECCS and containment spray pumps during the sump recirculation mode. The licensee stated that the conservatively calculated available NPSH following RG 1.1 after SRAS exceeded the required NPSH for the ECCS and containment spray pumps. The staff finds the licensee's evaluation acceptable because the licensee followed GR 1.1 and conservatively calculated the minimum available NPSH without crediting the CAP for the ECCS and containment spray pumps during the sump recirculation mode.

3.4.2.4 Minimum Containment Pressure Analysis for ECCS Performance

NUREG-0800, Standard Review Plan (SRP) 6.2.1.5 describes the minimum containment pressure analysis for ECCS performance capability. RG 1.157, Section 3.12.1 provides guidance for calculating the containment pressure response used for evaluating cooling effectiveness during the post-blowdown phase of a LOCA. The RG states that the containment pressure should be calculated by including the effects of containment heat sinks and operation of all pressure reducing equipment assumed to be available. The NRC staff, requested in a RAI, that the licensee describe the impact of increasing UHS temperature from 75 °F to 80 °F and the changes in M&E release on the minimum containment pressure analyses for ECCS performance as described in the above NRC documents. In response to SCVB-RAI-3 (Reference 9), the licensee stated that the M&E release does not impact the minimum containment pressure analyses for ECCS performance because the M&E input to the 10 CFR 50.46 ECCS performance analysis is developed by AREVA, the current fuel supplier, using the

10 CFR 50.46 large break LOCA analysis methodology. The licensee stated that the AREVA methodology was not affected by the Westinghouse LOCA M&E release error. The licensee stated that conservative assumptions were used in the AREVA methodology which maximizes containment heat removal for calculating the minimum containment pressure response following a large break LOCA. The NRC staff finds the licensee response acceptable.

MPS2 uses the design pressure of 54 psig as the value of P_a in the TS. Since the calculated containment peak pressure is less than the value of P_a, the NRC staff concludes that the containment leakage testing program is not adversely affected.

3.5 Revisions to the UFSAR Concurrent with the License Amendment

The licensee's thermal-hydraulic analysis of the SW demonstrated acceptable results with the following restrictions:

- The maximum allowable Differential Pressure (DP) for the 'A' RBCCW heat exchanger has been reduced from 10-psid to 8-psid.
- RBCCW heat exchangers must be cleaned at a 3-month interval.
- Switchgear room coolers, X-181 A/B, must be cleaned at an 18-month interval.
- For SW temperatures above 75 °F, it may be necessary to also open the RBCCW winter temperature control valves (2-SW-245, 2-SW-246, and 2-SW-247) to maximize flow to the RBCCW heat exchangers to support normal operation.
- An RBCCW supply temperature of 85 °F in Modes 1, 2, and 3 will be included in the UFSAR as a design requirement for the RBCCW system and the design basis reason with respect to Generic letter 96-06 shall be stated.

Implementation of the amendment to TS 3/4.7.11 shall also include a revision to the Updated Final Safety Analysis Report (UFSAR) to include the above stated restrictions. The NRC staff has reviewed the licensee's analysis provided in References 1 through 5 and finds that raising the temperature limit of LCO 3/4.7.11 to 80 °F while remaining in Modes 1, 2, or 3 is acceptable. Based on these findings, the NRC staff concludes that there is reasonable assurance that the requirements of 10 CFR Part 50 Appendix A Criterion 44 and the UFSAR will continue to be met.

3.6 Program Reviews

The licensee submitted program information for Fire Protection/Appendix R, In-Service Inspection, and In-Service Testing, and the NRC staff determined based on the information submitted that the impact on these programs was bounded by the SR information and is therefore acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (78 FR 51225). 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.

7.0 REFERENCES

1. Dominion Letter 13-227, dated May 3, 2013, Millstone Power Station Unit 2 LAR for Changes to TS 3/4.7.11 UHS (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13133A033).
2. Dominion Letter 13-227A, dated June 27, 2013, Millstone Power Station Unit 2 Supplement to LAR for Changes to TS 3/4.7.11 UHS (ADAMS Accession No. ML13198A271).
3. Dominion Letter 13-419, dated July 19, 2013, Millstone Power Station Unit 2 Response to Request for Additional Information Regarding LAR for Changes to TS 3/4.7.11 UHS. (ADAMS Accession No. ML13204A035).
4. Dominion Letter 13-438, dated July 30, 2013, Millstone Power Station Unit 2 Response to Request for Additional Information Regarding LAR for Changes to TS 3/4.7.11 UHS. (ADAMS Accession No. ML13213A024).
5. Dominion Letter 13-450, dated August 1, 2013, Millstone Power Station Unit 2 Response to Request for Additional Information Regarding LAR for Changes to TS 3/4.7.11 UHS. (ADAMS Accession No. ML13219A109).
6. NRC Letter to Mr. David A. Heacock, Dominion Nuclear Connecticut, Inc. dated July 18, 2013. (ADAMS Accession No. ML13197A401).
7. NRC Email to William Bartron dated July 23, 2013 (ADAMS Accession No. ML13206A021).
8. NRC Email to William Bartron dated July 26, 2013 (ADAMS Accession No. ML13213A067).
9. DNC Letter to NRC dated October 2, 2013, "Dominion Nuclear Connecticut. Inc. Millstone Power Station Unit 2 Response to Request for Additional Information Regarding License Amendment Request for Changes to Technical Specification 3/4.7.11, "Ultimate Heat Sink" (ADAMS Accession No. ML13281A809).

10. Letter from NRC to DNC dated August 30, 2006, "Kewaunee Power Station (Kewaunee), Millstone Power Station, Unit Nos. 2 And 3 (Millstone 2 And 3), North Anna Power Station, Unit Nos. 1 And 2 (North Anna 1 And 2), and Surry Power Station, Unit Nos. 1 and 2 (Surry 1 And 2) - Approval of Dominion's Topical Report DOM-NAF-3, "GOTHIC Methodology for Analyzing the Response to Postulated Pipe Ruptures Inside Containment" (TAC NOS. MC8831, MC8832, MC8833, MC8834, MC8835, and MC8836)", (ADAMS Accession No. ML062420511).
11. Letter from NRC to DNC dated May 31, 2001, "Millstone Nuclear Power Station, Unit No.2 - Issuance of Amendment Re: Ultimate Heat Sink Action Requirements (TAC No. MB0867)", (ADAMS Accession No. ML011410153).

Principal Contributors: G. Purciarello
A. Sallman

Date: April 18, 2014

April 18, 2014

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: REVISE TECHNICAL SPECIFICATION 3/4.7.11 ULTIMATE HEAT SINK (TAC NO. MF1779)

Dear Mr. Heacock:

The Commission has issued the enclosed Amendment No. 318 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2, in response to your application dated May 3, 2013, as supplemented by letters dated June 27, 2013, July 19, July 30, August 1, and October 2. 2013.

The amendment would revise Technical Specification (TS) 3/4.7.11, "Ultimate Heat Sink", to increase the current ultimate heat sink water temperature limit from 75 °F to 80 °F and change the TS Action to state, "With the ultimate heat sink water temperature greater than 80 °F, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours."

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/
James Kim, Project Manager
Plant Licensing Branch 1-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 318 to DPR-65
2. Safety Evaluation

cc w/encls: Distribution via Listserv

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R. Bellamy, RI

*See memo dated August 20, 2013
**See memo dated February 4, 2014

Accession No.: ML14037A408

OFFICE	LPL1-1/PM	LPL1-1/LA	SBPB/BC*	SCVB/BC**	EPNB/BC
NAME	JKim	KGoldstein	GCasto	RDennig	TLupold
DATE	2/12/14	2/12/14	8/20/13	2/4/14	2/12/14
OFFICE	SRXBB/BC	STSB/BC	AFPB/BC	OGC/NLO with comments	LPL1-1/BC
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