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**DOMINION NUCLEAR CONNECTICUT, INC.**
**MILLSTONE POWER STATION UNITS 1, 2, 3, AND ISFSI**
**10 CFR 50.59, 10 CFR 72.48 CHANGE REPORT FOR 2014**
**AND 2015, AND COMMITMENT CHANGE REPORT FOR 2015**

Pursuant to the provisions of 10 CFR 50.59(d)(2), the report for changes made to the facility for Millstone Power Station Unit 2 (MPS2), Unit 3 (MPS3), and Millstone Power Station (MPS) are submitted via Attachments 1, 2, and 3 respectively for the years 2014 and 2015. There were no changes made to the facility for Millstone Power Station Unit 1 (MPS1) or the Independent Spent Fuel Storage Installation (ISFSI).

Attachment 4 submits the commitment changes for MPS. This constitutes the annual Commitment Change Report consistent with the Millstone Power Station's Regulatory Commitment Management Program.

If you have any questions or require additional information, please contact Mr. Thomas G. Cleary at (860) 444-4377.

Sincerely,

B. L. Stanley
Director, Nuclear Station Safety and Licensing

IE47
NMSS26
NRR
NMSS

Attachments: 4

Commitments made in this letter: None.

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**Attachment 1**

# 10 CFR 50.59 REPORT FOR 2014 and 2015

**Millstone Power Station Unit 2**
**Dominion Nuclear Connecticut, Inc. (DNC)**

**MPS2**

**S2-EV-14-0002, Revision 0**

OPS-FH 202, Revision 2
TCC-MP-2014-012

**Change to Procedure OPS-FH-202, "Transfer Carriage and Upender Operations", to Allow Operator Action to Gather and Input Up-ender Position into Fuel Transfer System**

The activity involves a change in procedure OPS-FH 2.02 to use a new, manually provided reactor side up-ender vertical input to the fuel transfer system rather than the up-ender vertical proximity switch on Millstone Power Station Unit 2 (MPS2).

The up-ender vertical proximity switch (reactor side switch PRS_RV_A and PRS_RV_B) that indicates the up-ender is in the vertical position does not properly indicate. Indication from the up-ender that it is the vertical position is needed by the fuel transfer system so that interlocks are freed and fuel movement at the reactor side up-ender can be accomplished.

The activity evaluated is a change to OPS-FH 202 to allow manual (visual) determination that the up-ender is in the vertical position. The operator using the newly installed vertical switch bypass then provides the information to the balance of the fuel handling system.

For 10 CFR 50.59 reviewing purposes, per the guidance in NEI 96-07, the substitution of the automatic indication of vertical orientation to the fuel transfer system with manual operator verification of vertical orientation, and subsequently providing the input to the fuel transfer system via the vertical switch bypass should conservatively be treated as adverse and "screened in" therefore requiring a 10 CFR 50.59 evaluation.

The equipment affected is the fuel handling equipment. This equipment does not provide mitigation to plant accident conditions. Therefore failures associated with the equipment would not create accident scenarios with different results than those currently in the Final Safety Analysis Report.

The new operator action meets the NEI 96-07 guidance of an action unlikely to be mis-performed. Therefore, the change does not increase the probability of an accident.

Additional features (interlocks, underload/overload limits, etc.) within the fuel handling equipment designed to minimize the likelihood of fuel damage during movement are unaffected. The change, therefore, does not change refueling operations relationship to the fuel clad fission product barrier.

The activity does not involve a method of evaluation described in the safety analysis report.

**MPS2**

**S2-EV-14-0003, Revision 0**

MP2-14-01045, Revision 0
MPZ-UCR-2014-016

**Millstone Unit 2 Service Water Strainers M2L1AS, M2L1BS and M2L1CS Maximum Allowable Working Pressure Rerate**

The proposed activity is to rerate the Millstone Power Station (MPS) Unit 2 (MPS2) automatic service water strainers M2L1A, M2L1B and M2L1C from a maximum allowable working pressure of 150 psig at 100 degrees F to 100 psig at 100 degrees F. MPS operability determination OD 000569 documents that the MPS2 service water strainers L1A and L1C are operable, but not fully qualified, due to corrosion related wall thinning. The activity was required to justify full qualification for the strainers which accounts for the corrosion related wall thinning.

Final Safety Analysis Report (FSAR) Table 9.7-2 lists150 psig as the design pressure of the MPS2 SW strainers. This is actually the maximum allowable working pressure (MAWP) of the strainers as originally constructed. The required design pressure is 100 psig. The difference between the 150 psig MAWP and the 100 psig design pressure represents the margin above design pressure that existed when the strainers were constructed. DC MP2-014045 evaluates and accepts the strainer with the MAWP reduced/rerated from 150 psig to 100 psig. Per NEI 97-06, this change is adverse to the FSAR Table 9.7-2 described design function and therefore has the potential to increase the likelihood of a malfunction, create new accidents or otherwise meet the 10 CFR 50.59 evaluation criteria. No methods of evaluation are affected.

The evaluation has concluded that prior NRC review and approval is not required since the responses to the eight criteria were negative and the basis provided for each response. The justification for each response basis was that the service water strainer rerate activity continues to meet the applicable codes and that the service water system will perform the same functions as the existing system under normal and accident conditions.

**MPS2**

**E-14-00075-000 Revision 0**

MP-DC-000-MP2-13-01204 Revision 0

**MPS2 De-icing Line Retirement**

A de-icing line in the circulating water (CW) system at Millstone Power Station Unit 2 (MPS2) is installed from the condenser discharge tunnel to the intake structure. The purpose of the de-icing line is to provide operations the ability to supply warm water to the intake structure to prevent the formation of ice in the intake which could cause flow degradation. The de-icing line was part of the original design and construction of MPS2.

The de-icing line function is being eliminated since it has been determined to not be needed at MPS2. The trash rack system has already shown it is capable of maintaining flow inside the intake structure. In order for ice to obscure the intake structure, the ice would need to be at least 10 feet thick, which is not a credible event at a salt water plant. The intake structure entrance fluid velocity is below the minimum velocity needed to keep frazil ice suspended and submerged, therefore the frazil ice will rise to the surface prior to contact with the trash racks. There have been no incidences of frazil ice building up within the intake structure; therefore the de-icing line has never been used. For these reasons, the elimination of the de-icing function of the MP2 de-icing line does not result in any increase in the frequency or consequences of occurrence of an accident previously evaluated in the safety analysis report (SAR).

Eliminating the de-icing function does not affect the CW or the service water (SW) systems, nor any other system. The elimination of the de-icing function of the MP2 de-icing line cannot increase the likelihood of occurrence or consequences of a malfunction of system, structures, and components (SSCs) important to safety previously evaluated in the SAR.

The pressure boundary function of the de-icing line is being maintained as a flooding boundary and inspections to validate this function will remain in the program.

No new failure modes are introduced by this design change. The elimination of the de-icing function of the MP2 de-icing line does not create a possibility for an accident of a different type than any previously evaluated in the SAR.

Eliminating the de-icing function does not affect the CW or the SW System, nor any other system. No new failure modes of SSC important to safety are being introduced. The elimination of the de-icing function of the MP2 de-icing line cannot create a possibility for a malfunction of a SSC important to safety with a different result than any previously evaluated in the SAR.

The failure of the CW and SW systems, which the de-icing line is related to, are already evaluated in the Final Safety Analysis Report (FSAR) and accounted for such that the design basis limit for the fission product barrier could not be exceeded.

The elimination of the de-icing function does not involve, utilize, or affect any method of evaluation described in the SAR used in establishing the design bases or in the safety analysis.

For these reasons stated in the summary above, this design change can be implemented without prior NRC review and approval.

**MPS2**

**S2-EV-15-0001 Revision 0**

OP 2304J Rev. 001
ETE-MP-201 5-1089/Revision 000

**Change to procedure, "Boric Acid Addition to Chemical Volume and Control System (CVCS) from a Temporary Borated Water Source"**

This 10 CFR 50.59 evaluation reviewed a revision to operating procedure OP 2304J, "Boric Acid Addition to CVCS from a Temporary Berated Water Source". Procedure OP 2304J uses a procedurally controlled temporary modification to use temporary tanks as a source of borated water to be injected into the reactor coolant system during a refueling shutdown. The specific procedure changes that were evaluated by this 10 CFR 50.59 evaluation are as follows:

1. The boron concentration of the temporary borated water source is being increased from 3.5 weight percent boric acid to 4.5 weight percent boric acid. The maximum boron concentration specified in the Technical Requirements Manual and the Updated Final Safety Analysis Report is limited to 3.5 weight percent boric acid to ensure that the boron does not precipitate in the boric acid storage tanks and boric acid piping with ambient temperatures in the auxiliary building greater than 55°F.
2. The required minimum temperature in the auxiliary building is being increased from 55°F to 70°F during performance of OP 2304J. The minimum ambient temperature in the vicinity of the boric acid piping and the charging pumps is required to be greater than 70°F whenever berated water containing 4.5 weight percent boric acid is in the boric acid piping to ensure that the boron does not precipitate in the boric acid piping.

These proposed activities will not increase the probability of an accident previously analyzed or create the possibility of an accident of a different type. All of the Chapter 14 accident analyses are unaffected by this change and remain bounding. Use of this borated water flow path will be terminated by manual operator action should a SIAS (Safety Injection Actuation Signal) actuation occur. This manual action is required to avoid the reactor vessel boron concentration building up to the precipitation limit following a large break loss of coolant accident, potentially impacting long term cooling. Credit for this manual operator action is qualitatively justified based on the lower reactor coolant system decay heat boil-off rates and the relatively long timeframe post-loss of coolant accident when boron precipitation is of concern. In addition, the proposed activities will not increase the likelihood or consequences of a malfunction of systems, structures or components important to safety and will not create the possibility of a malfunction with a different result than previously analyzed. The only systems that are potentially impacted by this proposed activity are the boric acid, and charging systems. The evaluation concludes that these systems are not adversely impacted by the proposed activity. As such, the results of this 10 CFR 50.59 evaluation demonstrate that the increase in the boron concentration of the temporary borated water source can be implemented without prior NRC approval.

**MPS2**

**E-15-00047-000 Revision 0**

MP2-13-01154-000

The Millstone Power Station Unit 2 (MPS2) service water strainer backwash valves and actuators, 2-SW-90A/B/C, are being replaced due to end-of life degradation and parts obsolescence. The replacement actuators have a maximum functional pressure rating of 116 psi which is below the maximum instrument air supply piping pressure of 125 psi. Thus the change includes the installation of an air regulator in the individual instrument air supply lines to the service water strainer backwash valve actuators. The new air regulators are credited for protecting the replacement pneumatic actuators from potential over-pressurization. If the new air regulator were to fail, the actuator could become overpressurized, potentially damaging the actuator and preventing it from performing its function which is to open upon loss of air. Failure of the actuator as a result of an air regulator failure was considered to be a new failure mode and therefore had the potential to increase the likelihood of a malfunction or otherwise meet the 10 CFR 50.59 evaluation requirements. However, the evaluation determined the potential malfunction of the actuator failing to open would result in the same failure of the service water strainer backwash valve as previously considered, and there was not a more than minimal increase in the likelihood of a malfunction because the design continues to meet all applicable design codes and standards per NEI 96-07 Guidance.

The 10 CFR 50.59 evaluation concludes that prior NRC review and approval is not required because the responses to the eight evaluation criteria are negative, and the basis for each response was appropriately provided. In summary, the design change for the service water strainer backwash valves, actuators, and new pressure regulators, meets the design requirements, and that the service water system will continue to perform the same functions as the existing system under normal and accident conditions described in the safety analysis report.

**Attachment 2**

# 10 CFR 50.59 REPORT FOR 2014 and 2015

**Millstone Power Station Unit 3**
**Dominion Nuclear Connecticut, Inc. (DNC)**

**MPS3**

**S3-EV-14-0001, Revision 0**
**Performance of TDAFW Pump 3FWA*P2 Full-Flow Test During Operational Mode 1**

The activity evaluated is performance of SP 3622.9 Rev. 000, the "Auxiliary Feedwater Pump 3FWA*P2 Full Flow Test in Mode 1 (ICCE)". This evaluation also supports ETE-MP-2014-1016, Revision 0. The test is required to verify Technical Specification (TS) operability of the turbine driven auxiliary feedwater (TDAFW) pump after replacement of the steam admission valve and governor.

This evaluation was required because:

Full flow testing of the TDAFW pump is currently performed in mode 3 by surveillance procedure SP 3622.3. However, SP 3622.9 is a new full flow auxiliary feedwater (AFW) pump test to be performed in mode 1 near 100% power and this test is not specifically described in the Millstone Power Station Unit 3 (MPS3) Final Safety Analysis Report (FSAR). FSAR Chapter 14 refers to pre-operational testing of the auxiliary feedwater (AFW) system (e.g., Table 14.2-1, item 30 at ambient conditions) that is not applicable. FSAR Section 10.4.9.4 describes AFW pump testing with flow established by recirculating AFW to the demineralized water storage tank.

Further, the AFW system is being "utilized outside the reference bounds for design ..." as described in FSAR Chapter 10. The critical design function is to remove fission product decay heat and other residual heat form the reactor core to ensure design limits are met.

Testing the AFW system flowing to the steam generators (SGs) in mode 1 near 100% power can be considered "... outside the reference bounds for design". Additional information was acquired from Section 5.2.2.4 of the USA 50.59 Resource Manual, which is recommended for guidance by CM-AA-400.

The following excerpt covering tests not described in the FSAR (which applies here) indicates that an evaluation is not required if the test is bounded by another test that is described in the safety analysis report (SAR). The MPS3 FSAR does not identify such a bounding test. Further, the USA 50.59 Resource Manual specifically identifies that an evaluation is not required if the affected systems, structures, and components (SSCs) will be isolated from the facility. SP 3622.9 does not isolate the AFW system from the operating secondary system.

Tests and experiments that are not described in the FSAR do not require evaluation under 10 CFR 50.59 provided the test or experiment is bounded by tests and experiments that are described.

Similarly, tests and experiments not described in the FSAR do not require a 10 CFR 50.59 evaluation provided that affected SSCs will be appropriately isolated from the facility.

Because the test is not described in the SAR and the SSC is being utilized outside the reference bounds for design, a 10 CFR 50.59 evaluation is required.

The evaluation concluded that prior NRC review and approval is not required since the responses to the eight criteria were negative.

**MPS3**

**S3-EV-14-0003**

**RELOAD SAFETY EVALUATION MILLSTONE UNIT 3 CYCLE 17**

The proposed activities that are the subject of this evaluation as part of EVAL-ENG-RSE-M3C17, Revision 0 are:

1. Implementation of the revised Westinghouse cladding corrosion model,
2. Implementation of the increased maximum allowable reactor coolant system (RCS) Lithium limit from 4.2 ppm to 6.0 ppm, and
3. Implementation of the revised Rod Withdrawal at Power (RWAP) and Loss of Load/Turbine Trip (LOL/TT) analyses.

Evaluation Item 1 addresses the implementation of WCAP-12610-P-A, Addendum 2-A, "Westinghouse Clad Corrosion Model for ZIRLO and Optimized ZIRLO" as a replacement for the existing Westinghouse cladding corrosion model. This new model is used in the Millstone Power Station Unit 3 (MPS3) Cycle17 fuel rod design analysis performed by Westinghouse. A 10 CFR 50.59 evaluation is required since the method documented in WCAP-12610-P-A Addendum 2-A is replacing the current method cited in the MPS3 Final Safety Analysis Report (FSAR). The current fuel rod design methodology cited in the FSAR for cladding corrosion is documented in the original issue WCAP-12610 and Addendum 1-A: A 50.59 evaluation is conducted based on the application of the new methodology. The evaluation concluded that the proposed activity does not constitute a departure from a method of evaluation as the NRC has approved this method for use in the intended application for which it is applied. In the NRC safety evaluation Report (SER) for WCAP-12610 Addendum 2-A, the NRC found it acceptable that Westinghouse apply this method to fuel rod design analysis for ZIRLO and Optimized ZIRLO based fuel. As the Westinghouse RFA-2 design currently employed at MPS3 uses both ZIRLO and Optimized ZIRLO, the intended application is fulfilled. In addition, the database on which the new corrosion model was developed is based on plants that operated at conditions similar to MPS3. It is thus appropriate to apply this new model to predictions of MPS3 fuel. The Westinghouse fuel rod design analysis for MPS3 Cycle 17 has been conducted in accordance with the methodologies as approved by the NRC. Therefore the limitations and conditions as set forth in the NRC safety evaluation report have been fulfilled. It can therefore be concluded that this activity is not a departure from a method of evaluation and does not require NRC approval.

Evaluation Item 2 addresses the implementation of increasing the maximum allowable RCS Lithium concentration from 4.2 ppm to 6.0 ppm. This change could impact corrosion phenomena of reactor coolant system materials including general corrosion and primary water stress corrosion cracking (PWSCC).

Evaluation Item 3 addresses the implementation of the revised RWAP and LOL/TT analyses to correct a Westinghouse error in modeling the flow area of the pressurizer safety valves (PSVs) and power operated relief valves (PORVs) used in the analysis. The RWAP and LOL/TT analyses were required to be re-run to demonstrate all acceptance criteria were met after correcting the PORV and PSV flow area used in the analysis. Evaluation Items 2 and 3 under the first seven criteria of 10 CFR 50.59 determined the change can be implemented without prior NRC approval as the analyses continue to satisfy the all acceptance criteria described in the FSAR. In addition, the changes are not an initiator of any transient or malfunction, introduce no new failure modes or alternations of system, structures, and components (SSCs) and do not change or exceed a design basis limits for a fission product barrier (DBLFPB). NRC review and approval is not required prior to implementing the changes associated with the increased Lithium concentration or the revised RWAP and LOL/TT analyses for MPS3.

The three changes comprising this activity listed in this 10 CFR 50.59 evaluation have been shown to not impact the associated design function, exceed or alter a DBLFPB, or result in departure in a method of evaluation as described in the FSAR. Therefore, the revised Westinghouse cladding corrosion model, the increase in the maximum allowable RCS Lithium concentration, and the revised RWAP and LOL/TT analyses can be implemented under the provisions of 10 CFR 50.59 without prior NRC review and approval.

**MPS3**

**S3-EV-14-0004**

**MP3 TDAFW PUMP OVERSPEED MARGIN CHANGE**

The Millstone Unit 3 (MPS3) turbine-driven auxiliary feedwater (TDAFW) pump nominal operating speed was changed from 4,500 rpm to 4,400 rpm by adjusting the TDAFW pump governor mechanical and electrical stop. The reduction in speed decreased the flow delivered from the TDAFW pump.

The auxiliary feedwater (AFW) flow at the reduced turbine speed resulted in flow delivery rate lower than that modeled for the Station Blackout (SBO) event. That is considered an adverse change in Section III of the 50.59 screen review. For other events, the AFW flow rates associated with the lower terry turbine speed remained sufficiently high such that the minimum flow requirements for the specific events were met.

The activity evaluated herein was limited to the incorporation of the new, reduced TDAFW flow rate into the SBO thermal hydraulic analysis of record. While the minimum steam generator (SG) inventory occurring was seen to be lower with the lower flow rate, it was demonstrated that adequate SG inventory remains at all times. Therefore, a license amendment was not required.

The activity was a new analysis of the SBO event and the capacities of the equipment responding. There was no change in the frequency of the initiator as it is modeled to occur. Therefore, the activity did not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the safety analysis report (SAR).

The change in terry turbine speed was an adjustment made to the existing governor to increase margin to the overspeed trip reducing the likelihood of a TDAFW pump trip upon startup. Additionally, the new analysis did not credit additional equipment activation responding to the SBO. The expected response of other plant equipment, i.e. emergency diesel generators was proscribed by the definition of the event. Therefore, the activity did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of systems, structures, and components (SSCs) important to safety previously evaluated in the SAR.

In both the current analysis and the analysis of the event with the reduced AFW flow rate, the response of the TDAFW pump was a requirement for success, therefore, there was no change in the consequences of a malfunction of the SSC.

The equipment modeled to response to the SBO was specified in the definition of the event. The new SBO analysis makes no changes to the assumptions. Therefore, the activity did not create a possibility for an accident of a different type than any previously evaluated in the SAR.

While the minimum SG inventory occurring was seen to be lower with the lower flow rate, it was demonstrated that adequate SG inventory remains at all times. Therefore the activity did not result in more than a minimal increase in the consequences of an accident previously evaluated in the SAR.

The activity did not change the design basis limits for the fission product barriers and the new analysis demonstrated that these design basis limits were not challenged with the lower AFW flow rates. Therefore, the activity ddid not result in a design basis limit for a fission product barrier as described in the SAR being exceeded or altered.

The new evaluations were performed with the same methodologies as the existing SBO evaluation. Therefore the activity did not result in departure from a method of evaluation described in the SAR used in establishing the design bases or in the safety analyses and can be implemented under the provisions of 10 CFR 50.59 without prior NRC review and approval.

**MPS3**

**S3-EV-14-0005, Revision 0**

- OD000590 Revision 0

**Addition Of Compensatory Action to Provide Procedural Guidance to Control the Rate Of Opening/Closing of the Millstone Power Station Unit 3 (MPS3) Turbine Driven Auxiliary Feedwater Pump (TDAFWP) Discharge Flow Control Valves**

Operability determination, OD000590, added a compensatory action to provide procedural guidance to control the rate of opening/closing of the Millstone Power Station Unit 3 (MPS3) turbine driven auxiliary feedwater pump (TDAFWP) 3FWA*P2 discharge flow control valves 3FWA*HV32A-D & 3FWA*HV36A-D in order to minimize the potential for causing pressure spikes that result in lifting of the pump discharge relief valve 3FWA*RV45. The control valves are remote-manually controlled from the control room. The specified minimum allowed full stroke time is 15 seconds. The procedural guidance also states that the subject control valves may be opened-closed more quickly when needed to perform time-critical actions.

The evaluation determined that the opening/closing rate limit of the procedural guidance would not be challenged during throttling of the subject control valves to maintain steam generator (SG) inventory control for decay heat removal after the initial phase of design basis accidents (DBA's).

The evaluation also determined there are no accident analysis which credited operator action to open the TDAFWP discharge control valves and that the accident analyses which credit operator action to close the TDAFWP or motor driven auxiliary feedwater pump (MDAFP) discharge control valves is limited to the SG tube rupture (SGTR) margin to overfill (MTO) analysis (and associated SGTR radiological consequences analysis) and the mass and energy release analysis for a steam line break (SLB) or feedwater line break (FLB) inside containment. Both of these analyses credit operator action to isolate AFW flow to the affected SG; the intact SG's continue to receive auxiliary feedwater (AFW) flow for decay heat removal. Operating experience has shown that rapidly isolating AFW flow from the TDAFWP to a single SG does not result in a pressure spike that challenges the TDAFWP discharge relief valve.

The evaluation determined that while observing the procedural guidance to limit the open-close rate of the TDAFWP discharge control valves, the operators are still capable of performing all credited operator actions within the time frames that are assumed by the subject accident analyses. Also, there are no physical changes associated with this activity and there are no automatic control functions associated with the subject control valves. Based on this, the evaluation determined a negative response to each of the eight evaluation criteria and Prior NRC approval for this change is not required.

**Attachment 3**

# 10 CFR 50.59 REPORT FOR 2014 and 2015

**Millstone Power Station Units 2 and 3**
**Dominion Nuclear Connecticut, Inc. (DNC)**

**MPS Site**

**SG-EV-15-0002, Revision 0**

ETE-MP-201 5-1068
MP2-UCR-016
MP3-UCR-017

The activity being evaluated is the replacement of 345 kV obsolete and aging electro-mechanical and solid state protective relaying with newer state of the art microprocessor based digital relays. The replacement digital protective relays which were installed approximately 7 - 10 years ago are owned and maintained by the grid owner. The evaluation is being prepared to address the replacement of electro-mechanical and solid state relays with microprocessor based digital protective relays. Based on the change from electro-mechanical/solid state relays to digital based relays coupled with the change in the diverse method for performing the protective trip function, the 10 CFR 50.59 evaluation is being performed to address this change.

The replacement protective relays are digital in nature. However, to ensure diversity in the protection scheme, the primary and backup digital protective relays are of a different manufacturer and design. Similar to the previous design, either the primary or backup protective digital relay sensing a fault condition will result in the protective action of isolating the faulted 345 kV line. The digital relays were installed because the existing electro-mechanical and solid state relays were obsolete and aging and the reliability of the existing relays was reduced.

The 10 CFR 50.59 evaluation concluded that the following:

- The change did not result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the safety analysis report (SAR).
- The change did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a system, structure, and component (SSC) important to safety previously evaluated in the SAR.
- The change did not result in more than a minimal increase in the consequences of an accident previously evaluated in the SAR.
- The change did not result in more than a minimal increase in the consequences of a malfunction of a SSC important to safety previously evaluated in the SAR.
- The change did not create a possibility for an accident of a different type than any previously evaluated in the SAR.
- The change did not create a possibility for a malfunction of a SSC important to safety with a different result than any previously evaluated in the SAR.
- The change did not result in a design basis limit for a fission product barrier as described in the SAR being exceeded or altered.
- The change did not result in departure from a method of evaluation described in the SAR used in establishing the design bases or in the safety analyses.

The digital relays provide the same protective action as the relays which they replaced and do not introduce any new operator intervention or burden with the change. The digital relays are periodically checked in accordance with Table 1-1 of Standard PRC-005-2 and the Nuclear Electric Insurance Limited standard. Further, these relays provide remote alarm notification via the supervisory control and data acquisition (SCADA) system which is monitored by the grid

operator. The installation of the newer state of the art digital relays was performed to maintain the reliability of the transmission line protection system by replacing aging and obsolete electro-mechanical and solid state protective relays with newer state of the art microprocessor based digital relays. The newer digital relays are of a proven design and are used throughout by the grid owner for similar applications. These specific protective relays which provide the 345 kV transmission line protection have been in operation for approximately 7 to 10 years of service and have been shown to be reliable.

Both MPS2 and MPS3 Final Safety Analysis Reports (FSARs) require updating to reflect the replacement of the existing electro-mechanical and solid state protection relays with digital protection relays. The FSARs identify these protection relays as being electromechanical and solid state relays when in fact they are now digital based protection relays.

**MPS Site**

**SG-EV-15-0003, Revision 0**

ETE-MP-201 5-1068
MP2-UCR-016
MP3-UCR-017

This regulatory evaluation evaluates changes to the Millstone Power Station (MPS) Unit 2 (MPS2) and Unit 3 (MPS3) Final Safety Analysis Reports (FSARs) to resolve a legacy issue regarding a change on the number of available close-open cycles that the switchyard circuit breakers can perform. The MPS2 and MPS3 FSARs essentially state that:

Each (switchyard) power circuit breaker has a separate pneumatic supply unit capable of operating the breaker for five close-open operations after the loss of its pneumatic supply unit.

The switchyard circuit breakers were replaced by the grid owner circa 2005 with new breakers that per the manufacturers are capable of only four close-open cycles, however, the Licensee was unable to validate four full close-open cycles to so three close-open cycles was used for this evaluation.

The switchyard equipment impacted by this change is required for GDC-17 compliance and thus the change was considered potentially adverse warranting a regulatory evaluation to determine if NRC approval is required.

Consultations with the grid owner, Eversource, and with Dominion Electric Transmission, Dominion Virginia Power's grid owner were used to develop an expected worst case scenario for restoration of the MPS switchyard breaker close-open cycles. The worst case scenario for system restoration scenario assumed the switch yard breakers do not trip from the initiating disturbance. All switchyard breakers would have to be opened to clear the yard prior to energizing any of the 345 kV lines connected to MPS to avoid possible undesirable equipment energization. After energizing the transmission line, the switchyard breakers would be closed as necessary to restore offsite power to the normal station services transformer and reserve station services transformer. This scenario required only one open-close cycle whether the switchyard breakers tripped due to protective relay operation or operator action. Assuming a breaker fails to open when commanded (single failure) previously closed breakers may need to be opened to isolate the inoperable breaker requiring a second open cycle. At this point the breaker's spring is charged for closing which would complete the second open-close cycle when needed. It could be assumed that the switchyard would be restored to the extent practical at this point such that compressor power could be restored as only one offsite transmission line would be impacted by a faulted breaker. However, additional design margin still exists in that a sufficient air pressure remains to cycle all the switchyard breakers open close at least one more time providing adequate defense in depth.

A review of the MPS2 and MPS3 accident analyses indicates that for accidents where a loss of offsite power represents a potential limiting condition, the loss of offsite power is assumed and it is further assumed that offsite power is not restored and all required licensing conditions remain met. As such the consequences of an accident previously evaluated in the safety analysis report (SAR) are not increased, nor are the design basis limits for a fission product barrier adversely affected.

It is noted that the change in breaker cycles affects a mitigating activity and as such, it is not an accident initiator. Thus the frequency of occurrence of previously evaluated accidents is not increased and the changes do not result in the failure of systems, structures, and components

(SSCs) important to safety so no new accidents or failures of SSCs important to safety will occur. Similarly, accidents of a different type or a malfunction of an SSC with a different result were also determined not to occur. As SSCs are not impacted, the consequences of the failure of an SSC important to safety are also not increased and the design basis limits for the fission product barriers as described in the SAR are not changed.

The change does not represent a change in a method of evaluation.

Therefore, this change was determined not to result in more than a minimally adverse change and further NRC review is not required.

**MPS Site**

**SG-EV-15-0004, Revision 0**

ETE-MP-201 5-1068
MP2-UCR-016
MP3-UCR-017

This regulatory evaluation evaluates changes to the Millstone Power Station (MPS) Unit 2 (MPS2) and Unit 3 (MPS3) Final Safety Analysis Reports (FSARs) to resolve legacy issues regarding the time required to restore a "good element" (transmission system component or line) in the MPS switchyard. The MPS2 and MPS3 FSARs essentially state that:

*"The failure of any circuit breaker to trip initiates the automatic tripping of the adjacent breakers and thus may result in the loss of a line or generator for this contingency condition; however, power can be restored to the good element in less than one hour by manually isolating the fault with appropriate disconnect switches."*

The grid owner has indicated that due to regulatory and security requirements, a one hour response time cannot be met. Note that at the time the FSAR was prepared, Northeast Utilities was the Licensee and the grid owner. The current grid owner has proposed four hours as a response time, however, this time could not be validated so a less than eight hour response time was evaluated. This time was selected because it remained within the station black out (SBO) coping criteria limitations in accordance with the SBO rule as described in the MPS2 and MPS3 FSARs. Note that SBO equipment and coping strategies are not credited as part of this evaluation.

The switchyard equipment impacted by these changes are required for GDC-17 compliance and thus these changes were considered potentially adverse warranting a regulatory evaluation to determine if NRC approval is required.

For the May 25, 2014 loss of off-site power (LOOP) and dual unit trip event, offsite power was lost at approximately 07:01 and the north and south 345 kV busses were restored at approximately 10:20 (grid owner personnel were in the switchyard at the time of the event). On a separate event in July 2014, grid owner personnel were available and on site in approximately 3 hours. ISO New-England would be authorizing required switching orders while the grid operator was in route. Assuming that it takes approximately three hours to respond (July 2014 event) and three hours to perform switching activities (May 25, 2014 event) it would take less than eight hours to respond and restore a good element.

A review of the MPS2 and MPS3 accident analyses indicates that for accidents where a loss of offsite power represents a potential limiting condition, the LOOP is assumed and it is further assumed that offsite power is not restored and all required licensing conditions remain met. As such the consequences of an accident previously evaluated in the safety analysis report (SAR) are not increased, nor are the design basis limits for a fission product barrier adversely affected.

It is noted that this change is to mitigating activities and so are not considered accident initiators. Thus the frequency of occurrence of previously evaluated accidents is not increased and the changes do not result in the failure of systems, structures, and components (SSCs) important to safety so no new accidents or failures of SSCs important to safety will occur. Similarly, accidents of a different type or a malfunction of an SSC with a different result were also determined not to occur. As SSCs are not impacted, the consequences of the failure of an SSC important to safety are also not increased.

These changes do not represent a change in a method of evaluation.

The less than eight hour time limit is consistent with the SBO identified coping criteria so the ability of MPS2 or MPS3 to cope with an SBO event is also not challenged by this change.

Therefore this change was determined not to result in more than a minimally adverse change and further NRC review is not required.

**MPS Site**

**SG-EV-15-0005, Revision 0**

ETE-MP-201 5-1068
MP2-UCR-016
MP3-UCR-017

This document evaluates a change to the Millstone Power Station (MPS) Unit 2 (MPS2) and Unit 3 (MPS3) Final Safety Analysis Reports (FSARs) to update statements that switchyard is arranged in a breaker and a half configuration. With MPS Unit 1 (MPS1) decommissioned the functions of 345 kV switchyard breakers 3T and 6T were no longer required and the breakers were subsequently removed by the grid owner. The removal of these breakers resulted in the configuration of the switchyard changing from a full breaker and a half configuration to a partial breaker and a half and a partial double breaker configuration. FSAR MPS2 Appendix 1.A and section 8.1.2.1 and MPS3 section 8.2.2 state respectively:

The MPS2 safety analysis report (SAR) 1.A for GDC-17 states:

*The breaker-and-a-half switching arrangement in the 345 kV substation includes two full capacity main buses. Primary and backup relaying are provided for each circuit along with circuit breaker failure backup protection. These provisions permit the following:*

a. *Any circuit can be switched under normal or fault conditions without affecting another circuit.*
b. *Any single circuit breaker can be isolated for maintenance without interrupting the power or protection to any circuit.*
c. *Short circuits on any section of bus are Isolated without interrupting service to any element other than those connected to the faulty bus section.*
d. *The failure of any circuit breaker to trip within a set time initiates the automatic tripping of the adjacent breakers and thus may result in the loss of a line or generator for this contingency condition; however, power can be restored to the good element in less than one hour by manually isolating the fault with appropriate disconnect switches.*

MPS2 section 8.1.2.1 and MPS3 section 8.2.2 also state that the switchyard is configured in a breaker and a half configuration.

The switchyard equipment impacted by this change is required for GDC-17 compliance and thus the change was considered potentially adverse warranting a regulatory evaluation to determine if NRC approval is required.

A breaker and a half configuration consists of three breakers between the two switchyard busses, for MPS, the north and south 345 kV busses. In this configuration, an offsite transmission line or unit generator output line is connected between the first and second and then between the second and third breakers providing additional flexibility for switchyard alignment and reliability. In a double breaker alignment, only two breakers are installed between the north and south busses with only a single transmission line connection between the breakers. When MPS1 was decommissioned and the associated main generator and reserve station services transformer (RSST) connections removed there was no longer a need for the breaker and a half configuration for the associated lines. The FSAR change being evaluated affects components that perform functions necessary for compliance with GDC 17. Compliance with the GDC-17 criteria was determined to be maintained as discussed below:

When the MPS1 main generator and RSST connections were removed, the breaker and a half configuration for the associated lines was no longer required to maintain the same level of functionality for switchyard protection and alignments. Since the internal connections between the intermediate breakers had been removed the third breaker had tb function (the breaker and a half configuration left two switchyard breakers in series with no connection between) Therefore the criteria specified in MPS2 Appendix 1.A for GDC-17 compliance and by inference for MP3 GDC-17 compliance is maintained at the same level of reliability by the current breaker and a half/double bus configuration as was provided by the previous full breaker and a half configuration.

A review of the MPS2 and MPS3 accident analyses indicates that for accidents where a loss of offsite power represents a potential limiting condition, the loss of offsite power is assumed and it is further assumed that offsite power is not restored and all required licensing conditions remain met. As such the consequences of an accident previously evaluated in the SAR are not more than minimally increased, nor are the design basis limits for a fission product barrier adversely affected.

It is noted that the change in the switchyard configuration is not an accident initiator. The reliability and functionality of the switchyard connections were not adversely impacted by the change in configuration. Thus the frequency of occurrence of previously evaluated accidents is not more than minimally increased and the changes do not result in the failure of systems, structures, and components (SSCs) important to safety so no new accidents or failures of SSCs important to safety will occur. Similarly, accidents of a different type or a malfunction of an SSC with a different result were also determined not to occur. As SSCs are not impacted, the consequences of the failure of an SSC important to safety are also not more than minimally increased.

The change does not represent a change in a method of evaluation.

Therefore, this change was determined not to result in more than a minimally adverse change and further NRC review is not required.

**Attachment 4**

# COMMITMENT CHANGE REPORT FOR 2015

**Millstone Power Station**
**Dominion Nuclear Connecticut, Inc. (DNC)**

In 1991 senior Northeast Utilities leadership chartered a task force under the self-assessment process, whose purpose was to "address the issues of quality and timeliness of our determinations regarding operability and reportablity, as well as, communications with the NRC". One of the recommendations provided by the task force suggested improvements in the handling of differing professional opinions (DPOs). The level of guidance recommended resulted in new procedure NEO 2.25, in the absence of other existing processes and tools. Over time NEO 2.25, was superseded by new procedures. By August 2012, the procedure was called PI-MP-200-2004 "Differing Professional Opinion Resolution." At this time, a fleet decision was made to cancel PI-MP-200-2004 and incorporate the intent of the commitment into procedure PI-AA-200, "Corrective Action" - Attachment 6, which assigns a Medium Priority to any working environment issue that involves a DPO. However, in this process the commitment, as described in Regulatory Commitment Record (RCR) 25779, was no longer specifically met and reference to the RCR 25779 was not carried into PI-AA-200.

Review of the docketed correspondence (BI3917, 08/26/1991) did not indicate the specific reason for implementing NEO 2.2.5 beyond that the Task Force was chartered to look at this area for improvement. However, NRC letters dated 01/02/1992 and 01/22/1992 make reference to a meeting where actions were presented to the NRC and the NRC would continue to assess the actions against weaknesses identified during the Systematic Assessment of Licensee Performance process.

The correspondence BI3917 did note implementing a DPO process could "be done either as an explicit element of NEO 2.25, or by some broader vehicle (e.g., a new NEO) to which NEO 2.25 could refer. One element of the DPO resolution process is to have a point by point enumeration of the reasons for the DPO, and a response to each." In today's environment, there are several processes that individuals can use to raise issues such as differing professional opinions, which include raising the issue with management, entering the issue in the corrective action program, contacting Human Resources (HR), contacting Employee Concerns (ECP), or if needed, contacting the NRC. The preferred method would be to use the corrective action program, which is a robust, structured problem identification and resolution process that is periodically assessed by the NRC, INPO, and Nuclear Oversight for effectiveness. Additionally, an independent and confidential process is available and encouraged for use through the employee concerns program. To support this, procedure EC-AA-110 "Identifying and Addressing Nuclear Safety Concerns" was revised to further explain a DPO and set expectations for management, or ECP to resolve DPOs exactly as they would a nuclear safety concern, essentially treating either equally. While this commitment was not addressing an adverse condition, it was an action taken to address weaknesses identified by the NRC during the SALP process and, based on correspondence, which was being assessed by the NRC as a long-term corrective action (NRC letters dated 01/02/1992 and 01/22/1992). Based on these mature processes, RCR 25779 specified actions are no longer necessary and RCR 25779 can be closed.