Northeast
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September 21, 1994

Docket Nos. 50-245
50-336
50-423
B14961

Re: 10CFR2.201

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

Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3
Reply to a Notice of Violation
Inspection Report Nos. 50-245/94-18; 50-336/94-17; 50-423/94-16

In a letter dated August 11, 1994,[1] the NRC Staff transmitted a Notice of Violation (NOV) relating to NRC Inspection Reports Nos. 50-245/94-18, 50-336/94-17, and 50-423/94-16. The report discusses the results of the safety inspection conducted from April 6, 1994, to May 17, 1994, and July 28, 1994, to August 4, 1994, at Millstone Station. Based on the results of the Staff's inspection, five violations of NRC requirements were identified.

The Staff requested that NNECO respond within 30 days of the date of the letter transmitting the NOV. Based on discussions with the Staff, the due date for this response was extended to September 21, 1994. Accordingly, Attachment 1 to this letter provides NNECO's reply to the NOV pursuant to the provisions of 10CFR2.201.

In the August 11, 1994, letter, the NRC Staff requested that NNECO discuss our assessment of the May 13, 1994, inadvertent engineered safeguards actuation during infrequently performed test SP 2613C and our corrective actions to prevent recurrence. As stated in the NRC Inspection Report, Section 2.5, these issues will be specifically addressed in the response to the plant information report (PIR) which was initiated to investigate this incident. Although investigation is in progress, closure has been delayed due to unit outage and startup activities. At this

(1) R. W. Cooper II letter to J. F. Opeka, "Notice of Violation (NRC Combined Inspection 50-245/94-18; 50-336/94-17; 50-423/94-16)," dated August 11, 1994.

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point in the investigation, we agree that the understanding of the test termination criteria appears to warrant increased emphasis. This and other potential corrective actions will be determined when the PIR investigation is completed. We expect the PIR investigation to be completed on or before October 31, 1994, at which time the results will be presented to the Millstone Unit No. 2 NRC Resident Inspector.

If you have any questions regarding information contained herein, please contact Mr. D. N. Harris at (203) 437-5903.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka
Executive Vice President

Attachments

cc: T. T. Martin, Region I Administrator
J. W. Andersen, NRC Acting Project Manager, Millstone Unit No. 1
G. S. Vissing, NRC Project Manager, Millstone Unit No. 2
V. L. Rooney, NRC Project Manager, Millstone Unit No. 3
P. D. Swetland, Senior Resident Inspector, Millstone Unit Nos. 1, 2, and 3

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Attachment 1

Millstone Nuclear Power Station, Unit Nos. 1, 2, and 3

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September 1994

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**RESTATEMENT OF VIOLATIONS:**

A. Millstone Unit 1 Technical Specification 6.8.1.a requires that written procedures shall be established and implemented covering the activities recommended in Appendix A of Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, dated February 1978. RG 1.33, Step 1.c, recommends written procedures for the control of equipment. Administrative control procedure ACP-QA-2.12, "System Valve Alignment Control," was established pursuant to the above.

Procedure ACP-QA-2.12, Step 6.1.1, requires that independent verification shall be performed of repositioned valves in safety-related systems following maintenance or surveillance activities. Important safety-related systems requiring independent position verification of valves are listed in station form SF-227-1. The emergency service water and standby diesel generator systems are listed on form SF-227-1.

Contrary to the above, independent position verification was not performed on repositioned valves in the emergency service water and standby diesel generator systems following maintenance or surveillance activities, as evidenced by the following examples:

1. On April 30, 1994, during performance of special procedure SP 94-1-54, "Service Water and Emergency Service Water Thermal Hydraulic Test," emergency service water system valves 1-LP-14A and 1-LP-30A were repositioned at the direction of the system engineer without an independent position verification of the valve manipulations until May 13, 1994; approximately ten days after the emergency service water system was returned to an operable status.

2. On April 8, 1994, during performance of special procedure SP 94-1-66, "Emergency Diesel Generator Retest," diesel generator cooling water valve 1-DGCW-7 was opened at the direction of the system engineer and left in this abnormal position without an independent position verification of the valve manipulation until

May 4, 1994; approximately one day after the emergency diesel generator was returned to an operable status.

B. 10CFR50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality be prescribed by and implemented in accordance with documented instructions, procedures, and drawings appropriate to the circumstances.

Pursuant to the above, Administrative Control Procedure ACP-QA-3.10, "Preparation, Review, and Disposition of Plant Design Change Records (PDCRs)," prescribes detailed instructions for making plant design changes. Steps 4.14 and 4.16.2.1, respectively, of procedure ACP-QA-3.10 require that station procedures be updated to reflect the modifications before engineering release for operations, or that changes to surveillance procedures must be implemented within three months or before the next surveillance interval falls due, whichever occurs first.

Contrary to the above, when Unit 1 plant systems were modified by plant design change requests, operating procedures and annunciator alarm response forms were not changed prior to system turnover, and a surveillance procedure was not changed within three months or before the next surveillance interval fell due, as evidenced by the following examples:

1. Plant modification PDCR 1-25-93, which changed control board data recorders 640-27 and 640-28 (reactor vessel pressure), was released by engineering for operations on April 26, 1994. As of May 2, 1994, the associated annunciator alarm response form for panel 903 A-2, window 6-4, was not changed to reflect the modification.

2. Plant modification PDCR 1-25-93, which changed control board data recorder L/TR-1340 (isolation condenser), was released by engineering for operation on April 13, 1994. As of May 3, 1994, the associated annunciator alarm response forms for panel 903 A-2, windows 4-4 and 7-3, and procedure OP-307, "Isolation Condenser System," were not changed to reflect the modification.

3. Plant modification PDCR 1-70-93, which changed the normal position of emergency diesel generator cooling water valve 1-DGCW-7 from "closed" to "open," was released by engineering for operation on April 14, 1994. As of May 2, 1994, Operations Form 338-1,

"Standby Diesel Generator Valve Lineup," was not changed to reflect the modification.

4. Plant modification PDCR 1-58-93, which removed local service water system pressure indicator PI-4-46, was released by engineering for operation on April 29, 1994. As of May 25, 1994, when surveillance procedure SP-623.19, "Emergency Service Water Operational Readiness Test," was scheduled to be performed, removal of pressure indicator PI-4-46 was not reflected in the procedure.

C. 10CFR Part 50, Appendix B, Criterion III, "Design Control," requires that measures shall be established to assure that the design bases for those systems for which the Appendix applies are correctly translated into procedures. Millstone 2 Final Safety Analysis Report Section 9.4, "Reactor Building Closed Cooling Water System," specifies that the maximum design temperature at the outlet of the system heat exchangers is 85 degrees Fahrenheit (°F) when cooled by service water up to maximum temperature of 75°F nd Section 6.5, "Containment Air Recirculation and Cooling System," specifies that cooling water is supplied at 85°F maximum to the cooling coils by the reactor building closed cooling water system.

Contrary to the above, design basis temperature limits of the service water and reactor building closed cooling water systems specified in Section 9.4 and 6.5 of the Millstone 2 Final Safety Analysis Report was not correctly translated into procedures, as evidenced by the following examples:

1. Procedure OP-2330A, "Reactor Building Closed Cooling Water," did not contain controls for the heat exchanger outlet temperature limit of 85°F;

2. Procedure OP-2325A, "Circulating Water," did not contain thermal backwashing precautions regarding service water system or reactor building closed cooling water heater exchanger outlet temperature limits of 75°F and 85°F, respectively; and

3. The reactor building closed cooling water heat exchanger high temperature annunciator on main control board C06F and response procedure (OP 2387 E-140) contained an alarm setpoint 10°F higher than the design temperature limit of 85°F.

As a result, on May 24, 1993, the system temperature limits were exceeded during a main condenser thermal backwashing evolution, thereby causing a reactor plant trip.

D. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires measures to be established to assure that the cause of significant conditions adverse to quality is determined and corrective actions are taken to preclude repetition.

On January 26, 1993, the licensee docketed corrective actions for a violation cited in NRC Inspection Report 50-423/92-23 in which the work control process was inadequately implemented such that a component was not retested following maintenance because the job description and actual work performed were not clearly documented as required by administrative procedures. These corrective measures included providing guidance to station personnel emphasizing the need for proper documentation of work and the need for adequate review of work orders prior to releasing to Operations.

On March 8, 1992, the license docketed corrective actions for violations cited in NRC Inspection Report 59-423/92-28 in which bypass jumpers for scaffolding were installed for greater than three months without review by the Plant Operations Review Committee (PORC) as required by administrative procedures. These corrective actions included a commitment that procedure changes would be ma to facilitate tracking of long-term installations.

Contrary to the above, the corrective actions taken to preclude repetition of these NRC identified violations were inadequate, as evidenced by, the following recurrent events:

1. As of April 26, 1994, the guidance provided to station personnel regarding proper documentation of work was not effective; in that, emergency diesel generator cooling isolation valve 3SWP-AOV39A did not have a required retest specified following maintenance because the removal and replacement of the valve was not documented on the work order.

2. As of April 5, 1994, the procedure changes made to assure proper review of long-term scaffolding installations were not effective; in that, nine bypass jumpers installed for greater than three months had not been reviewed by the PORC.

E. Millstone Unit 1 Technical Specification (TS) 2.1.2.A.1.b, "Fuel Cladding Integrity," requires that during operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the Average Power Range Monitor (APRM) neutron flux scram trip setting (Run Mode) shall be modified by a multiplication factor consisting of the ratio of actual FRP/MFLPD.

Millstone Unit 1 TS 2.1.2.B.1.b, "Fuel Cladding Integrity," requires that during operation with a MFLPD greater than the FRP, the APRM rod block trip setting shall be modified by a multiplication factor consisting of the ratio of actual FRP/MFLPD.

Millstone Unit 1 TS 3.1.A, "Reactor Protection System," requires a minimum of two APRM neutron flux scram channels to be operable per trip system with trip level settings per TS 2.1.2.A. If the minimum number of operable channels cannot be met for both trip systems, the licensee must either initiate insertion of operable rods and complete insertion of all operable rods within four hours; or reduce power level to the IRM range and place the mode switch in the "Startup/Hot Standby position within eight hours.

Millstone Unit 1 TS 3.2.C., "Protective Systems," requires a minimum of one APRM neutron flux rod block channel to be operable per trip system with trip level settings per TS 2.1.2.B, whenever the reactor mode switch in the RUN position. If the minimum number of channels cannot be met for both trip systems, the systems shall be tripped.

Contrary to the above, from approximately 10:30 a.m., on July 27, 1994 to 11:50 a.m. on July 28, 1994, the TS requirements for the APRM neutron flux scram and rod block functions listed above were not met, as evidenced by the following examples.

1. With the MFLPD greater than the FRP, the APRM neutron flux scram and rod block trip settings were not modified by a multiplication factor consisting of the ratio of actual FRP/MFLPD.

2. The minimum number of APRM neutron flux scram and rod block instrument channels per trip system were not operable with trip level settings per TS 2.1.2.A and

2.1.2.B, respectively, and an insertion of operable control rods was not initiated and completed within four hours; power level was not reduced to the IRM range and the mode selector switch placed in the "Startup/Hot Standby" position within eight hours; and the rod block trip systems were not tripped.

**REASON FOR THE VIOLATION (Violation A)**

These events were caused by a failure to adequately implement the requirements for independent verification of valve positions.

**CORRECTIVE STEPS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED (Violation A)**

Operations performed an independent verification of system valve positions in accordance with the system valve lineup list. A review of valve lineup sheets for safety systems for which independent valve position verifications were previously completed and on which special procedures were subsequently performed was undertaken. All valves identified by this review were independently verified to be in the correct position.

For Item 2, the valve lineup form was changed to indicate the proper position for valve 1-DGCW-7 (Open).

**CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS (Violation A)**

For Items 1 and 2, Millstone Unit No. 1 Engineering management will discuss the proper controls that are necessary to control valve manipulations within special procedures with personnel in the Millstone Unit No. 1 Technical Support Group, including requirements for independent verification of valve positions.

Millstone Unit No. 1 Operator training will discuss Items 1 and 2 of this violation during requalification training. This discussion will be completed by March 31, 1995.

**DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED (Violation A)**

NNECO considers the immediate corrective actions sufficient to correct this condition. Therefore, NNECO is in full compliance with respect to these issues.

**REASON FOR THE VIOLATION (Violation B)**

The cause for Items 1 and 2 was the failure to properly verify that the procedures were revised. The required procedure changes were believed to have been verified prior to system turnover.

Item 3 was caused by a failure to properly identify all procedure revisions that were needed to support the issuance of a jumper bypass. This resulted in a failure to revise the emergency diesel generator (EDG) valve lineup sheet during implementation of a design change. The valve position was being controlled by a jumper bypass and valve lineup sheet. The jumper bypass form required the procedures that were revised during issuance of the jumper to be listed. However, the valve lineup sheet was not listed.

Item 4 involves a surveillance procedure SP-623.19, "Emergency Service Water Operational Readiness Test," which did not reflect the removal of service water system pressure indicator PI-4-6. This event was caused by an oversight during the PDCR process.

A contributing factor was lack of adequate planning and timely preparation of PDCRs required for implementation during the Cycle 14 refueling outage.

**CORRECTIVE STEPS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED (Violation B)**

For Item 1, the annunciator alarm response form for panel 903 A-2, window 6-4 was corrected.

For Item 2, the annunciator alarm response form for panel 903 A-2, windows 4-4 and 7-3 were corrected, and OP-307 was revised.

For Item 3, a plant information report (PIR 1-94-220) was initiated on May 4, 1994, to fully investigate this event and to recommend corrective actions. The valve lineup form has been corrected to reflect the proper position of 1-DGCW-7. The "Jumper, Lifted Lead, and Bypass Control" procedure was significantly strengthened to correct this and other observed weaknesses. A sign-off block for the Requestor's Department Head was added to ensure that "All procedure changes implemented for jumper device installation have been reviewed for applicability."

For Item 4, surveillance procedure SP-623.19 was corrected.

Planning and scheduling for the Cycle 15 refueling outage have already begun. Work Planning and Control (WP&C) is holding

outage meetings every other week to ensure scheduling for the next refueling outage is being appropriately assessed. Lessons learned from the Cycle 14 refueling outage are being incorporated into the planning efforts for the Cycle 15 outage by WP&C.

**CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS (Violation B)**

Cycle 15 refueling outage projects will be frozen at least 4 months prior to the start of the refueling outage, as specified in the "Outage Management" procedure (OM-1), to allow Engineering and Operations to properly coordinate the implementation and closure of design change packages. OM-1 contains guidelines for managing the outage. These include freezing project assignments, and ensuring that all PDCRs are PORC approved well in advance of starting the refuel outage, and limiting the projects to be worked to those which are necessary.

**DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED (Violation B)**

For Item 1, the annunciator alarm response form for panel 903 A-2, window 6-4 was corrected on May 3, 1994.

For Item 2, the annunciator alarm response forms for panel 903 A-2, windows 4-4 and 7-3 were corrected on May 3, 1994, and procedure OP-307 was revised on August 17, 1994.

For Item 3, the EDG valve lineup form was updated on May 5, 1994, to indicate the proper position of 1-DGCW-7.

For Item 4, surveillance procedure SP-623.19 was corrected on June 1, 1994. Therefore, NNECO is in full compliance with respect to all four of the identified items.

**REASON FOR VIOLATION (Violation C)**

Millstone Unit No. 2 operating procedures were developed utilizing the Final Safety Analysis Report, technical manuals, design drawings and experience from the other operating units at Millstone Station. All relevant information needed to successfully operate each system was placed into the procedures. The upper design temperatures of the service water and reactor building closed cooling water (RBCCW) systems were inadvertently omitted from the procedures.

**CORRECTIVE STEPS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED (Violation C)**

Service water, RBCCW, and condenser thermal backwashing procedures have been revised to include the system design temperature limits. Millstone Unit No. 2 has since successfully completed condenser thermal backwashing without exceeding design temperature limits.

**CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS (Violation C)**

Millstone Unit No. 2 operating procedures are being reviewed and upgraded as part of a station-wide procedure upgrade project. This upgrade process includes a review of design bases information. Applicable design bases information is included in the upgraded procedures. To further enhance the upgrade process and ensure appropriate design bases information is included in procedures, system engineers are reviewing selected primary system operating procedures during the upgrade process.

**DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED (Violation C)**

Since the operating procedures for service water, RBCCW and condenser thermal backwashing have been revised, Millstone Unit No. 2 is currently in full compliance.

**REASON FOR VIOLATION (Violation D)**

Item 1 was caused by a breakdown in communication between supervisors and workers. The job leader did not clearly define the job scope of actual work to be performed on valve 3SWP*AOV39A as required by the work order procedure. The job leader failed to assure that the maintenance area isolation was adequate for work to be performed. Also, the job leader did not document in the Automated Work Order (AWO) the actual work performed on valve 3SWP*AOV39A. In a response to a previous NRC Notice of Violation, NNECO committed to brief all Millstone Station personnel involved in the AWO process on lessons learned from that cited violation, and specifically on the need for proper documentation of work and adequate review of the work completion section of AWOs, prior to release to Operations. The job leader involved in this event was not employed at Millstone Station when the training was given and, as such, did not receive the training.

Item 2 was caused by confusion over assigned responsibilities, and lack of ownership and accountability for the jumper review process.

**CORRECTIVE STEPS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED (Violation D)**

For Item 1, the job leader involved in this event was counseled by management concerning management's expectation on this issue. Additionally, discussions were held with Millstone Unit No. 3 Maintenance personnel to reinforce management expectations concerning accuracy of job descriptions in work orders and requirements for adequate and accurate documentation of work performed. It was emphasized that accurate job descriptions are necessary to ensure comprehensive boundaries are established to maintain plant, equipment, and personnel safety. Accurate descriptions of actual work performed are necessary to ensure comprehensive and thorough retest are performed to validate equipment operability. Corrective actions from the previous similar event regarding expectations and the need to accurately document actual work performed within work orders were incorporated into a revision of the work control procedure. The new work control procedure became effective shortly after this event occurred.

Also, corrective actions from a previous similar event were incorporated into a revision of the work control procedure. The new work control procedure became effective shortly after this event occurred. A human performance evaluation was also performed to identify the root cause and contributing causes to determine additional corrective actions that may be needed to prevent recurrence.

For Item 2, management assigned the Millstone Unit No. 3 administrative staff the responsibility for updating the bypass jumper tracking system and providing reports to department managers. Also, the PORC secretary will be provided the list of bypass jumpers requiring PORC review on a monthly basis to facilitate proper review of bypass jumpers. In addition, scaffolding is now controlled under a new program with clearly defined responsibility for tracking.

**CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS (Violation D)**

NNECO has created a task force to perform a self-assessment on work control issues including performing work outside the scope

of work orders. Results from this task force will be available for resolution by October 21, 1994.

**DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED (Violation D)**

NNECO believes that the corrective actions specified above, and implemented in April 1994, are sufficient to prevent recurrence of this type of event. Therefore, NNECO is in full compliance.

**REASON FOR VIOLATION (Violation E)**

For Item 1, the cause of this event is attributed to failure of the operators and Reactor Engineering personnel to adequately monitor thermal limits and possible xenon transients following a power reduction.

For Item 2, the cause of this event is attributed to failure of the operators to correlate the APRM operability requirement with the technical specification requirements for the case when MFLPD exceeds FRP.

**CORRECTIVE STEPS THAT HAVE BEEN TAKEN AND THE RESULTS ACHIEVED (Violation E)**

Upon identification of MFPLD being greater than FRP, prompt corrective actions were taken to review core power shape and determine adjustments required to control rod pattern and core flow. The adjustments to the control rod pattern and core flow which restored MFLPD to less than FRP took less than 30 minutes to complete.

Operating Procedure 204, "Power Operations," was revised by adding steps to monitor MFLPD following power reductions in addition to the daily fuel surveillance. The operator logs were modified to clearly specify actions required when MFLPD exceeds FRP.

Operations Department management conducted briefings with operating shifts to establish clear management expectations for monitoring core thermal limits and requirements for entering Limiting Conditions for Operation (LCOs). Operations has issued a new instruction that clearly defines expectations and identifies specific responsibilities for Operations and Reactor Engineering personnel. The procedure will become effective on September 30, 1994.

**CORRECTIVE STEPS THAT WILL BE TAKEN TO AVOID FURTHER VIOLATIONS (Violation E)**

Plant management has initiated a self-assessment plan to evaluate the roles and responsibilities of and interaction between Reactor Engineering and Operations personnel. The plan will identify whether any additional qualifications or training are required to ensure full implementation of these roles and responsibilities and ensure adequate monitoring of thermal limits and full compliance with technical specifications. This self-assessment plan will be completed by November 27, 1994.

A technical specification change is being evaluated to clarify actions and sequence of actions to be taken when thermal limits are exceeded. This technical specification change will be initiated internally by December 20, 1994.

**DATE WHEN FULL COMPLIANCE WILL BE ACHIEVED (Violation E)**

NNECO considers the immediate corrective actions to be sufficient to prevent recurrence of this type of event. The long term corrective actions will further enhance this process. As such, NNECO is in full compliance.