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Dominion

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U.S. Nuclear Regulatory Commission
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Washington, DC 20555

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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 2008 AND 2009,
AND THE COMMITMENT CHANGE REPORT FOR 2009

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) and Unit 3 (MPS3) are submitted via Attachments 1 and 2 respectively for the years 2008 and 2009. There were no changes made to the facility for Millstone Power Station Unit 1 (MPS1) and the ISFSI.

During 2009, there were no commitment changes for MPS1, MPS3 or ISFSI, however, there were eleven commitment changes made for MPS2. Attachment 3 provides the submittal of these changes for MPS2. 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. William D. Bartron at (860) 444-4301.

Sincerely,

R. K. MacManus
Director, Nuclear Station Safety and Licensing

IE47
NRR

Attachments: 3

Commitments made in this letter: None.

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Attachment 1

10 CFR 50.59 REPORT FOR 2008 AND 2009

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

Millstone Power Station Unit 2

S2-EV-06-0003 Revision 02

FSC 08-MP2-019

DCR M2-05009 Revision 01

Replacement of Emergency Core Cooling System (ECCS) Sump Strainer per Generic Letter (GL) 2004-02 (GSI-191)

Revision 0 of Design Change Record (DCR) M2-05009 replaced the ECCS sump strainer, increasing the flow area of the strainer from approximately 100 square feet to approximately 6000 square feet. The screen mesh size of the ECCS sump strainer was reduced from 3/32 inch to 1/16 inch. In response to generic letter GL 2004-02, the previous ECCS sump strainer was evaluated and determined to be unsatisfactory based on the issues identified in the generic letter. The new ECCS sump strainer was installed in the fall 2006 refueling outage in anticipation of meeting the generic letter requirements. Revision 1 of this design change implemented new methods developed by NEI, the PWR Owners Group, and Dominion for evaluating debris generation and transport, including chemical effects. Revision 1 of the DCR also implemented new methods developed to address the potential impact on the long term decay heat removal due to debris blockage of structures, systems, and components (SSCs) downstream of the ECCS sump strainer. Revision 2 to the 10 CFR 50.59 evaluation evaluates the new ECCS sump strainer based on the methods developed to address the issues raised in GL 2004-02.

The ECCS sump strainer is a passive component credited during the recirculation phase of a loss of coolant accident (LOCA). The design function of the strainer is to prevent potentially damaging debris from entering the ECCS and containment spray (CS) systems while minimizing the head loss across the screen to ensure satisfactory ECCS and CS pump operation with the pump suction aligned to the containment sump. No other SSC is affected by this change. The ECCS sump strainer cannot cause an accident. It will not increase the probability of an accident previously analyzed in the Updated Final Safety Analysis Report. The sump strainer is designed to the codes and standards appropriate for its safety related function. The removal of the divider screen design feature in the new ECCS sump strainer does not increase the likelihood of a malfunction previously evaluated or create the possibility of a malfunction with a different result than previously analyzed. Adequate net positive suction head margin for the ECCS and CS pumps has been demonstrated considering the debris loading on the replacement ECCS sump strainer. The debris loading on the SSCs downstream of the replacement sump strainer will not jeopardize the core or containment heat removal safety functions. There is no increase in the likelihood of a failure of the SSCs. These activities did not increase the dose consequences or the challenges to the fission product barriers for any previously analyzed accident. With the exception of the LOCA, all of the Chapter 14 accident analyses are unaffected by this change and remain bounding. The Loss of Coolant accident was reanalyzed. The peak clad temperature increased by 2°F to 1825°F, well below the established 10 CFR 50.46 acceptance criteria of 2200°F.

Millstone Power Station Unit 2

S2-EV-08-0001 Revision 00

OP 2304H Revision 001-00

Boric Acid Addition to Chemical Volume and Control System (CVCS) from Spent Fuel Pool Cask Laydown Pit Infrequently Conducted or Complex Evolution (ICCE)

This 10 CFR 50.59 evaluation reviewed a revision to operating procedure OP 2304H, "Boric Acid Addition to CVCS from Spent Fuel Pool Cask Laydown Pit (ICCE)." This procedure uses a procedurally controlled temporary modification to use the spent fuel pool cask laydown pit as a source of borated water to be injected into the reactor coolant system (RCS) during a refueling shutdown. This evaluation is a revision of a 10 CFR 50.59 evaluation originally approved and performed in 2003 (Evaluation Number S2-EV-03-0009) to address potential leakage of the cask pit gate seal and the effects of this leakage on the operation of the spent fuel pool cooling system.

The activities proposed by this procedure will not increase the probability of an accident previously analyzed or create the possibility of an accident of a different type. These proposed activities will not increase the dose consequences or challenges to the fission product barriers for any previously analyzed accident. All of the Chapter 14 accident analyses are unaffected by this change and remain bounding. Use of this additional borated water flow path will be terminated by manual operator actions should a large break loss of coolant accident (LOCA) occur. This manual action is required to avoid the reactor vessel boron concentration building up to the precipitation limit following a large break LOCA, potentially impacting long term core cooling. Credit for this manual operator action is qualitatively justified based on the lower RCS decay heat boil off rates and the relatively long timeframe post-LOCA when boron precipitation is of concern. In addition, the proposed activities will not create the possibility of a malfunction of a structure, system, or component important to safety and will not create the possibility of a malfunction with a different result than previously analyzed. The only systems potentially impacted are the boric acid system, charging system, and spent fuel cooling system. The evaluation concludes these systems are not adversely impacted by the proposed activities.

Millstone Power Station Unit 2

S2-EV-08-0002 Revision 00

FSC 08-MP2-007

Updated Final Safety Analysis Report (UFSAR) Change: Clarify Assumptions Regarding Main Steam Line Break (MSLB) Dose Analysis

UFSAR Section 14.1.5.3 and Table 14.1.5.3-1 were updated to reflect a revised MSLB dose analysis. The previously approved analysis credited Enclosure Building Filtration System (EBFS) for mitigation of MSLB in containment. The revised analysis did not credit EBFS and also used a more appropriate release model for secondary side bulk liquid from the affected steam generator.

Not crediting EBFS resulted in an increase in dose from the affected steam generator release of primary to secondary liquid. This increase in dose is less than the available margin and is considered to be no more than a minimal increase.

While there was a dose increase due to not crediting EBFS, there was a dose reduction due to the new secondary side bulk liquid release model. The previous model did not credit containment for holdup; it released the activity direct to the environment. The new model credits containment and associated Technical Specification leak rates; these parameters were previously credited in the analysis for the primary to secondary leak rate release.

Doses reported in the UFSAR for MSLB in containment were not changed. This ensured that no additional margin was gained from this change.

This change was solely an update to the UFSAR. There are no changes to any structures, systems, or components (SSCs) or procedures as a result of this change. This change did not increase the frequency of occurrence of an accident or malfunction previously analyzed in the UFSAR. It 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 UFSAR. The change did not introduce any new failure modes that create the possibility of an accident of a different type or a malfunction with a different result than previously analyzed. There are no additional challenges to fission product barriers.

The net result of both changes is a reduction in offsite and control room doses. While one change (not crediting EBFS) does result in an increase in the consequences of an accident previously evaluated in the UFSAR, that increase was no more than a minimal increase.

Millstone Power Station Unit 2

S2-EV-08-0004 Revision 00

LBDCR 07-MP2-010 Revision 00

Changes to the Millstone Unit 2 (MPS2) Technical Requirements Manual (TRM)

This10 CFR 50.59 evaluation was written to support changes made to the MPS2 TRM. One of the changes made was the introduction of parallel yet distinct terminology into the TRM which distinguishes the terms for requirements, surveillances and functional equipment from the terms used in plant Technical Specifications (TS). These terms are defined in a new section, 1.0. Their use is made consistent with plant TS in that they are capitalized when used in the text of the TRM. This is considered a non-intent change which resulted in textual changes in a number of TRM sections. A table of the parallel terms follows:

Technical Specification Term	Technical Requirements Manual Term
Technical Specification (TS)	Technical Requirements Manual (TRM)
Limiting Condition for Operation (LCO)	Technical Requirement (TR)
Surveillance Requirement (SR)	Technical Surveillance Requirement (TSR)
Specification	Requirement
Operable/Operability	Functional/Functionality

Section 3/4.0, "Technical Requirement (TR) And Technical Surveillance Requirement (TSR) Applicability", was added to provide guidance for situations that might occur that went beyond the individual TRM sections.

Another change was the removal of the requirement to submit special reports should Meteorological Monitoring Instrumentation (TRM Section 3.3.3.4) be out of service for time periods of 7 and 30 days, or should the reactor vessel head vents systems (TRM Section 3.4.11) be non-functional for a period exceeding 30 days. Instead of the reporting requirement, non-functionality of this equipment will be entered into Millstone Power Station's corrective action program. The instrumentation system does not directly interact with systems used to control the plant or to respond in the event of an accident, and the reactor vessel head vents are not credited nor connected to any system credited in any design basis events in the safety analysis. Therefore, the 10 CFR 50.59 evaluation questions were answered in the negative for these changes.

Finally, the TRM change contained several instances in which the shutdown requirement for degraded equipment has been eliminated and replaced with an action to manage the risk impact in accordance with the requirement of 10 CRF 50.65(a)(4). The affected sections were TRM 3.1.2.2, "Boron Flow Paths - Operating," TRM 3.1.2.4, "Charging Pumps – Operating," TRM 3.1.2.6, "Boric Acid Pumps – Operating," and TRM 3.1.2.8, "Borated Water Sources –Operating." These changes were only for configurations in which the change did not adversely impact a structure, system, or component (SSC) credited in the Updated Final Safety Analysis Report (UFSAR)

Chapter 14 analysis. It was noted that one of the Borated Water Sources, the Refueling Water Storage Tank, mentioned in TRM 3.1.2.8 is credited in the safety analysis, but to the extent that it is credited, it is governed by a TS which is unaffected by this TRM change package. None of the TRM changes described above created a new type of event nor resulted in an event with a different result, and is not a departure from any evaluation methodology as described in the UFSAR. The consequences and frequency of accidents as presented in the UFSAR were unaffected.

Millstone Power Station Unit 2

S2-EV-09-0001 Revision 00

FSC MP2-UCR-2009-002 Revision 00

Chapter 14 Revision to the Millstone Unit 2 (MPS2) Updated Final Safety Analysis Report (UFSAR)

A UFSAR change was generated to revise and update the Chapter 14 event write-ups of the MPS2 UFSAR relative to the credit for and description of charging pumps for the inadvertent opening of the Power Operated Relief Valve (PORV) and small break Loss of Coolant Accident (LOCA) events. Regarding the UFSAR Section 14.6.1.6 inadvertent opening of the PORV event analysis results write-up, the statement that charging and safety injection was adequate to prevent core uncovery was changed to indicate that high pressure safety injection, by itself, is adequate to prevent core uncovery. Additionally, the UFSAR general description of the small break LOCA transient was changed to delete charging from the range of break size scenario descriptions. This deletion was made to be consistent with the description of the analysis contained in the following section which correctly indicates no charging pump flow is credited in the analysis of the event. This small break LOCA UFSAR change is an editorial clarification that does not involve a change to the small break LOCA analysis of record for MPS2.

Regarding the analysis results section of the UFSAR inadvertent opening of the PORV event write-up, both short-term and long-term results are described as follows:

The short-term results of the inadvertent opening of the PORV analysis presented in UFSAR Section 14.6.1.6 demonstrates that the departure from nucleate boiling ratio (DNBR) and fuel centerline melt design limits are not violated as a result of the rapid reactor coolant system (RCS) depressurization caused by the opening of both PORVs. The core average temperatures, pressure, level and flow rates following inadvertent opening of the PORV analysis are modeled using the NRC approved computer simulation code PTSPWR. A second computer code uses these predicted core average conditions to determine the hot channel most limiting DNBR. The Thermal Margin/Low Pressure (TM/LP) reactor trip generates a reactor trip in time to prevent violation of these fuel design limits. The reactor trip occurs approximately eight seconds after the PORV opens (at zero seconds). The computer simulation of this accident is terminated by twenty seconds, which is long enough to demonstrate that the DNBR and fuel centerline melt acceptance criteria are not violated. The analysis of this short-term portion of the event has not changed. The existing analysis of record assumed that the charging pumps were disabled and letdown was maximized at the initiation of the event to maximize the RCS depressurization, and therefore, the challenge to the DNBR fuel design limits. As such, charging was not credited for this short-term portion of the inadvertent opening of the PORV event.

The long-term results of the inadvertent opening of the PORV event presented in UFSAR Section 14.6.1.6, which demonstrated that charging and the safety injection system was adequate to prevent core uncovery, was supported by a calculation comparing the inventory discharge rate out both PORVs to the inventory addition rate of the high pressure safety injection (HPSI) and charging pumps. The change made was also supported by a comparison of the inventory discharge rate out both PORVs to the inventory addition rate of the HPSI system, without taking credit for the charging pumps. This more conservative comparison demonstrates that core uncovery would not occur with HPSI by itself following the inadvertent opening of the PORV event. Since the revised analysis method yielded more conservative results by increasing the potential for core uncovery, this change to an element of the method of evaluation did not result in a departure from the method of evaluation described in the UFSAR.

This change did not increase the probability of an accident previously analyzed in the UFSAR, as it does not increase the likelihood of the PORV inadvertently opening causing a decrease in RCS inventory. The change did not result in more than a minimal increase in the likelihood of a malfunction previously evaluated. The change also did not introduce any new failure modes that create the possibility of an accident of a different type or a malfunction with a different result than previously analyzed. These activities do not increase the dose consequences for any previously analyzed accident. These activities do not increase the challenges to fuel design limits for the inadvertent opening of the PORV event, hence the fuel fission product barrier is not adversely impacted for this or any other previously analyzed accident. The challenges to the RCS and containment fission product barriers for this or any other previously analyzed accident remain unchanged. All of the UFSAR Chapter 14 accident analyses continue to meet the established acceptance criteria.

Millstone Power Station Unit 2

S2-EV-09-0002 Revision 00

MMOD DM2-00-0200-09 Revision 00

OP 2305B Revision 00-001

Temporary Credit for Local Manual Action Outside Control Room, Refueling Water Storage Tank (RWST) Purification Subsystem

Millstone Unit 2 (MPS2) RWST inventory requires purification to remove radioactive impurities and to attenuate Refueling Outage 19 (2R19) doses. MPS2 contains an RWST to Spent Fuel Pool (SFP) purification sub-system described in the Updated Final Safety Analysis Report (UFSAR). However, the SFP purification system is classified as non-safety related (NSR) and portions are non-seismically qualified. This condition was identified in Condition Report CR324869, "RWST Inventory May be Cross Tied IAW 2305 to Non-Safety Related Equipment." Because of RWST inventory diversion concerns, administrative controls were put in place to preclude RWST purification via the SFP purification system by red tagging stop valve 2-RW-27.

Temporary Plant Change MMOD DM2-00-0200-09 is based upon the view that current design requires 2-RW-27 to be in a closed position during Modes 1, 2, 3 and 4 and that this required closed position was the technical bases for the purification sub-system's NSR classification. This MMOD was written to provide the technical bases for RWST purification via the SFP purification sub-system during 2R19. This was accomplished, in part, by crediting local manual action outside the control room to isolate 2-RW-27 during a plant transient that would result in a manual or automatic reactor trip or during a seismic event. The temporary plant change also changed the safety classification boundary within the purification pump suction piping from valve 2-RW-27 to contingency valves 2-RW-25, 2-RW-28B, and 2-RW-19 to address 2-RW-27 active failure.

The following changes were made to operating procedure OP 2305B, "RWST Purification," Revision 00-001, in support of MMOD DM2-00-0200-09:

- Added an administrative limit for minimum required measured RWST inventory, when the RWST to SFP purification sub-system pathway is aligned to support RWST purification. This administrative limit is intended to compensate for potential RWST inventory diversion due to a postulated purification sub-system pressure boundary failure.
- Included a requirement to stop the running purification pump and to isolate 2-RW-27 via a dedicated operator given a plant transient that results in a manual or automatic reactor trip, or if a seismic event is self-evident to the dedicated operator. This measure is intended to limit RWST inventory diversion to the purification system given a postulated purification sub-system pressure boundary failure.
- Added a prerequisite to close suction valve 2-RW-28B if standby purification pump P14A is used for RWST purification, and a prerequisite to close suction valve 2-RW-25 if standby purification pump P14B is used for RWST purification. These measures are intended to preclude the possibility of two purification pumps assisting in RWST inventory diversion given the potential for seismic

event relay chatter concurrent with a postulated purification sub-system pressure boundary failure.

This 10 CFR 50.59 evaluation concluded that the temporary plant change and associated procedure changes have no adverse impact upon accidents, malfunctions, fission product barriers, or operator response to a design basis event and operational transients. There was not a more than minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component important to safety previously evaluated in the UFSAR because the credited manual action outside the control room satisfied the following criteria derived from NEI 96-07, R1:

- The action (including required completion time) is reflected in plant procedures and operator training programs,
- The licensee has demonstrated that the action can be completed in the time required considering aggregate affects, such as workload and environmental conditions expected to exist when the action is required,
- The evaluation of the change considers the ability to recover from credible errors in performance of manual actions and the expected time required to make such a recovery,
- The evaluation considers the effect of the change on plant systems.

Millstone Power Station Unit 2

S2-EV-09-0003	Revision 00
EVAL-ENG-RSE-M2C20	Revision 01
LBDCR 09-MP2-025	Revision 00
FSC MP2-UCR-2009-021	Revision 00

Cycle 20 In-Core Instrument (ICI) Thimble Tube Replacement

Revision 01 of the Millstone Unit 2 (MPS2) Cycle 20 Reload Safety Evaluation (RSE) describes two changes made for MPS2 Cycle 20 operation. These changes are an increase in the primary system lithium concentration performed to support true constant 7.2 pH operations. The second change is an evaluation of the impact of an error in the installation of replacement ICI thimble tubes that left twenty-six (26) of the forty-five (45) thimble tubes 1.375 inches shorter than designed.

The increase in lithium was performed to comply with EPRI chemistry recommendations and was reviewed in accordance with 10 CFR 50.59 separately. The screen for FSC MPS2-UCR-2009-020, which incorporated the increase in allowed lithium concentration into the MPS2 Updated Final Safety Analysis Report (UFSAR), found that no evaluation was required. Therefore, while the increase is mentioned as a change in terms of the RSE itself, no further consideration of this aspect was required in this evaluation.

New physics and setpoint analyses were performed by AREVA to quantify the potential impact of the shortened thimble tubes resulting in a misalignment of the associated twenty-six (26) ICI assemblies. AREVA found that by using existing methodologies they were able to accommodate any potential effect of the misaligned ICIs within the discretionary conservatism that was present between the cycle specific thermal hydraulic behavior and existing plant Limiting Condition for Operation and Limiting Safety Systems Settings setpoints.

This 10 CFR 50.59 evaluation concluded that the analyses performed to evaluate the mispositioned ICIs did not adversely impact the frequency or results of the safety analysis presented in the UFSAR and did not introduce a new accident or structures, systems or components failure mode not currently analyzed in the UFSAR. The analyses were performed with the current licensed analytical methods used by AREVA in support of MPS2. The change resulted in no change to the values of or the approach to the design basis limits of the three fission product barriers.

Attachment 2

10 CFR 50.59 REPORT FOR 2008 AND 2009

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

Millstone Power Station Unit 3

S3-EV-07-0002 Revision 0

FSC 07-MP3-058

Changes Associated with the Millstone Unit 3 (MPS3) Charging System Alternate Minimum Flow (AMF) Line Modifications

Modifications to the MPS3 Charging System AMF Line were completed during Refueling Outage 3R11, along with follow-on calculations. These modifications re-routed the AMF lines, removing the relief valves, and adding manual isolation valves downstream of Motor Operated Valves 3CHS*MOV8512A & B. Design Change Record (DCR) M3-05002 justified these plant changes. This Updated Final Safety Analysis Report (UFSAR) Change was initiated to include UFSAR changes identified as part of the Design Change implementation. It was determined that Safety Evaluation S3-EV-07-002, previously reviewed and approved as part of the DCR M3-05002 design change, fully bounds these UFSAR changes.

Affected sections of the UFSAR were updated to reflect the removal of the relief valves from the AMF lines, the change in the maximum time available to open the Power Operated Relief Valve blocking valves based on the revised inadvertent Safety Injection (SI) analysis, and the changes in net positive suction head. Additionally, a note was added to the bottom of Table 6.3-1 to indicate nominal curves were used for the Inadvertent SI analysis, and to update Figures 15.5-1, 15.5-2, and 15.5-3 accordingly.

The changes to the MPS3 AMF path for the charging pumps replaced the relief valves in the AMF path with a locked open globe valve. This results in flow to the Refueling Water Storage Tank when a SI Signal is present. Westinghouse evaluated the change and concluded that the change met the criteria for the MPS3 UFSAR Chapters 6 and 15 events. The change did not create a new type of event or result in an event with a different result. It has no negative impact on any fission barrier, does not increase the consequences of an event, and does not result in any departure from any evaluation methodology described within the UFSAR.

Millstone Power Station Unit 3

S3-EV-08-0002 Revision 0

OD MP3-016-07 Revision 004

Operability Determination (OD) MP3-016-07: Gas Accumulation in High Pressure Safety Injection (SIH) Discharge Piping

OD MP3-016-07, Revision 004, addressed the condition where an unacceptable amount of gas was found in the SIH discharge piping while performing surveillances for gas voids in the Emergency Core Cooling System (ECCS). This OD contained several compensatory measures. The guidance for implementing compensatory measures associated with venting containment penetration 98 via vent valve 3SIH*V967 inside containment is not contained within an approved plant procedure. As such, that compensatory measure was reviewed in accordance with 10 CFR 50.59.

Compensatory measures associated with venting containment penetration 98 involve performing Ultrasonic Testing (UT) of the SIH common discharge header piping outside the containment penetration at least once per four days. If the UT identified that the containment penetration was less than an established criterion, a containment entry would be made within three days to vent the penetration at 3SIH*V967. While the penetration is being vented in accordance with these compensatory measures, the potential exists for the diversion of some SIH flow through a one-quarter inch vent port to containment instead of the reactor coolant system (RCS) cold leg injection points. An engineering review determined that the potential for diversion of SIH flow to containment is sufficiently limited such that the required minimum SIH flow injected into the cold legs would continue to meet the flow rate required to satisfy the Updated Final Safety Analysis Report (UFSAR) Chapter 15 Loss of Coolant Accident (LOCA) and non-LOCA analyses assumptions. This engineering determination was based on using actual ECCS pump performance as opposed to the degraded pump performance. This assumption is considered reasonable since recent past (since January 2004) in-service test pump performance indicates the SIH pumps have been performing very close to their reference head values. No additional pump degradation is anticipated in the near-term future while this compensatory measure is in place.

The proposed change does not increase the probability of an accident previously analyzed in the UFSAR as it will not increase the likelihood of the ECCS inadvertently actuating causing an increase in RCS inventory. The proposed change does not result in more than a minimal increase in the likelihood of a malfunction previously evaluated. The proposed change does not introduce any new failure modes that create the possibility of an accident of a different type or a malfunction with a different result than previously analyzed. These proposed activities do not increase the dose consequences or the challenges to the fission product barriers for any previously analyzed accident. The UFSAR Chapter 15 accident analyses are unaffected by this change, and remain bounding.

Millstone Power Station Unit 3

S3-EV-08-0003 Revision 0

DCR M3-07021 Revision 00

DCN DM3-00-0269-07

LBDCR 08-MP3-015

Changes to the Millstone Unit 3 Control Room Emergency Ventilation (CREV) System

This activity modified the control room ventilation system for accident conditions. There were two distinct modifications: disable the automatic control room pressurization on a Control Building Isolation (CBI) signal by the control room envelope pressurization system, and automatically initiate the CREV system in the Recirculated Outside Filtered Air Mode on a CBI signal. The first modification was approved under Millstone Unit 3 Technical Specification (MPS3 TS) Amendment 232 (Accession Number ML061990025) which removed the control room envelope pressurization system from the MPS3 TS. This change was addressed under the license amendment request submittal, and is not addressed by this evaluation. License Basis Document Change Request (LBDCR) 08-MP3-015 updated the MPS3 Technical Requirements Manual in accordance with these design changes.

The second modification was to automatically initiate CREV on a CBI signal. The modification provides additional margin to the Fuel Handling Accident Control Room dose analysis. The change did 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 and does not result in any departure from any evaluation methodology as described in the Updated Final Safety Analysis Report (UFSAR).

The modification did result in more than a minimal increase of an accident described in the UFSAR as it existed at the time the evaluation was performed. That was due to the modification being combined with the Stretch Power Uprate (SPU). However, the consequences resulting from the modification are less than those documented in the SPU License Amendment request. Those results were reviewed by the NRC and found to be acceptable (MPS3 TS Amendment 242, Accession Number ML082180137). Therefore, this modification did not result in an increase in consequences as presented in the UFSAR resulting from the SPU License Amendment request.

Millstone Power Station Unit 3

S3-EV-08-0005	Revision 0
EVAL-ENG-RSE-M3C13	Revision 00
FSC MP3-UCR-2009-001	Revision 00

Incorporation of Westinghouse PAD 4.0 Fuel Performance Model to Millstone Unit 3 Updated Final Safety Analysis Report (UFSAR) References

The reload fuel rod design for Millstone Unit 3 (MPS3) Cycle 13 was performed by Westinghouse using their NRC-approved PAD 4.0 models, which was not then referenced in the MPS3 UFSAR.

Westinghouse uses their NRC-approved fuel performance models to predict the performance of reload batches of fuel, and verify that the fuel will comply with all pertinent design criteria for the projected operating conditions. These evaluations ensure the fuel integrity is maintained throughout the projected operating life of the fuel. The same models may also be used to the power level at which fuel centerline melt occurs, and to generate fuel temperature and pressure data that may be used as inputs to the safety analyses and the Loss of Coolant Accident (LOCA) analysis.

For MPS3 Cycle 13, Westinghouse used updated fuel performance models to perform reload design evaluations for MPS3. The models were revised to provide calculation results that are more consistent with in-reactor experience. The new models are based on a wide range of Pressurized Water Reactors (PWR) fuel designs and operating conditions that bound the MPS3 fuel design and typical reload parameters, so application of these models to evaluate MPS3 fuel is appropriate. The fuel performance code that incorporates the new models also uses numerous models that were incorporated into the MPS3 UFSAR (by reference). No changes to the fuel rod design criteria (e.g. limits on rod internal pressure, clad stress and strain, clad fatigue, or clad corrosion) or to safety analysis limits were made in association with these model changes.

As discussed in the industry guidelines for 10 CFR 50.59 implementation given in NEI 96-07 Revision 1, when 10 CFR 50.59 evaluations are performed only to support the adoption of a new methodology, criteria 10 CFR 50.59(c)(2)(i-vii) are not applicable. Accordingly, this evaluation of the implementation of the new Westinghouse fuel performance models addresses only criterion 10 CFR 50.59(c)(2)(viii).

In general, the use of new models or methods to evaluate fuel compliance with the design criteria cannot be implemented under 10 CFR 50.59 without prior NRC review and approval. However, the new models that were used for the reload evaluation for MPS3 were submitted to the NRC for review by the fuel vendor (Westinghouse). The NRC had completed its review of the models and approval was granted in April 2000 for Westinghouse to apply the new models to fuel licensing applications. As described in Section 4.3.8 of NEI 96-01, Revision 1 "Guidelines for 10 CFR 50.59 Implementation," replacing a methodology described in the UFSAR (e.g., the old fuel performance models) with a new NRC-approved methodology that is being used for its intended application within the limitations of the safety evaluation report (design evaluations and licensing of Westinghouse fuel) is not considered a departure from a method of evaluation described in the UFSAR. Use of these new fuel performance models is therefore an acceptable change in methodology under 10 CFR 50.59, and no separate submittal for prior NRC review is required by Dominion. The MPS3 UFSAR has since been updated to incorporate these new models.

Some MPS3 accident analyses, such as the current Large Break LOCA analysis of record, are based on data generated with the PAD 4.0 fuel performance models. Implementation of these models was approved by the NRC on a forward fit basis, so any accident evaluations based on fuel temperatures and pressures predicted with older models that may remain in the MPS3 UFSAR do not require reanalysis. The new fuel performance models predict lower fuel rod gas pressures, particularly at the end of life, and slightly lower fuel temperatures than were calculated with the older models. As lower fuel temperatures and gas pressures provide more margin to the design limits, any accident analyses based on the older models would conservatively bound the results of analyses using input data calculated with new models.

Millstone Power Station Unit 3

S3-EV-09-0001 Revision 00

FSC MP3-UCR-2009-007 Revision 00

Updated Final Safety Analysis Report (UFSAR) Update in the Calculated Maximum Containment Structure Liner Temperature

The maximum containment structure liner temperature was initially determined using the assumption that the limiting conditions for the peak containment temperature would also be limiting for the peak liner temperature analysis. Based on new insight, a revised analysis using more limiting initial condition assumptions for the limiting liner temperature scenario (the double ended rupture of the main steam line at hot zero power) has been performed. The analysis has concluded that the new peak liner temperature is marginally higher than the previously reported value (245.4°F versus 241°F).

The change was determined to be adverse to the design function of the containment liner. The accident mitigation design function for the containment liner to provide a leak tight membrane and to provide support for the quench spray and recirculation spray nozzle lines was potentially affected.

An increase in the calculated peak containment liner temperature from 241°F to 245.4°F resulted from the revised containment liner analysis. However, the revised peak temperature was below the design temperature limit of 280°F. Therefore, the containment liner is capable of performing its design function.

The change did not result in an increase in the frequency of occurrence of an accident or increase the likelihood of occurrence of a malfunction of a structure, system, or component important to safety. The proposed change does not create a new type of accident or create the possibility of a malfunction with a different result. It has no negative impact on any fission product barrier, does not increase the consequences of an event or malfunction and does not result in any departure from any evaluation methodology described in the UFSAR.

Attachment 3

COMMITMENT CHANGE REPORT FOR 2009

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

Implement NRC specified criticality accident requirements to allow use of NRC issued 10CFR50.68(b)(1) exemption for Millstone 2 (MPS2)

The NRC issued a new 10 CFR 50.68 regulation in November 2006, specifically so an exemption would not be needed. The new 10 CFR 50.68 can be complied with by MPS2, and therefore the exemption is no longer needed. The NRC stated that with the new regulation, licensees with an exemption (like Millstone) can choose to comply with the new rule, rather than use the exemption.

Per the current Licensing basis, the original commitments were required by the NRC to allow use of the 10 CFR 50.68(b)(1) exemption. These commitments were made by Dominion in letters to the NRC, serial numbers 04-771A, 04-592, and in the NRC letter (TAC MC5056) to Dominion, which stated the exemption was issued. These commitments were documented in Regulatory Commitment Records (RCR's) 42920-42925 and 42931-42935 (11 total commitments). The deletion of these commitments still preserves compliance, since the underlying basis of compliance is maintaining acceptable criticality accident requirements. In the past, when casks were placed in the MPS2 spent fuel pool for dry storage operations, Millstone had to meet both the 10 CFR 50.68(b)(1) exemption requirements and the 10 CFR 72.124 criticality accident requirements. The new 10 CFR 50.68(c) rule allows the 10 CFR 72.124 criticality accident requirements to be sufficient for 10CFR50.68 compliance. The NRC stated in the federal register for the new 10 CFR 50.68 rule, that compliance with Part 72 criticality accident requirements is essentially the same as compliance with 10 CFR 50.68 accident requirements. The criticality accident requirements of 10 CFR 72.124 are required to be met in order to be able to use a cask for ISFSI applications. Therefore, by meeting 10 CFR 72 criticality accident requirements, this preserves compliance, the exemption and its associated commitments can be deleted.

Using the 10 CFR 50.68(b)(1) exemption and its associated requirements has the same safety effect as using the new 10 CFR 50.68 rule, therefore the change does not negatively impact the ability of an SSC to perform its safety function or negatively impact the ability of personnel to ensure the SSC is capable of performing its intended safety function

Additionally, the commitments were not:

a) Explicitly credited as the basis for a safety decision in an NRC SER, or
b) Made in response to an NRC Bulletin or Generic Letter, or
c) Made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204, or
d) Identified as a long term corrective action in response to an NRC Notice of Violation, or
e) Made to minimize recurrence of an adverse condition such as long-term corrective action stated in an LER.

Therefore, the commitments can be changed without prior notification to the NRC.