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U.S. Nuclear Regulatory Commission
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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 2016,
AND COMMITMENT CHANGE REPORT FOR 2016

Pursuant to the provisions of 10 CFR 50.59(d)(2), the report for changes made to the facility for Millstone Power Station Unit 1 (MPS1), Unit 2 (MPS2), and Unit 3 (MPS3) are submitted via Attachments 1, 2, and 3 respectively for 2016. The report for changes made to the facility for both MPS2 and MPS3 is submitted via attachment 4. There were no changes made to the Independent Spent Fuel Storage Installation (ISFSI).

Attachment 5 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. Jeffry A. Langan at (860) 444-5544.

Sincerely,

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

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NMSS26
NRR
NMSS

Attachments: 5

Commitments made in this letter: None.

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Attachment 1

10 CFR 50.59 REPORT FOR 2016
MILLSTONE POWER STATION UNIT 1

Millstone Power Station Unit 1
Dominion Nuclear Connecticut, Inc. (DNC)

MPS1

MPS1-EVAL-2016-0006

Design Change (DC) MP1-15-01006, Millstone Unit 1 ESST Transformer Replacement

This 50.59 evaluation reviewed the replacement of the Millstone Power Station Unit 1 (MPS1) Emergency Station Service Transformer (ESST) with a new, smaller transformer. The replacement transformer is sized to power the reduced loading of MPS1 in its defueled condition compared to the originally installed ESST that had been sized for an operating unit. The screen determined that an evaluation was required to further review the reduction in size of the transformer compared to the ESST (margin reduction) described in the Defueled Safety Analysis Report (DSAR), and to evaluate the elimination of the ESST deluge system.

This change does not result in a more than minimal increase in the consequences or frequency of any accident analysis presented in the DSAR. There is no impact on the current evaluation of fuel handling accidents or availability of other power sources. This change does not result in a more than minimal increase in the likelihood of failure or consequences of failure of any safety system component (SSC).

The activity provides an additional means to selected fuel pool island loads by adding a tap box to allow for the back feeding of MPS1, 480V Bus FAC-B2, from a portable diesel generator. This additional tap box is not required for any accident analysis and has no adverse effect on any accident analysis. The existing ESST contains a deluge fire suppression system that is not included with the replacement transformer. A fire protection review determined that the Fire Hazards Analysis (FHA) is impacted by this change. The fire protection review determined that the replacement transformer does not require an automatic suppression system.

The DSAR will be updated by Safety Analysis Report (SAR) change request number MP1-DFCR-2015-001 to reflect the new name established for the replacement transformer. Additionally, updates will be made to reflect the performance ratings of the new transformer and abandonment of the ESST deluge system.

The proposed change does not create a new type of event or result in an event with a different result. It has no negative impact on any fission product barrier, does not increase the consequences of an event and does not result in any departure from any evaluation methodology described in the Final Safety Analysis Report (FSAR).

Therefore, this evaluation has determined the proposed changes may be implemented per applicable procedure without obtaining a License Amendment.

MPS1

MPS1-EVAL-2016-0008

23 KV Switch / Circuit Breaker Replacement on Unit 1

This 50.59 evaluation reviewed the replacement of the MPS1 Trayer switch with a new G&W switch that contains digital voltage sensing and digital protective relaying. The screen determined that an evaluation was required to address the likelihood of malfunction, consequences of malfunction, and malfunctions of a different type with respect to the electric system function to supply reliable power for the permanently defueled condition at MPS1. The 10 CFR 50.59 evaluation concluded 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 DSAR.
- The change did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated In the DSAR.
- The change did not result in more than a minimal increase in the consequences of an accident previously evaluated in the DSAR.
- The change did not result in more than a minimal increase in the consequences of a malfunction of a SSC important to safely previously evaluated in the DSAR.
- The change did not create a possibility for an accident of a different type than any previously evaluated In the DSAR.
- 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 DSAR.
- The change did not result in a design basis limit for a fission product barrier as described in the DSAR being exceeded or altered.
- The change did not result in departure from a method of evaluation described in the DSAR used in establishing the design bases or in the safety analyses.

No change is required to the DSAR because the types of protective relaying are beyond the level of detail described by the DSAR for the electrical system.
Therefore, this evaluation has determined the proposed changes may be implemented per applicable procedure without obtaining a License Amendment.

Attachment 2

10 CFR 50.59 REPORT FOR 2016
MILLSTONE POWER STATION UNIT 2

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

MPS2

MPS2-EVAL-2016-0004, Revision 0

EOP 2541 Standard Appendix 42 "Restoring Spent Fuel Pool Cooling", Revision 001-00

EOP 2541 Standard Appendix 42 "Restoring Spent Fuel Pool Cooling" is being changed. Step 8, to adjust reactor building closed cooling water (RBCCW) flow through the spent fuel pool (SFP) heat exchanger(s) and maintain SFP temperature less than 120°F, is being split into two steps (one for each RBCCW header) with additional detail added directing the opening of the 2-RB-8.1A/B valves from either the control room (as an action/expected response step) or in the auxiliary building locally at the value (as a response not obtained step). A third step taken from existing step 8 is added to maintain SFP temperature less than 120°F. In addition, a caution statement is being added prior to new Steps 8 and 9 regarding the plant equipment operator travel path in the -25 foot auxiliary building elevation to the RBCCW heat exchanger area. In particular the caution recommends travel from the stairs to the right (west side) of the removable hatch cover adjacent to the spent fuel resin tank valves wall due to potentially elevated dose rates from the emergency core cooling system (ECCS) vertical pipes on the left (east side) near the condensate recovery tank. This will reduce the potential radiological dose received by the plant equipment operator while restoring SFP cooling.

Valves 2-RB-8.1A/B are air operated valves (AOVs) that receive a signal to automatically close on a safety injection signal, isolating RBCCW flow to the heat exchanger. This automatic isolation terminates SFP heat removal and directs more RBCCW flow through the containment air recirculation (CAR) cooling units during the ECCS injection phase. Between 4 and 5 hours following a loss of coolant accident (LOCA), EOP 2532, EOP 2534, EOP 2536, EOP 2540D and EOP 2541 Standard Appendix 4 directs the operator to restore SFP cooling via EOP 2541 Standard Appendix 42.

AOVs 2-RB-8.1A/B do not have safety related backup air. Since instrument air cannot be credited to reposition valves from the control room following a LOCA or other design basis event, these valves must be opened in the auxiliary building by a plant equipment operator. In accordance with NUREG-0737 Item II.B.2, the calculation documenting the radiological doses to the plant equipment operator to restore SFP in the auxiliary building was updated. The updated calculation concludes that the operator can accomplish this action without exceeding the NUREG-0737 Item II.B.2 allowable dose.

The NEI 96-07 10 CFR 50.59 definition of safety analysis includes "supporting Updated Final Safety Analysis Report analyses that demonstrate safety system component design functions will be accomplished as credited in the accident analysis." Because restoration of SFP cooling is required following a LOCA and a change to a calculation was required to demonstrate that restoration of SFP cooling could be accomplished without exceeding the NUREG-0737 II.B.2 allowable dose, this 50.59 evaluation was required.

The additional emergency operating procedure (EOP) guidance to restore SFP cooling following an event does not increase the probability or consequences of any event previously evaluated in the updated final safety analysis report (UFSAR). It does not introduce the possibility of an accident of a different type or a malfunction with a different result. The proposed EOP changes do not result in a design basis limit for a fission product barrier being exceeded or altered, or result in departure from a method of evaluation described in the Safety Analysis Report (SAR) used in establishing the design bases or in the safety analyses.

Attachment 3

10 CFR 50.59 REPORT FOR 2016
MILLSTONE POWER STATION UNIT 3

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

MPS3

EVAL-ENG-RSE-M3C18, Rev. 0

Reload Safety Evaluation Millstone Unit 3 Cycle 18

The proposed activity subject to evaluation considers final resolution of Westinghouse Nuclear Safety Letter (NSAL) 14-5 which communicates the lower than expected critical heat flux (CHF) results obtained at Westinghouse's ODEN CHF test loop. Specifically, the activity requires an additional flow quality check on Departure from Nucleate Boiling Ratio (DNBR) results obtained using the WRB-2M and WRB-2 CHF correlations within their NRC-approved applicability ranges. When a DNBR analysis is shown to exceed the applicable quality threshold, a conservative margin reduction (penalty) is applied which effectively uses up all of the DNBR margin to the approved 95/95 DNBR limit. The flow quality check is performed to verify DNBR results in the non-conservative sub-region identified in NSAL-14-5 are not obtained.

The quality check is considered an addition to the WRB-2M (WCAP-15025-P-A) and WRB-2 (WCAP-10444-P-A) methodologies, which constitutes a change in methodology that requires a 10 CFR 50.59 evaluation under criterion (viii). Review of the methodology change determined the additional check on flow quality does not constitute a departure from a method of evaluation since the check and its associated penalty application yield conservative (more limiting) or essentially the same results as compared to previous analyses which do not incorporate the check.

Further, the proposed activity does not require evaluation under 10 CFR 50.59 criterion (i-vii). The test data supporting WRB-2M and WRB-2 described in WCAP-15025-P-A and WCAP-10444-P-A remains valid and unchanged. As such, the correlations are accurate with regards to the range of applicability for the correlation and the licensed DNBR limits. This activity does not require a change to a design basis limit for a fission product barrier (DBLFPB) or exceed a DBLFPB.

In conclusion, NRC review and approval is not required to implement the additional flow quality check on DNBR results obtained using the WRB-2M and WRB-2 CHF correlations, as the activity does not result in a departure from a methodology as described in the FSAR. This is the only activity requiring a 10 CFR 50.59 evaluation to support implementation the MPS3 Cycle 18 (M3C18) reload design discussed in EVAL-ENGRSE-M3C18, Rev. 0. Therefore, EVAL-ENG-RSE-M3C18, Rev. 0 can be implemented under the provisions of 10 CFR 50.59.

MPS3

ETE-MP-2016-1033, Revision 0

Impact of Millstone Unit 3 ISFSI Fuel Handling Process

The activity that is being evaluated is the Technical Requirements Manual (TRM) Section 3/4.9.7 which prohibits loads in excess of 2,200 pounds from travel over fuel assemblies in the storage pool. The TRM currently provides no exceptions for a single failure proof load lift. The TRM is information incorporated by reference into the FSAR per FSAR Section 1.11 which is subject to the change controls of 10 CFR 50.59. MPS3 independent spent fuel storage installation (ISFSI) operations in the SFP introduce a heavy load lift of the dry storage canister (DSC) top shield plug which weighs 7,230 pounds that occurs over a fuel loaded transfer cask/dry storage canister (TC/DSC) that sits in the cask pit of the SFP which is not located in the area where fuel assemblies are stored in the storage pool spent fuel racks. The lift of the DSC top shield plug, which is in excess of 2,200 pounds meets single failure proof criteria in accordance with the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

A change to the technical requirements and bases for TRM section 3/4.9.7 is being made to include an exception for single failure proof lifts in the cask pit in accordance with the guidelines of NUREG-0612.

The 10 CFR 50.59 evaluation concluded the following:

- ISFSI operations at MPS3 do 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 fuel handling accidents evaluated in FSAR Chapter 15 in the spent fuel pool include the dropping of a spent fuel assembly onto another fuel assembly in the SFP, the drop of a non-spent fuel assembly component into the SFP and spent fuel cask drop accidents (SFCDAs).

- ISFSI operations at MPS3 do not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety evaluated in the SAR. The spent fuel shipping cask crane (SFSCC) is the only component allowed to perform the lift of a heavy load over a fuel loaded TC/DSC in the cask pit of the storage pool and the SFSCC and associated rigging is designed to meet single failure proof requirements in accordance with NUREG-0612.

- ISFSI operations at MPS3 do not result in more than a minimal increase in the consequences of an accident previously evaluated in the SAR. The fuel handling accidents evaluated in FSAR Chapter 15 in the SFP applicable to this activity include the drop of a non-spent fuel assembly component into the SFP and SFCDAs. ISFSI operations introduce heavy loads which are performed by single failure proof lifts and are therefore are not credible. No additional targets are affected by the accidents.

- ISFSI operations at MPS3 do not result in more than a minimal increase in the consequences of a malfunction of a SSC important to safety evaluated in the SAR. The SFSCC and associated rigging meets single failure proof requirements and therefore load lifts using this crane are not credible and a malfunction of this crane is not required to be postulated. Failure of the spent fuel bridge and hoist is the initiating accident for a fuel handling accident in the fuel building. ISFSI operations do not introduce any increase in the consequences of a malfunction of this SSC.

- ISFSI operations at MPS3 do not create a possibility for an accident of a different type than previously evaluated in the SAR. Absent any new accident initiators or any other new results from the malfunction of an SSC important to safety, there can be no accident of a different type created.

- ISFSI operations at MPS3 do 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 SFSCC and associated rigging meet the single proof criteria resulting in load drops not being credible. ISFSI operations have no impact on any mechanism which could cause a fuel assembly to become disengaged from the fuel handling tool and the spent fuel bridge and hoist is prohibited from lifting a heavy load in excess of 2,200 pounds over a fuel loaded TC/DSC in the cask pit.
- ISFSI operations at MPS3 do not result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered. The drop height of a spent fuel assembly on to another spent fuel assembly in the analysis of record bounds the drop height of a fuel assembly on to fuel assemblies in a TC/DSC. For ISFSI operations, the DSC top shield plug is a single failure proof lift and therefore a drop of this component is not credible resulting in any ruptured fuel assembly rods.
- ISFSI operations at MPS3 have no impact on any analytical methods that are described in the FSAR that demonstrate the design meets the design bases or that the safety analyses is acceptable.

Attachment 4

10 CFR 50.59 REPORT FOR 2016
MILLSTONE POWER STATION UNITS 2 & 3

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

MPS2 & 3

MPS3-EVAL-2016-0005

Temporary Modification, 345kV Switchyard Breaker Fault Clearing Time

The activity being evaluated is changing the 345 kilovolt (kV) transmission network protection system fault clearing times associated with breaker failure scenarios described in Millstone Power Station (MPS) Unit 2 (MPS2) and MPS Unit 3 (MPS3) Final Safety Analysis Reports (FSARs). This is associated with a temporary design change to breaker trip circuitry. The evaluation was prepared to address the impact to 345kV transmission network stability after a fault occurs and a switchyard breaker fails to open on a trip signal with a delay in the breaker failure scheme trip signal provided to another switchyard breaker.

Breaker clearing times are utilized as inputs to the transmission network stability analysis consistent with the requirements of the governing standards. An increase in fault clearing times will have an adverse affect on FSAR described protection system functions by increasing power swing of connected generators during transients encroaching on the transmission network generator rotor angle and voltage stability limits.

The MP2 and MP3 FSARs state "With any of the four Millstone 345kV transmission circuits out of service, the plant remains stable for any three-phase fault normally cleared (four cycles) or any one-phase fault with delayed clearing (eight cycles)." The temporary proposed change results in a slight increase in the breaker failure fault clearing time from 8 cycles to 8.66 cycles. ISO-NE reported that stability studies show that the system continues to remain stable and Millstone generators remain online after increasing the default clearing times (breaker failure scheme timing) to 8.66 cycles.

The offsite power system serves as the preferred electric power source to the MPS operating units to permit functioning of structures, systems, and components important to safety. The system provides sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents. This system has suitable redundancy in design and operation to provide reliable service to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies per General Design Criteria 17 of Appendix B to 10 CFR 50. The Safety Analysis Report (SAR) credits transmission system protective relaying systems for maintaining continuity of offsite power during various system transients by successfully isolating faulted equipment including consideration of failures of protective system components.

The 10 CFR 50.59 evaluation concluded 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 SAR.
- The change did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a safety system 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.

Attachment 5

COMMITMENT CHANGE REPORT FOR 2016
MILLSTONE POWER STATION

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

Thermal Performance testing of Service Water Cooled Heat Exchangers at MPS3 - RCR-42037

In letters dated May 6, 1998 and February 28, 2001, Northeast Nuclear Energy Company (NNECO), the former owner of Millstone Power Station (MPS), submitted supplemental information regarding the MPS Unit 3 (MPS3) response to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment." Those letters addressed the commitment to perform thermal performance testing of certain safety-related heat exchangers associated with Item II of GL 89-13. Dominion Nuclear Connecticut Inc. (DNC) changed the commitment identified in the May 6, 1998 letter. DNC will perform either thermal performance testing or frequent, regular maintenance of safety-related heat exchangers associated with Item II of GL 89-13. Either method is acceptable per GL 89-13, Supplement 1, dated April 1, 1990.

MPS Unit 2 (MPS2) currently performs frequent, regular maintenance in lieu of thermal performance testing of safety-related heat exchangers associated with Item II of GL 89-13. Maintenance frequency changes are based on engineering evaluation of operational history, previous baseline test data, and/or subsequent test data.

For both MPS2 and MPS3, the method of ensuring the heat transfer capability of these heat exchangers will be based on engineering evaluation of operational history, previous baseline test data, and/or subsequent test data. This method of evaluation is consistent with the guidance in GL 89-13 and Supplement 1 for Item II to ensure the optimum maintenance frequencies are established for the safety-related heat exchangers cooled by service water.

The information provided in response to GL 89-13 was referenced in the application and subsequent safety evaluation associated with license renewal of MPS. DNC considers the changes consistent with the guidance in GL 89-13 and Supplement 1 for Item II. Changes to the commitments associated with GL 89-13 and license renewal have been performed and documented in accordance with DNC's commitment change procedure.