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

February 7, 2017

Mr. Daniel G. Stoddard
Senior Vice 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: TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.1.3.1.2 FOR CONTROL ELEMENT ASSEMBLY 39 (CAC NO. MF8935)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 333 to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2), in response to your application dated December 14, 2016.

The amendment revises the MPS2 Technical Specifications (TSs) for MPS2 by adding a note to TS Surveillance Requirement (SR) 4.1.3.1.2, control element assembly (CEA) freedom of movement surveillance, such that CEA 39 may be excluded from the remaining quarterly performance of the SR in Cycle 24. The amendment allows the licensee to delay exercising CEA 39 until after repairs can be made during the next outage.

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, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
1. Amendment No. 333 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. 333
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 December 14, 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. 333 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 30 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 Facility Operating License
and Technical Specifications

Date of Issuance: February 7, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 333

MILLSTONE POWER STATION, UNIT NO. 2

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 a marginal line indicating the area 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 a marginal line indicating the area of change.

Remove	Insert
3/4 1-21	3/4 1-21

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. 333 are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

ACTION: (Continued):

C. CEA Deviation Circuit inoperable.	C.1 Verify the indicated position of each CEA to be within 10 steps of all other CEAs in its group within 1 hour and every 4 hours thereafter or otherwise be in MODE 3 within the next 6 hours.
D. One or more CEAs untrippable. OR Two or more CEAs misaligned by ≥ 20 steps.	D.1 Be in MODE 3 within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 Verify the indicated position of each CEA to be within 10 steps of all other CEAs in its group at the frequency specified in the Surveillance Frequency Control Program AND within 1 hour following any CEA movement larger than 10 steps.

------------------ NOTE ------------------

SR 4.1.3.1.2 is not required to be performed for CEA 39 for the remainder of Cycle 24.

4.1.3.1.2 Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core 10 steps in either direction at the frequency specified in the Surveillance Frequency Control Program.

4.1.3.1.3 Verify the CEA Deviation Circuit is OPERABLE at the frequency specified in the Surveillance Frequency Control Program by a functional test of the CEA group Deviation Circuit which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 10 steps (indicated position).

4.1.3.1.4 Verify the CEA Motion Inhibit is OPERABLE by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents regulating CEAs from being inserted beyond the Transient Insertion Limits specified in the CORE OPERATING LIMITS REPORT:

a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 31 days, and

b. At the frequency specified in the Surveillance Frequency Control Program.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 333

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 December 14, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16354A424), Dominion Nuclear Connecticut, Inc. (Dominion, the licensee) requested a change to the Technical Specifications (TSs) for the Millstone Power Station, Unit No. 2 (MPS2). The proposed amendment would add a note to TS Surveillance Requirement (SR) 4.1.3.1.2, control element assembly (CEA) freedom of movement surveillance, such that CEA 39 may be excluded from the remaining quarterly performance of the SR in Cycle 24. The proposed amendment would allow the licensee to delay exercising CEA 39 until after repairs can be made during the next outage.

2.0 REGULATORY EVALUATION

2.1 Regulatory Discussion

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation, (3) SRs, (4) design features, and (5) administrative controls.

SRs in 10 CFR 50.36 are 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 limiting conditions for operation will be met.

The regulations in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," include the following General Design Criteria (GDC) applicable to CEA design requirements:

- GDC 26, "Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of

the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

- GDC 27, "Combined reactivity control systems capability." The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

- GDC 28, "Reactivity limits." The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

- GDC 29, "Protection against anticipated operational occurrences." The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences."

2.2 System Description

The CEAs in MPS2 are used for reactivity control. The CEAs are divided into nine control groups: two shutdown groups and seven regulating groups, which are designated Groups 1 through 7 with CEA 39 in Regulating Group 7. The shutdown groups ensure that sufficient negative reactivity is available to support a reactor trip or normal shutdown. Insertion limits on Regulating Groups 1 through 7 are established, and the CEA positions are monitored and controlled during initial criticality and power operation to ensure that the power distribution and reactivity limits are preserved. During reactor startup, regulating CEA groups are withdrawn and operated in a predetermined sequence with a predetermined amount of position overlap. During power operations, the CEAs are normally fully withdrawn, except to complete SR 4.1.3.1.2 to demonstrate CEA freedom of movement, or as required for plant maneuvers or to respond to certain abnormal plant conditions.

The control element drive mechanism (CEDM) is an electromechanical device that converts electrical energy into mechanical motion. The CEDM coils provide the magnetic flux that

operates the mechanical parts of the drive within the pressure housing. Motion of these parts engage, lift, and release the latching devices, which translate the motion of the gripper assembly to the CEDM drive shaft.

Each CEDM has a lift coil, upper gripper coil (UGC), pull down coil, load transfer coil, and lower gripper coil, which comprise the coil stack assembly. These coil stack assemblies produce magnetic fields, which control the magnetic jack assemblies, causing the jacks to engage, hold, move, or release the CEAs. Each CEDM is capable of withdrawing, inserting, holding, or tripping (releasing) its CEA from any point within its 137-inch stroke. Under normal operating conditions, a CEA is not in motion, and the CEA is held in place by the UGC, which is continuously energized.

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Change

Current TS SR 4.1.3.1.2 states:

> Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core 10 steps in either direction at the frequency specified in the Surveillance Frequency Control Program.

Revised SR 4.1.3.1.2 would state:

> --NOTE--
> SR 4.1.3.1.2 is not required to be performed for CEA 39 for the remainder of Cycle 24
> --
>
> Verify CEA freedom of movement (trippability) by moving each individual CEA that is not fully inserted into the reactor core 10 steps in either direction at the frequency specified in the Surveillance Frequency Control Program

3.2 U.S. Nuclear Regulatory Commission (NRC or the Commission) Staff Evaluation

The licensee requested to revise the MPS2 TS. The proposed revision would eliminate exercising CEA 39 for SR 4.1.3.1.2 for the remainder of operating Cycle 24, currently scheduled to end on April 1, 2017. This would effectively allow CEA 39 to not be exercised during the final remaining quarterly performance of SR 4.1.3.1.2.

Potential Causes of Failure

During its review of data from the most recent CEA freedom of movement surveillance performed on October 27, 2016, the licensee discovered that the UGC low current ripple for CEDM 39 was found to have noticeably increased. The licensee has not determined a definitive cause; however, three potential causes have been assessed, including power switch components, a capacitor, and the UGC. The licensee stated that the most likely cause, based on industry experience and consultation with Westinghouse, is degradation of the UGC. The licensee also states that the majority of industry failures are related to coil insulation breakdown

and described how this problem becomes worse over time while being energized. Therefore, as a precaution, the licensee has moved CEA 39 from the UGC to the lower gripper coil. Upon transfer to the lower gripper coil, the UGC was de-energized and remains de-energized, unless needed for urgent plant response.

The licensee states that CEA 39 remains capable of meeting the requirements of TS 3.1.3.1, as it remains trippable and within 10 steps of other CEAs in its group. The licensee states that with the UGC de-energized and the lower gripper coil energized, CEA 39 will remain aligned and withdrawn with its group as required by TS 3.1.3.1.

The purpose of SR 4.1.3.1.2 is to verify that the CEAs are free to move (i.e., trippable). This surveillance is accomplished with the control room operator moving an individual CEA, 10 steps in either direction. Successful movement of the CEA confirms no mechanical binding exists. The licensee states that results from the last quarterly performance of SR 4.1.3.1.2 on October 27, 2016, showed freedom of movement.

If required to perform the last quarterly freedom of movement surveillance in Cycle 24 on CEA 39, the UGC would need to be re-energized to move the CEA. If the UGC fails, the CEA could drop into the core, resulting in a reactivity transient and subsequent power reduction, and a plant shutdown if the CEA is unrecoverable.

The licensee states that the control element drive system is designed to ensure that electrical problems will not prevent insertion of a CEA into the core when the reactor trip circuit breakers (RTCBs) are opened. The RTCBs open upon an automatic or manual reactor trip signal, removing all power from both control and holding circuitry. All coils on each CEDM subsequently de-energize, resulting in all CEAs inserting into the core. This design is fail-safe in that a loss of power, regardless of whether a reactor trip signal has been generated, will result in the CEAs inserting into the core. The licensee states that each CEA is magnetically coupled to its associated gripper coil, and there is no physical (mechanical) coupling between the CEDM circuit/coil and the associated gripper. Any postulated failure mechanism that could prevent rod insertion (such as mechanical binding of the CEA itself) is not influenced or impacted by coil failure or control or holding circuitry failures. The licensee found no postulated failure mechanisms where the coil or associated circuitry could physically prevent rod insertion once the RTCBs have opened.

The NRC staff also considered plant operating experience as part of its review of the licensee's proposed license amendment. In the fall of 2016, CEA 39 successfully passed its last freedom of movement surveillance, and while not stated in the LAR, there is no operating history of CEAs getting stuck. A licensee event report search of all Combustion Engineering Pressurized-Water Reactors (CE PWRs) showed only a single case of a CEA stuck due to binding (Calvert Cliffs Nuclear Power Plant, Unit No. 1, letter dated June 5, 2006, ADAMS Accession No. ML061580322) with the most likely cause determined to be the presence of debris in or on top of a fuel assembly guide tube. The NRC staff did not identify any other operating experiences to suggest that there is a failure mode that would prevent the CEA from inserting.

Based on the above, the NRC staff finds that the licensee has adequately assessed the failure modes and possible causes of the identified failure of the CEDM for CEA 39. Given the design of the system, plant operating experience, as well as the results of the last quarterly performance of SR 4.1.3.1.2, the NRC staff finds it is likely that CEA 39 will insert into the core if a reactor trip signal is generated.

Nuclear Safety Risk Insights

The licensee stated that the probabilistic risk assessment model presumes insertion of one-half or more of the control rods is needed to achieve hot zero power (reference: NUREG/CR-5500, Volume 10 (ADAMS Package Accession No. ML022060194)). Therefore, a common cause failure of roughly 35 CEAs is necessary to fail the reactivity control function. Since only one of the 73 CEAs will not be exercised during the last remaining quarterly surveillance prior to the next MPS2 refueling outage, the impact on the reactivity control function, and thus, core damage frequency and large early release frequency, is negligible. Given that the remaining 72 CEAs will be tested, the NRC staff finds that this will provide confidence that a common cause condition does not exist.

Reactivity Impact

The current Final Safety Analysis Report (FSAR) Chapter 14 safety analysis assumes the most reactive CEA fails to insert on a reactor trip signal. Given that CEA 39 is still expected to insert following receipt of a reactor trip signal, the NRC staff finds that the existing FSAR Chapter 14 safety analysis remains bounding. However, the potential impact on shutdown margin (SDM) of the hypothetical failure of the highest reactivity combination of CEA 39 and a second CEA failing to insert on reactor trip was analyzed by the licensee. TS 3.1.1.1, "SHUTDOWN MARGIN - (SDM)," specifies that SDM shall be within the limit specified in the CORE OPERATING LIMITS REPORT (COLR). The licensee stated that the Cycle 24 COLR SDM operating limit is 3.6 percent ΔK/K [delta-K/K where K is k-effective, the effective multiplication factor) in Modes 3 through 5. The licensee performed a parametric study from a Cycle 24 core exposure of 11,500 megawatt days per metric-ton uranium (MWD/MTU) to the end of Cycle 24 to determine the minimum SDM that would exist following a reactor trip, assuming the highest reactivity worth combination of CEA 39 and a second CEA fails to insert. The calculated minimum SDM for this scenario is 3.7 percent ΔK/K, which is above the 3.6 percent ΔK/K SDM requirement in the COLR. The calculated SDM value bounds operation for the remainder of MPS2 Cycle 24 operation. The licensee concluded that the SDM in excess of the COLR limit of 3.6 percent ΔK/K exists for the remainder of MPS2 Cycle 24 operation, even if CEA 39 fails to insert into the core during a reactor trip. The licensee stated that calculations were performed using NRC-approved methodologies used to generate the COLR and to perform the TS surveillances. The NRC staff finds that the licensee's calculations assured adequate existing SDM, assuming both CEA 39 and the single highest reactivity worth CEA failed to insert. The models and methods utilized for the SDM calculation are also used to perform the TS surveillances. Based on the above, the NRC staff concludes that the licensee has adequately calculated the SDM available if CEA 39 fails to trip and that the SDM available in that case would be greater than the SDM required by the COLR.

Administrative Controls and Compensatory Actions

The licensee described administrative controls and compensatory actions that have been established to minimize the frequency of energizing the CEA 39 UGC and potentially causing further degradation during the remainder of Cycle 24 operation. The administrative controls ensure operator movement of CEA 39 is not performed without knowledge of the current condition of the CEA 39 UGC degradation. The licensee stated the following:

- MPS2 operations has issued a standing order to limit, but not prohibit, the use of CEAs. Guidance has been provided for changing power levels while limiting Regulating Group 7 motion for the remainder of Cycle 24 operation. Specifically, the use of CEDM 39, and therefore, Regulating Group 7, will be limited to a plant response as directed by abnormal operating procedures or planned power reductions in excess of 15 percent.

- MPS2 operations has placed a tag on the CEA motion control switch identifying that CEA 39 is on the lower gripper coil and the potential for actuating the automatic CEA timer module (ACTM) trouble alarm upon motion of Regulating Group 7.

- Reactivity plans have been developed by Reactor Engineering for downpowers to 95, 90, and 85 percent power without the use of CEAs for axial shape index (ASI) control. The reactivity plans utilize a combination of RCS boration/dilution and ramp rate control in order to minimize axial xenon perturbations and maintain ASI within its COLR limits.

The NRC staff determined that the above administrative controls and compensatory actions are appropriate and consistent with a degraded UGC, and the proposed TS change will help prevent accidentally energizing the UGC.

3.3 NRC Staff Conclusion

The proposed change to TS SR 4.1.3.1.2 excludes CEA 39 from the last remaining quarterly performance during Cycle 24. Based on its review, the NRC staff concludes that the licensee has provided adequate information for its justification of excluding CEA 39 from SR performance for the remainder of Cycle 24. Specifically, the licensee has (1) adequately assessed the failure modes and possible causes of the identified failure of the CEDM for CEA 39, (2) established that CEA 39 will insert into the core with reasonable certainty if a reactor trip signal is generated, (3) demonstrated there is confidence that a common cause condition does not exist, and (4) shown there is adequate SDM, even if CEA 39 fails to insert into the core. Additionally, the NRC staff concludes that elimination of the exercising of CEA 39 will reduce the likelihood of a potential rod drop and plant transient. Therefore, the NRC staff concludes that the elimination of exercising CEA 39 for SR 4.1.3.1.2 for the remainder of Cycle 24 is acceptable and that there is reasonable assurance that the requirements of GDC 26, 27, 28, 29, and 10 CFR 50.36 will continue to be met.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC's regulations in 10 CFR 50.92, "Issuance of amendment, state that the NRC may make a final determination that a license amendment involves no significant hazards consideration (NSHC) if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As provided below, the NRC has made a final determination concerning whether the license amendment involves a significant hazards consideration.

An evaluation of the issue of no significant hazards consideration is presented below:

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

   Response: No.

   The proposed amendment would exclude CEA 39 from SR 4.1.3.1.2 for the remainder of MPS2 Cycle 24 operation. The function of CEA 39 is to provide negative reactivity addition into the core upon receipt of a signal from the Reactor Protection System (RPS). CEA 39 was demonstrated to be moveable and trippable during the last performance of SR 4.1.3.1.2. Since the functionality of CEA 39 has not been affected, the assumptions and conclusions of the Final Safety Analysis Report (FSAR) Chapter 14, Safety Analysis, are not affected by this license amendment request.

   The misoperation of a CEA, which includes a CEA drop event, has been evaluated in the MPS2 FSAR and found acceptable. The proposed change would minimize the potential for inadvertent insertion of CEA 39 into the core by eliminating the requirement to place the CEA on the UGC to perform freedom of movement testing. The proposed change does not significantly increase the probability of a failure of a CEA to insert into the core on a reactor trip or the probability of an inadvertent CEA drop into the core at power.

   No modifications are proposed to the RPS or associated Control Element Drive Mechanism (CEDM) system logic.

   Based on the above, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

The proposed amendment would exclude CEA 39 from SR 4.1.3.1.2 for the remainder of MPS2 Cycle 24 operation. CEA 39 was demonstrated to be moveable and trippable during the last performance of SR 4.1.3.1.2; therefore, the functionality of CEA 39 has not been affected. The proposed change will not introduce any new design changes or systems that can prevent the CEA from performing its specified safety function to insert on a reactor trip. The current MPS2 FSAR safety analysis considers the drop of a CEA into the core as an initiating event. This change does not alter assumptions made in the FSAR Chapter 14 safety analysis.

Based on the above, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed amendment would exclude CEA 39 from SR 4.1.3.1.2 for the remainder of MPS2 Cycle 24 operation. SR 4.1.3.1.2 is intended to verify freedom of movement of CEAs (i.e., trippable). CEA 39 was demonstrated to be moveable and trippable during the last performance of SR 4.1.3.1.2. The physical and electrical design of the CEAs, and past operating experience, provides high confidence that CEAs remain trippable whether or not exercised during each SR interval. Eliminating further exercise of CEA 39 for the remainder of MPS2 Cycle 24 operation does not directly relate to the potential for CEA binding to occur. The current MPS2 FSAR safety analysis is unaffected by this license amendment request and there is no reduction in the margin of safety. There is no known failure mechanism (e.g., crud deposition) that would preclude the CEA from inserting into core on a valid trip signal or loss of power.

Based on the above, the proposed amendment does not involve a significant reduction in the margin of safety.

Based on the above evaluation, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that NSHC is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

5.0 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 provided comments via e-mail on January 19, 2017, which were considered in the NRC staff's review. The NRC staff's response to the State official's comments are shown below and can also be viewed in ADAMS at Accession No. ML17031A188.

State Official Comment

1. Millstone 2 surveillance frequencies, including SR 4.1.3.1.2 are controlled by TS Surveillance Control Program. Is there a reason that this change, a de facto surveillance frequency change, is not being evaluated under this program?

NRC Staff Response

The license amendment request (LAR) is for a one-time exclusion for a single control element assembly (CEA) and not a proposed change to the CEA surveillance frequency in the Surveillance Control Program. Once the actual problem (which the licensee suspects is the control element drive mechanism upper gripper coil) is identified and repaired during the upcoming outage, the existing surveillance frequency will still be applied to all CEAs. Note that the surveillances on the other 72 CEAs are still being performed per the Surveillance Control Program.

State Official Comment

2. Dominion's NSHC response to question 1 states, "CEA 39 was demonstrated to be moveable and trippable during the last performance of SR 4.1.3.1.2. Since the functionality of CEA 39 has not been affected..." The purpose of SR 4.1.3.1.2 is to determine if there are any conditions that have resulted in mechanical binding of the CEA. Dominion has not (see 1) determined that the surveillance interval can be extended; therefore, it is not clear to me how they can justify that the rod remains operable without performing the required surveillance within the specified surveillance interval. (TS 4.0.1) What is the basis for assuming the rod remains operable for a period beyond the surveillance interval? Does Dominion have rod failure data? How have they justified that end of cycle failure mechanisms such as thermally induced twisting/bowing are not present? Is there predictive data that indicates other signs of mechanical bowing? In short, the assessment provided seems to focus on why the UGC condition does not affect trippability but does not address basis for assuming that the CEA is not affected by other unrelated mechanisms beyond the analyzed surveillance interval.

NRC Staff Response

The NRC staff agrees with the comment that the LAR focuses on why problems with the upper gripper coil would not affect trippability and does not discuss other unrelated mechanisms. In the fall of 2016, CEA 39 did successfully pass its last freedom of movement surveillance, and while not stated in the LAR, there is no history of CEAs getting stuck. A Licensee Event Report search of all Combustion Engineering Pressurized-Water Reactors (CE PWRs) showed only a single case of a CEA stuck due to binding (Calvert Cliffs, Unit 1, April 8, 2006; ADAMS Accession No. ML061580322) with the most likely cause determined to be the presence of debris. The NRC staff did not identify any operating experience documents to suggest that there is a failure mode that would prevent the CEA from inserting.

While the licensee expects this CEA to be trippable for the remainder of Cycle 24 (approximately 49 days past the extended surveillance requirement date), it also addressed the hypothetical failure of the highest reactivity combination of CEA 39 and a second CEA failing to insert on reactor trip. This analysis showed that the shutdown margin (SDM) available if CEA 39 fails to trip would be greater than the required SDM as specified in the MPS2 Core Operating Limits Report (COLR). In addition, in the Probabilistic Risk Assessment models (used for core damage frequency (CDF) and large early release frequency (LERF)), success is defined for all PWRs as the insertion of one-half or more of the control rods into the core in a roughly checkerboard pattern. Based on this definition, one CEA out of 73 total CEAs failing to insert is considered negligible (for the purposes of CDF and LERF calculations). The NRC staff determination is that exclusion of a single CEA from the freedom of movement surveillance (thus rendering the CEA inoperable) does not pose a safety concern.

State Official Comment

3. How is the UGC being maintained de-energized? Does this method require any actions outside the control room to re-energize in order to move rods or can the LGC [lower gripper coil] perform this function alone? If so, does this affect any TCOAs [time critical operator actions] for an ATWS [anticipated transient without scram] or other events requiring rapid manual insertion of control rods? I did not see this addressed under Administrative Controls.

NRC Staff Response

The lower gripper coil alone cannot move the rod (including performance of the surveillance). The LAR states that MPS2 Operations has issued a standing order to limit but not prohibit the use of CEAs. It states that a tag has been placed on the CEA motion control switch identifying that (1) CEA 39 is on the lower gripper coil and (2) the potential for actuating the Automatic CEA Timer Module (ACTM) Trouble alarm upon motion of Regulating Group 7. The NRC staff also understands from the licensee that the UGC was de-energized using the ACTM computer and is being verified de-energized twice a shift. To move

the rod requires no action outside the control room; the UGC will automatically re-energize if the rod is called on to move, provided the system does not detect an error. In the case of events requiring rapid manual insertion of control rods, the system is designed so that loss of power (by opening the reactor trip circuit breakers) to the coils (upper, lift, lower, etc.) results in the rod dropping into the core by gravity.

The NRC staff finds that the concerns identified by the State official do not impact the staff's safety conclusions for the proposed license amendment and that there is reasonable assurance that the activities proposed in the LAR can be conducted without endangering the health and safety of the public and will be conducted in compliance with the Commission's regulations.

6.0 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 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 (82 FR 157), and there has been no public comment on such finding. 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.

7.0 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 Contributor: R. Beaton

Date: February 7, 2017

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 – ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.1.3.1.2 FOR CONTROL ELEMENT ASSEMBLY 39 (CAC NO. MF8935) DATED FEBRUARY 7, 2017

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