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Dominion®

June 27, 2016

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

Serial No.	16-214
NRA/WDC	R0
Docket No.	50-336
License No.	DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR LICENSE AMENDMENT REQUEST TO REVISE ECCS TS 3/4.5.2 AND FSAR CHAPTER 14 TO REMOVE CHARGING (CAC NO. MF7297)

In a letter dated January 25, 2016, Dominion Nuclear Connecticut, Inc. (DNC) requested an amendment to Facility Operating License No. DPR-65 for Millstone Power Station Unit 2 (MPS2). The proposed amendment would revise MPS2 Technical Specification (TS) 3.5.2, "Emergency Core Cooling Systems, ECCS Subsystems - T_{avg} ≥ 300°F," to remove the charging system and eliminate Surveillance Requirement 4.5.2.e. The proposed amendment would also revise MPS2 Final Safety Analysis Report (FSAR) Chapter 14 relative to the long-term analysis in Section 14.6.1, "Inadvertent Opening of a Pressurized Water Reactor Pressurizer Pressure Relief Valve," and would clarify the existing discussion regarding the application of single failure criteria. An update to the associated TS Bases was included to address the proposed change. In an email dated May 12, 2016, the Nuclear Regulatory Commission (NRC) transmitted a request for additional information (RAI) related to the amendment request. DNC agreed to respond to the RAI by June 30, 2016.

The attachment to this letter provides DNC's response to the NRC's RAI.

Should you have any questions in regard to this submittal, please contact Wanda Craft at (804) 273-4687.

Sincerely,

M. 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 27TH day of June, 2016.

My Commission Expires: 5-31-18.

Vicki L. Hull
Notary Public

A001
NRR

Commitments made in this letter: None.

Attachment:

Response to Request for Additional Information for License Amendment Request to Revise ECCS TS 3/4.5.2 and FSAR Chapter 14 to Remove Charging

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ATTACHMENT

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION FOR LICENSE AMENDMENT REQUEST TO REVISE ECCS TS 3/4.5.2 AND FSAR CHAPTER 14 TO REMOVE CHARGING

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2

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RAI – 1

In Attachment 1 of the LAR, Section 3.1.2, it states that "...a review of the MPS2 Analyses of Record (AOR) for the FSAR Chapter 14 events was performed and concluded that flow from charging pumps is not credited for event mitigation." Section 14.2.7, "Loss of Normal Feedwater Flow," in the updated final safety analysis report (UFSAR) shows charging flow in response to Pressurizer level control in the sequence of events (Tables 14.2.7-3 and 14.2.7-4). Please confirm that flow from the charging pumps is not credited to mitigate the Loss of Normal Feedwater Flow accident.

DNC Response

The FSAR Section 14.2.7, "Loss of Normal Feedwater Flow," event identifies charging flow initiation, but does not credit the charging system for event mitigation. The loss of normal feedwater event is analyzed to determine both the resulting minimum steam generator water level and the maximum pressurizer water level.

FSAR Table 14.2.7-3 provides the sequence of events for the Loss of Normal Feedwater event analyzed to determine the minimum steam generator inventory. In this scenario, normal automatic operation of the charging pumps is assumed in response to pressurizer level dropping below the program setpoint. This automatic action has negligible impact on the minimum steam generator inventory results of this event.

FSAR Table 14.2.7-4 provides the sequence of events for the Loss of Normal Feedwater event analyzed to determine the maximum pressurizer water level. As noted in this table, the maximum pressurizer water level occurs prior to the start of the charging pumps in response to level deviation. Therefore, the automatic operation of the charging pumps does not impact the maximum pressurizer level results of this event.

Based on the above discussion, the flow from the charging pumps is not credited to mitigate the Loss of Normal Feedwater Flow event.

RAI – 2

In Attachment 1 of the LAR, Figure 2 Pressurizer Level Indication, the pressurizer level is at 100% for about 35 minutes. The new long-term Inadvertent Opening of Pressurizer Pressure Relief Valve (IOPPRV) analysis would be the only UFSAR Chapter 14 event that fills the pressurizer and can have a pressurizer safety valve (PSV) or power-operated relief valve (PORV) discharge water or a two-phase liquid/vapor mixture. UFSAR Section 4.3.7 states that the structural analysis for the PSV and PORV discharge piping system was re-analyzed as part of the pressurizer replacement. Describe how the current structural analysis for the PSV and PORV discharge piping system supports the discharge of water or a two-phase liquid/vapor mixture associated with the new long-term IOPPRV analysis.

DNC Response

The current hydrodynamic analysis for the PSVs and the PORVs bounds the IOPPRV for piping and support loads for a saturated two-phase mixture discharge through a PSV or two PORVs. The current hydrodynamic analysis is bounding because of the higher operating pressure, the larger momentum change (due to the rapid valve opening) and since the current analysis considers multiple valves opening at the same time.

The current hydrodynamic analysis for PSV and PORV piping considers multiple valves (either two PORVs or three safety valves) discharging at the same time. The IOPPRV transient is bounded by this consideration since it assumes one stuck open PSV or two open PORVs.

For both the current structural analysis and the IOPPRV analysis, the initial lift pressure is approximately 2575 psia and 2472 psia for the PSVs and the PORVs, respectively. The largest loads experienced by the associated piping and supports occur at the initial opening of a PSV or a PORV due to the rapid fluid momentum change. The PSV analysis considers a 25 millisecond opening time and the PORV analysis considers a 160 millisecond opening time. The pressure wave caused by the valve opening travels at the speed of sound through the piping system. Some additional conservatism is noted in the PORV analysis as empirical data indicates that the PORV opening time is actually a minimum of 253 milliseconds.

For the IOPPRV transient, after approximately four minutes, the pressurizer will fill with a saturated two-phase mixture. The pressurizer pressure at this point in the transient is approximately 1150 psia or less than half of the pressure used in the current structural analysis of record. When the saturated two-phase mixture in the pressurizer reaches the stuck open PSV or PORVs, there is a momentum change as the flow transitions from steam to a steam/water mixture of decreasing quality. This transition (and the associated momentum change) occurs relatively slowly compared to the initial valve opening event. Therefore, the momentum change and loads experienced by the piping and supports during the transition from steam to the two-phase mixture, is less than the initial momentum change and loads experienced during the rapid opening of the PSV or PORVs.

Once the saturated two-phase mixture passes through the seats of the PSV or PORVs, the saturated two-phase mixture will flash into a larger vapor fraction two-phase mixture (40% void fraction or greater). The reduced density of the two-phase mixture exerts loads on the piping and supports that are less than the current hydrodynamic loads.

RAI – 3

In Attachment 1 of the LAR, Figure 2 Pressurizer Level Indication, what causes the rapid decrease in pressurizer level from 100% to ~85% at about t=2100s?

DNC Response

As depicted in Figure 2 of the LAR, the pressurizer level swells with the open PPRV and reaches a level indication of 100%. The plant remains in a fairly slow boil-off phase while mass is lost out the PPRV. Redistribution of core voids throughout the Reactor Coolant System (RCS) begins during the time period starting at approximately 2100 seconds, and culminates in the 2B loop seal clearing at approximately 2400 seconds. When the 2B loop seal clears, a slug of liquid enters the core, collapses voids, and significantly reduces pressurizer level. All other loop seals remain plugged for the duration of the transient.

RAI – 4

Attachment 1 of the LAR, Section 4.3, states the proposed change to UFSAR Section 14.6.1.1: "the limiting event is obtained by assuming the inadvertent opening of a pressurizer safety valve which bounds the capacity of two pressurizer power-operated relief valves." From Reference 5 (ANF-87-161 Rev. 0), the limiting event for the short-term departure from nucleate boiling (DNB) analysis is the inadvertent opening of both PORVs. Based on the data in UFSAR Chapter 4, the capacity of 1 PSV (at set pressure) is less than the minimum capacity of 2 PORVs. Provide additional *justification to support why opening of one PSV will bound two PORVs for both the short-term and long-term IOPPRV event.*

DNC Response

In Section 4.1.4 of Attachment 1 of the LAR, Dominion identified that the revised long-term analysis assumed an open PSV flow area of 0.01767 ft^2. This compares to the maximum capacity of a PORV, specified as 160,650 lbm/hr at 2397 psia, which equates to a smaller flow area of 0.01730 ft^2 for both PORVs. Thus, the PSV capacity bounds the capacity of two PORVs. In addition, as noted in Attachment 1, Page 17 of the LAR which discusses the proposed change to FSAR Section 14.6.1.1, the existing short-term analysis of the IOPPRV event [References 5 and 6 of the LAR] is not affected by this change since that analysis assumed a capacity that bounds the maximum capacity of either one PSV or two PORVs (an area equivalent to 0.0213 ft^2).

Note, per Section III of the ASME code (Article 9 through the summer 1970 addenda), the rated minimum capacity of the PSV provided in Table 4.3-11 of the FSAR is 90% of the actual capacity of the valve, whereas the PORV minimum capacity provided in Table 4.3-10 of the FSAR does not apply the 10% conservatism because the PORVs are not ASME code relief valves. In addition, the valve capacities assumed in the short-term and long-term analyses are significantly higher than the minimum values presented in the FSAR tables.

RAI – 5

Describe the key SRELAP-5 input parameters and assumptions used for the long-term IOPPRV analysis. Also, discuss the conservativism used in the analysis inputs and the operator actions assumed during this event.

DNC Response

The long-term analysis of the IOPPRV for MPS2 was performed in accordance with the NRC-approved S-RELAP5 Small Break Loss of Coolant Accident (SBLOCA) methodology described in EMF-2328(P)(A), Revision 0, including Supplement 1 [References 1 and 2 of the LAR].

The SRELAP-5 input parameters and assumptions are identified in Tables 1 and 2. The analysis employed the following key assumptions to ensure a conservative calculation.

- The initiation of the High Pressure Safety Injection (HPSI) system was delayed for 10 seconds following safety injection actuation (SIAS) activation as shown in Table 1 below. However, no HPSI flow was delivered until system pressure fell below the HPSI pump shutoff head. This occurred more than 85 seconds after the HPSI system was available. Table 2 below shows the minimum HPSI flow rates with two HPSI pumps, which conservatively includes pump degradation. The Low Pressure Safety Injection (LPSI) system was included in the model. However, the pressure did not fall low enough to allow LPSI injection.

- The heat generation rate in the S-RELAP5 reactor core model was determined from reactor kinetics equations with actinide and decay heat as prescribed by 10 CFR 50 Appendix K for conservatism.
- The reactor SCRAM is conservatively delayed based on the low pressurizer pressure thermal margin/low pressure (TM/LP) floor or minimum reactor trip and includes Technical Requirements Manual maximum delays for Reactor Protection System (RPS) circuitry and Control Element Assembly (CEA) coil delay.

The sole operator action assumed in the analysis is the following:

- Consistent with the SBLOCA analysis submitted for approval [Reference 4 of the LAR], the reactor coolant pumps continue to run until manually tripped two minutes after the loss of subcooling in the cold leg.

Table 1
System Parameters and Initial Conditions

Reactor Power, MWt	2754[1]
Axial Power Shape	EOC, peaked high in core
Peak LHR, kW/ft	15.1
Radial Peaking Factor (1.69 plus uncertainties)	1.854
RCS Flow Rate, gpm	360,000
Pressurizer Pressure, psia	2250
Core Inlet Coolant Temperature, °F	549
SIT Pressure, psia	214.7
SIT Fluid Temperature, °F	120
SIT Water Volume, ft^3	1135
Maximum SG Tube Plugging Level per SG, %	5.87
SG Secondary Pressure, psia	880
MFW Temperature, °F	435
AFW Flow Rate per SG, gpm	72
AFW Temperature, °F	70
Low-Low SG Level Setpoint, % Narrow Range Span	0
AFW Delay, sec	240
HPSI and LPSI Fluid Temperature, °F	140
TM/LP Reactor Trip Setpoint (RPS), psia	1700
Reactor Protection System Delay Time on TM/LP trip, sec	0.9
SCRAM CEA Holding Coil Release Delay Time, sec	0.5
SIAS Activation Pressurizer Pressure Setpoint (Harsh Environment Conditions), psia	1500
HPSI Pump Delay Time on SIAS, sec	10
MSSV Lift Pressure and Tolerance	Nominal + 3%

[1] Includes 2.0% measurement uncertainty

Table 2
Minimum HPSI Flow Rates with Two HPSI Pumps

RCS Pressure (psia)	HPSI into RCS Loop (gpm)			
	Loop 1A	Loop 1B	Loop 2A	Loop 2B
200	262	262	262	262
300	248	248	248	248
500	217	217	217	217
700	183	183	183	183
900	141	141	141	141
1000	118	118	118	118
1050	103	103	103	103
1100	86	86	86	86
1150	60	60	60	60
1190	30	30	30	30
1204	0	0	0	0

RAI – 6

By submittal dated March 17, 2006 (ADAMS Accession No. ML060790325), DNC requested a license amendment to change pressurizer water level limits in MPS2 TS 3.4.4. The 2006 LAR states, "The maximum level limit prevents filling the pressurizer during FSAR Chapter 14 anticipated operational occurrences, ensuring that the pressure relief devices (PORVs or pressurizer safety valves) can control pressure by steam relief rather than water relief, thereby preventing a challenge to the integrity of the RCS fission product barrier." The limiting event with respect to pressurizer overfill was the Loss of Normal Feedwater event at the time of the 2006 LAR.

The new long-term IOPPRV analysis is inconsistent with the assumptions of the pressurizer not filling used in the 2006 LAR and the NRC safety evaluation dated January 30, 2007 (ADAMS Accession No. ML062920334) to justify changes to TS 3.4.4. Provide discussion to support the maximum pressurizer level in TS 3.4.4 with the new long-term IOPPRV analysis that allows the pressurizer to fill.

DNC Response

The maximum water level limit specified in TS 3.4.4 is sufficient to prevent filling the pressurizer during FSAR Chapter 14 anticipated operational occurrences that require the pressurizer relief devices (PORVs or PSVs) for event mitigation. In these events, proper function of the relief or safety valves is ensured when discharging steam rather than a two-phase liquid. Controlling RCS pressure by steam relief rather than water relief ensures that these relief devices will properly close, thereby avoiding a consequential loss of the reactor coolant pressure boundary integrity. This ensures that these anticipated operational occurrences will not progress to a more serious plant condition without other faults occurring independently.

As noted in MPS2 FSAR Section 14.6.1.1, the IOPPRV event results in a rapid depressurization of the primary coolant system and does not use the PORVs or PSVs for pressure control and/or event mitigation as discussed below. Instead, the event is initiated by the failure (inadvertent opening) of one or more PORVs or PSVs due to an electrical or mechanical failure. If the initiating event involves the inadvertent opening of the PORV(s), MPS2 Emergency Operating Procedure EOP 2525, "Standard Post-Trip Actions," directs the operator to close the associated PORV block valve(s). However, if the initiating event involves the inadvertent opening of a PSV as described in the current MPS2 FSAR, there are no operator actions available to isolate the affected valve. As shown in Attachment 1, Figure 2 of the January 25, 2016 LAR, the IOPPRV event progresses to a point where the pressurizer fills approximately four minutes after the initiation of the event. Without the ability to isolate a stuck open PSV, the pressurizer level will increase to the point where the pressurizer will fill regardless of the initial pressurizer water level. This thermal-hydraulic phenomenon is independent of whether charging pumps are credited for event mitigation and is a consequence of the initiating PORV or PSV failure.

As noted in Attachment 1, Figure 1 of the LAR, the RCS pressure at the time the pressurizer fills is approximately 1150 psia. This RCS pressure is below that required to open the unaffected PSV and PORVs to prevent RCS overpressurization. As such, the challenge to the RCS pressure boundary will not be increased beyond that caused by the initiating event. Therefore, though this IOPPRV event will result in two-phase discharge out the inadvertently opened PSV, the event will not progress to a more serious plant condition without other faults occurring independently. See also the response to RAI-2.

Based upon the above discussion, the maximum pressurizer level in existing TS 3.4.4 is justified. As such, the 2006 LAR and the NRC safety evaluation dated January 30, 2007 (ADAMS Accession No. ML062920334) remain valid.