UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 3, 2017

Mr. Daniel G. Stoddard
Senior Vice President and Chief Nuclear Officer
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Innsbrook Technical Center
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SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 – CORRECTION TO SAFETY EVALUATION FOR LICENSE AMENDMENT NO. 331 RE: REVISION TO EMERGENCY CORE COOLING SYSTEM TECHNICAL SPECIFICATIONS AND FINAL SAFETY ANALYSIS REPORT, CHAPTER 14, TO REMOVE CHARGING PUMP FLOW (CAC NO. MF7297)

Dear Mr. Stoddard:

On December 22, 2016, the U.S. Nuclear Regulatory Commission issued Amendment No. 331 (Agencywide Documents Access and Management System Accession No. ML16308A485) to Renewed Facility Operating License No. DPR-65 for the Millstone Power Station, Unit No. 2 (MPS2). This amendment revised the MPS2 Technical Specifications (TSs) to remove the requirements for the charging pumps to be operable in TS 3.5.2, "Emergency Core Cooling Systems, ECCS Subsystems $T_{avg} \geq 300°F$," by eliminating Surveillance Requirement 4.5.2.e from the TSs. The amendment also revises the MPS2 final safety analysis report relative to long-term analysis of the inadvertent opening of a pressurized-water reactor pressurizer pressure relief valve event and clarified the existing discussion regarding the application of single failure criteria.

Subsequent to the issuance of Amendment No. 331, your staff pointed out errors in the safety evaluation (SE) supporting this amendment. We agree that there were errors in the SE, resulting in the need for clarification. The resulting revisions to the SE do not impact the technical basis or safety conclusion as originally issued in Amendment No. 331. Enclosed please find the corrected pages 8, 9, and 10 of the SE, with side bars highlighting the areas of correction.

If there are any questions regarding this matter, please contact me at 301-415-1030 or Richard.Guzman@nrc.gov.

Sincerely,

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosures:
Corrected Pages 8, 9, and 10

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requirements of GDC 10. The inadvertent opening of two PORVs was previously evaluated as part of the IOPPRV event for MPS2 as discussed in FSAR Section 14.6.1. The IOPPRV reanalysis identified that inadvertent opening of a PSV is the limiting event rather than an inadvertent opening of two PORVs. As discussed in section 3.1.1 of this SE, the flow area for one open PSV bounds the combined flow area of two pressurizer PORVs. Thus, the NRC staff finds that the inadvertent opening of a PSV as the limiting event remains consistent with the previous analysis with respect to core uncovery, fuel clad temperature, SAFDL, and the event non-escalation criteria.

Since the pressurizer fills during the long-term reanalysis, the failed valve will pass water and/or a two-phase liquid/vapor mixture. The MPS2 PSVs and PORVs are not qualified to relieve water and must be assumed to fail in the open position after water relief. Since the initiating event is a mechanical failure of a PSV, passing water through the failed PSV has no additional impact on the operability of the valve (it will not close even if it does not pass water). However, the reanalysis shows that while the pressurizer is filled, RCS pressure is much less than the valve's opening setpoint, so a non-failed valve (PORV or PSV) would not open during the analyzed IOPPRV transient. As such, there would be no further challenge to this RCS fission product barrier filling the pressurizer beyond that caused by the initiating IOPPRV event. Therefore, the NRC staff finds that (1) the reanalysis provides reasonable assurance that the IOPPRV event would not generate a more serious plant condition without other faults occurring independently, and (2) there is no additional challenge to the RCS boundary in addition to the initiating event, thereby satisfying the SRP 15.6.1 acceptance criteria regarding event escalation limitation.

3.1.3 Block Valves to Isolate the PORVs in Water or Water- Steam Conditions

As noted in the existing MPS2 FSAR 14.6.1, the IOPPRV event is initiated by the inadvertent opening of one or more PORVs or PSVs due to an electrical or mechanical failure. DNC indicated in its response to RAI-6 (ADAMS Accession No. ML16182A037) that if the initial event involves the inadvertent opening of the PORV (IOPORV), MPS2 Emergency Operating Procedure (EOP)-2525, "Standard Post-Trip Actions," directs the operator to close the associated PORV block valve(s). The IOPORV event, an AOO, is a depressurization event. The steam releases from the open PORVs which results in a decrease in the RCS pressure. If operators do not take appropriate actions to terminate the RCS depressurization by either closing the PORV or its block valve, the safety injection (SI) system will be actuated when the RCS pressure decreases to the low pressurizer RPS signal. Injection of the HPSI pump flow following the SI actuation signal could fill the pressurizer and lead the PORV and the associated block valve to discharge water or steam-water mixtures. Similar to the above PSV analysis, after the PORV and its block valve discharge water or steam-water mixture, the valves are assumed to fail open, unless they are qualified for water or steam-water mixture releases. For the IOPORV event, the SRP 15.6.1 guidance states that in meeting the acceptance criterion, the event must not generate a more serious plant condition without other faults occurring independently.

Like the above PSV analysis, passing water through the initially failed PORV, has no impact on operability of the valve. However, unlike the above PSV analysis, there is an initially operable block valve. To satisfy the requirement of TMI [Three Mile Island] Action item II.D.1, the PORV block valve(s) must be qualified to ensure that a stuck-open relief valve can be isolated, thereby terminating an SBLOCA. At the NRC staff's request, DNC provided plant data showing that the PORV block valve(s) will close, on demand, in water or a steam-water mixture condition. The NRC staff's review of the plant data is discussed in the following section.

NRC Staff Evaluation of the Licensee's Plant Data for PORV Block Valves

NRC GL 89-10 recommended that each nuclear power plant establish a program to demonstrate that safety-related MOVs are capable of performing their design basis functions. Program features include analysis of worst case system demands on MOV operation, MOV setup to meet demands, and demonstration via dynamic testing that the MOV will perform its safety-related function. GL 96-05 superseded GL 89-10 and requested plants to establish a program or ensure the effectiveness of the current program by periodically verifying that MOVs continue to be capable of performing their safety-related function. The PORV block valves are within the scope of the licensee's MOV program.

During the implementation phase of GL 89-10, the industry realized that there was a population of MOVs that cannot be dynamically tested in situ due to various hardships such as system configuration, as low as reasonably achievable (ALARA) concerns, inaccessibility, testing could cause system or component damage, and excessive personnel hazards. In response, the Electric Power Research Institute (EPRI) initiated efforts to develop a computational methodology to be used in demonstrating the design basis capability of MOVs when valve specific test data is not available. The EPRI Performance Prediction Methodology (PPM) includes computer models, software, and hand calculation models to predict individual valve performance. The PPM model is based on extensive testing of several model valves under various conditions and evaluating the test results with the predicted computational value. The final product was the EPRI MOV PPM Program. The key elements of the PPM program are:

1) System Flow Model to predict the differential pressure and fluid pressure for pumped flow and blowdown system configurations
2) Gate Valve Model to predict the thrust required to operate gate valves and potential damage at sliding surfaces
3) Globe Valve Model to predict the thrust required to operate globe valves
4) Butterfly Valve Model to predict the torque required to operate butterfly valves

The results of the EPRI PPM efforts were captured in topical report TR-103237 "EPRI MOV Performance Prediction Program" and submitted by the Nuclear Energy Institute (NEI) on February 22, 1994 (Revision 0) and November 3, 1995 (Revision 1) to the Nuclear Regulatory Commission (NRC) for evaluation. NRC staff completed the safety evaluation on March 15, 1996 (ADAMS Accession No. ML15142A761) and accepted the PPM methodology described in the topical report, with certain conditions and limitations.

The PORV block valves at MPS2 are a normally open Velan 2 ½ inch flexible wedge gate valve with a Limitorque SMB-00 actuator. Due to system design and other hardships, the PORV block valves are difficult to dynamically test at projected design pressure and flow. The licensee elected to use the PPM method to validate the MPS2 PORV block valves to be operationally ready to perform their safety function.

The EPRI PPM gate valve computer model predicts the thrust required to operate gate valves throughout their stroke up to initial wedging under specified fluid conditions and differential pressure. The model uses theoretical equations that address fluid loading on the disk as well as the detailed mechanical interaction between the stem, disk, guides and seat. Valve internal information (dimensions and materials) are required to develop the model as well as piping configuration and fluid conditions for the system flow model. The use of the PPM gate valve model assumes that the valve is in good condition. However, the staff notes that it is necessary for model users to ensure that an adequate internal valve preventive maintenance program is established for the thrust or torque requirements predicted by the model to remain valid. End users are also cautioned that aging conditions or valve degradation can influence valve performance, which may or may not be accelerated by nonstandard orientations. The NRC staff approved the gate valve PPM model with certain conditions and limitations as noted in its safety evaluation March 15, 1996 (ADAMS Accession No. ML15142A761).

DNC responded to the NRC staff's follow-up RAI (RAI-1) in its supplement dated October 12, 2016 (ADAMS Accession No. ML16291A508) to demonstrate that the MPS2 pressurizer PORV block valves can be credited for closure under conditions predicted from the analysis of an IOPORV by performing a PPM calculation for the PORV block valve MOVs. The calculation was completed using EPRI MOV PPM software program version 3.5. This version has been approved by the NRC staff in a letter dated April 2, 2015 (ADAMS Accession No. ML15075A012). The NRC staff reviewed the calculation results, parameters selected, model valve, current MOV settings, and the projected flow and pressure conditions and finds the analysis provides reasonable assurance that the PORV block valves will close on demand. This conclusion is based on the PPM gate valve model prediction.

To ensure that the PPM calculation for PORV block valves remains valid, MPS2 has a preventive maintenance program that maintains valve internals in good condition. In addition, the PORV block valve MOVs are periodically verified via diagnostic testing to ensure the MOVs have positive margin and are operationally ready. Diagnostic data is also used to monitor actuator performance and degradation such as stem nut wear, stroke time, torque switch repeatability, and spring pack operation. The NRC staff concludes that the PORV block valve MOVs will close under conditions predicted from the analysis of an IOPORV.

In addition, based on the review discussed above, the NRC staff finds that the plant data provides reasonable assurance that the block valves could be closed in water or a water-steam mixture condition.

3.1.4 Long Term IOPPRV Piping Structure Consideration

The PSVs and PORVs are connected to nozzles on the top of the pressurizer vessel. As discussed in FSAR Section 4.3.5, the discharge from the PSVs and PORVs is piped to a quench tank where it is cooled and condensed by water in the tank. Because the current FSAR clearly states that steam is discharged from the pressurizer safety and relief valves, the NRC staff considered that the hydrodynamic analysis may address short-term steam discharge only. Therefore, the NRC staff noted that the current IOPPRV structural analysis may not support the

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