August 11, 2000

Mr. Stephen E. Scace, Director
Nuclear Oversight and Regulatory Affairs
Northeast Nuclear Energy Company
PO Box 128
Waterford, CT 06385

SUBJECT: NRC's MILLSTONE INSPECTION REPORT NOS. 05000336/2000-008 AND 05000423/2000-008

Dear Mr. Scace:

On July 1, 2000, the NRC completed inspections at your Millstone Units 2 & 3 reactor facilities. The enclosed reports present the results of these inspections. The results were discussed on August 3, 2000, with Messrs. M. Brothers and R. Necci and other members of your staff.

These inspections were an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspections consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The NRC identified five Unit 2 and one Unit 3 issues that were evaluated under the risk significance determination process and were determined to be of very low safety significance (Green). These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection reports. These issues were determined to involve violations of NRC requirements. Consistent with the NRC Enforcement Policy, the violations are not cited. If you contest these noncited violations, you should provide a response within 30 days of the date of these inspection reports, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Millstone facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

James C. Linville, Chief
Projects Branch 6
Division of Reactor Projects

Docket Nos.: 05000336, 05000423
License Nos.: DPR-65, NPF-49

Enclosures: (1) NRC Inspection Report 05000336/2000-008
(2) NRC Inspection Report 05000423/2000-008

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ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	05000336
License No.:	DPR-65
Report No.:	05000336/2000-008
Licensee:	Northeast Nuclear Energy Company
Facility:	Millstone Nuclear Power Station, Unit 2
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	May 14, 2000 - July 1, 2000
Inspectors:	D. P. Beaulieu, Senior Resident Inspector, Unit 2 P. C. Cataldo, Resident Inspector, Unit 2 S. R. Jones, Senior Resident Inspector, Unit 2 P. Frechette, Physical Security Inspector, Division of Reactor Safety (DRS) K. Jenison, Senior Project Engineer, Division of Reactor Projects M. Modes, Senior Reactor Inspector, DRS T. A. Moslak, Health Physicist, DRS G. C. Smith, Senior Physical Security Inspector, DRS
Approved by:	James C. Linville, Chief Projects Branch 6 Division of Reactor Projects Region I

SUMMARY OF FINDINGS

IR 05000336/2000-008 on 05/14-07/01/2000; Millstone Nuclear Power Station; Unit 2. Fire Protection, Maintenance Rule Implementation, Maintenance Risk Assessment and Emergent Work Evaluation, Operability Evaluations, Post Maintenance Testing.

The inspection was conducted by resident and regional inspectors. This inspection identified five green issues, all of which were noncited violations. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (see Attachment 1).

Cornerstone: Mitigating Systems

- **Green.** The NRC identified that the licensee had not adequately maintained fire fighting strategies, which could reduce the effectiveness of manual fire fighting. This failure to adequately maintain manual fire fighting implementing procedures as required by Unit 2 Technical Specification 6.8.1.f is being treated as a non-cited violation. Because manual fire suppression is the principal method of fighting fires only in areas where safe-shutdown equipment trains are separated by at least three-hour rated fire barriers, the Fire Protection Significance Determination Process characterizes a reduction in manual fire suppression effectiveness alone as a condition of very low safety significance. (Section 1R05)

- **Green.** The NRC found that inadequate instructions for filling the chilled water system following maintenance led to the common-cause failure of both vital DC switchgear cooling trains due to air binding of the associated vital chilled water pumps. This failure to adequately implement procedures for filling the chilled water system as required by Unit 2 Technical Specification 6.8.1.a is being treated as a non-cited violation. Evaluation using the NRC Significance Determination Process revealed that the safety significance of this common cause failure of vital DC switchgear cooling was very low because the exposure time was short, the normal cooling system was in operation, the compensatory measures for loss of cooling were proceduralized, and the vital DC switchgear cooling trains are only initiated for events involving a loss of offsite power or safety injection. (Section 1R12.1)

- **Green.** The NRC found that the licensee inappropriately authorized performance of a work order for replacement of the “D” reactor coolant pump seal when reactor coolant system (RCS) level was above the elevation of the seal. Although RCS level was below the seal prior to removal, the inadequate control of maintenance activities resulted in control room operators being unaware that an opening in the RCS existed during RCS draining activities. This failure to adequately establish and implement procedures for control of maintenance activities as required by Unit 2 Technical Specification 6.8.1.a is being treated as a non-cited violation. The NRC evaluated this condition using the Shutdown Operations Significance Determination Process and concluded that the condition was of very low safety significance because the licensee had planned and implemented appropriate controls to reduce RCS level below the opening created by the seal removal. The NRC also found that the licensee’s corrective action plan for this condition was inadequate in that it did not address the work control process. (Section 1R13.1)

- **Green.** The NRC identified that the licensee had not provided adequate justification for operability of the reactor building closed cooling water (RBCCW) system when multiple thermal relief valves lifted during pump starts under conditions simulating a loss of normal power. The licensee had determined that lifting of RBCCW relief valves was acceptable once three relief valves that had failed to reseat during testing were gagged. However, the NRC found that the licensee had failed to take adequate corrective actions to address the increased probability of failure of the RBCCW system due to loss of inventory through relief valves that fail to reseat. This violation of Criterion XVI, “Corrective Action,” of 10 CFR Part 50, Appendix B, is being treated as a non-cited violation. Because the condition was addressed prior to Unit 2 startup from refueling by gagging other relief valves, no actual loss of safety function occurred, and the Significance Determination Process screened this condition as one of very low safety significance. (Section 1R15.1)

- **Green.** The NRC identified that the licensee had not implemented measures to ensure adequate train independence for the reactor building closed cooling water (RBCCW) system. This violation of Criterion XI, “Test Control,” of 10 CFR Part 50, Appendix B, is being treated as a non-cited violation. Because no loss of function of the train separation valves was identified, no actual loss of safety function occurred, and the Significance Determination Process screened this condition as one of very low safety significance. (Section 1R19.1)

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Report Details

SUMMARY OF UNIT 2 STATUS

The plant entered the inspection period in Operational Mode 6, Refueling, with core reload complete and the reactor vessel head in place. On May 30, 2000, the plant entered Operational Mode 2, Startup, for low-power physics testing and subsequent power ascension. The plant was shutdown briefly on May 31, 2000, to correct control rod problems (Sections 1R14.2 and 4OA3). On June 4, 2000, the plant experienced an uncomplicated automatic reactor trip from 65 percent power during main turbine control system testing (Sections 1R12.3 and 1R14.1). Operators performed a reactor startup later on June 4, 2000, conducted a routine post-refueling power ascension to 100 percent power, and maintained the plant at 100 percent power through the end of the inspection period.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity)

1R04 Equipment Alignment

a. Inspection Scope

Inspectors performed the following partial system alignment checks:

Following the performance of surveillance tests affecting auxiliary feedwater (AFW) system valves, the inspector verified that valves associated with the “A” motor-driven AFW pump and the turbine-driven AFW pump were properly aligned using procedure SP 2610C, “Auxiliary Feedwater System Lineup Verification,” and system piping and instrumentation diagrams 25203-26002 and 25203-26005.

With the “C” charging pump in operation and the “B” charging pump out of service for maintenance, the inspector verified that the “A” charging pump was correctly aligned using procedure SP 2601B, “Boric Acid Flowpath Verification, Facility 1,” and system piping and instrumentation diagram 25203-26017.

b. Findings

There were no findings identified.

1R05 Fire Protection

a. Inspection Scope

The inspector toured the following plant areas to evaluate the operational status of the fire suppression systems protecting these areas, the condition of penetration seals and other fire barriers, and the control of transient combustible materials located in these areas: (1) Auxiliary Building -45' General Area, Fire Zone A-1A; (2) Auxiliary Building - Reactor Building Closed Cooling Water Pump and Exchanger Area, Fire Zone A-1B; (3) Auxiliary Building Waste Tank Pump Room, Fire Zone A-1C; (4) Auxiliary Building -5' General Area, Fire Zone A-1G, and; (5) Auxiliary Building Volume Control Tank Room, Fire Zone A-1H.

b. Findings

The inspector found that the licensee had not adequately maintained fire fighting strategies, which could reduce the effectiveness of manual fire fighting. However, since manual fire suppression is the principal method of fighting fires only in areas where safe-shutdown equipment trains are separated by at least three-hour rated fire barriers, the Fire Protection Significance Determination Process characterizes a reduction in manual fire suppression effectiveness alone as a condition of very low safety significance (Green).

As an example of this condition, the volume control tank room, Fire Zone A-1H, contains a hydrogen supply line to the volume control tank that was not mentioned in the associated fire fighting strategy. The hydrogen supply isolation valve within the area, valve 2-CH-107, is normally open. In the event of a fire in this room, valve 2-GAH-30, which is located in the turbine building, would be closed or verified closed to isolate the hydrogen.

Although there have been six changes to the fire fighting strategies since December 1999, the inspector found that the strategies had not been previously updated since 1993. This was because the Design Control Manual had not previously required an assessment of whether the strategies were impacted by plant modifications. In 1995, the licensee had created a draft update to the strategies, but, due to differing views within licensee management whether the strategies needed to kept up to date, the draft update was never approved and implemented. The failure of the strategies to reflect the hydrogen line in the volume control tank room was a deficiency addressed in the 1995 draft strategy, but it was never issued. Although the licensee has a copy of the 1995 draft strategy, the licensee does not have a current understanding of what other deficiencies may exist. The licensee initiated Condition Report (CR) M2-00-1796 to document the concerns with the fire fighting strategies. As an initial corrective action, the licensee changed the fire fighting strategy to reflect the hydrogen line.

Unit 2 Technical Specification 6.8.1.f requires that written procedures be established, implemented and maintained covering the fire protection program implementation. Figure 6.1, “Fire Protection Program Responsibilities and Implementing Procedures Matrix,” of the Millstone Fire Protection Program Manual identifies the Fire Fighting Strategy Manual as an implementing procedure for fire response. The Millstone Fire Protection Manual section 4.6.2, "Fire Fighting Strategies," specifies that the strategies address plant systems that should be disabled or managed to reduce the damage potential during a fire and the location of local and remote controls for such actions. Contrary to the above, the fire fighting strategy for Fire Zone A-1H was not maintained in that the hazard posed by the hydrogen supply line was not identified. This violation of Technical Specification 6.8.1.f is being treated as a Non-Cited Violation **(NCV 50-336/2000-008-01)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), in that the associated condition was of very low safety significance and was entered in the licensee’s corrective action program.

1R07 Heat Sink Performance

a. Inspection Scope

The inspector observed the performance of data collection activities under procedure EN 21246, “Thermal Performance Test of Unit 2 RBCCW Heat Exchangers X-18A/B.” The inspector also reviewed Proto-Power Corporation Calculation No. 00-067, Rev. A, “Analysis of X-18A and X-18B Thermal Performance Test Results,” and verified that the acceptance criteria were satisfied with consideration for instrument inaccuracies and measurement uncertainties.

b. Findings

There were no findings identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

The inspector reviewed eddy current data taken from steam generator tubes containing indications of sludge (R116C35) and manufacturing buffing marks (R159C54) and interviewed the individual responsible for coordinating the inspections. The inspector also reviewed data from a tube chosen at random (R62C25) and a row 3 tube (R3C60).

The inspector reviewed the loose part analysis for an item identified as a steel dowel originally located in the vicinity of R20C27. Due to sludge lancing of the steam generator the part moved to R140C93 where bobbin data was acquired. The inspector reviewed the bobbin data. The inspector reviewed Northeast Nuclear Energy Company’s (NNECO) revision of the computer aided sorting criteria, used to supplement the second analysis team, so it would capture the originally missed loose part data. The inspector reviewed a randomly chosen reanalysed tube data.

The inspector reviewed the current 10 year inspection interval plan which was submitted to the NRC for review on July 2, 1996. The inspector interviewed the Inservice Inspection Coordinator for Millstone Unit 2 and discussed the statement contained in the cover letter of July 22, 1998, transmitting the Safety Evaluation Report (SER), in which the NRC staff noted that "NNECO is not performing inservice volumetric, surface, and visual examinations ... using sample schedules as described in Section XI". In addition the SER in paragraph 2.2.2 states that NNECO is meeting the Code for schedule, except for the volumetric examination of reactor pressure vessel closure head studs, required by Examination Category B-G-1, Item B6.20. The inspector reviewed the “Millstone Unit 2 Interval 3 - Category Summary Report” which indicated the stud exams were not deferred but completed with the following periodicity: Period 1 = 25%, Period 2 = 49%, and Period 3 = 103% (spare included in sample set). The inspector discussed the need to clarify the record in this regard with NNECO management and the Regulatory Affairs coordinator.

The inspector reviewed the results of the ultrasonic inspection of Pressurizer Relief Valve Nozzle-to-Head PR-NTH-1 (DWG 25203-29527 SH 15) which reported a spot indication at 18% of the Distance Amplitude Correction (DAC) curve in addition to three other previously reported indications. The inspector reviewed the disposition of the indication by re-examination with 45 degree and 60 degree techniques supplemented by 0 degree technique.

The inspector also reviewed the ultrasonic examination results for Pressurizer Safety Valve Nozzle-to-Top-of-Head (PR-NTH-3). The inspector reviewed the inspection report which originally reported a 2 inch long flaw at midwall with both a 45 and 60 degree technique. The inspector reviewed the disposition of this indication by re-examination and its return to service as acceptable under ASME Table IWB 3512-1.

The inspector reviewed the only ASME work performed this outage requiring radiography and interviewed the Radiography Coordinator at NNECO. The work was for the weld of the cap of Spare Containment Penetration 48; used to bring support lines into containment for the purposes of sludge lancing the Steam Generators. This was implemented as part of an ASME Section XI Repair and Replacement Plan; reviewed by the inspector. The testing was first done by penetrant; the results of which were reviewed by the inspector. The inspector reviewed the Safety Evaluation (S2-EV000-0017) and the three radiographs which were done in conformance with ASME Section III 1971. The inspector also reviewed the code reconciliation with USAS B31.7, the code of record for construction for Millstone Unit 2. The inspector reviewed welding procedure WPS 001 Rev 3 which was used to join the A 333 Grade 6 or A 155 Gr KCF-70 material. Qualification of a welder dated 1/27/95, by radiography, was also reviewed.

The inspector reviewed “Weld Repair of Spool Piece P012 (6"-JGD-4) in the Service Water System 2326A” used to replace base metal degradation on the outlet flange. The inspector also reviewed “Repair and Replacement Plan to Replace Snubber for Pipe Support HGR-416014A”, “Repair Casing of TDAFW Pump P4" due to pump casing surface irregularities and areas of pitting identified by Ingersoll-Dresser Pump Corp., “Repair Hinge and Stop Pin Holes for Valve 2-SW-13B” in the service water system, and “Repair Flange Bolting for Spectacle Flange at Penetration 85".

NNECO has not implemented a non-Code repair in approximately 4 years and the interviewed NNECO staff could not remember any non-Code repairs on risk significant systems. The only example of a non-Code repair was the repair by clamping of the Class 3 "B" Service Water system discharge piping (line 24"-JGD-6, Spool SK 923) from the Reactor Building Closed Cooling Water system heat exchangers. The leak was originally caused by a split in the PVC lining which resulted in an area approximately 2" in diameter eroded with a pinhole leak. This temporary repair was removed during the current outage and the piping replaced. The inspector reviewed the evaluation and relief request performed in conformance with the guidance in NRC Generic Letter 90-05 and discussed the repair with the Unit 2 Service Water System Engineer as well as any generic implications it might have. The inspector ascertained, through discussion with NNECO staff, that the Unit 3 Service Water System Engineer had been informed of the leak in Unit 2.

b. Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

.1 Common Cause Failure of Both Vital DC Switchgear Cooling Trains

a. Inspection Scope

The inspector reviewed maintenance rule implementation and corrective actions associated with the common-cause failure of both vital DC switchgear cooling trains. The inspector reviewed work order M2-00-04399 and Operating Procedure OP 233OC, "Chilled Water Systems".

b. Issues and Findings

Inadequate venting following maintenance affecting the chilled water system led to the common-cause failure of both vital DC switchgear cooling trains due to air binding of the associated vital chilled water pumps. This common-cause failure was found to be of very low safety significance because the exposure time was short, the normal cooling system was in operation, the compensatory measures for loss of cooling were proceduralized, and the vital DC switchgear cooling trains are only initiated for events involving a loss of offsite power or safety injection.

After draining a heat exchanger in the non-vital chilled water system for maintenance under work order M2-00-04399, the heat exchanger was filled by initiating chilled water flow to flush the air to the chilled water surge tank. However, the vital chilled water system shares piping with the non-vital chilled water system for cooling the DC switchgear rooms, and some of the air collected in the idle suction piping for the vital chilled water pumps near the interface with the non-vital chilled water system. Approximately 60 hours after the heat exchanger was filled, operators shut down Unit 2 from 100 percent power for refueling on April 21, 2000. About 50 hours after the shutdown, operators found that the “A” chilled water pump became air bound when it was placed in service for routine testing of standby equipment. Subsequently, operators found that the “B” chilled water pump also became air bound when it was placed in service for testing. The licensee documented these failures in condition reports (CRs) M2-00-0956 and M2-00-0979.

The inspector found that, although draining the heat exchanger was within the scope of the work order, the work order did not provide instructions for filling the heat exchanger nor measures for verifying the chilled water system was properly filled and vented. Also, Operating procedure OP 2330C, “Chilled Water System,” provided instructions for filling and venting large portions of the chilled water system, but these instructions were not implemented immediately following the maintenance.

The inspector and a Region I Senior Reactor Analyst (SRA) evaluated this condition using the NRC's Significance Determination Process. The chilled water system was explicitly modeled in the NRC's risk assessment model for Millstone Unit 2, so the SRA performed an analysis assuming that both vital chilled water pumps were inoperable for the 60 hour exposure time and that the probability of operator failure to recover DC switchgear room cooling was 0.01. Because the vital chilled water system would only be initiated by events involving a loss of offsite power or safety injection actuation, the loss of offsite power initiating event was most significant for this analysis. The analysis results indicated a change in core damage frequency of 8.7 E-07 per year, which the NRC classifies as a condition of very low safety significance (green).

Unit 2 technical specification 6.8.1.a requires that written procedures be established, implemented, and maintained for the activities described in Appendix A of Regulatory Guide (RG) 1.33, "Quality Assurance Program Requirements (Operation)." Section 3 of RG 1.33, Appendix A describes filling of various safety related systems, including the auxiliary building heating and ventilation system. The chilled water system is part of the auxiliary building heating and ventilation system. The licensee had neither established an appropriate procedure to fill the heat exchanger that was drained for maintenance nor implemented the existing procedure OP 2330C to fill and vent appropriate portions of the chilled water system following the maintenance. This failure to adequately implement a procedure covering the filling of the chilled water system is a violation of Technical Specification 6.8.1.a. This violation is being treated as a Non-Cited Violation **(NCV 50-336/2000-008-02)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), in that the associated condition was of very low safety significance and was entered in the licensee's corrective action program.

The inspector reviewed Revision 10 to the maintenance rule action plan for the chilled water system, which was approved on June 27, 2000. The inspector found that the failures of the vital chilled water pumps were appropriately classified as maintenance-preventable functional failures and that unavailability time accrued as a result of the condition was evaluated. The chilled water system has not satisfied its performance criteria since initial implementation of the maintenance rule and remained in a(1) status. Previous failures involved the chillers, so this failure of the chilled water pumps does not indicate that previous corrective actions have been inadequate.

.2 Lifting of Reactor Building Closed Cooling Water Relief Valves

a. Inspection Scope

The inspectors reviewed maintenance rule implementation associated with relief valves failing to reseat in the reactor building closed cooling water (RBCCW) system. The inspector verified that the condition was correctly classified as a maintenance-preventable functional failure.

b. Findings

There were no findings identified.

Findings regarding licensee corrective actions to address the RBCCW relief valve lifting are discussed in Section 1R15.1 of this report.

.3 <u>(Closed) LER 50-336/2000-010-00: Reactor Trip during Testing of Turbine Power-Load Unbalance Circuit</u>

a. <u>Inspection Scope</u>

The inspector reviewed the maintenance rule implementation and corrective actions associated with a failed push-button test switch for the power/load unbalance trip circuit of the turbine electro-hydraulic control (EHC) system, which resulted in an uncomplicated automatic reactor trip on June 4, 2000. The inspector verified that the failure was correctly classified as a maintenance-preventable functional failure based on General Electric Technical Information Letter 1212-2, "Plant Scram Frequency Reduction Features for BWR and PWR Nuclear Turbines with MK I or MK II EHC Controls," issued January 27, 1997, which described the failures of the push button-test switch at other facilities, and the licensee's cancellation of an engineering work request to perform a modification to address the concern. The inspector also verified that appropriate near-term corrective actions were implemented prior to returning the main turbine to service.

b. <u>Findings</u>

There were no findings identified.

Weak licensee performance in addressing the longstanding industry problem with the push-button test switch is discussed in Section 4OA2.

1R13 <u>Maintenance Risk Assessment and Emergent Work Evaluation</u>

.1 <u>Inadequate Control of Reactor Coolant Pump Seal Replacement Work</u>

a. <u>Inspection Scope</u>

The inspector reviewed work controls implemented for the replacement of the "D" reactor coolant pump (RCP) seal while the reactor coolant system level was being drained to the centerline of the hot leg nozzle.

b. <u>Findings</u>

The licensee inappropriately authorized performance of a work order for replacement of the "D" RCP seal when reactor coolant system (RCS) level was above the elevation of the seal. Although RCS level was below the seal prior to removal, the inadequate control of maintenance activities resulted in control room operators being unaware that an opening in the RCS existed during RCS draining activities. Because the potential for a loss of decay heat removal or a significant loss of RCS coolant inventory was minimal, this failure to implement adequate work controls was of very low safety significance. The licensee's corrective action plan for this condition was inadequate in that it did not address the work control process.

Following "D" RCP seal replacement work on May 18, 2000, the licensee's nuclear oversight organization documented concerns with the adequacy of work controls for the seal replacement work in condition report (CR) M2-00-1487. The inspector reviewed work order M2-99-14057, "'D' Reactor Coolant Pump Seal Replacement," and the Shift

Manager's Log for the period from May 17 to May 18, 2000, and the licensee's root cause investigation report for CR M2-00-1487. The work order documents that a senior reactor operator performing work control functions authorized work to replace the "D" RCP seal at 9:45 p.m. on May 17, 2000. At that time, the Shift Manager's Log indicates that RCS level was stable at 70 inches above the hot leg centerline, which is a level that would not support the seal replacement work. During activities to drain the RCS level to the hot leg centerline on May 18, 2000, the maintenance supervisor verified that RCS level had been drained low enough to support RCP seal replacement and completed the removal of the "D" RCP seal. However, the licensee had scheduled the seal replacement work to begin after completing RCS draining activities and the licensee found that the control room operators were unaware of the opening in the RCS pressure boundary created by the seal removal while they were draining the RCS. This condition was a concern because the procedural actions to address an uncontrolled loss of RCS coolant inventory or a loss of shutdown cooling involve raising RCS level, which could have resulted in a loss of RCS makeup inventory through the opening.

The inspector and a Region I Senior Reactor Analyst (SRA) evaluated this condition using the NRC's Shutdown Operations Significance Determination Process because the condition degraded the ability of the licensee to add RCS inventory while the RCS was closed and at reduced inventory. However, the NRC concluded that the condition was of very low safety significance (Green) because the licensee had planned and implemented appropriate controls to reduce RCS level below the opening created by the seal removal. Therefore, the potential for a loss of decay heat removal or a significant loss of RCS coolant inventory was minimal.

The licensee's root cause investigation identified the root cause of the incorrect sequence of activities as repeated failures of verbal communications among outage management, maintenance, and operations personnel. The root cause investigation report also noted that it was not uncommon for operations personnel to release work early and place responsibility for verifying that plant conditions supported the work with the maintenance supervisor. The report also documented that this practice was consistent with procedure U2 WC 1, "Unit 2 Work Control Process." Consequently, the licensee did not identify any corrective actions associated with their work control process.

However, in Appendix D to the Northeast Utilities Quality Assurance Program Topical Report, the licensee commits to utilize the guidance of Regulatory Guide (RG) 1.33, "Quality Assurance Requirements (Operation)," and ANSI N18.7-1976/ANS 3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," which RG 1.33 endorses. Section 5.2.6, "Equipment Control," of ANSI N18.7 states that, prior to granting permission for maintenance, designated operating personnel shall verify that the equipment can be released. Unit 2 Technical Specification 6.8.1.a requires that written procedures be established, implemented, and maintained for the activities described in Appendix A of RG 1.33, "Quality Assurance Program Requirements (Operation)." Section 9 of RG 1.33, Appendix A, describes general procedures for control of maintenance, including obtaining permission for work. Step 1.5.32 of procedure U2 WC 1 specifies that a senior reactor operator licensed individual authorize release of work. Procedure U2 WC 1 was not adequately established and implemented in that operators failed to ensure plant conditions would allow RCP seal replacement prior to authorizing the work, which is a violation of Technical Specification 6.8.1.a. This violation is being treated as a Non-Cited Violation

(NCV 50-336/2000-008-03), consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), in that the associated condition was of very low safety significance and was entered in the licensee's corrective action program.

The inspector found that the licensee did not identify appropriate corrective actions for this condition in that multiple levels of the licensee's organization accepted early release of work without appropriate controls to ensure plant conditions would allow its safe performance. The licensee's corrective action plan did not identify any changes in their work control process to address this condition. The inspector discussed this concern with the manager responsible for corrective actions. Subsequently, the licensee reopened their root cause investigation for CR M2-00-1487.

.2 Emergent Work on "B" Charging Pump

a. Inspection Scope

The inspector reviewed work controls implemented to manage risk when emergent work to repair a failed discharge relief valve on the "B" charging pump was performed on a day where high-risk surveillance activities were scheduled. The inspector verified that the licensee managed risk to an acceptable level in accordance with their on-line maintenance procedures.

b. Findings

There were no findings identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 Automatic Reactor Trip During Weekly Surveillance of the Main Turbine Control System

a. Inspection Scope

In response to an automatic reactor trip from 65 percent power during turbine control system testing on June 4, 2000, the inspector performed a detailed review of plant process computer data to verify operator response was in accordance with emergency operating procedure (EOP) 2525, "Standard Post-Trip Actions," and EOP 2526, "Reactor Trip Recovery."

b. Findings

There were no findings identified.

.2 Reactor Shutdown Complicated by Control Rod Malfunction

a. Inspection Scope

The inspector observed operator performance during a planned reactor shutdown and subsequent dropped rod event that occurred on May 31, 2000. One or more dropped rods were anticipated by the operators based on problems experienced with control rods during the low power physics testing conducted the previous day. The inspector verified that operator actions were performed in accordance with procedure OP 2206, “Reactor Shutdown,” and the abnormal operating procedure addressing dropped control rods, AOP 2556, “CEA Malfunctions.”

b. Findings

There were no findings identified.

1R15 Operability Evaluations

.1 Lifting of Reactor Building Closed Cooling Water System Relief Valves

a. Inspection Scope

The inspectors reviewed Technical Evaluation M2-EV-00-0034, which provided the licensee’s technical justification for acceptance of the lifting of the thermal relief valves on the heat exchangers of the reactor building closed cooling water (RBCCW) system due to pressure transients from RBCCW pump starts under conditions simulating a loss of normal power.

b. Findings

The inspector found that the licensee had not provided adequate justification for operability of the RBCCW system when multiple thermal relief valves lifted during pump starts. This condition had very low safety significance because the problem was identified and corrected while the plant was shutdown for refueling.

Each of the heat exchangers cooled by the RBCCW system has an associated thermal relief valve that protects the heat exchanger from over-pressure due to thermal expansion of water when the heat exchanger is isolated for maintenance. Normally, the RBCCW system pressure remains below the setpoint of these relief valves during system operation. However, maintenance activities during the refueling outage introduced air into the RBCCW system that operators were unable to remove by venting. During the performance of the “A” train loss of normal power test on May 7, 2000, two thermal relief valves lifted and failed to reseat. The licensee determined that the air trapped in the RBCCW system caused a minor water hammer during system transients, such as pump starts, that caused multiple relief valves to lift.

Despite licensee attempts to better vent the system, subsequent testing of both RBCCW trains showed that the RBCCW relief valves lifted each time the RBCCW pumps were started under conditions simulating a loss of normal power. During subsequent testing, an additional relief valve lifted and failed to reseat. The licensee determined that the lifting of the RBCCW system thermal reliefs following a postulated loss of offsite power did not render the RBCCW system inoperable because, with the exception of the three relief valves that failed to reseat and were subsequently gagged shut, the relief valves functioned properly by lifting and reseating.

The inspector reviewed the licensee's technical evaluation and did not agree that the RBCCW system was operable with the thermal relief valves lifting following a loss of normal power for the following reasons:

(1) Testing showed that relief valves would lift each time the RBCCW pumps were started under conditions simulating a loss of normal power. The RBCCW system is a closed system with limited inventory, and the installed inventory makeup system is not designed to function following a loss of normal power. Therefore, RBCCW system operability hinged upon the ability to maintain inventory by proper seating of the thermal relief valves.

(2) The relief valves were neither procured nor tested to ensure each valve would reliably reseat under conditions representative of those experienced following system transients. The Millstone Unit 2 Final Safety Analysis Report states that the thermal relief valves provide over-pressure protection when components are isolated. Therefore, the routine lifting of relief valves near in-service components and heat exchangers was not within the design basis of the RBCCW system.

After discussing the concerns with NRC management and technical personnel in both Region I and headquarters, the inspector informed the licensee of the NRC position that the licensee's technical evaluation did not provide a sufficient technical basis to support RBCCW system operability. The licensee documented this concern in Condition Reports M2-00-1609 and M2-00-1741. The licensee's near-term corrective actions to address the NRC's concerns included preparing a safety evaluation and gagging all affected RBCCW relief valves prior to entering Mode 2, Startup. The inspector found this corrective action acceptable to address RBCCW operability. Because the deficiency was addressed prior to Unit 2 startup, the finding did not represent an actual loss of safety function of the RBCCW system. Therefore, the risk associated with the licensee's initial inadequate corrective actions was determined to be very low and was characterized as Green by the Significance Determination Process.

The failure of the licensee to take adequate initial corrective actions to address RBCCW relief valves lifting in the event of a loss of normal power is a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions." This violation is being treated as a Non-Cited Violation **(NCV 50-336/2000-008-04)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), in that the associated condition was of very low safety significance and was entered in the licensee's corrective action program.

.2 Spent Fuel Pool Cooling Operability for Cycle 14

a. Inspection Scope

The inspector reviewed Operability Determination (OD) MP2-021-99 (for Cycle 13), and ODs MP2-004-00, Revision 2 (for Cycle 14), which addressed calculated stress at the spent fuel pool cooling pump discharge nozzle exceeding the vendor allowable stress at the design operating temperature of 150°F. The inspector verified that an adequate basis was presented for the continued operability of the spent fuel pool cooling pumps for the limiting design basis spent fuel pool temperature.

b. Findings

There were no findings identified.

.3 “B” Charging Pump

a. Inspection Scope

The inspector reviewed Operability Determination MP2-025-00, which addressed reliability of the “B” charging pump discharge relief valve following repairs to correct the valve’s failure in an open position during surveillance testing. The inspector verified that the licensee had an adequate basis for the continued operability of the “B” charging pump in that:

(1) The relief valve exhibited indications of seat leakage and was repaired.

(2) Minor seat leakage causes the relief valve to lift at pressures well below its set pressure.

(3) The licensee had detected the relief valve seat leakage by a slight reduction in flow from the “B” charging pump during a surveillance test the day prior to the relief valve failure and the subsequent surveillance test was performed to identify the cause.

b. Findings

There were no findings identified.

1R17 Permanent Plant Modifications

Spent Fuel Pool Cooling Modification

a. Inspection Scope

The inspector reviewed Design Change Record (DCR) M2-00007, “Spent Fuel Pool Cooling Analysis for 2R13 and 2C14,” which revised the Millstone Unit 2 spent fuel pool cooling design basis for refueling outage 13 and operating cycle 14. Using simplified calculational methods, the inspector verified that the calculated cycle-specific spent fuel pool temperature changes for various spent fuel pool cooling system operating conditions were accurate. The inspector also verified that necessary administrative controls (e.g., emergency operating procedure changes and Technical Requirements Manual revisions) were implemented to ensure values used in the design basis would bound actual operating values for important parameters.

b. Findings

There were no findings identified.

1R19 Post Maintenance Testing

.1 Reactor Building Closed Cooling Water System Train Separation

a. Inspection Scope

The inspector reviewed measures implemented to ensure adequate separation of the two independent reactor building closed cooling water (RBCCW) system trains following maintenance activities.

b. Findings

During routine Shift Manager's Log reviews, the inspector identified that the operational "B" RBCCW train lost coolant inventory during draining of the "A" RBCCW train when the plant was in operational mode 6, refueling. The licensee found that the loss of inventory was caused by the failure to maintain one of several large-diameter, air-operated butterfly valves that provide train separation fully closed when the valves were placed in manual for tagging.

The inspector evaluated the measures the licensee had implemented to ensure train separation when the system is returned to service. Because each RBCCW system train is a closed system with limited inventory makeup capability, a loss of inventory would render the train inoperable. Certain initiating events (e.g., pipe whip following a high energy line break inside containment) could cause failure of the pressure boundary for one RBCCW train, and inadequate train separation could lead to failure of the redundant train. The licensee documented NRC concerns with the adequacy of train separation in condition report M2-00-1543.

Although the train separation valves have functioned properly, the inspector found that the licensee had no periodic or post-maintenance test to verify adequate train separation. This condition is a violation of Criterion XI, "Test Control," of 10 CFR Part 50, Appendix B. Because no actual degradation of train separation valves was identified, the condition was evaluated through the Significance Determination Process as a condition of very low safety significance (green). This violation is being treated as a Non-Cited Violation **(NCV 50-336/2000-008-05)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), in that the associated condition was of very low safety significance and was entered in the licensee's corrective action program.

.2 "B" Charging Pump Testing

a. Inspection Scope

The inspector observed the post-maintenance bench testing of valve 2-CH-325, the "B" charging pump discharge relief valve, which had previously stuck open during testing of the "B" charging pump, and verified that valve operation was acceptable. The inspector also verified that post-maintenance testing associated with automated work order M2-00-12912, "'B' Charging Pump Discharge Relief Valve," in combination with a surveillance test performed by operators adequately demonstrated operability of the "B" charging pump.

b. Findings

There were no findings identified.

1R20 Refueling and Outage Activities

.1 Refueling Outage Inspections

a. Inspection Scope

The inspectors reviewed the following activities related to the Unit 2 refueling outage for conformance to applicable procedural and technical specification requirements, and witnessed selected evolutions.

- shutdown risk evaluations
- second planned reduced reactor coolant system inventory operation period
- reactor startup and power ascension

b. Findings

There were no findings identified.

.2 Containment Closeout

a. Inspection Scope

During plant heatup, the inspectors performed a detailed walkdown of containment just prior to plant entry into operational mode 4, hot shutdown, to verify that equipment and debris that could affect the operability of the containment sumps had been removed.

b. Findings

There were no findings identified.

.3 Reactor Core Performance

a. Inspection Scope

The inspectors compared video records of fuel assembly placement with the core fuel loading plan to verify a sample of five fuel assemblies had been properly located in the core. The inspector observed a portion of low power physics testing and verified that measured core physics parameters (e.g., critical boron concentration, isothermal temperature coefficient, control rod worth, and moderator temperature coefficient) were consistent with the fuel vendor's analysis for operating cycle 14.

b. Findings

There were no findings identified.

1R22 Surveillance Testing

.1 Loss of Normal Power Test

a. Inspection Scope

On May 20, 2000, the inspector observed the performance of surveillance procedure SP 2613H, "Loss of Normal Power Testing Facility 2." The inspector reviewed the results of the test and verified that they satisfied the specified acceptance criteria and the associated technical specification requirements.

b. Findings

There were no findings identified.

.2 Main Steam Containment Isolation Valve Testing

a. Inspection Scope

The inspector observed the preparation and performance of stroke testing on the No. 2 steam generator atmospheric dump valve, 2-MS-190B, on June 28, 2000. The inspector verified proper control of electrical jumpers utilized in support of the testing. The inspector reviewed the test results documented in OPS Form 2610E-5, "Main Steam System Valve Stroke and Timing IST," for valve 2-MS-190B and valve 2-MS-65B, the No. 2 steam generator main steam isolation bypass valve, both of which are containment isolation valves. The inspection activities included verification of compliance with applicable in-service testing acceptance criteria, technical specifications, and the component's design bases.

b. Findings

There were no findings identified.

2. RADIATION SAFETY
Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

During the period May 22-26, 2000, the inspector conducted the following activities to determine the effectiveness of access controls to radiologically significant areas for the Unit 2 refueling outage:

All locked high radiation areas in Unit 2 were physically checked and the keys inventoried. Independent measurements were made of radiation levels within radiologically controlled areas (RCAs) at Unit 2 including those areas of the Containment Building and Auxiliary Building whose status would change from a high radiation area to a Technical Specification Locked High Radiation Area following the resumption of power operations. Survey data and barricades/postings to high radiation areas located in these buildings were verified.

On May 25, 2000, the inspector reviewed the corrective actions taken in response to the identification by a technician of a 50 R/hour hot spot on an overhead drain line in the “A” Safeguards Room (CR-M2-00-1566). The inspector verified that the area was properly reposted as a Technical Specification Locked High Radiation Area and that appropriate controls were implemented to alert personnel of the change in the area’s radiological conditions.

b. Findings

There were no findings identified.

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed the effectiveness of various controls to minimize and equalize personnel exposure for activities conducted during the Unit 2 refueling outage. Performance of selected work groups was reviewed including those groups performing reactor disassembly/assembly and snubber inspections. The inspector attended post-job ALARA debriefings for the seal replacement on the “D” Reactor Coolant Pump and motor operated valve testing/maintenance. The inspector attended the pre-job RWP briefing for placement of the missile shield over the Unit 2 reactor vessel. Individual exposure records were reviewed including those for declared pregnant workers and for individuals that were multi-badged. The inspector interviewed selected workers and technicians performing tasks in the containment and auxiliary buildings to determine if radiological controls were being properly implemented.

At Unit 3, the inspector observed preparations for workers to enter a potentially Very High Radiation Area in the containment building (at full power) for replacing a movable incore detector. The inspector attended the pre-job planning meeting and RWP/confined space briefing on May 24 and 25, 2000, respectively. Work coordination activities, survey data, and individual exposure results were reviewed. For this task, the inspector interviewed selected workers on their knowledge of the relevant RWP, dosimetry set points, and job-site radiological conditions.

b. Findings

There were no findings identified. Actual collective exposure attributed to the Unit 2 outage was 88 person-rem versus the outage estimate of 140 person-rem.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

Calibration and maintenance records were reviewed for various portable survey instruments, contamination monitors, area radiation monitors, airborne radioactivity samplers/monitors and electronic dosimeters in use during the Unit 2 refueling outage.

The inspector reviewed the licensee's actions following the identification on May 24, 2000 that the audible and visual alarms for the dose and dose rate set points for a number of electronic dosimeters (EDs) were inadvertently turned off as a result of an error in vendor supplied software (CR-M2-00-1511). Included in this review were verifications that individuals using the disabled EDs did not exceed their RWP dose limits and that there was no substantial potential for a worker to exceed regulatory limits when wearing the affected ED. The inspector confirmed that the affected EDs were removed from service, that the licensee investigated the matter to the extent practical, and notified the vendor and other ED users of the problem. The inspector verified that a testing program was completed by the licensee prior to placing the EDs in service that provided reasonable assurance that the EDs would perform their intended function. The inspector observed source testing of in-service EDs. The inspector randomly chose an in-service ED and verified that it alarmed at the prescribed dose/dose rate set points. The inspector observed testing of affected EDs and confirmed that the dose/dose rate readouts and elapsed time alarm were not affected by the software problem.

b. Findings

There were no findings identified.

3. SAFEGUARDS

Physical Protection [PP]

3PP1 Access Authorization

a. Inspection Scope

The following activities were conducted to determine the effectiveness of the licensee's behavior observation portion of the personnel screening and fitness-for-duty programs:

Five supervisors representing the Maintenance, Radiation Protection, Operations, System Engineering and Instrumentation & Control Departments were interviewed, on June 13 and 14, 2000, regarding their understanding of behavior observation responsibilities and the ability to recognize aberrant behavior traits. Two (2) Access Authorization/ Fitness-for-Duty self-assessments, an audit, and event reports and loggable events for the four previous quarters were reviewed, during this inspection. On June 13 and 14, 2000, five (5) individuals, who perform escort duties, were interviewed to establish their knowledge level of those duties. Behavior observation training procedures and records were also reviewed.

b. Findings

There were no findings identified.

3PP2 Access Control

a. Inspection Scope

The following activities were conducted during the period June 12-15, 2000 to verify that the licensee had effective site access controls, and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area:

A random sample of twenty (20) personnel, granted unescorted access to the protected and vital areas, was checked to assure that they were properly screened, identified and authorized. Site access control activities were observed, including personnel and package processing through the search equipment at the north and south access points during peak ingress periods on June 14, 2000, and vehicle searches, on June 15, 2000. On June 15, 2000, testing of all access control equipment; including metal detectors, explosive material detectors, and X-ray examination equipment, was observed. The Access Control event log, an audit, and three (3) maintenance work requests were also reviewed.

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

.1 Physical Protection Performance Indicators

a. Inspection Scope

The inspectors reviewed the licensee's programs for gathering and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment Performance Indicators. The review included the licensee's tracking and trending reports, personnel interviews and security event reports for the Performance Indicator data submitted from the second quarter of 1997 through the first quarter of 2000.

b. Findings

There were no findings identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed engineering action plans and condition reports to verify that problems requiring corrective actions were captured at an appropriate threshold and identified corrective actions were commensurate with the significance of the problem.

b. Findings

The licensee demonstrated weak problem resolution in that a work request to address a longstanding industry problem with the push-button test switch for the power/load unbalance trip circuit of the turbine electro-hydraulic control (EHC) system was canceled. General Electric Technical Information Letter 1212-2, "Plant Scram Frequency Reduction Features for BWR and PWR Nuclear Turbines with MK I or MK II EHC Controls," issued January 27, 1997, described that failures of the push button-test switch at other facilities was resulting in frequent trips of nuclear and fossil units. The licensee had developed Engineering Work Request 2-94-00269 to improve the reliability of the test circuit, but the request was canceled during a review of open work requests in January 2000. The licensee's nuclear oversight organization identified weaknesses in the process used to identify work requests for cancellation and documented their concern in Condition Report M2-00-0792, which was initiated on April 4, 2000. The reactor trip caused by failure of the push-button test switch on June 4, 2000, is discussed in Section 1R12.3.

The inspector also found that the licensee did not identify appropriate corrective actions for a condition involving early release of work without appropriate controls to ensure plant conditions would allow its safe performance (Section 1R13.1).

4OA3 Event Follow-up

a. Inspection Scope

The inspector reviewed the licensee's actions in response to control rod problems during low power physics testing on May 30 and 31, 2000, that were manifest as slipped and dropped control rods.

b. Findings

There were no findings identified.

4OA4 Cross-cutting Issues

Human Performance Problems

a. Inspection Scope

The inspector reviewed human performance related to the control and implementation of maintenance activities.

b. Issues and Findings

The inspector found that operations released work without appropriate controls to ensure plant conditions would allow its safe performance (Section 1R13.1). The inspector also identified that work planning, maintenance, and operations personnel failed to establish and implement appropriate maintenance instructions to ensure a system partially drained for maintenance would be properly refilled, which resulted in the common cause failure of redundant safety system trains (Section 1R12.1).

4OA5 Other

.1 Initiating Events Performance Indicators

a. Inspection Scope

The inspector reviewed licensee event reports, monthly operating reports, plant process computer power history information, and NRC inspection reports to identify significant plant power changes and conditions associated with plant scrams that occurred between the unit's restart from an extended outage on May 9, 1999, and March 31, 2000 . The inspector compared this information with the licensee's reported value for the first quarter of calendar year 2000 for the following performance indicators (PIs):

- Unplanned scrams per 7000 critical hours
- Scrams with a loss of normal heat removal
- Unplanned power changes per 7000 hours

b. Findings

The inspector found that, due to a misunderstanding of the guidance, the licensee did not count a manual reactor scram involving main steam isolation to terminate a steam leak in the turbine building on May 25, 1999, in the PI value for scrams with a loss of normal heat removal. After reevaluating the guidance, the licensee determined that the May 25, 1999, scram had been incorrectly classified and documented this issue in Condition Report M2-00-1896. The NRC classified this issue as a minor discrepancy because the total number of scrams with a loss of normal heat removal in the last 12 quarters was 2, which is in the licensee response band (Green).

.2 Performance Indicator Data Collecting and Reporting (TI 2515/144)

a. Inspection Scope

Unit 2 performance indicators (PI) for the first quarter calendar year 2000 and beyond were reviewed to ensure that the licensee had a clear understanding of the PI definitions, data reporting elements, calculational methods, definitions of terms and clarifying notes. The sample included unplanned power changes per 7000 hours, safety system availability and functional failures, emergency response organization drill participation, occupational exposure control effectiveness and protected area security equipment performance index. Further, the review verified that the licensee's process was capable of producing accurate PIs, in accordance with the guidance in NEI 99-02, “Performance Indicators.”

There were no findings identified.

4OA6 Meetings, including Exit

.1 Regional Engineering Inspection Exit Meeting

The inspector presented the inspection results to Mr. Paul Grossman, Manager of Engineering, and other members of the licensee management at the conclusion of the inspection on May 24, 2000. The licensee acknowledged the conclusions presented.

The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Security Inspector Exit Meeting

The inspectors met with licensee representatives at the conclusion of the security inspection on June 15, 2000. At that time, the purpose and scope of the inspection were reviewed, and the preliminary findings were presented. The licensee acknowledged the preliminary inspection findings.

.3 Resident Inspector Exit Meeting

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection. The licensee acknowledged the findings presented.

ITEMS OPENED AND CLOSED

Opened and Closed During this Inspection

NCV 05000336/2000-008-01	fire fighting strategy for Fire Zone A-1H was not maintained in that the hazard posed by the hydrogen supply line was not identified (1R05)
NCV 05000336/2000-008-02	failure to adequately implement a procedure covering the filling of the chilled water system (1R12.1)
NCV 05000336/2000-008-03	failure to adequately establish and implement a procedure covering control of maintenance work (1R13.1)
NCV 05000336/2000-008-04	failure of the licensee to take adequate corrective actions to address RBCCW relief valves lifting in the event of a loss of normal power (1R15.1)
NCV 05000336/2000-008-05	failure of the licensee to implement any periodic or post-maintenance test to verify adequate RBCCW train independence (1R19.1)

Previous Items Closed

50-336/2000-010-00	LER	Reactor Trip during Testing of Turbine Power-Load Unbalance Circuit (1R12.3)

LIST OF ACRONYMS USED

AWO	automated work order
AFW	auxiliary feedwater
CR	condition report
DCR	design change record
DAC	distance amplitude correction
ED	electronic dosimeter
EHC	electro-hydraulic control
EOP	emergency operating procedure
EWR	engineering work request
NNECO	Northeast Nuclear Energy Company
OD	operability determination
OP	operating procedures
PI	performance indicators
RBCCW	reactor building closed cooling water
RCA	radiologically controlled area
RCP	reactor coolant pump
RCS	reactor coolant system
RG	Regulatory Guide
RWP	radiation work permit
SER	safety evaluation report
SRA	senior reactor analyst
TLD	thermoluminescent dosimetry

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	05000423
License No.:	NPF-49
Report No.:	05000423/2000-008
Licensee:	Northeast Nuclear Energy Company
Facility:	Millstone Nuclear Power Station, Unit 3
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	May 14, 2000 - July 1, 2000
Inspectors:	A. C. Cerne, Senior Resident Inspector, Unit 3 B. E. Sienel, Resident Inspector, Unit 3 P. C. Cataldo, Resident Inspector, Unit 2 P. Frechette, Physical Security Inspector, Division of Reactor Safety (DRS) K. Jenison, Senior Project Engineer, Division of Reactor Projects P. Kaufman, Senior Reactor Inspector, DRS A. Lohmeier, Reactor Inspector, DRS T. A. Moslak, Health Physicist, DRS S. Pindale, Reactor Inspector, DRS G. C. Smith, Senior Physical Security Inspector, DRS
Approved by:	James. C. Linville, Chief Projects Branch 6 Division of Reactor Projects Region I

SUMMARY OF FINDINGS

IR 05000423/2000-008 on 05/14-07/01/00; Millstone Nuclear Power Station; Unit 3 Surveillance Testing.

The inspection was conducted by resident and regional inspectors. This inspection identified one green issue, which was a noncited violation. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (see Attachment 1).

Cornerstone: Mitigating Systems

- **Green.** Technical Specification 4.0.5 requires that inservice testing of check valves be conducted at maximum credited design basis flow. Unit 3 Surveillances SP 3622.7-1 and SP 3622.3-5 established acceptance criteria for auxiliary feedwater (AFW) pump discharge check valve operability below this criteria. This is a violation of Millstone Unit 3 TS 6.8.1, Procedures, and is being treated as a Non-Cited Violation. (Section 1R22)

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Report Details

SUMMARY OF UNIT 3 STATUS

The plant began the inspection period operating at 100 percent power. On May 18 operators reduced power to approximately 85% due to equipment and weather-related problems at the intake. Following restoration of equipment and improved weather conditions, on May 19 operators restored power to approximately 100%, where it remained through the end of the inspection period.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity)

1R02 Evaluation of Changes, Tests or Experiments

a. Inspection Scope

The inspectors reviewed 18 safety evaluations associated with the initiating events, mitigating systems, and barrier integrity cornerstones to verify that changes to the facility or procedures as described in the UFSAR were reviewed and documented in accordance with 10 CFR 50.59. Safety evaluations were selected based upon the safety and risk significance of the changes.

The inspectors also reviewed 16 applicability reviews (10 CFR 50.59 safety screens) for changes, tests and experiments for which the licensee determined that a safety evaluation was not required. This review was performed to verify that the licensee's threshold for performing safety evaluations was consistent with 10 CFR 50.59.

Finally, the inspectors reviewed a sample of condition reports documenting problems identified by the licensee in the corrective action program related to safety evaluations to verify the effectiveness of corrective actions.

A listing of the 10 CFR 50.59 safety evaluations, safety screens, and condition reports reviewed is provided in the List Of Documents Reviewed section at the end of this inspection report.

b. Findings

There were no findings identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspector conducted partial walkdowns of both trains of the supplementary leak collection and release system (SLCRS) equipment, including filters, fans, and instrumentation. This field inspection was conducted immediately prior to the removal of the train "A" engineered safety features (ESF) building air conditioning unit (ACU) for planned preventive maintenance activities. Since both the ESF ACUs and the SLCRS equipment provide separate safety-related ventilation support to common areas in the ESF building, the existence of any degraded SLCRS equipment conditions would represent not only a concern with Unit 3 technical specification (TS) 3.6.6 compliance, but also a potential problem with redundant, safety-train component operability. The inspector also reviewed operations procedure, OP3314B, for the risk configuration of both fan and ACU inoperability with the planned equipment scheduled for removal from service.

As part of this review, the inspector did discuss with licensee personnel the observation that a SLCRS key was required to access the door to one train of SLCRS equipment, while the door to the other train was unlocked. Both the Operations shift manager and the responsible system engineer confirmed that this access configuration was acceptable. A trouble report was written to repair the latching mechanism on the door that was locked, since neither of these SLCRS equipment rooms requires key-entry control.

b. Findings

There were no findings identified.

1R05 Fire Protection

.1 Fire Protection Area Inspections

a. Inspection Scope

The inspector conducted inspection-tours of both the East and West Motor Control Center/Rod Control Air Conditioning Unit rooms in the auxiliary building and the North and South Emergency Generator Enclosures in the emergency diesel generator (EDG) building. These areas contain redundant sets of electrical and ventilation support equipment and were inspected for fire protection design features at separate time periods, both before and after the conduct of surveillance activities of the housed, safety-related equipment (e.g., the EDGs). During this tour, the inspector examined the fire suppression and detection equipment located in the area and compared the fire protection capabilities with that described in the Millstone Unit 3 Fire Protection Evaluation report for each of the four fire areas (i.e., AB-8; AB-6, Zone B; EG-3, Zones A & B; EG-4, Zones A & B) inspected.

b. Findings

There were no findings identified.

.2 (Closed) URI 50-423/99-02-07: Fire Safe Shutdown Analysis Design Bases. On January 15, 1999, an inadvertent actuation of the carbon dioxide (CO_2) fire suppression system occurred in the cable spreading room. Following the discharge, CO_2 was found to have migrated into the east and west switchgear rooms, located directly below the cable spreading room, rendering the auxiliary shutdown panel area in the west switchgear room uninhabitable. The NRC's initial inspection of this event is documented in Section U3.O2.1 of Inspection Report 50-423/99-02.

The NRC Standard Review Plan Branch Technical Position 9.5-1 assumes that the east and west switchgear rooms remain habitable following a cable spreading room fire with CO_2 suppression discharge. Because a fire in the cable spreading room may render control room equipment inoperable, an alternate remote shutdown method was provided via equipment in the switchgear rooms. Unresolved Item 99-02-07 was opened pending the results of the licensee's investigation to determine whether this CO_2 migration was outside the plant's design basis.

The licensee has conducted several analyses and tests to determine the cause for the gas migration. At the close of the report period, the licensee continued to evaluate the test results. In order to provide further information, the licensee plans to perform a tracer gas test. The technical specifications (TS) currently prevent such a test because it will pressurize the cable spreading room to a pressure that exceeds the pressure of the adjacent control room envelope area. Therefore, the licensee submitted technical specification change request 3-16-99 on February 1, 2000, which is currently under NRC staff review. Following this testing, scheduled to be completed by the end of 2000, the licensee will determine whether the plant was outside the design basis. If so, the licensee will report this condition to the NRC as required by 10 CFR 50.72 and 73.

Following the January 15, 1999, event, the licensee locked out the automatic CO_2 fire suppression system in the cable spreading room. The system remains isolated and is controlled to prevent manual initiation of the system, as well. Therefore, while the historical design basis question has not been answered, the inspector does not have a concern with the present configuration of the system.

The licensee has continued to investigate the migration of CO_2 into the switchgear rooms to determine whether a condition outside the design basis existed. The licensee indicated they will report this condition, if 10CFR50.72 or 73 applicability is determined. Since no current design basis question exists with the system's current configuration, unresolved item **URI 50-423/99-02-07** is **closed**. If a condition outside the design basis is subsequently identified, the inspector will review the associated licensee event report in accordance with the NRC inspection program.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspector observed licensed operator requalification training (LORT) of a Unit 3 mixed/administrative crew, using the Unit 3 simulator to mimic a scenario involving a condenser tube leak and a loss of instrument air, with the subsequent reactor trip recovery activities complicated by the degraded instrument air conditions. Operator use of both abnormal and emergency operating procedures (AOPs/EOPs) was verified, with a revision to AOP 3557, " Secondary Chemistry" (Revision 6), that resulted from a real Unit 3 trip in 1998 exercised as part of the scenario. The inspector conducted discussions with the licensee's operator training personnel during the conduct of this simulator session, as necessary to understand the objectives of the observed training.

b. Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed licensee condition reports generated since March 2000 for maintenance rule applicability. Several condition reports (CRs) were selected for a detailed examination, that included a review of the affected system's maintenance rule scoping document; first quarter system health report; maintenance rule functional failure (MRFF) determination and a(1) action plan, if applicable. The inspector discussed recent revisions to the radiation monitors scoping document and a(1) action plan with the system engineer.

The following CRs were reviewed:

- M3-00-0757: "B" Instrument Air compressor tripped due to low oil pressure
- M3-00-0797: Failed "B" Control Building Chilled Water (HVK) valve strokes
- M3-00-0800: While stroking HVK valves two valves exceeded limits
- M3-00-1250: Containment atmosphere radiation monitor caused unplanned LCO entry

The licensee stated that although the CR M3-00-1250 also was listed as requiring an evaluation in the maintenance rule database, an action request had not been attached to the CR to effect the evaluation. The licensee documented this issue in CR M3-00-1742. This issue does not constitute a violation of the maintenance rule because even if a MRFF is subsequently identified, no change in the system's status will occur, i.e., the system has already been classified as a(1).

b. Findings

There were no findings identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope

After surveillance testing of the main steam system (MSS) supply valves to the turbine driven auxiliary feedwater (TDAFW) pump, the licensee identified a leakage trend on one of the three steam admission valves, 3MSS*AOV31B, as documented in condition report (CR) M3-00-1365.

The inspector confirmed that this work activity was properly controlled in accordance with technical specification 3.6.1.1. requirements and that all valves were then restored to their normal operational alignment, with independent verification of valve position, as required.

The inspector reviewed licensee troubleshooting activities associated with a recurrent problem with a turbine generator combined intermediate valve, 3MSS-CIV5, which periodically has stuck closed during weekly surveillance testing performed in accordance with TS 3.3.4 for turbine overspeed protection.

The inspector also reviewed licensee recommendations for continued testing, contingency plans for a unit downpower to remove the turbine from service to effect repairs, an assessment of the risks involved, and longer-term plans for CIV component replacement with upgraded parts that would allow online replacement of sticking solenoid valves.

b. Findings

There were no findings identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The resident inspector attended various planning meetings and reviewed design change record (DCR) M3-99039 and supporting safety evaluations, SG-EV-0006, Revisions 2 and 3, relative to modifications of the Unit 3 electrical distribution system in support of GDC 17, Appendix R, and Station Blackout electrical power for Unit 2 systems and components. The modifications also include the physical separation of the Unit 1 systems and components that are currently credited to supply the Unit 2 support functions discussed above. In addition, the inspector reviewed various condition reports (CRs), including M3-00-1446 and M3-00-1447, that were initiated by the licensee in response to various deficiencies identified during the design process. The inspector evaluated the impact that the identified deficiencies had on current plant operations, i.e., operability impact, and whether the deficiencies impacted the scope and implementation of the plant modifications.

Additionally, region-based inspectors conducted an inspection of a sample of plant modifications during the week of June 12 - 15, 2000. The inspectors selected and reviewed portions of 23 permanent plant modifications from design changes that were completed at Millstone Unit 3 since 1996. The selection was based on risk insights from the Millstone 3 probabilistic risk assessments and the impact on the reactor safety cornerstones. The inspection focused on complementary inspectable areas under the reactor safety cornerstones of initiating events, mitigation systems, and barrier integrity. The modifications included safety related piping and components, as well as changes to plant operating procedures. The inspectors reviewed selected portions of the modification packages that included safety evaluation screening forms, 10 CFR 50.59 safety evaluations, design calculations, setpoint changes, and results of post-modification testing. Where appropriate, the inspectors discussed the scope and extent of the modifications, technical factors associated with the changes, and implementation of the changes with the responsible engineering personnel. In addition, the inspectors reviewed a sample of condition reports that documented problems identified by the licensee in the corrective action program relative to permanent plant modifications to verify the effectiveness of corrective actions. A listing of the permanent modifications and condition reports that were reviewed is provided in the List Of Documents Reviewed section at the end of this inspection report.

b. Findings

There were no findings identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspector observed and/or reviewed the completed documentation for post maintenance testing (PMT) performed on the “C” reactor plant closed cooling water pump motor and station blackout diesel generator control room air conditioning unit. The inspector reviewed the scope of the work activities and verified that the PMTs planned were appropriate to verify restoration of the systems. The inspector also reviewed completed work orders and conducted equipment walkdowns using the system valve lineup after testing was completed to verify acceptable system restoration.

Additionally, a field inspection and verification of the PMT valve lineup for the service water (SWP) system were conducted after completion of planned maintenance and the operational test on the “D” SWP pump. The inspector also observed the plant equipment operators restore the alignment of the hypochlorite system to normal operating conditions following the pump swaps related to the PMT activities.

b. Findings

There were no findings identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspector witnessed field activities associated with the following surveillance tests for components in systems representing significant contributors to the prevention of core damage in design-basis accident scenarios.

- SP 3446A.9 Slave Relay Testing - Train B (Section 4.24, Safety Injection)
- SP 3610A.7 Residual Heat Removal Train “A” Valve Operability Test
- SP 3626.7-1 Service Water Pump 3SWP*P1D Operational Readiness Test

The completed data sheets were reviewed for SP 3446A.9 to verify the equipment met procedural acceptance criteria and was operable consistent with technical specification requirements. Restoration from SP 3626.7-1 was observed in the intake structure to confirm procedural compliance, as-left equipment operability, and control of the system allowed outage times in accordance with TS requirements. The inspector also witnessed SP 3610A.7 to verify adequate controls of the local valve operation and restoration of the RHR train “A” total flow full-stroke setting for the valve.

Additionally, portions of surveillance procedures SP 3622.3-5, Turbine Driven Auxiliary Feedwater (TDAFW) Pump Mode 3 Full Load Test, and SP 3622.7-1, Cold Shutdown Check Valve Full Stroke Testing For Auxiliary Feedwater System, associated with the inservice testing of auxiliary feedwater system components were evaluated, along with the test data and acceptance criteria, to check the safety system availability consistent with the licensee performance indicator data discussed in Section 4OA1 of this inspection report.

b. Findings

During a review of auxiliary feedwater (FWA) system availability performance indicator data, the inspector identified that full flow testing of motor driven FWA pump discharge check valves did not meet the requirements of Millstone Unit 3 Technical Specification (TS) 4.0.5. TS 4.0.5 requires, through reference, that inservice testing of FWA check valves be conducted at maximum credited design basis flow. The maximum credited design basis flow is the largest flow rate for which credit is taken in a safety analysis for a component in any flow configuration. Unit 3 Surveillances SP 3622.7-1 and SP 3622.3-5 established acceptance criteria (190 gpm - indicated) based on calculation 98IST-01641, which did not meet the requirements of TS 4.0.5. The inspector determined that based on FSAR Figures 10.4-11, 15.2-18 and 24, and Table 15.2-1 (rev 91-12, 6/92) secondary system pressure during a main feedline rupture would decrease to between 820 psig and 600 psig, causing flow to the intact steam generators to exceed 230 gpm per check valve. In addition to the system pressure band of between 820 and 600 psig not being considered in the indicated calculation, portions of the band fall below the analyzed pump operating curves 10.4.11 and 12. Based on ancillary licensee test data the inspector was able to determine that the FWA check valves were “available”.

Subsequently, the licensee provided for review current FWA check valve testing data that support the licensee’s determination of operability for the subject check valves. Nevertheless, the failure to establish adequate surveillance test criteria, as identified by NRC inspection and documented above, is a violation of TS 6.8.1, Procedures. This violation is being treated as a Non-Cited Violation (NCV), in accordance with Section VI.A of the NRC's Enforcement Policy (**NCV 05000423/2000-008-06**). This issue was entered into the Millstone corrective action process as Condition Report M3-00-1418.

2. RADIATION SAFETY
Occupational Radiation Safety [OS]

2OS2 ALARA Planning and Controls

Refer to NRC Inspection Report 05000336/2000-008, Section 2OS2 for specific details.

3. SAFEGUARDS

Physical Protection [PP]

3PP1 Access Authorization

Refer to NRC Inspection Report 05000336/2000-008, Section 3PP1 for specific details.

3PP2 Access Control

Refer to NRC Inspection Report 05000336/2000-008, Section 3PP2 for specific details.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

.1 Physical Protection Performance Indicators

Refer to NRC Inspection Report 05000336/2000-008, Section 4OA1.2 for specific details.

4OA2 Identification and Resolution of Problems

One inspection activity in a previous section of this report had implications regarding the licensee's evaluation of problems, as follows:

- Section 1R12 - The licensee failed to perform a maintenance rule functional failure evaluation following the identification of a problem with a containment atmosphere radiation monitor. This demonstrated weak problem evaluation.

4OA5 Other

.1 (Closed) LER 50-423/99-006: failure of both reactor plant aerated drains safety-related air driven sump pumps during a technical requirements manual surveillance. The licensee corrective measures addressed the inadequate preventive maintenance program that was identified to be the root cause of the equipment failure. The licensee commenced monthly surveillance testing of the pumps. In September 1999, this periodic testing identified another pump failure, with a different root cause that is discussed in IRs 50-423/99-09 & 99-12. During the latter inspection, the inspector determined that the inadequate qualification testing of the pumps resulted in a Non-Cited Violation (NCV 50-423/99-12-13).

Since the last pump failure in September 1999, monthly testing of the pumps has revealed no operability concerns. The licensee continues with plans to replace both existing air driven pumps, during the next Unit 3 refueling outage in 2001, with a new submersible, electric driven pump design. The inspector reviewed design change record (DCR) M3-00004 and safety evaluation E3-EV-00-0009 for this planned modification and observed a plant operation review committee meeting at which it was discussed. As committed in LER 99-006, the existing safety-related, air driven sump pumps have continued to be tested and maintained in an operable status. This **LER** is **closed**.

.2 (Closed) LER 50-423/2000-001: power operation with the ultimate heat sink temperature below its minimum design basis. Based on the licensee's compensatory and corrective actions, this was determined to be a weather related, minor issue. This **LER** is **closed**.

.3 Unit 3 Specific PI Verification

a. Inspection Scope

Unit 3 performance indicators (PI) for the first quarter calendar year 2000 and beyond were reviewed to ensure that the licensee had a clear understanding of the PI definitions data reporting elements, calculational methods, definitions of terms and clarifying notes. The sample included unplanned power changes per 7000 critical hours, safety system availability and functional failures, emergency response organization drill participation, occupational exposure control effectiveness and protected area security equipment performance index. Further, the review verified the licensee's process was capable of producing accurate PIs, in accordance with the guidance in NEI 99-02, Performance Indicators.

b. Findings

There were no findings identified.

4OA6 Meetings, including Exit

.1 Final Millstone Assessment Panel (MAP) Meeting

On June 14, 2000, the NRC Millstone Assessment Panel (MAP) convened to perform its final, quarterly review of Millstone performance and activities. The MAP members discussed the NRC senior management decision to return Millstone Units 2 and 3 to normal regulatory oversight. The MAP members also discussed the MAP Charter to ensure that all actions had been completed. A consensus was reached that the MAP had satisfied the commitments of its Charter and the MAP was disbanded.

.2 Plant Performance Review Meeting

On June 20, 2000, a meeting open for public observation was held at Millstone Station between NU and the NRC to discuss the results of the NRC's plant performance review as described in the NRC's March 31, 2000 letter. Slides from the meeting are attached to this report.

.3 Regional Engineering Inspection Exit Meeting

The regional engineering inspectors presented their inspection results to Mr. P. Grossman, Plant Engineering Director, and other members of licensee management at the conclusion of the permanent plant modification and evaluation of changes, tests, and experiments inspection on June 15, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.4 Resident Inspector Exit Meeting

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection. The licensee acknowledged the findings presented.

ITEMS OPENED AND CLOSED

Opened and Closed During this Inspection

NCV 05000/423-008-06	failure to establish adequate surveillance test criteria (1R22)

Previous Items Closed

50-423/99-02-07	URI	Fire Safe Shutdown Analysis Design Bases (1R05)
50-423/99-006	LER	failure of both reactor plant aerated drains safety-related air driven sump pumps during a technical requirements manual surveillance (4OA5)
50-423/2000-001	LER	power operation with the ultimate heat sink temperature below its minimum design basis (4OA5)

LIST OF DOCUMENTS REVIEWED FOR 1R02 and 1R17

10 CAR 50.59 Safety Evaluations

S3-EV-98-0021, Modification DAR M3-98008, "3RSS*P1A,B,C and D Seal Water Coolers Modification"
S3-EV-97-0147, Modification DAR M3-96055, "Reactor Coolant Pump Internals Replacement"
S3-EV-97-0163, FAR Change Request 97-MP3-219, Revision 1, "R.S. Suction Relief Valve Capacity"
S3-EV-97-0415, Modification DAR M3-96068, "Safety Injection System Pipe Support Modifications"
S3-EV-97-0488, Modification DAN. DM3-00-1560-97, "Disc Guide Modification for Valves 3SIL*MV8809A/B"
S3-EV-97-0528, Modification DAN. DM3-00-1653-97, "Valve Modifications for 3QSS*MOV34A/B"
S3-EV-97-0583, Modification DAR M3-97111, "Installation of Screens in the SW Inlet of the R.S. Heat Enchanters"
E3-EV-98-0006, Modification DAR M3-98007, "Reduced A.W. Flow Rates, New FAR Chapter 15 Analyses"
S3-EV-98-0011, Various steam generator tube rupture emergency operating procedure changes
S3-EV-98-0099, Modification DAR-M3-98022, "Main Feedwater Pump 3FWS-P1 Vibration"
S3-EV-98-0151, Procedure OP 3306, "Containment Recirculation Spray System"
S3-EV-98-0231, DAN. DM3-00-0961-98, "3RCS*MV8098 Installed With Flow Over Valve Disc"
S3-EV-99-0001, Modification MMOD M3-98039, "MP3 Service Water Piping Modifications for RFO6"
S3-EV-99-0009, Modification DAR M3-99004, "Replacement of Turbine-Driven A.W. Pump Rotating Assembly and Governor Valve 3MSS*MCV5 Stem Material Replacement"
S3-EV-99-0011, Modification DAR M3-99003, "Reactor Coolant Pump No. 3 Seal Leakoff Piping Reroute"
S3-EV-99-0034, Modification DAR M3-99014, "Service Water Pump Internal Mechanical Seals"
S3-EV-99-0083, Modification DAR M3-99029, "Throttle Position of 3SWP*P2A/B Discharge Valves"
S3-EV-00-0023, Modification DAR M3-00002, "Installation of Mechanical Seals in Unit 3 Service Water System"

<u>10 CAR 50.59 Safety Screens</u>

DAR M3-00022, "Main Steam Generator Feedring/J-Tube Inspection and Repair"
DAR M3-00023, "Motor-Operated Valve Wedge Replacement"
DAR M3-99019"Steam Generator Tube Stabilizers"
DAR M3-96061, "3SIL*HCV943A/B Replacement"
SP 3606.3, Revision 13, "Containment Recirculation Pump 3RSS*P1C Operational Test"
OP 3304A, Revision 27, Change 3, "Charging and Letdown"
OPS Form 3301D-4, Revision 4, Change 1, "RCP Oil System Valve Lineup"
OP-3304C, Revision 19, Change 5, "Primary Makeup and Chemical Addition"
OP-3322, Revision 19, Change D1, "Auxiliary Feedwater System"
OP-3308, Revision 11, "High Pressure Safety Injection"

DAN. DM3-00-0794-98, “Bypass 3QSS-P1A/B Auto Trip on Low RWST Temperature”
DAN. DM3-00-0515-99, “Revise Control Building Chiller Condenser SW Low Flow Alarm/Trip”
DAN. DM3-01-1744-97, “3RHS*FCV610 & 3RHS*FCV611 Trip Coil Setting”
MMOD M3-99027, “MSIV Trip Circuit Relay Modification”
MMOD M3-99032, “Time Delay of CDS and CCP Low Flow Trip of 3CDS-CHL1A,B,&C”
Temporary Modification 3-98-004, “Arcor Application on Valve Assembly 3SWP*TV35A”

Permanent Plant Modifications

Event Initiators

DAR-M3-98022, “Main Feedwater Pump 3FWS-P1 Vibration”
DAR M3-99004, “A.W. Pump Rotating Assembly and 3MSS*MCV5 Stem Replacement”
DAR M3-99014, “Service Water Pump Internal Mechanical Seals”
DAR M3-99029, “Throttle Position of 3SWP*P2A/B Discharge Valves”
DAR M3-97111, “Installation of Screens in the SW Inlet of the R.S. Heat Enchanters”
DAR M3-98008, “3RSS*P1A,B,C and D Seal Water Coolers Modification”
DAR M3-00002, “Installation of Mechanical Seals in Unit 3 Service Water System”
MMOD M3-98039, “MP3 Service Water Piping Modifications for RFO6”

Barrier Integrity

DAR M3-00022, “Steam Generator Feedring/J-Tube Inspection and Repair”
DAR M3-00004, “Installation of Non-Safety Related Underdrain System Pump in ESF Building”
DAR M3-99019, “Steam Generator Tube Stabilizers”
EOP 35 FR - P.1/P.2, “Response to thermal pressurized thermal shock emergency procedures”
EOP 35 ES - 3.1/3.2/3.3, “Steam generator tube rupture cooldown emergency procedures”
FAR Change Request 97-MP3-219, Revision 1, “R.S. Suction Relief Valve Capacity”
DAR M3-99003, “Reactor Coolant Pump No. 3 Seal Leakoff Piping Reroute”
DAR M3-96055, “Reactor Coolant Pump Internals Replacement”
DAR M3-96061, “3SIL*HCV943A/B Replacement”

Mitigation Systems

DAR M3-00023, “Motor-Operated Valve Wedge Replacement”
DAN. DM3-00-1560-97, “Disc Guide Modification for Valves 3SIL*MV8809A/B”
DAN. DM3-00-1653-97, “Valve Modifications for 3QSS*MOV34A/B”
DAN. DM3-00-0961-98, “3RCS*MV8098 Installed With Flow Over Valve Disc”
OP 3306, “Containment Recirculation Spray System”
DAR M3-96068, “Safety Injection System Pipe Support Modifications”

Condition Reports

CR M3-98-1100, “Five of Seven MMODs Required Safety Evaluations”
CR M3-98-3689, “Auxiliary Feedwater Pump Being Run to Cool ‘D’ A.W. Containment Penetration”
CR M3-99-0101, “Procedure CBM 105 Not Given Adequate Initial Safety Screen”
CR M3-99-2292, “Audit MP-99-A03: Inadequate Safety Screens”
CR M3-99-2293, “Audit MP-99-A03: Testing Methodology for Primary Equipment Snubbers”
CR M3-99-2471, “Design Engineering Screen was Inappropriately Completed”
CR M3-00-0270, “DCM Requires Calculation Change Notices”

CR M3-00-0937, “Nonconforming Conditions Dispositioned as Rework Have Not Been Fixed for Over One Year”
CR M3-99-2625, “Audit MD-99-A03: Recommendation for Plant Modification at Unit 3 Post-Modification Testing”
CR M3-98-4504, “Acceptance Criteria and Allowable Value for Calculation SP-3GS-4, Revision 2, has no Basis”
CR M3-99-2960, “The Volume of Borated Water Required by TS is Greater Than the Volume of a Single Tank”
CR M3-99-1299, “Safety Evaluation Screen Package Needs Improvement”

LIST OF ACRONYMS USED

ACU	air conditioning unit
AOP	abnormal operating procedure
CIV	containment isolation valve
CO_2	carbon dioxide
CR	condition report
DAR	design change record
EDG	emergency diesel generator
ESF	engineered safety features
EOP	emergency operating procedure
FWA	auxiliary feedwater
IR	Inspection Report
LORT	licensed operator requalification training
MAP	Millstone Assessment Panel
MSS	main steam system
MRFF	maintenance rule functional failure
PI	performance indicator
PMT	post maintenance testing
SLCRS	supplementary leak collection and release system
SWP	service water
TDAFW	turbine driven auxiliary feedwater
TS	technical specification

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And

RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.