NOTICE OF VIOLATION

Northeast Nuclear Energy Company
Millstone Nuclear Power Station
Unit 2

Docket No. 50-336
License No. DPR-65

During an NRC inspection conducted from April 13, 1998, through May 8, 1998, violations of NRC requirements were identified. In accordance with NUREG-1600, "General Statements of Policy and Procedure for NRC Enforcement Actions," the violations are listed below.

A. 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in the applicable design documents.

Millstone Unit 2 Technical Specification 6.13, "Systems Integrity," requires that, "The licensee shall implement a program to reduce leakage from systems outside containment that would, or could, contain highly radioactive fluids during a serious transient, or accident, to as low as practical levels."

10 CFR 50.55a, "Codes and Standards," Section (f), "Inservice Testing Requirements," requires that such valves be included in the ASME Section XI, inservice testing requirements (IST) leak testing program.

ASME Section XI, Article IWV-2200(a) classified such valves as Category A valves (i.e., "valves for which seat leakage is limited to a specified maximum amount in the closed position of fulfillment of their function.") Paragraph IWV-3421 required that such "Category A valves shall be leak tested...in a manner that demonstrates functionally adequate seat tightness..." (i.e., at a rate less than that which would cause the design-basis offsite or control room accident dose limits to be exceeded).

Contrary to the above, two examples were identified where the licensee was not performing leakage testing of safety-related valves in systems that could contain highly radioactive fluids during an accident are:

(1) ECCS containment sump isolation valves, 2-CS-16.1A&B, were not surveillance leakage tested per the above stated requirements. Additionally, two modifications requiring disassembly were performed on these valves and no post-modifications leakage testing was performed.

(2) ECCS suction isolation valves from the refueling water storage tank (RWST), 2-CS-14A&B and 2-CS-13.1A&B, were not surveillance leakage tested per the above stated requirements.

This is a Severity Level IV violation (Supplement I).

B. 10 CFR Part 50, Criterion XVI, "Corrective Action," states, in part, that, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." It also requires that, "In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition."

Section 1.1 of Station Procedure NGP3.05, "Nonconformance Reports", requires that, "The NCR is used to document and disposition nonconforming, materials, parts, components or services..."

The following three examples are contrary to the requirements listed above:

(1) The licensee failed to adequately determine the root cause of the corrosion of the 316 L stainless steel material of service water pump P5C's column in several cases over a period of several years and, therefore, failed to take appropriate corrective actions to preclude repetition.

(2) While conducting maintenance activities on the "A" reactor building component cooling water (RBCCW) heat exchanger in February 1998, the licensee identified, but failed to take prompt corrective action and issue a nonconformance report (NCR) to formally identify that incorrect washers of various sizes and materials were installed on the "C" RBCCW heat exchanger head during previous maintenance activities.

(3) The licensee installed a bypass jumper to the alarm contacts to prevent control room nuisance alarms without attempting to determine the root cause of the ground fault alarms. The reason for the jumper device was to eliminate the alarm. The alarm originated in the non-1E section of the alternate power supply to safety-related panel VA-40.

This is a Severity Level IV violation (Supplement I).

C. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires that, "Measures shall be established to assure that the applicable regulatory requirements and the design basis...are correctly translated into specifications, drawings, procedures, and instructions."

ASME Code Section VIII, Article UG-134, "Pressure Setting of Pressure Relief Devices," (a), states, in part, "When a single pressure relieving device is used, it shall be set to operate at a pressure not exceeding the maximum allowable working pressure of the vessel [the design pressure]."

Contrary to the above, the design requirements of Section VIII, Article UG-134(a), for pressure relief devices, were not correctly translated into the design for the RBCCW heat exchangers' relief valves' setpoints. The Code required that the relief valves' setpoints be no higher than the design pressure of 150 psig. The licensee incorrectly raised the setpoints to 165 psig.

This is a Severity Level IV violation (Supplement I).

D. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," states, in part, that, "Measures shall be established to assure that the applicable regulatory requirements and the design basis...are correctly translated into specifications, drawings, procedures, and instructions." It further states, in part, that, "Design changes, including field changes, shall be subjected to the design control measures commensurate with those applied to the original design..."

Contrary to the above, the licensee performed changes to the design of the P-41 "B" and "C" high pressure service injection (HPSI) pump seals without performing updates of the associated design drawings.

This is a Severity Level IV violation (Supplement I).

E. 10 CAR Part 50, Appendix B, Criterion III, "Design Control," states, in part, "The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews...." It further states, in part, that "design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design...."

The following five findings are contrary to the requirements listed above:

(1) The licensee failed to properly control the design and verification of pipe support calculation M2505194-01649-C2, Rev. 0, since Attachments were not properly signed by the individuals who performed and checked the calculation.

(2) The resolution of Generic Letter (GL) 87-02/USI A-46 at Millstone Unit 2 is provided by the "Generic Implementation (GIP) for Seismic Verification of Nuclear Plant Equipment," or GIP procedure. Contrary to the GIP requirements, cantilevered spans for cable trays Z25AA10 and Z24AA10 located in the containment building exceeded the maximum permissible spans and, therefore, should have been identified as outlier to the GIP and evaluated accordingly.

(3) The licensee failed to provide documented objective evidence to support the technical basis of Engineering Evaluation M2-EV-96-0061, Rev. 0, page 3 of 3, performed in support of Design Change Notice (DCN) No. DM2-00-1466-96. No specific reference to the calculation that would support the statement on page 3 of 3, "Fault current available over the entire length of the power circuit is adequate to actuate the trip element of any breaker with an instantaneous trip setting up to, and including, the HI setting," was included. Also, the statement that "Coordination reviews of 480-vac MCC circuits and upstream devices are based upon the largest breaker installed in the MCC...," was not referenced to the relevant coordination study. Section 6.0, "References," did not include any coordination study or calculations.

(4) As part of Jumper Device Control Sheet No. 2-96-052, a temporary diesel generator was installed to provide power to safety-related loads and to allow for an extended outage of the normal emergency diesel generator (EDG) "B." However, the provisions for feeding the safety loads from the temporary diesel generator did not include

consideration of protection and protective relaying features consistent with normal operation when using the safety-related diesel generator. Since the temporary generator step up transformer secondary winding was connected in delta, there was no source to detect a ground fault for protective relaying to operate, which differed from the grounding provided by the normal diesel generator.

The failure to include relevant protection requirements could result in undue exposure of the safety-related equipment while connected to the temporary diesel generator. The team concluded that the licensee had not conducted a complete engineering evaluation.

(5) Jumper Device Control Sheet No. 2-95-016, was for the replacement of EDG potential transformer fuses. Five (5)-amp fuses of a different type and make were installed for "B" EDG potential transformers in place of the previous 6-amp fuses. The evaluation of loadings failed to consider actual loading, but instead, reflected on 40 percent of the fuse rating, which may not have been adequate. The selection of the fuse was justified on the basis that coordination was "not required," however; the fuse should have coordinated with the potential transformer high-voltage fuses and should also have provided transformer protection. There was no discussion concerning the presence of any downstream fuses with which the fuse should also coordinate. Also, there was no evidence that any required coordination under energizing inrush conditions was considered.

This is a Severity Level IV violation (Supplement I).

F. 10 CFR 50.59, "Changes, Tests, and Experiments," (a)(1) states, in part, that the holder of a license authorizing operation of a production or utilization facility may make changes in the facility as described in the safety analysis report...without prior Commission approval, unless the proposed change, test, or experiment involves...an unreviewed safety question and (b)(1) states, in part, that the licensee shall maintain records of changes to the facility. These records must include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question.

Contrary to the above, the licensee made minor changes to the FSAR drawings in late 1997, but failed to perform safety evaluations pursuant to the requirements of (a)(i) as evidenced by the following examples:

(1) FSAR Figure 11.01-04 Sheet 1, P&ID 25203-260211 Sheet 1, "Aerated Liquid Radwaste System," was revised by Maintenance Support Engineering Evaluation (MSEE) DCN DM2-00-1102-97, "Resolution of Drawing Discrepancies for Radiation Monitoring Loop RM-9116 (UIR 3389)." A written safety evaluation was not performed.

(2) FSAR Figure 11.01-02, Sheet 1, P&ID 25203-26020, Sheet 2, "Aux Building Drains," was revised by MSEE DCN DM2-00-1104-97, "Drawing Update for Radiation Monitoring Loop RM-9049" (UIR 3352). A written safety evaluation did not envelope the change.

This is a Severity Level IV violation (Supplement I).

G. 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, that activities affecting quality be prescribed by and accomplished in accordance with documented procedures appropriate to the circumstances.

SP-EE-261, "Design Standards for Modification of Control Panels at Connecticut Yankee, Millstone Units 1, 2, and 3," Attachment 2, Section 1.1, "Instrument/Display Labels," requires the use of a delimiter between the device designator (e.g., "TI" for temperature indicator) and instrument loop.

Contrary to the above, SP-EE-261 was not followed for changes made to control room panel labels implemented over an indeterminate period before April 20, 1998. Specifically, the delimiter was a dash for all non-RG 1.97 Post-Accident Monitoring (PAM) devices, and a color coded dot for PAM instruments. Some non-PAM indicators were color coded, and some PAM indicators had a black dash. Some indicators such as the nuclear instruments had color coded labels (i.e., "A," "B," "C," and "D") above the instruments rather than using a dot on the label. The control room label deviations from the standard was indicative of a failure to perform adequate HFE reviews for changes.

This is a Severity Level IV violation (Supplement I).

H. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that ...measures shall be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components."

Contrary to the above the following five examples were identified where safety-related equipment was modified without ensuring the suitability of the new equipment for its intended use.

(1) PDCR 2-039-94 modified the auxiliary feedwater automatic initiation system but did not ensure that the electromagnetic interference (EMI) generated by the new equipment did not adversely affect other safety-related equipment.

(2) Three non-quality assurance (QA) bushings were installed in 4.16 kV safety-related 4.16 Switchgear cubicle A407 for the "C" Service Water Pump at Emergency Bus 24D, Facility Z2, without performing adequate suitability of application evaluation for the non-QA equipment. The acceptance of the non-QA devices was performed on the basis of a review that considered only a few of the critical characteristics for establishing equivalency.

(3) PDCR 2-050-93, dated July 13, 1995, installed two safety-related isolating transformers in an alternate feed path to safety-related equipment but failed to evaluate the electrical circuit changes introduced by the transformers. Because of the addition of the new transformers, the circuit impedance was substantially changed, which would have an effect on the voltage regulation and the short circuit profiles. The lack of required evaluations and/or calculations could jeopardize the operation of both redundant safety

divisions of vital ac power. While the main path of power would not be affected, both redundant alternate paths were affected.

Safety Evaluation (SE) No. SE-2-050-93, failed to include any objective evidence of an evaluation of the new failure modes introduced by the installation of two safety-related isolating transformers in alternate feed paths to safety-related equipment. For example, the SE Issue 3.2.1, "Effect on the probability that mitigating equipment will fail," was incorrectly annotated as "The credible failure modes are unchanged," which failed to recognize the fact that any failures associated with the new transformers would constitute new failure modes.

(4) PDCR 2-009-95 failed to provide an evaluation of impact of changing from an inverter type power supply to a transformer type power supply to safety-related circuits for the "A"and "B" Hydrogen Analyzer power circuits. These circuits were disconnected from VA10 and VA20 buses (fed from inverters) and reconnected to VA30 and VA40 buses (fed from transformers), to obtain higher short circuit current to provide for adequate coordination. The inverter type power supply is credited with a higher reliability, constituted by the dc battery source.

PDCR 2-009-95 also failed to provide an evaluation of impact of increasing the inverters frequency tolerance bandwidth from 1 percent to 2 percent, to provide objective evidence that indicated that the new frequency setting was tolerable and did not have any undesired effects in the operation of the connected safety-related instrumentation.

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Northeast Nuclear Energy Company is hereby required to submit a written statement or explanation within 30 days of receipt of the letter transmitting this Notice of Violation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Director, Special Projects Office, Office of Nuclear Reactor Regulation, and a copy to the NRC Resident Inspector at the Millstone Nuclear Power Station Unit 2. This reply should be clearly marked as a "Reply to a Notice of Violation," and should include the following information for each violation (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the required time specified in this Notice of Violation, an Order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated at Rockville, Maryland
this 12th day of August, 1998