Dominion Nuclear Connecticut, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060
Web Address: www.dom.com

Dominion®

~~PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390~~

January 26, 2016

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No. 15-571
NLOS/TFO R0
Docket No. 50-336
License No. DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2 LICENSE AMENDMENT REQUEST
SPENT FUEL POOL HEAT LOAD ANALYSIS

By letter dated July 21, 2010, Dominion Nuclear Connecticut Inc. (DNC) submitted a license amendment request (LAR) to the Nuclear Regulatory Commission (NRC) to revise Technical Specification (TS) 3/4.9.3.1, "Decay Time," for Millstone Power Station Unit 2 (MPS2). DNC proposed to reduce the minimum decay time for irradiated fuel prior to movement in the reactor vessel from 150 hours to 100 hours. On June 4, 2013, the NRC approved the LAR under License Amendment 315.

By letter dated April 29, 2015, the NRC informed DNC of an apparent violation (AV) pertaining to changes made by DNC under 10 CFR 50.59 to support the July 21, 2010 LAR. Specifically, the NRC determined that changes made to the MPS2 Final Safety Analysis Report (FSAR) Section 9.5 and MPS2 TS Bases Section 3/4.9.3 should have obtained prior NRC approval.

As a result of the Alternate Dispute Resolution (ADR) between DNC and the NRC staff, Confirmatory Order (CO) EA-13-188, dated August 26, 2015, requires DNC to submit a LAR requesting NRC approval of the spent fuel pool (SFP) heat load analysis and any associated TS changes.

In response to CO EA-13-188, DNC is requesting NRC approval of proposed changes to TS Bases 3/4.9.3 and FSAR Section 9.5. The SFP heat load analysis which supports the proposed TS Bases and FSAR changes are being submitted in support of the proposed changes.

Attachment 1 provides a description and assessment of the proposed changes to TS Bases 3/4.9.3 and FSAR Section 9.5. Attachments 2 and 3 provide the marked-up TS Bases page and FSAR pages, respectively. Attachment 4 provides the analyses which support the proposed changes to the FSAR. Attachment 4 contains information proprietary to Holtec International. Therefore, it is requested that the information be withheld from public disclosure in accordance with 10 CFR 2.390. The Holtec International affidavit and application for withholding are provided in Attachment 5.

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NRR

Attachment 4 contains ~~proprietary information that is being withheld from public disclosure under 10 CFR 2.390.~~ Upon separation of Attachment 4, this letter is decontrolled.

There have been no license amendments submitted since 2010 that affect or are affected by the proposed changes. The proposed changes to TS Bases 3/4.9.3 and FSAR Section 9.5 do not involve a significant hazards consideration pursuant to the provisions of 10 CFR 50.92. The proposed changes have been reviewed and approved by the Facility Safety Review Committee.

Issuance of this amendment is requested by January 31, 2017 with the amendment to be implemented within 60 days of NRC approval.

In accordance with 10 CFR 50.91(b), a copy of this request is being provided to the State of Connecticut.

If you have any questions regarding this request, please contact Wanda Craft at (804) 273-4687.

Sincerely,

Mark D. Sartain
Vice President – Nuclear Engineering

Vicki L. Hull
NOTARY PUBLIC
Commonwealth of Virginia
Reg. # 140542
My Commission Expires May 31, 2018

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Mark D. Sartain, who is Vice President – Nuclear Engineering of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 26TH day of January, 2016.

My Commission Expires: 5-31-18.

Vicki L. Hull
Notary Public

Attachments:

1. Description and Assessment of Proposed Changes to TS Bases 3/4.9.3 and FSAR Section 9.5.
2. Mark-up to Technical Specification Bases Page
3. Mark-up to FSAR Section 9.5 Pages
4. Holtec Analyses Supporting Proposed Changes to the FSAR
5. Holtec International Affidavit and Application for Withholding

cc: U.S. Nuclear Regulatory Commission
Region I
2100 Renaissance Blvd
Suite 100
King of Prussia, PA 19406-2713

Richard V. Guzman
NRC Senior Project Manager
U.S. Nuclear Regulatory Commission, Mail Stop 08 C2
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Millstone Power Station

Director, Radiation Division
Department of Energy and Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

ATTACHMENT 1

Description and Assessment of Proposed Changes to TS Bases 3/4.9.3 and FSAR Section 9.5

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

1.0 Summary Description

By letter dated July 21, 2010, Dominion Nuclear Connecticut Inc. (DNC) submitted a license amendment request (LAR) (Reference 7.1) to the Nuclear Regulatory Commission (NRC) to revise Technical Specification (TS) 3/4.9.3.1, "Decay Time," for Millstone Power Station Unit 2 (MPS2). DNC proposed to reduce the minimum decay time for irradiated fuel prior to movement in the reactor vessel from 150 hours to 100 hours. On June 4, 2013, the NRC approved the LAR under License Amendment 315 (Reference 7.2).

On July 9, 2014, the NRC informed DNC that the MPS2 change pertaining to decay time may be non-conservative based on the un-reviewed status of the removal of heat load considerations. As a result, Operations Standing Order SO-14-021 was implemented to prohibit use of the 100-hour decay time. SO-14-021 remains in place pending resolution of this LAR.

By letter dated April 29, 2015 (Reference 7.3), the NRC informed DNC of an apparent violation (AV) pertaining to changes made by DNC under 10 CFR 50.59 to support the July 21, 2010 LAR. Specifically, the NRC informed DNC that changes made to the MPS2 Final Safety Analysis Report (FSAR) Section 9.5 and MPS2 TS Bases Section 3/4.9.3 should not have been made without obtaining a License Amendment.

As a result of the Alternate Dispute Resolution (ADR) between DNC and the NRC staff, Confirmatory Order (CO) EA-13-188, dated August 26, 2015 (Reference 7.4), requires DNC to submit a LAR requesting NRC approval of the spent fuel pool (SFP) heat load analysis and any associated TS changes.

In response to CO EA-13-188, DNC is requesting NRC approval of proposed changes to TS Bases Section 3/4.9.3 and FSAR Section 9.5. The SFP heat load analyses which support the proposed TS Bases and FSAR changes are being submitted in support of the proposed changes.

2.0 Description of Proposed Changes

2.1 <u>Proposed Change to TS Bases 3/4.9.3</u>

DNC proposes to revise TS Bases Section 3/4.9.3 to remove reference to the SFP heat load analysis. Specifically, TS Bases Section 3/4.9.3 will be revised as follows (deleted text is struck through):

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel ~~ensures that sufficient time has elapsed to allow the decay heat load of the fuel to be within the assumptions of the spent fuel pool heat load analysis. This minimum requirement also~~ ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products so that the calculated radiological dose consequences of the fuel handling accident are bounding.

Attachment 2 provides the mark-up to TS Bases Section 3/4.9.3 based on the 2010 revision in effect when the changes were performed under 10 CFR 50.59. DNC is not proposing a revision to TS 3/4.9.3.1, "Decay Time," as previously approved by License Amendment 315. The conclusions of the NRC safety evaluation from License Amendment 315 remain valid.

2.2 Proposed Changes to FSAR Section 9.5

DNC proposes to revise FSAR Section 9.5 to allow fuel movement to start 100 hours after reactor subcriticality and proceed at an average rate of six assemblies per hour provided the Reactor Building Closed Cooling Water (RBCCW) temperature to the SFP cooling and shutdown cooling (SDC) heat exchangers (HXs) is maintained at less than or equal to 75°F. If 75°F cooling water is not achievable, fuel movement at an average rate of six fuel assemblies per hour could be permitted at 150 hours after subcriticality and then only with RBCCW temperatures less than or equal to 85°F. The proposed changes to FSAR Section 9.5 also address some typographical errors. Attachment 3 provides the marked-up FSAR Section 9.5 pages based on the 2010 revision in effect when the changes were performed under 10 CFR 50.59.

The SFP heat load analyses which support the proposed TS Bases and FSAR changes are provided in Attachment 4 for review.

3.0 SFP Cooling System

FSAR Section 9.5 describes the maximum heat load experienced by the SFP cooling system under various fuel load scenarios. The heat load is based on an assumed core offload beginning at 150 hours decay time and proceeding at a fuel movement rate of four assemblies per hour. Under these assumptions, the SFP bulk temperature can be maintained at or below the maximum normal operating temperature of 150°F specified in Technical Requirements Manual (TRM) Section 3/4.9.3.2 with an RBCCW temperature of less than or equal to 85°F.

The MPS2 TRM Section 3/4.9.3.3 specifies that the reactor remain in Mode 5 or 6 until decay time is greater than or equal to 616 hours or SFP heat load is less than 10.16 x 10^6 BTU/hr (British Thermal Units per Hour). This ensures that adequate cooling is available to maintain the SFP bulk temperature less than or equal to 150°F should a single failure occur in the SFP cooling system during plant operation.

The MPS2 TRM Section 3/4.9.3.2 limits SFP temperature to less than or equal to 150°F at all times. This ensures:

- Bulk water temperature will not exceed 200°F with the loss of SFP cooling assumed following a design basis Loss of Coolant Accident (LOCA). This ensures the post-LOCA function of the SFP cooling system.

- Temperature and humidity above the pool remain compatible with personnel comfort and safety requirements.

- Design temperature of the SFP cooling system, liner/building structures, and racks are not exceeded.

TRM Section 3/4.9.3.2 Bases identifies that the SFP cooling system and/or SDC system heat removal capabilities are required to be sufficient to maintain SFP bulk water temperature less than or equal to 150°F during Mode 5, Mode 6, or when the reactor vessel is defueled.

4.0 Technical Evaluation

4.1 Proposed Revision to TS Bases Section 3/4.9.3

DNC proposes to revise TS Bases Section 3/4.9.3 to remove reference to the SFP heat load analysis.

The unrevised TS Bases Section 3/4.9.3 specifies that the minimum requirement for reactor subcriticality prior to movement of irradiated fuel (decay time) is based on both the decay heat load limitations (FSAR Section 9.5.2) and the radiological dose consequences of the fuel handling accident (FHA) (FSAR Section 14.7.4). DNC has determined that the decay heat load limitations of the SFP heat load analysis do not meet the criteria of 10 Code of Federal Regulations (CFR) 50.36(c)(2)(ii) and therefore should be removed from TS Bases 3/4.9.3.

10 CFR 50.36(c)(2)(ii) specifies that a TS limiting condition for operation (LCO) of a nuclear reactor must be established for each item meeting one or more of four criteria.

> (ii) A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one or more of the following criteria:
>
> (A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
>
> (B) *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (C) *Criterion 3.* A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
>
> (D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Criterion 1

The SFP heat load limitation from TS Bases Section 3/4.9.3, which supports TS 3/4.9.3.1 specifications on decay time, does not cover installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. Therefore, Criterion 1 is not satisfied.

Criterion 2

The NRC Final Policy Statement on "Technical Specification Improvements for Nuclear Power Reactors" discussion for Criterion 2 of 10 CFR 50.36(c)(2)(ii), as discussed in 58 Federal Register (FR) 39132, references American National Standards Institute (ANSI) N18.2 condition II, III, or IV (or equivalent) events presented in chapters 6 and 15 (or equivalent chapters) of the FSAR that either assume or present a challenge to the integrity of a fission product barrier. For MPS2, the equivalent events and the equivalent FSAR chapters are the moderate frequency, infrequent, and limiting faults events presented in FSAR Chapter 14. SFP heat load does not affect any of these events.

Referencing decay heat load limitations in the SFP heat load analysis in TS Bases Section 3/4.9.3 is not consistent with the NRC final policy statement and should be removed. Removing the decay heat load limitations from the TS Bases Section 3/4.9.3 does not change the design or function of the SFP cooling system. It also does not change the applicability of the thermal-hydraulic analysis which demonstrates that the temperature limits of the SFP are met with increased heat loads due to reduced time to fuel movement and a higher rate of fuel movement. Therefore, Criterion 2 is not satisfied.

Criterion 3

The SFP heat load limitation does not affect a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Therefore, Criterion 3 is not satisfied.

Criterion 4

The SFP heat load limitation does not affect a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. The design and testing of systems designed to maintain the SFP temperature within established limits is not affected by the proposed change. The SFP heat load analyses submitted demonstrate that the impact of reduced decay time on SFP decay heat load is offset by the reduced cooling water temperatures such that the maximum normally allowed pool temperature is not exceeded. The peak temperature of the SFP during a loss of cooling event is unaffected and the peak temperature of the fuel cladding, or along the fuel, remains within acceptable limits. Therefore Criterion 4 is not satisfied.

Conclusion

The decay heat load limitations of the SFP heat load analysis do not satisfy any of the four criteria of 10 CFR 50.36(c)(2)(ii) and therefore, should be removed from TS Bases 3/4.9.3. This proposed amendment is consistent with that previously approved by the NRC under License Amendment 240 (Reference 7.5) and License Amendment 315 (Reference 7.2).

4.2 Proposed Revision to FSAR Section 9.5

The MPS2 FSAR considers the decay time of irradiated fuel in two analyses. Section 14.7.4, "Radiological Consequences of a Fuel Handling Accident (FHA)," lists decay time among many assumptions used to determine the radiological consequences of FHAs in the SFP and inside containment during reactor refueling. Section 9.5, "Spent Fuel Pool Cooling (SFPC) system," includes in-reactor decay time as an assumption used to determine the decay heat rate of the fuel most recently discharged to the SFP. MPS2 FSAR Section 14.7.4 is not affected by this LAR. The FHA for MPS2 assumes a minimum decay time of 100 hours using the alternate source term (AST) methodology, which was approved by the NRC under License Amendments 284 and 298 (Reference 7.6 and 7.7). The scope of the proposed amendment is limited to MPS2 FSAR Section 9.5 for the SFP heat load analysis.

DNC proposes to revise FSAR Section 9.5 to bound fuel movement to start 100 hours after reactor subcriticality and proceed at an average rate of six assemblies per hour provided RBCCW temperature to the SFP cooling and shutdown cooling heat exchangers is maintained at less than or equal to 75°F. If RBCCW temperature exceeds 75°F, fuel movement at an average rate of six fuel assemblies per hour could start at 150 hours after subcriticality and then only with RBCCW temperatures less than or equal to 85°F. The proposed changes to FSAR Section 9.5 also address some typographical errors. Attachment 3 provides the marked-up FSAR Section 9.5 pages.

A reanalysis of Holtec Report Number HI-971778 was performed in 2009 (Reference 7.8) to determine if the SFP heat load decay time addressed in FSAR Section 9.5 and TS 3/4.9.3.1 could be revised from 150 hours to 100 hours to align with the 100-hour decay time for a FHA specified in FSAR Section 14.7.4. The assumptions of this analysis (Reference 7.8) are conservative with regard to the current fuel and core design. Any future changes related to fuel or core design will be dispositioned as part of the normal reload design change process. Additionally, Holtec Report Number HI-2094491 (Reference 7.9) was performed in 2010 and compared to results from previous Holtec Report Number HI-981901 (Reference 7.10) performed in 1998 to assess thermal hydraulic impacts. The reduced decay time would allow DNC to move fuel from the reactor vessel to the SFP earlier during refueling outages. The reanalysis determined that reducing the decay time from 150 hours to 100 hours, and increasing the assumed average fuel assembly movement rate from four to six assemblies per hour, is supported when RBCCW temperature to the SFP cooling and shutdown cooling heat exchangers is maintained at less than or equal to 75°F. DNC requests NRC

approval of a proposed revision to FSAR Section 9.5 to reflect the SFP heat load reanalysis and allow for a decay time of 100 hours.

The reanalysis to support the change in decay time from 150 hours to 100 hours used the same method of evaluation described in FSAR Section 9.5.2.1 (i.e., ORIGEN 2) to assess heat load. Although the reanalysis methods remain unchanged, inputs were changed. Specifically, the start for irradiated fuel movement was changed from 150 hours to 100 hours after subcriticality and the assumed fuel movement rate was changed from four assemblies per hour to an average rate of six assemblies per hour. In order to offset the reduced time to start of fuel offload and increase in rate of fuel offload while maintaining SFP bulk water temperature below the maximum normal temperature limit of 150°F, credit was taken for reduced RBCCW temperatures.

NUREG 0800, Standard Review Plan (SRP), supports using reduced RBCCW temperatures to control SFP temperatures. Specifically, Revision 2 of SRP 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," notes the following related to SFP system design cooling capacity:

> The largest heat load placed on the SFPCCS [spent fuel pool cooling and cleanup system] heat exchangers is imposed by refueling offloads, which are deliberate, planned evolutions. As a result, if necessary for adequate cooling of the fuel, factors that increase heat load (e.g., power increases, decay time reductions, or storage capacity increases) may be offset by operational factors that reduce heat load (e.g., longer decay times or transfer of fewer fuel assemblies to the SFP) or that increase heat removal capability (e.g., scheduling offloads for periods of reduced ultimate heat sink temperature or optimizing cooling system performance).

Therefore, crediting reduced RBCCW temperature is an acceptable method for controlling SFP cooling temperatures and is consistent with the NRC safety evaluation from License Amendment 315 (Reference 7.2).

Analyses Summary

The SFP heat load reanalysis documented in Holtec Report Number HI-2094491 (Reference 7.9) contained over 80 different system alignments and temperature combinations. Results were compared to the previous Holtec Report Number HI-981901 (Reference 7.10), which supported the original 150-hour decay time. The complete original and revised analyses are included in Attachment 4. Key cases are summarized below for comparison.

The RBCCW flow rates remain constant at 1100 gpm (gallons per minute) for the SFP cooling HXs and 3000 gpm to the SDC HX for all scenarios.

- Normal Refueling - Both HI-981901 and HI-2094491 Case 2.1.1(a) analyzed two SFP cooling pumps and two SFP cooling HXs with a flow rate of 850 gpm each (SFP side).

- Full Core Offload as a Normal Event - HI-981901 Case 2.1.2(b2) and HI-2094491 Case 2.1.2(l) analyzed one train of SDC with a flow rate of 3000 gpm and flow split between the Refueling Pool / Reactor Vessel (RP/RV) and the SFP. At the end of the full core offload, after the last assembly has been discharged to the SFP, the flow to the RP/RV is still at the minimum value of 1000 gpm and the flow to the SFP is 1900 gpm.

- Emergency Full Core Offload - HI-981901 Case 2.1.3 and HI-2094491 Case 2.1.3(n) analyzed one train of SDC at a flow rate of 1900 gpm to the SFP.

Table 1
Comparison of Results from Select Cases of
Previous Holtec Analysis HI-981901 and Revised Holtec Analysis HI-2094491

Pool Condition	Previous RBCCW Temperature (°F)	Previous SFP Temperature for 150 hour hold time (°F)	Revised RBCCW Temperature (°F)	Revised SFP Temperature for 100 hour hold time (°F)
Normal Refueling	85	129.9	85 75	135.0* 125.1
Full Core Offload as a Normal Event	85	141.6	85 75	152.5*,† 143.2
Emergency Full Core Offload	85	146.5	85 75	154.5*,† 144.5

*Provided for comparison purposes only. Procedures will prohibit fuel movement at less than 150 hours without RBCCW below 75°F

† Results do not support remaining below bulk temperature limits so this condition is prohibited by procedure.

For all three cases, the results demonstrate that with an RBCCW temperature of less than or equal to 75°F, the SFP bulk water temperature will remain below the 150°F limit with fuel movement starting at 100 hours after reactor shutdown.

Since the emergency core offload event represents the limiting heat load in the SFP reanalysis, an additional analysis was performed to verify that local water temperatures were less than boiling and local fuel cladding temperatures were acceptable. Holtec Report Number HI-2094491, Appendices C and D (Reference 7.9), conservatively assumed the bulk SFP temperature was 154.5°F prior to entry into the bottom of the spent fuel rack cell rather than the 144.5°F temperature calculated with 75°F RBCCW temperature. Results of this analysis showed that at the maximum emergency core offload heat load of 38.8 x 10^6 BTU/hr, the maximum local SFP water temperature along a fuel assembly is 178°F and the maximum local fuel clad temperature is 234°F. These temperatures are below the local saturation temperature of 240°F at the top of the fuel assembly. Therefore, no local boiling will occur within the fuel rack cells.

The analyses show that with fuel movement beginning at 100 hours after reactor shutdown and proceeding at an average rate of six assemblies per hour, an RBCCW temperature less than or equal to 75°F will ensure SFP bulk temperature remains below the maximum normal operating temperature limit of 150°F specified in TRM 3/4.9.3.2. In addition, fuel movement beginning at 150 hours after reactor shutdown and proceeding at an average rate of six assemblies per hour, is acceptable with RBCCW temperature less than or equal to the maximum operating temperature of 85°F.

RBCCW temperature is influenced by the ultimate heat sink temperature. The NRC approved License Amendment 318 (Reference 7.11) to allow MPS2 to increase the ultimate heat sink water temperature limit from 75°F to 80°F. License Amendment 318 does not affect the proposed amendment. If the ultimate heat sink temperature does not enable the RBCCW temperature to meet the 75°F or 85°F requirement, fuel movement would not be allowed until the RBCCW temperature requirement is met. Additionally, there have been no license amendments submitted since 2010 that affect or are affected by the proposed changes.

DNC has implemented administrative controls in the form of operating procedures to ensure minimum SFP cooling temperatures are met prior to moving fuel. As stated in NRC safety evaluation letter dated June 4, 2013 (Reference 7.2), using procedural controls to allow a variable subcriticality time (decay time) as a function of RBCCW inlet temperature, is acceptable. This approach has been accepted by the NRC staff as an appropriate means to control SFP temperature, based on the review criteria contained in SRP Section 9.1.3.

5.0 Regulatory Evaluation

The proposed amendment revises MPS2 TS Bases Section 3/4.9.3 "Decay Time" to remove discussion related to the SFP heat load analysis. The unrevised TS Bases 3/4.9.3 states that the minimum time that the reactor must be subcritical prior to movement of irradiated fuel (decay time) is based on both the decay heat load limitations and the radiological dose consequences of the FHA. DNC has determined that the decay heat load limitations of the SFP heat load analysis do not meet the criteria of 10 CFR 50.36(c)(2)(ii) and therefore can be removed from TS Bases Section 3/4.9.3.

DNC proposes to revise MPS2 FSAR Section 9.5 to bound fuel movement to start 100 hours after reactor subcriticality and proceed at an average rate of six assemblies per hour provided RBCCW temperature to the SFP cooling and SDC HXs is maintained at less than or equal to 75°F. If RBCCW temperature exceeds 75°F, fuel movement at an average rate of six fuel assemblies per hour could start at 150 hours after subcriticality and then only with RBCCW temperatures less than or equal to 85°F. The proposed changes to FSAR Section 9.5 also address some typographical errors.

DNC requests NRC approval of the proposed changes to TS Bases Section 3/4.9.3 and FSAR Section 9.5. The SFP heat load analyses which support the proposed TS Bases and FSAR changes are submitted as Attachment 4.

5.1 No Significant Hazards Consideration

The NRC has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated; or 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in a margin of safety. DNC has evaluated whether or not a significant hazards consideration (SHC) is involved with the proposed amendment. A discussion of these standards as they relate to this amendment request is provided below.

Criterion 1

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

Response: No.

The proposed amendment affects some assumptions in the MPS2 FSAR related to the performance of the SFP cooling system and cooling of the fuel in the refueling pool. However, the existing design limits for the SFP remain unchanged. Reducing the decay time from 150 hours to 100 hours prior to allowing fuel movement at an increased average rate of six fuel assemblies per hour does not adversely affect SFP design or operation, provided proposed RBCCW temperature limits are satisfied. The proposed amendment does not change the design or function of the SFP cooling system and is consistent with that previously approved by the NRC under License Amendment 240 (Reference 7.5).

The proposed amendment does not affect the temperature limits of the SFP. The thermal-hydraulic analyses supporting the amendment show that the SFP temperature limits continue to be met with increased heat loads due to reduced time to fuel movement and a higher rate of fuel movement. SFP heat load is not an initiator of any accident discussed in Chapter 14 of the MPS2 FSAR. The proposed amendment does not affect the capability of plant structures, systems, or components (SCCs) to perform their design function and does not increase the probability of a malfunction of any SSC.

The MPS2 FSAR Chapter 14 accident analyses, including the FHA presented in FSAR Section 14.7.4, are not affected by the proposed amendment. The proposed amendment does not increase the probability of a FHA, change the assumptions in the FHA, or affect the conclusions of the current FHA analysis of record. The current FHA analysis of record assumes a minimum 100-hour decay time, which is consistent with the minimum allowable decay time assumed in the thermal-hydraulic analyses that support this amendment. The dose results of the FHA analysis are unchanged, and remain within applicable regulatory limits.

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

Criterion 2

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

Response: No.

The proposed amendment would revise the minimum allowed start time to begin fuel movement from 150 hours to 100 hours after reactor subcriticality and increase the maximum allowable rate of fuel assembly movement from an average of four assemblies per hour to an average of six assemblies per hour. The revised decay time limit and fuel offload rates do not create the possibility of a new type of accident because the methods for moving fuel and the operation of equipment used for moving fuel are not changed. The proposed amendment does not add or modify any plant equipment. The design and testing of systems designed to maintain the SFP temperature within established limits are not affected by the proposed change. The proposed amendment does not create any credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing basis.

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

Criterion 3

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

Response: No.

The FHA analysis of record already accounts for irradiated fuel with at least 100 hours of decay. This approved analysis has shown that the projected doses will remain within applicable regulatory limits. Therefore, the proposed amendment does not reduce the margin of safety of the currently approved FHA analysis of record.

The SFP heat load analyses submitted demonstrate that the impact of reduced decay time on SFP decay heat load is offset by the reduced cooling water temperatures such that the maximum normally allowed pool temperature is not exceeded. The slight 1.6°F increase in SFP temperature for full core off-load as a normal event (for 100 hour hold time with 75°F RBCCW temperature) is not a significant change and remains below the maximum normally allowed SFP temperature of 150°F. The peak temperature of the SFP during a loss of cooling event is unaffected and the peak temperature of the fuel cladding, or along the fuel, remains within acceptable limits. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Conclusion

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. DNC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

6.0 Environmental Considerations

A review has determined that the proposed amendment would not change requirements with respect to use of a facility component located within the restricted area as defined by 10 CFR 20, or an inspection or surveillance requirement. DNC has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released off-site, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 References

7.1 Letter dated July 21, 2010, from J. Alan Price (DNC) to USNRC, "Dominion Nuclear Connecticut, Inc. Millstone Power Station Unit 2 License Amendment Request to Revise Technical Specification 3/4.9.3.1 "Decay Time,"' (ADAMS Accession No. ML102240064)

7.2 Letter dated June 4, 2013, from J. Kim (USNRC) to D. Heacock, "Millstone Power Station Unit No. 2 – Issuance of Amendment RE: Revise Decay Time Technical Specification (TAC No. ME4367)," (ADAMS Accession No. ML13072B341), (License Amendment 315)

7.3 Letter dated April 29, 2015, from S. Morris (USNRC) to D. Heacock, "Millstone Power Station Unit 2 – Inspection Report 05000336/2015201, Investigation Report No. 1-2012-008, and Apparent Violations," (ADAMS Accession No. ML15119A028)

7.4 Letter dated August 26, 2015, from S. Morris (USNRC) to D. Heacock, EA-13-188, "Confirmatory Order Related to NRC Report No. 05000336/2015201 and OI Report 1-2012-008; Millstone Power Station Unit 2," (ADAMS Accession No. ML15236A207)

7.5 Letter dated February 10, 2000, from J. Zimmerman (USNRC) to S. Scace, "Millstone Nuclear Power Station, Unit No. 2 – Issuance of Amendment RE: Relocation of Technical Specifications (TAC No. MA6081)," (ADAMS Accession No. ML003684825), (License Amendment 240)

7.6 Letter dated September 20, 2004, from V. Nerses (USNRC) to D. Christian, "Millstone Power Station, Unit No. 2 - Issuance of Amendment Re: Selective Implementation of Alternate Source Term (TAC No. MB6479)," (ADAMS Accession No. ML042360671), (License Amendment 284)

7.7 Letter dated May 31, 2007, from J. Hughey (USNRC) to D. Christian, "Millstone Power Station, Unit No. 2 - Issuance of Amendment Regarding Alternate Source Term (TAC No. MD2346)," (ADAMS Accession No. ML071450053), (License Amendment 298)

7.8 Holtec Report Number HI-971778, Revision 2, "Heat Load From the Spent Fuel Pool for 3 Core Unload Scenarios," August 26, 2009

7.9 Holtec Report Number HI-2094491, Revision 0, "Thermal-Hydraulic Analysis of Millstone Unit 2 Spent Fuel Pool with Increased Fuel Transfer Rate and Reduced In-Core Hold Time," February 24, 2010

7.10 Holtec Report Number HI-981901, Revision 0, "Spent Fuel Pool Thermal-Hydraulic Analysis for Millstone Unit 2," November 6, 1998, Including August 17, 2012 Addenda A and August 15, 2012 Addenda B

7.11 Letter dated April 18, 2014, from J. Kim (USNRC) to D. Heacock, "Millstone Power Station Unit No. 2 - Issuance of Amendment Re: Revise Technical Specification 3/4.7.11 Ultimate Heat Sink (TAC NO. MF1779)," (ADAMS Accession No. ML14037A408), (License Amendment 318)

ATTACHMENT 2

MARK-UP TO TECHNICAL SPECIFICATION BASES PAGE

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel ~~ensures that sufficient time has elapsed to allow the decay heat load of the fuel to be within the assumptions of the spent fuel pool heat load analysis. This minimum requirement also~~ ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products so that the calculated radiological dose consequences of the fuel handling accident are bounding.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment to the environment will be minimized. The OPERABILITY, closure restrictions, and administrative controls are sufficient to minimize the release of radioactive material from a fuel element rupture based upon the lack of containment pressurization potential during the movement of irradiated fuel assemblies within containment. The containment purge valves are containment penetrations and must satisfy all requirements specified for a containment penetration.

Containment penetrations, including the personnel airlock doors and equipment door, can be open during the movement of irradiated fuel provided that sufficient administrative controls are in place such that any of these containment penetrations can be closed within 30 minutes. Following a Fuel Handling Accident, each penetration, including the equipment door, is closed such that a containment atmosphere boundary can be established. However, if it is determined that closure of all containment penetrations would represent a significant radiological hazard to the personnel involved, the decision may be made to forgo the closure of the affected penetration(s). The containment atmosphere boundary is established when any penetration which provides direct access to the outside atmosphere is closed such that at least one barrier between the containment atmosphere and the outside atmosphere is established. Additional actions beyond establishing the containment atmosphere boundary, such as installing flange bolts for the equipment door or a containment penetration, are not necessary.

Administrative controls for opening a containment penetration require that one or more designated persons, as needed, be available for isolation of containment from the outside atmosphere. Procedural controls are also in place to ensure cables or hoses which pass through a containment opening can be quickly removed. The location of each cable and hoses isolation device for those cables and hoses which pass through a containment opening is recorded to ensure timely closure of the containment boundary. Additionally, a closure plan is developed for each containment opening which includes an estimated time to close the containment opening. A log of personnel designated for containment closure is maintained, including identification of which containment openings each person has responsibility for closing. As necessary, equipment will be pre-staged to support timely closure of a containment penetration.

ATTACHMENT 3

MARK-UP TO FSAR SECTION 9.5 PAGES

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

9.5 SPENT FUEL POOL COOLING SYSTEMS~~SUBCRITICALHOURS~~

9.5.1 DESIGN BASES

9.5.1.1 Functional Requirements

or the equivalent heat load

The function of the spent fuel pool cooling system is to remove decay heat generated by spent fuel assemblies stored in the pool by limiting the temperature of the borated pool water to an acceptable level, thereby ensuring the cladding integrity of stored spent fuel assemblies. In order to ensure that spent fuel pool water temperature limits and fuel integrity are maintained two different refueling operations are analyzed: (1) Normal refueling involves movement of a maximum of 80 fuel assemblies ~~or 14.35 x 10^6 BTU/HR~~ from the reactor vessel to the spent fuel pool, and (2) A full core offload of all 217 fuel assemblies from the reactor vessel to the spent fuel pool. A full core offload comprises of an end-of-cycle full core offload or a mid-cycle emergency full core offload. In the event of a full core offload, the spent fuel pool water temperature will be limited to 150°F. This would utilize one train of the shutdown cooling system for the limiting emergency full core offload. Under less limiting full core offload conditions, the Spent Fuel Pool Cooling system or Spent Fuel Pool Cooling supplemented by the Shutdown Cooling system may be used, provided that a Spent Fuel Pool temperature of less than 150°F is maintained

The spent fuel pool cooling system is provided with a cleanup system for maintaining the purity and clarity of water in the spent fuel pool, refueling pool and the refueling water storage tank after completion of refueling operations. The cleanup systems limit operating personnel radiation exposures from these sources by reducing the concentrations of radioactive constituents introduced into these waters.

9.5.1.2 Design Criteria

The spent fuel pool cooling system is designed in accordance with the following criteria:

a. The system shall be designed to ensure adequate decay heat removal capability under normal and postulated accident conditions.

b. The system shall be designed with the ability to permit appropriate periodic inspection and testing of components important to decay heat removal.

c. The system shall be provided with suitable shielding for radiation protection.

d. Reliable and frequently tested monitoring equipment to detect conditions that may result in loss of decay heat removal shall be provided.

e. The system shall be designed with an adequate spent fuel pool makeup system with appropriate provisions for a backup system for filling the pool.

f. Design shall prevent significant reduction in fuel storage cooling water inventory due to equipment failure, maloperation or accident conditions.

g. Appropriate containment and cleanup system for spent fuel pool cooling water shall be provided.

9.5.2 SYSTEM DESCRIPTION

9.5.2.1 System

The spent fuel pool cooling system, shown in Figure 9.5-1 is designed to remove decay heat generated by stored spent fuel assemblies by circulating the borated pool water through a heat exchanger unit consisting of two heat exchangers in parallel. Cooling water for the heat exchangers is supplied by the RBCCWS as described in Section 9.4. The two motor-driven pumps take suction from the spent fuel pool at elevation 35 feet 6 inches, one foot below the normal operating level of elevation 36 feet 6 inches and at elevation 13 feet 0 ~~inces~~ through the lower suction line. The normal depth of the spent fuel pool is approximately 38.5 feet of water. The spent fuel pool cooling water is returned to the bottom of the pool through three he penetrate the pool walls approximately one foot below the normal operating water opposite side of the pool from the supply headers to provide convective circulation of the pool water through the stored fuel assemblies.

inches

proceeds at an average rate of 6

Normal Refueling (Fuel Shuffle)¶

between 100 and

(fuel shuffle)

The first design basis for the spent fuel pool cooling (SFPC) system is for normal refueling. Fuel movement to the SFP is assumed to start ~~after~~ 150 hours of decay, and ~~proceed at a rate of 4~~ fuel assemblies per hour. The case analyzed is refueling at the end of plant life to maximize fuel inventory and therefore decay heat of spent fuel in the SFP. Decay heat calculations are performed with ORIGEN2.

, started at 150 hours,

This

14.52

~~The~~ normal refueling produces a maximum conservative heat loading of ~~14.35~~ x 10^6 BTU/HR. With both trains of SFPC in service, spent fuel pool water temperature will be maintained to less than ~~130~~°F.

and 85°F RBCCW

Insert 1

131

A single active associated failure analysis of the spent fuel pool cooling system shows the limiting failure to be failure of RBCCW valve 2-RB-8.1A or 8.1B. Failure of either valve closed results in loss of cooling flow to the associated SFPC heat exchanger. This failure is more limiting than the loss of a SFPC pump. Failure of RBCCW valve 2-RB-8.1A or 8.1B in the closed position would result in a SFP bulk water temperature ~~of 172°F~~, if the failure occurred at maximum heat load. ~~To ensure that 150°F is not exceeded,~~ Should this limiting single failure occur, 1 train of the shutdown cooling (SDC) system may be used to cool the SFP, or the SDC system may be used to supplement the SFPC system. Used in this fashion, depending on heat load, between 0 to 1000 gpm of SDC flow would be able to maintain SFP water temperature ≤ 150°F. See Section 9.3 for a description of the SDC. Since the SDC system will eventually be re-aligned for its ECCS requirements for Mode 4, the SFPC system must be capable of sustaining a single failure without assistance from SDC upon entry into Mode 4. Analysis shows that once the SFP heat load drops to 10.16 x 10^6BTU/HR, even if the limiting SFPC single failure occurs, the SFPC system will be able to maintain SFP bulk water temperature ≤ 150°F. At least 616 hours of being subcritical or a heat load of 10.16 x 10^6 BTU/HR is required prior to re-entry into Mode 4 from a refueling outage.

greater than 150°F but less than 200°F

Key SFP Cooling and SDC parameters used in this analysis are:

Suitable redundancy is ensured by the shutdown risk program. ¶

Credited SFP cooling flow is 850 gpm per SFP cooling pump.

Credited RBCCW flow to each SFP cooling heat exchanger is 1100 gpm ~~at 85°F~~.

Credited RBCCW flow to SDC heat exchangers is 3500 gpm ~~at 85°F~~.

Insert 2

SFP and SDC heat exchangers have a 1% tube plugging penalty.

Periodic maintenance of the Spent Fuel Pool Cooling ~~stem~~ system and associated supporting systems is required. During the performance of maintenance activities it may be necessary to remove system components and portions of various support systems from service. While these components and systems are out of service, the ability of the Spent Fuel Pool Cooling system to meet single failure criteria will be limited. These maintenance activities do not conflict with system design or licensing basis. ~~The~~ Therefore the requirement to maintain the SFP ≤ 150°F can be met during modes 1 through 4 with a single train of SFP Cooling. During refueling, for a fuel shuffle, under limiting postulated conditions, 2 trains of SFP cooling are sufficient to maintain SFP bulk water temperature ≤150°F. During refueling, for a fuel shuffle, under limiting postulated conditions, 1 train of SFP cooling will need to be supplemented by Shutdown Cooling to maintain SFP bulk water temperature ≤ 150°F. Eventually during refueling, 1 train of SFP cooling is sufficient to maintain SFP bulk water temperature ≤ 150°F. For the final refuel outage, this point is reached 616 hours after shutdown or when the SFP heat load is at or below 10.16 x 10^6 BTU/HR, when 1 train of SFP cooling is sufficient to maintain SFP bulk water temperature ≤150°F. A single train of Shutdown Cooling is sufficient to remove all decay heat from both the reactor vessel and the SFP.

Full Core Offload as Normal Refueling

The second design basis for the SFP cooling system is for a full core offload as normal refueling. ~~Fuel movement to the SFP is assumed to start after 150 hours of decay, and proceed at a rate of 4 fuel assemblies per hour, until all 217 fuel assemblies are in the spent fuel pool and the reactor vessel is empty of fuel.~~ Insert 3 The case analyzed is the final core offload at ~~end~~ the end of plant life to maximize the fuel inventory, and therefore decay heat, of spent fuel in the pool. At end of plant life, decay heat removal is needed for a total of 1343 fuel assemblies and 3 consolidated fuel storage boxes. The number of fuel assemblies analyzed in the heat load analysis bounds the number of fuel assemblies which can be physically stored in the SFP. Decay heat calculations are performed with ORIGEN2.

Full core offload as a normal activity is acceptable because there are 2 redundant trains of Shutdown Cooling (SDC). One train of SDC is sufficient to ensure that the Spent Fuel Pool (SFP) can be maintained at a bulk water temperature ≤150°F even with the entire core offloaded to the SFP. The other train of SDC is available as backup, should there be a failure in the operating SDC train. In the event that maintenance activities during an outage cause the redundant train of SDC to be unavailable, the Containment Spray (CS) pump has sufficient capacity (Design Temperature of 300°F and Design Head of 390 ft at 1600 GPM) in this circumstance to cool the SFP and to ensure that SFP temperature is maintained ≤150°F. The CS pump has no required function with

Move paragraph to Ⓑ

the core defueled. The CS pump supplemented by a SFP cooling train (if required) will be the means of backup to SDC during SDC maintenance activities. Suitable redundancy is ensured per the Shutdown Risk program.

~~The last full core offload at end of plant life, calculated in the above manner, produces a maximum conservative heat loading of 29.88 x 10^6 BTH/hr.~~ Insert 4 With 1 train of SDC available during the core offload to cool both the reactor vessel and the core, spent fuel pool water temperature will be maintained to less than 150°F. One train of SDC is assumed to be capable of delivering 3000 gpm, however the split of SDC flow going to the reactor vessel, or cooling the SFP will change during the core offload as the heat load shifts from the core to the SFP. With this limiting heat load, at the start of the offload most of the SDC flow will go to the reactor vessel (approximately 2000 gpm), with lesser SDC flow going to the SFP (approximately 1000 gpm). By the end of the offload most of the SDC flow will go to the SFP (approximately 1900 gpm), with lesser SDC flow going to the reactor vessel (>1000 gpm). See Section 9.3 for a description of the SDC system.

Key SDC parameters used in this analysis are:

- Credited SDC flow is 3000 gpm per LPSI pump, with a maximum of 1900 gpm diverted to cooling the SFP.
- Credited RBCCW flow to SDC heat exchangers is 3500 gpm ~~at 85°F~~. Insert 5
- SFP and SDC heat exchangers have a 1% tube plugging penalty.

While the above analysis is for the limiting heat load case at end of plant life, it is acceptable to use SFP cooling, or SFP cooling supplemented by lesser amounts of SDC, during any portion of the core offload, provided that spent fuel pool bulk water temperature can be maintained below 150°F.

Ⓑ→

Therefore, for a full core offload as normal refueling, the system is conservatively designed to maintain the water in the spent fuel pool water below 150°F during all normal conditions or single active failure conditions.

Emergency Full Core Offload

The third design basis for the spent fuel pool cooling system is for an emergency full core offload. ~~Fuel movement to the SFP is assumed to start after 150 hours of decay, and proceed at a rate of 4 fuel assemblies per hour, until all 217 fuel assemblies are in the SFP and the reactor vessel is empty of fuel.~~ Insert 6 The case analyzed is during the final fuel cycle at end of plant life to maximize the fuel inventory, therefore decay heat, of spent fuel in the pool. At end of plant life, decay heat removal is needed for a total of 1343 fuel assemblies and 3 consolidated fuel storage boxes. The number of fuel assemblies analyzed in the heat load analysis bounds the number of fuel assemblies which can be physically stored in the spent fuel pool. Decay heat calculations are performed with ORIGEN2. Significant conservatism is used in the calculation of the decay heat values. A conservatively short decay time of 36 days is chosen for the previous batch of discharge

fuel, and 400 days decay for the previous discharge batch, to maximize decay heat of fuel stored in the pool.

~~The emergency full core offload during the last fuel cycle of operation, calculated in the above manner, produces a maximum conservative heat loading of 34.7 x 10^6 BTH/hr.~~ Insert 7 There is no single failure requirement for emergency core offload calculations, per our current licensing basis. With 2 trains of SDC available during the emergency core offload, approximately 1900 gpm of SDC flow is available to cool the SFP during the entire core offload. With 1900 gpm of SDC flow available to cool the SFP, the SFP water temperature will be maintained to less than 150°F. See Section 9.3 for a description of the SDC system.

Key SDC parameters used in this analysis are:

- Credited SDC flow is 3000 gpm per LPSI pump, with a maximum of 1900 gpm diverted to cooling the SFP.
- Two trains of SDC are assumed to be available.
- Credited RBCCW flow to SDC heat exchangers is 3500 gpm ~~at 85°F~~.

Insert 8

- SFP and SDC heat exchangers have a 1% tube plugging penalty.

While the above analysis is for the limiting heat load case at end of plant life, it is acceptable to use SFP cooling, or SFP cooling supplemented by lesser amounts of SDC, during any portion of the emergency core offload, provided that SFP water can be maintained below 150°F.

Therefore, for an emergency full core offload, the system is conservatively designed to maintai the water in the SFP below 150°F.

Also since the emergency core offload represents the limiting heat load in the SFP, an addition analysis is performed to verify that the local water temperatures and local fuel cladding temperatures were less than boiling. Results of that analysis show that at the maximum emergency core offload heat load, the maximum local SFP water temperature ~~is 163°F~~, and the maximum local fuel clad temperature ~~of 198°F~~. These temperatures are ~~well~~ below the local boiling temperature.

along the fuel assembly is less than 178°F

is less than 234°F

, at the top of the fuel assemblies, of 240°F

Normal makeup to the spent fuel pool water inventory is supplied from the primary water storage tank at a rate of 50 gpm. The 50 gpm makeup capability is adequate to provide water at a rate greater than normal water loss due to evaporation and any system leakage. The maximum makeup capability of this permanently installed system is approximately 200 gpm. Backup makeup for the pool is from the RWST by using one Refueling Pool Purification Pump to transfer the water at a rate of approximately 125 gpm. The RWST contents can also be transferred by the use of one LPSI pump and the interconnecting piping between the spent fuel pool cooling system and shutdown cooling system. A makeup capability of 3000 gpm for 15 minutes is possible from this source. Makeup at approximately 100 gpm can be supplied by an Auxiliary Feedwater Pump taking suction from the Condensate Storage Tank. An alternate backup source for makeup is

obtained from the fire protection system by using temporary hose connections. Flow is at a rate of 200 gpm.

All leakage and drains from the spent fuel pool cooling system components are collected in the open drains system and processed as aerated liquid waste as described in Subsection 11.1.3. The maximum system leakage for an extended period of time for which adequate processing by the RWS is possible is 50 gpm.

Sufficient monitoring equipment is provided to detect and alert operating personnel to conditions that may result in loss of decay heat removal capability. Table 9.5-1 lists the monitoring equipment provided.

A low-flow alarm will alert operating personnel that there is either a low pool water level or one or both cooling water pumps has failed to operate. Excessive radiation levels are detected by local area radiation monitors as described in Section 7.5.6.

To prevent significant reduction in the fuel storage cooling water inventory, all connections to the pool penetrate the pool walls near the normal operating water level so that the pool cannot be gravity drained by leaking pumps, valves, etc. The cooling water return piping which extends to the pool bottom is provided with anti-siphon devices.

Subsection 5.4.3 describes the way the spent fuel pool has been designed to prevent cooling water inventory loss. The spent fuel pool liner leak monitoring and detection system consists of a series of channels welded behind each seam of the pool liner. Any leakage through the pool liner seams is collected in the channels which are piped to the open drain system. The drain header contains a level switch which annunciates in the main control room to alert operating personnel to any leakage. The drain header is provided with valves to isolate the leak and is of a small pipe size (1/2 in.) to prevent significant water loss prior to isolation. The location of the leaks in the pool liner seams is accomplished by pressurizing the channels with air and observing air bubbles rising to the pool water surface. Subsections 5.4.3.1.9 and 9.8.4 describe the provisions incorporated in the station design to preclude and/or limit the consequences of a dropped fuel shipping cask.

Boron concentration in the spent fuel pool water is maintained consistent with Technical Specification requirements. The design of the spent fuel storage racks to preclude criticality and the bases for the safe geometry are described in Subsection 9.8.2.

A cleanup system consisting of pumps, filters, and a demineralizer is provided to maintain the purity and clarity of the water in the spent fuel pool, refueling pool, and the RWST after completion of reactor refueling operations. The cleanup system is designed to remove corrosion and fission products introduced into these waters by leaking fuel assemblies and mixing with reactor coolant during refueling operations. The purity and clarity of these waters are maintained to limit operating personnel doses. The radiation levels are closely monitored during refueling operations to establish the allowable exposures times for personnel in accordance with 10 CFR Part 20.

To facilitate the removal of accumulated dust from the surfaces of both the spent fuel pool and refueling pool, skimmer assemblies with a pump and filters are provided for each pool.

9.5.2.2 Components

A description of the components of the spent fuel pool cooling system is given in Table 9.5-2.

9.5.3 SYSTEM OPERATION

9.5.3.1 Normal Operation

During normal operation of the pool, both pumps and heat exchangers are in continuous service. As the decay heat generated by the spent fuel decreases, one pump is stopped. As the decay heat further decreases, operation of the system is intermittent as required to limit the pool water temperature to less than or equal to 150°F.

As described in Section 9.5.2.1, during refuel operations, the spent fuel cooling system, with assistance from the shutdown cooling system when needed, is capable of maintaining spent fuel pool water temperature ≤150°F for a normal refueling through (and including) the end of plant life refueling. This includes allowance for a single active failure of the spent fuel pool cooling system, or shutdown cooling system, as appropriate.

As described in Section 9.5.2.1, the spent fuel pool heat load must decay to a value of 10.16×10^6 BTU/HR, for the SFP cooling system to be capable of withstanding the worst single active failure, and maintain spent fuel pool water temperature ≤ 150°F. For the time period before the SFP heat load drops to 10.16 MBTU/hr, should the limiting single failure occur, 1 train of the shutdown cooling (SDC) system may be used to cool the SFP, or the SDC system may be used to supplement the SFP cooling system, to maintain spent fuel pool water temperature ≤ 150°F.

As a result of this need to depend on the SDC for potential SFP cooling single failure per Technical Requirements Manual (TRM), entry into Mode 4 following a refueling is not allowed until either:

(a) 616 hours has passed since subcriticality for the fuel bundles remaining in the spent fuel pool which were discharged from the previous refueling, of ≤ 80 fuel bundles, or

(b) the heat load to the SFP is less than 10.16 x 106 BTU/hr. At a SFP heat load of 10.16×10^6 BTU/hr, adequate cooling is available to maintain the SFP bulk water temperatures less than or equal to 150°F should a single failure occur in the SFP cooling system.

These TRM limits assure that the shutdown cooling system would be available for supplemental cooling and not committed to its emergency core cooling system functions.

While water temperatures up to 150°F are allowed in the spent fuel pool, the demineralizers in the spent fuel pool cleanup system should not be exposed to spent fuel pool water temperatures in

excess of 140°F, to prevent demineralizer resin degradation. Alarms and procedural controls are used to prevent spent fuel pool water from reaching the demineralizer resin prior to the temperature reaching 140°F.

A portion of the spent fuel pool cooling water flow, approximately 125 gpm, is passed through the cleanup system. Operation of this system as well as the skimmer system is intermittent as required to maintain the clarity and purity of the spent fuel pool water. During refueling operations, one refueling water purification pump and one spent fuel pool filter are used for purification of the water in the refueling pool, while the other pump and filter are lined up for purification of the spent fuel pool water. The demineralizer will be used as required for either service. At the completion of the refueling operations, the pumps are used for transferring the remaining refueling water from the pool to the RWST after the water level is lowered to the elevation of the reactor vessel flange.

9.5.3.2 Abnormal Operation

If a full core offload is placed in the spent fuel pool, the cooling capacity for this additional spent fuel is provided by connecting the spent fuel pool cooling system with the shutdown cooling system, if additional cooling is required to limit the pool water temperature. The shutdown cooling system is placed into service by manual initiation.

9.5.3.3 Emergency Conditions

In the event that a serious leak develops in the spent fuel pool liner, makeup water is supplied to the pool from the primary makeup water system by manual initiation from the 14 foot 6 inches level of the auxiliary building. Should the leakage exceed the 50 gpm normal makeup capability, additional makeup is available from the RWST via the refueling water purification system, and the fire protection system by temporary hose connections.

Following a postulated LOCA, RBCCW flow to the spent fuel pool cooling system is stopped by the automatic closing of valves in the cooling water discharge piping from the heat exchangers on SIAS. The availability of cooling water, normally supplied to the spent fuel pool system, allows for greater heat removal capability in the engineered safety features components. As discussed later in Subsection 9.5.4.1, spent fuel pool cooling loss for a period of 9.76 hours will raise the pool water temperature to 212°F. RBCCW cooling to the spent fuel pool cooling heat exchangers is restored 4 to 5 hours after the start of the postulated LOCA when the heat load on the RBCCW system is substantially reduced.

For the normal case where spent fuel is stored in the spent fuel pool, it is unlikely that a seismic event will cause loss of cooling water flow. The lines from the spent fuel pool to the suction of the spent fuel pool cooling pumps and from the spent fuel pool heat exchangers to the spent fuel pool have been designed and analyzed to Seismic Category I requirements and the remainder of the system, including the spent fuel pool cooling pumps, heat exchangers and their connecting piping, has been analyzed and found to be in accordance with Seismic Category I requirements. This provides assurance that cooling water would be available from the spent fuel pool cooling system.

In the event that the cooling water flow is lost, makeup water is available from one of the two seismic Category I sources. The seismic category 1 makeup sources for the spent fuel pool are:

The auxiliary feedwater system (AFW) interconnection to the spent fuel pool cooling system. This allows the use of any AFW pump and the inventory of the condensate storage tank as makeup to the SFP. This path is nominally designed to deliver 100 gpm by a flow orifice (FO-8954).

The low-pressure safety injection (LPSI) system interconnection to the spent fuel pool cooling system. This allows the use of a LPSI pump and the inventory of the refueling water storage tank (RWST). The LPSI interconnection is also capable of taking water from the refueling cavity during outages when the inventory from the RWST has been utilized to fill the refueling cavity.

The events or malfunctions that may require makeup to the spent fuel include the long-term loss of spent fuel pool cooling heat removal capability, the failure of a steam generator nozzle dam with the fuel transfer tube isolation valve (2-RW-280) open or the refueling pool cavity seal with the fuel transfer tube isolation valve (2-RW-280) open. The catastrophic failure of the cavity seal is not considered to be a credible event, but has been analyzed and is dealt with the unit procedures. Failures of the spent fuel pool structure are not considered to be credible. The aforementioned events or malfunctions could allow the spent fuel to boil thus requiring makeup. The maximum required makeup would be approximately ~~81~~ 85 gallons per minute which would be for core offload conditions.

In the case where a full core offload is stored in the spent fuel pool, it is improbable that cooling water flow will be lost entirely due to the redundancy of the cooling equipment. In the event that the cooling water flow is lost while a full core offload is being stored in the spent fuel pool, makeup water may be available from the refueling water storage tank, refueling cavity, or the condensate storage tank.

9.5.4 SYSTEM AVAILABILITY AND RELIABILITY

9.5.4.1 Special Features

The spent fuel pool purification system can be used as a backup to maintain the RWST contents above 50°F.

The spent fuel pool cooling pumps and heat exchanger units can be supplemented by the shutdown cooling system.

The most serious failure of the system would be the complete loss of the spent fuel pool water inventory. To protect against this possibility, all connections to the pool enter near to or above the pool operating water level, so that the pool cannot be gravity drained through leaking valves, piping and equipment maloperation. Piping which extends to the bottom of the pool is provided with antisiphoning devices.

Reliable monitoring equipment, as described in Subsection 9.5.2.1 is provided to detect and alert plant operating personnel to any conditions which could result in loss of decay heat removal capability.

~~During refueling, following the loss of all cooling, the minimum times to reach boiling conditions in the SFP during Normal Refueling, Full Core Offload as Normal and Emergency Full Core Offload are 9.12 hours, 3.8 hours and 3.0 hours, respectively. For Normal Refueling, Full Core Offload as Normal and Emergency Full Core Offload, the initial conditions heat loads and temperatures for time to boil times are 14.35 x 10^6 BTU/hr, 29.88 x 10^6 BTU/hr and 34.70 x 10^6 BTU/hr and 129.9°F, 141.6°F and 146.5°F respectively.~~ Insert 9

in Modes 1 through 4

To conservatively estimate the time for the SFP bulk water temperature to reach the SFPC system temperature of 200°F, the SFP bulk water temperature is assumed to be 150°F at 616 hours after ~~subcriticality of~~ SFP heat load is at or below 10.16 x 10^6 BTU/HR. If loss of SFP cooling is assumed to occur at that time, the SFP bulk water temperature will reach the SFPC system temperature of 200°F in 7.87 hours or 212°F in 9.76 hours. Alternately, for SFP starting bulk water temperatures of 130°F and 117.7°F it takes 11 hours and 12.96 hours respectively to heat the SFP to 200°F.

and

,

Sufficient time is available for operating personnel to locate and correct the malfunction.

That portion of the spent fuel pool cooling system that is used in seismic Category I makeup consists of the section of the spent fuel pool heat exchangers outlet piping downstream of check value 2-RW-8 as shown on Figure 9.5-1. HCC-11 has been designed and analyzed to seismic Category I requirements and additionally all connections have been analyzed up to and including the first seismic restraint beyond the isolation value in each connecting line. Following is a list of the line and valve numbers that constitute the Category I makeup:

Piping

10" - HCC-11, 6" - HCC-11

Valves

2-RW-8, 2-RW-15, 2-RW-65, 2-RW-66, 2-RW-67, 2-RW-71, 2-RW-76, 2-RW-119, 2-RW-213, 2-RW-222

9.5.4.2 Tests and Inspections

Components of the spent fuel pool cooling system are nondestructive tested in accordance with the applicable codes as listed in Table 9.5-2.

All pumps in the spent fuel pool cooling system were manufacturer shop performance tested to demonstrate compliance with design head and capacity requirements. A performance curve for the spent fuel pool cooling pumps is shown in Figure 9.5-2.

Prototype filter cartridge testing was conducted by the filter manufacturer confirming a filter efficiency of 92 weight percent when tested with fine Arizona air dust.

All system components are visually inspected and manually adjusted if required to ensure proper installation and arrangement.

The completely installed spent fuel pool cooling system was preoperation tested prior to startup. The detailed test procedure is described in Section 13.

The components of the spent fuel pool cooling system are located in a low-radiation area which permits access for periodic testing and maintenance.

9.5.5 REFERENCES

9.5-1 Letter from W. G. Counsil to Director of Nuclear Reactor Regulation, "Millstone Nuclear Power Station, Unit No. 2 Control of Heavy Loads Near Spent Fuel," dated July 17, 1978.

INSERT 1

A normal refueling with fuel movement starting after 100 hours of decay, and proceeding at an average rate of 6 assemblies per hour over the refueling is acceptable as long as RBCCW temperature is maintained at or below 75°F. The resulting maximum conservative heat loading in the spent fuel pool is 16.18×10^6 BTU/hr. In the event that more than 80 fuel assemblies need to be moved to the spent fuel pool the evolution may still be defined as a Fuel Shuffle as long as the entire core is not offloaded and the maximum heat load in the spent fuel pool remains less than or equal to this value.

With RBCCW temperature maintained at or below 75°F the spent fuel pool cooling system can maintain the spent fuel pool temperature at approximately 125°F with both pumps and heat exchangers in service.

INSERT 2

- Credited RBCCW temperature to SFP and SDC heat exchangers is 75°F for fuel movement starting after 100 hours of decay and 85°F for fuel movement starting after 150 hours of decay.

INSERT 3

Fuel movement to the SFP is assumed to start after 100 hours with a RBCCW temperature of less than or equal to 75°F or after 150 hours with RBCCW temperature of less than or equal to 85°F. The fuel movement is assumed to proceed at an average rate of 6 assemblies per hour until all 217 fuel assemblies are in the SFP and the reactor vessel is empty of fuel.

INSERT 4

The last full core offload at end of plant life, calculated in the above manner, produces a maximum conservative heat loading of 34.59 MBTU/hr with a 100 hour core hold time and produces 30.90 MBTU/hr with a 150 hour core hold time.

INSERT 5

- Credited RBCCW temperature to SDC heat exchangers is 75°F for fuel movement starting after 100 hours of decay and 85°F for fuel movement starting after 150 hours of decay.

INSERT 6

Fuel movement to the SFP is assumed to start after 100 hours with a RBCCW temperature of less than or equal to 75°F or after 150 hours with RBCCW temperature of less than or equal to 85°F. The fuel movement is assumed to proceed at an average rate of 6 assemblies per hour until all 217 fuel assemblies are in the SFP and the reactor vessel is empty of fuel.

INSERT 7

The emergency full core offload during the last fuel cycle of operation, calculated in the above manner, produces a maximum conservative heat loading of 39.40 MBTU/hr with a 100 hour core hold time and produces 35.72 MBTU/hr with a 150 hour core hold time.

INSERT 8

- Credited RBCCW temperature to SDC heat exchangers is 75°F for fuel movement starting after 100 hours of decay and 85°F for fuel movement starting after 150 hours of decay.

INSERT 9

During refueling, following the loss of all cooling, the minimum times to reach boiling conditions in the SFP occur when the core offload begins at 100 hours after shutdown. During Normal Refueling, Full Core Offload as Normal and Emergency Full Core Offload the minimum times to reach boiling are conservatively determined to be in excess of 6.4 hours, 3.1 hours and 2.8 hours, respectively.

ATTACHMENT 5

HOLTEC INTERNATIONAL AFFIDAVIT AND APPLICATION FOR WITHHOLDING

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

HOLTEC INTERNATIONAL

Holtec Center, One Holtec Drive, Marlton, NJ 08053
Telephone (856) 797-0900
Fax (856) 797-0909

SENT VIA E-MAIL

November 12, 2015

Dominion Corporate Licensing
Innsbrook Technical Center, 2SE
Glen Allen, VA 23060

Attention: Ms. Diane E. Aitken
Dominion Corporate Licensing
E-mail: diane.aitken@dom.com
Phone: (804) 273-2694

Subject: Transmittal of Proprietary Reports to USNRC for Millstone Nuclear Power Station

Reference: [1] Holtec Report No. HI-971778, "Heat Load from the Spent Fuel Pool for 3 Core Unload Scenarios," Revision 2.
[2] Holtec Report No. HI-981901, "Spent Fuel Pool Thermal- Hydraulic Analysis for Millstone Unit 2," Revision 0, including Addenda A & B.
[3] Holtec Report No. HI-2094491, "Thermal-Hydraulic Analysis of Millstone Unit 2 Spent Fuel Pool with increased Fuel Transfer Rate and reduced in core hold time," Revision 0.

Dear Ms. Aitken,

Holtec has been requested to provide justification for withholding of proprietary information for Holtec reports Dominion intends to transmit to the USNRC, pertaining to Dominion's Millstone Nuclear Power Station. Attached to this letter is an Affidavit, requesting withholding of Holtec's propriety information from disclosure for reference documents [1], [2], and [3]. Please enclose this Affidavit with any potential transmittals to ensure proper withholding of Holtec's proprietary information.

If you have any questions or further need for support, please do not hesitate to contact me.

Sincerely,

Rick Trotta
Program Manager

Office: (856) 797-0900 x3720
E-mail: R.Trotta@Holtec.com

Attached: [Att. 1] Affidavit 1908-001-AFF, dated November 12, 2015 (applicable to Holtec Reports HI-971778R2, HI-981901R0, and HI-2094491R0), 5 pages.

CC: Kimberly Manzione [HI], Joy Russell [HI]

AFFIDAVIT PURSUANT TO 10 CFR 2.390

I, Kimberly Manzione, being duly sworn, depose and state as follows:

(1) I have reviewed the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.

(2) The information sought to be withheld is information provided in reference documents listed below as noted in Holtec letter 1908-001. These documents have been requested for transmittal by the USNRC and have been transmitted directly by Dominion's Millstone Nuclear Power Plant staff. These Enclosures contains Holtec Proprietary information:

- Holtec Report No. HI-971778, "Heat Load from the Spent Fuel Pool for 3 Core Unload Scenarios," Revision 2, dated 8/26/2009.
- Holtec Report No. HI-981901, "Spent Fuel Pool Thermal- Hydraulic Analysis for Millstone Unit 2," Revision 0, dated 11/6/1998, including Addenda A & B.
- Holtec Report No. HI-2094491, "Thermal-Hydraulic Analysis of Millstone Unit 2 Spent Fuel Pool with increased Fuel Transfer Rate and reduced in core hold time," Revision 0, dated 2/24/2010.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a, 4.b and 4.e above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for

maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

STATE OF NEW JERSEY)
) ss:
COUNTY OF BURLINGTON)

Kimberly Manzione, being duly sworn, deposes and says:

That she has read the foregoing affidavit and the matters stated therein are true and correct to the best of her knowledge, information, and belief.

Executed at Marlton, New Jersey, this 12th day of November, 2015.

Kimberly Manzione
Licensing Manager
Holtec International

Subscribed and sworn before me this 12 th day of November, 2015.

Maria C. Massi

MARIA C. MASSI
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires April 25, 2020