December 21, 1999

Mr. R. P. Necci, Vice President
Nuclear Oversight and Regulatory Affairs
c/o Mr. D. A. Smith, Manager - Regulatory Affairs
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, Connecticut 06385

SUBJECT: NRC COMBINED INSPECTION 05000336/99012, 05000423/99012

Dear Mr. Necci:

On November 22, 1999, the NRC completed an inspection at Millstone Units 2 & 3 reactor facilities. The enclosed report presents the results of that inspection.

During the seven weeks covered by this inspection period, your conduct of activities at the Millstone facilities was generally characterized by safety-conscious operations, sound engineering and maintenance practices, and careful radiological work controls. Both units remained at power, in operational Mode 1, throughout the inspection period.

Based on the results of this inspection, the NRC identified three Level IV violations of NRC requirements, one of which related to the Unit 3 design control inadequacy in the qualification of a safety-related sump pump modification. The other two involved examples at Unit 2 where operating procedures were not performed as written. In addition, the NRC determined that ten Level IV violations of NRC requirements, which were associated with conditions that are described in Licensee Event Reports, occurred prior to 1999. All of these violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the NRC Enforcement Policy. The NCVs are described in the subject inspection report. If you contest the violation or severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 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 placed in the NRC Public Document Room (PDR).

Sincerely,

ORIGINAL SIGNED BY:

James C. Linville, Director
Millstone Inspection Directorate
Region I

Docket Nos. 05000336 and 05000423

Enclosure: NRC Combined Inspection Report 05000336/99012 and 05000423/99012

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket Nos.:	05000336	05000423
Report Nos.:	99012	99012
License Nos.:	DPR-65	NPF-49

Licensee: Northeast Nuclear Energy Company
P. O. Box 128
Waterford, CT 06385

Facility: Millstone Nuclear Power Station, Units 2 and 3

Inspection at: Waterford, CT

Dates: October 5, 1999 - November 22, 1999

Inspectors: D. P. Beaulieu, Senior Resident Inspector, Unit 2
A. C. Cerne, Senior Resident Inspector, Unit 3
P. C. Cataldo, Resident Inspector, Unit 1
S. R. Jones, Resident Inspector, Unit 2
B. E. Sienel, Resident Inspector, Unit 3
R. L. Fuhrmeister, Senior Reactor Engineer, DRS
K. M. Jenison, Project Engineer, Millstone Inspection Directorate

Approved by: James Linville, Director
Millstone Inspection Directorate
Office of the Regional Administrator
Region I

EXECUTIVE SUMMARY

Millstone Nuclear Power Station
Combined Inspection 50-336/99-12; 50-423/99-12

Operations

- At Unit 2, operators did not adequately anticipate the effect of xenon during a planned power increase from 84 to 100 percent power. Consequently, this routine power increase was performed in a manner that resulted in the reactor's axial power distribution value exceeding the limit specified in the operating procedure, the receipt of multiple reactor protection system pre-trip alarms for local power density, and the unplanned need to add multiple batches of concentrated boric acid to restore axial power distribution within the specified limit. The failure to adequately implement the operating procedure to maintain the axial power distribution value within the specified limit, as required by Technical Specification 6.8.1, is being treated as a Non-Cited Violation. (NCV 50-336/99-12-01). (Section U2.O1.2)

- The licensee reported on February 18, 1998, that testing of the Unit 3 Turbine Driven Auxiliary Feedwater Pump (TDAFWP) had not fully met specific American Society of Mechanical Engineers (ASME) Section XI in-service testing requirements. This condition resulted from a failure to perform biennial position indication verification tests of certain TDAFWP solenoid operated valves and TDAFWP performance testing outside an accepted two percent band for rotational speed. This violation of Millstone Unit 3 Technical Specification 4.0.5 is being treated as a Non-Cited Violation (NCV 50-423/99-12-09). (Section U3.O8.1)

- On February 20, 1998, the licensee reported that they had identified historical failures to provide weep holes and other modifications to some safety related conduit and junction boxes located within the containment and auxiliary buildings. Without these modifications the equipment was potentially degraded and unable to meet the post Loss of Coolant Accident (LOCA) design basis described in the Unit 3 Final Safety Analysis Report (FSAR). This condition resulted from a failure to establish and implement an adequate 10 CFR 50.49 analysis to ensure that the Unit 3 FSAR design basis was maintained. This violation of 10 CFR 50 Appendix B, Criterion III, Design Control, is being treated as a Non-Cited Violation (NCV 50-423/99-12-10). (Section U3.O8.2)

- On March 18, 1998, the licensee reported that they had identified certain manual valves with an active safety function and several check valves that were historically not adequately tested in accordance with American Society of Mechanical Engineers (ASME) Section XI. ASME Section XI testing requirements are implemented through Unit 3 Technical Specification (TS) 4.0.5, to ensure that equipment performance criteria assumed in the Final Safety Analysis Report are met. This violation of TS 4.0.5 is being treated as a Non-Cited Violation (NCV 50-423/99-12-11). (Section U3.O8.3)

- At Unit 3, the licensee reported on October 13, 1998, that it had operated for a period of approximately three days with an inoperable Engineered Safety Function (ESF) channel (steam generator level) not in the tripped condition as required by Unit 3 Technical Specification (TS) 3.3.2. The cause of the condition was a failure of a P-14 bistable input power supply. Upon discovery, the failure was corrected. Failing to place the inoperable ESF instrument channel in a tripped condition is a violation of Unit 3 TS 3.3.2 and is being treated as a Non-Cited Violation (NCV 50-423/99-12-12). (Section U3.O8.5)

- The licensee reported on October 20, 1998, that historical surveillance testing of the Unit 3 Loose Parts Monitoring (LPM) system had not met Technical Specification (TS) requirements for range and accuracy. This violation of TS 3.3.3.8 is being treated as a Non-Cited Violation (NCV 50-423/99-12-13). (Section U3.O8.6)

Maintenance

- At Unit 2, following surveillance testing and operation of the "B" emergency diesel generator (EDG) on July 7, 1999, the licensee failed to restore the automatic voltage regulator to the position specified in the associated surveillance procedure. As a result, the "B" EDG output voltage was well below normal at its next start and was close to rendering the "B" EDG inoperable. The failure to adequately implement the surveillance procedure is being treated as a Non-Cited Violation. (NCV 50-336/99-12-02) (Section U2.M1.2)

Engineering

- At Unit 2, the licensee reported on January 30, 1997, that the actual vital chilled water flow to the east and west DC switchgear room coolers was lower than the indicated flow and that the reduced flow would have resulted in room temperatures exceeding final safety analysis report post-accident design temperatures. This condition resulted from a failure to adequately implement design controls to ensure that required chilled water flow in the vital switchgear room coolers was correctly translated into specifications, drawings and procedures. This violation of 10 CFR 50, Appendix B, Criterion III, Design Control, is being treated as a Non-Cited Violation (NCV 50-336/99-12-03). The licensee's corrective actions were found acceptable. Licensee Event Report 50-336/96-43-00 is closed. (Section U2.E8.1)

- At Unit 2, the NRC inspected a 1997 unresolved item (URI) and identified two instances where the licensee failed to establish and implement adequate fire protection program procedures. One instance was a failure to establish an adequate procedure for damper track inspections and a second instance was a failure to provide a method for the inspection of fire rated assemblies, fire barriers, and fire penetration seals. These two procedural inadequacies constitute a violation of Technical Specification 6.8.1 and are being treated as a Non-Cited Violation (NCV 50-336/99-12-04). Corrective Actions were found to be adequate. URI 50-336/97-84-02 is closed. (Section U2.E8.5)

- At Unit 2, the licensee identified in 1998 that the "A" AFW pump was not capable of meeting its design basis flow rate following a modification performed on its impeller. The licensee's corrective actions were found acceptable. This violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, is being treated as a Non-Cited Violation (NCV 50-336/99-12-05). Licensee Event Reports 50-336/98-04-00, 01, & 02 are closed. (Section U2.E8.6)

- The licensee identified in 1998 that a postulated loss of a Unit 2 service water pump without operator action to trip the associated reactor building closed cooling water (RBCCW) pump could cause RBCCW system temperatures to exceed the design basis values established in the Unit 2 Final Safety Analysis Report. The resulting reduction in room cooling capability could have impacted equipment operability in the other train. This violation of 10 CFR 50, Appendix B, Criterion III, Design Control, is being treated as a Non-Cited Violation (NCV 50-336/99-12-06). The licensee's corrective actions were found acceptable. Licensee Event Report 50-336/98-06-00 is closed. (Section U2.E8.9)

- At Unit 2, the licensee reported in 1998 that the auxiliary feedwater (AFW) system did not meet the revised loss of normal feedwater (LONF) safety analysis for the most limiting single failure. The postulated most limiting single failure could have resulted in less AFW flow to the steam generators than was credited in the Final Safety Analysis Report for a LONF event. This violation of 10 CFR 50 Appendix B, Criterion III, Design Control, is being treated as a Non-Cited Violation (NCV 50-336/99-12-07). The licensee's corrective actions were found acceptable. Licensee Event Reports 50-336/98-22-00, 01, & 02 are closed. (Section U2.E8.10)

- The licensee's failure to assure that the qualification testing for use of the air-driven sump pumps in a safety related application met all design control requirements was a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. However, based upon the license's subsequent corrective actions, promptly and deliberately implemented after the pump failure identified on September 23, 1999, and consistent with the NRC Enforcement Policy, this failure is being treated as a non-cited violation (NCV 50-423/99-12-14). (Section U3.E8.1)

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REPORT DETAILS

Summary of Unit 2 Status

Unit 2 entered the inspection period in Operational Mode 1, power operation, with the plant at 100 percent power. Operators reduced power to levels between 80 and 90 percent on the following three occasions: main turbine control valve testing on October 23, 1999; thermal backwash of circulating water bays from November 6 through 8, 1999; and cleaning of the "A" circulating water bay from November 15 through 18, 1999. At the conclusion of the inspection period, the plant remained in operation at 100 percent power.

U2.I Operations

U2 O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspector conducted frequent reviews of ongoing plant operations, including observations of operator evolutions in the control room; walkdowns of the main control boards; tours of the Unit 2 radiologically controlled area and other buildings housing safety-related equipment; and observations of several management planning meetings.

The inspector observed procedural adherence and conformance with technical specification requirements during routine operation at power, surveillance testing activities in the control room, and one reactor power increase. In general, the inspectors continued to note thorough turnovers and good communication practices among operators in the control room. However, the inspectors observed weak operator control of reactor parameters during the reactor power increase. This weakness is discussed in greater detail in the next section of this report (Section U2 O1.2). Also, operators failed to properly restore the "A" emergency diesel generator following surveillance testing (Section U2 M1.2).

O1.2 Weak Operator Control during Reactor Power Increase

a. Inspection Scope (71707)

The inspector observed a routine reactor power increase that occurred on November 8, 1999. The inspector reviewed relevant operating procedures and discussed operator actions with the responsible shift manager and the Unit 2 Operations Manager.

b. Observations and Findings

Following completion of circulating water bay thermal backwashing on November 8, 1999, operators initiated a return to 100 percent power at a planned rate of 7 percent per hour in accordance with procedure OP 2204, "Plant Power Changes." The reactor was initially at 84 percent power with nearly stable xenon reactivity. The control room operators initiated the power increase by adding positive reactivity through the addition of water to dilute the reactor coolant system boron concentration. The power increase caused a further addition of positive reactivity by depleting the xenon poison in the reactor. The xenon depletion occurred predominately in the lower half of the reactor, which is typical of power increases. This condition caused power production to shift toward the lower half of the reactor.

Step 4.1.6 of procedure OP 2393, "Core Power Distribution Monitoring and Control," specifies that operators maintain the axial shape index (ASI) between 0.020 and 0.040 during reactor power transients. [The ASI value is a measure of the axial power distribution in the reactor. Positive values for ASI indicate that power production is greater in the lower half of the reactor; negative values for ASI indicate that power production is greater in the upper half of the reactor.] As specified in Step 4.1.9 of procedure OP 2393, the control operators periodically compensated for the shift in power production to the lower half of the reactor by withdrawing the regulating control rods to shift power production back toward the upper half of the reactor. This action maintained ASI values within the specified band.

Approximately three hours after initiating the power increase, the reactor was at 99 percent power with all control rods fully withdrawn. The operators logged that they intended to allow continued xenon depletion to bring the reactor to 100 percent power. However, the reactor reached the 100 percent power limit specified in procedure OP 2204 while the xenon depletion was continuing at a significant rate. Because the operators were concerned about the positive reactivity addition that would be caused by about 40 gallons of pure water already in the charging pump suction piping due to the previous dilution, the operators decided not to add boric acid via the charging pumps. Instead, the operators inserted control rods to halt the reactor power increase.

The control rod insertion caused the ASI value monitored by the operators on the reactor protection system to increase from 0.039 to 0.043, which exceed the limit specified in procedure OP 2393. Concurrently, the operators received intermittent local power density pre-trip alarms on multiple reactor protection system channels. The local power density calculated in the reactor protection system is a function of reactor power and axial power distribution. The pre-trip alarm provides margin for operators to adjust reactor parameters before an actual reactor trip is initiated by the reactor protection system. Subsequently, the operators reduced reactor power to 97 percent power and restored the axial power distribution to within the limit specified in procedure OP 2393 by adding multiple batches of concentrated boric acid to the reactor coolant system. The licensee documented in Condition Report M2-99-2905 that the effects of xenon during this evolution to increase reactor power required the addition of substantial amounts of boric acid to lower and stabilize reactor power.

The inspector determined that operators did not adequately anticipate the effect of xenon during the planned power increase from 84 to 100 percent power. Consequently, this routine power increase was performed in a manner that resulted in the reactor's ASI value exceeding the limit specified in procedure OP 2393, the receipt of multiple RPS pre-trip alarms for local power density, and the unplanned need to add several batches of concentrated boric acid to restore ASI within the specified limit.

Technical Specification 6.8.1 requires that written procedures be established and implemented covering activities listed in Appendix "A" of Regulatory Guide 1.33, which includes power operation and process monitoring, such as reactor power distribution monitoring and control. Therefore, the failure to adequately implement the reactor axial power distribution limits specified in procedure OP 2393 is a violation. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. This issue was documented in Condition Report M2-99-2905.

c. Conclusions

The NRC concluded that operators did not adequately anticipate the effect of xenon during a planned power increase from 84 to 100 percent power. Consequently, this routine power increase was performed in a manner that resulted in the reactor's axial power distribution value exceeding the limit specified in the operating procedure, the receipt of multiple RPS pre-trip alarms for local power density, and the unplanned need to add multiple batches of concentrated boric acid to restore axial power distribution within the specified limit. The failure to adequately implement the operating procedure to maintain the axial power distribution value within the specified limit, as required by Technical Specification 6.8.1, is being treated as a **Non-Cited Violation. (NCV 50-336/99-12-01)**

U2.II Maintenance

U2 M1 Conduct of Maintenance

M1.1 General Maintenance Observations

a. Inspection Scope (62707/61726)

During routine plant inspection tours, the inspectors observed, on a random sampling basis, maintenance and surveillance activities to evaluate the propriety of the activities and the functionality of systems and components with respect to technical specifications and other requirements.

b. Observations and Findings

The inspectors reviewed maintenance work orders and surveillance procedures and interviewed licensee field personnel to verify the adequacy of work controls and surveillance testing. The inspector observed a portion of activities performed under the following automated work orders (AWOs) and surveillance procedures:

- AWO M2-99-00299 — "A" Emergency Diesel Generator Service Water Flow Transmitter Preventive Maintenance
- Procedure SP 2401D — Reactor Protection System Matrix Logic and Trip Path Relay Test
- Procedure SP 2613L — "B" Emergency Diesel Generator Slow Start Test
- Procedure SP 2601 — "Borated Water Sources and Flow Path Verification"

The inspector found that maintenance work was being performed in accordance with approved work orders present at the work site. A review of the work packages found that they were complete with respect to work authorizations, procedures, and inspection requirements. Surveillance testing was performed in accordance with approved procedures that demonstrated acceptable performance of equipment with respect to technical specification requirements.

c. Conclusions

The inspectors concluded that the work performed under the listed maintenance work orders and surveillance testing conducted in accordance with the listed procedures were acceptable.

M1.2 Low Output Voltage from the "B" Emergency Diesel Generator

a. Inspection Scope (93702/61726)

The inspector reviewed an event where the "B" emergency diesel generator output voltage was low when it was started for surveillance testing on August 5, 1999.

b. Observations and Findings

After a slow start of the "B" emergency diesel generator (EDG) on August 5, 1999, operators found that its output voltage did not satisfy the minimum voltage specified in surveillance procedure SP 2613L, "Diesel Generator Slow Start Operability Test, Facility 2." At startup, the control room operators observed that the "B" EDG output voltage was 4050 volts on the control panel gage. The surveillance procedure specified a minimum voltage of 4100 volts. The associated Technical Specification surveillance requirement specifies that the "B" EDG output voltage reach 97 percent of its design voltage of 4160 volts (approximately 4036 volts) within 15 seconds.

The control room operators determined that the voltage regulator was functioning properly and continued the EDG run. However, the operators declared the "B" EDG inoperable based on the failure to satisfy the surveillance test acceptance criterion and requested an engineering disposition of this condition through Condition Report (CR) M2-99-2179.

Based on plant process computer data that indicated the output voltage was below the voltage required for automatic loading of the vital bus onto the "B" EDG, the engineering department concluded that the "B" EDG would not have performed its design function for the previous 32 days since its last surveillance run. On August 11, 1999, the licensee reported this condition in accordance with 10 CFR 50.72(b)(1)(ii), as a condition while the plant was in operation that resulted in the plant being outside its design basis. However, upon further review, the licensee determined that the "B" EDG would have performed its design function because: (1) its output voltage was adequate to actuate a relay which permits automatic loading of the vital bus onto the "B" EDG; and (2) the relay setpoint is at the Technical Specification required voltage. Unlike the plant process computer voltage data, the relay setpoint is calibrated as a safety-related component and is, therefore, considered more accurate. Accordingly, on August 17, 1999, the licensee notified the NRC that they were retracting their previous notification of this event.

The inspector evaluated the licensee's reportability evaluation associated with this event. Since both the control board indication and the relay setpoint confirmed that the "B" EDG output voltage satisfied the Technical Specification required voltage, the inspector determined that the licensee's decision to retract the notification was correct.

The inspector also evaluated the cause of the low output voltage from the "B" EDG. The licensee's reportability evaluation noted that the "B" EDG voltage regulator was not readjusted following the previous surveillance run in July. Step 4.1.72 of surveillance procedure SP 2613L specifies that the operator adjust the automatic voltage regulator to maintain 4160 volts prior to securing the "B" EDG. However, plant process computer data from July 7, 1999, indicates that the "B" EDG output voltage was not readjusted following operation in parallel with the grid, which was at about 4050 volts. Technical Specification 6.8.1.c requires written procedures to be implemented for surveillance activities, which includes the "B" EDG surveillance procedure. The failure to adequately implement procedure SP 2613L to restore the "B" EDG output voltage to its normal value is a violation of Technical Specification 6.8.1.c. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. Although corrective actions addressing this issue were not included with CR M2-99-2179, appropriate corrective actions were recommended in CR M2-99-2380.

c. Conclusions

Following surveillance testing and operation of the "B" EDG on July 7, 1999, the licensee failed to restore the automatic voltage regulator to the position specified in the associated surveillance procedure. As a result, the "B" EDG output voltage was well below normal at its next start and was close to rendering the "B" EDG inoperable. The failure to adequately implement the surveillance procedure is being treated as a **Non-Cited Violation (NCV 50-336/99-12-02).**

U2 M8 Miscellaneous Maintenance Issues

M8.1 (Closed) LER 50-336/97-20-02 and 97-20-03; Boron Injection Flow Paths

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this licensee event report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

On August 31, 1998, the licensee initially reported that the surveillance procedure for verifying the proper alignment of the boron injection flow path was inadequate. The licensee identified this inadequate procedure in response to NRC violations and an NRC restart issue associated with inadequate procedures.

The inspector conducted an on-site review of this LER and its updates, Unit 2 Technical Specification (TS), licensee response to NRC violations 50-336/96-08-07 and 50-336/96-08-08, associated Unit 2 surveillance and operating procedures, Adverse Condition Reports (ACR M2-97-0649, M2-97-0890, and M2-97-2565), and surveillance procedures 2601A-1 and 2601F-1, "Borated Water Sources and Flow Path Verification". In addition, a limited review of additional TS related testing activities was performed. The licensee's corrective actions in response to NRC violations 50-336/96-08-07 and 50-336/96-08-08 resulted in the identified/reported conditions in this LER. The corrective actions documented by the licensee for this LER and its associated corrective actions were determined to be adequate.

The root cause of the condition described in the LER was a failure to properly incorporate TS surveillance requirements into plant surveillance procedures. Because the technical discoveries reported in this LER and its updates were directly related to corrective actions for a previous NRC violation, no additional violation will be issued. LER 50-336/97-20-02 & -03 are **closed.**

U2.III Engineering

U2 E8 Miscellaneous Engineering Issues

E8.1 (Closed) LER 50-336/96-43-00; Chilled Water Flow in the Vital Switchgear Room Coolers

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this licensee event report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On January 30, 1997, with Unit 2 in Mode 6, the licensee identified that the actual vital chilled water flow to the east and west DC switchgear room coolers was approximately 29 percent lower than the indicated flow. The flow error was discovered during the performance of a post-maintenance test and resulted from a calibration error on flow elements FI-8891 and FI-8893. The licensee initially discovered this condition in 1991 and performed an evaluation to determine the minimum indicated flow required to ensure the proper post-accident flow of 27 gallons per minute. However, the system surveillance procedure was not revised to reflect the higher indicated flow requirement. The licensee analyzed that the reduced flow would have resulted in room temperatures exceeding final safety analysis report (FSAR) assumed post-accident design temperatures.

The inspector conducted an on-site review of the LER, Unit 2 Technical Specifications (TS), associated Unit 2 operating and surveillance procedures, and corrective actions documented in the licensee's corrective action process. In addition, a limited review of additional setpoint calibration activities was performed. The licensee's corrective actions in 1991 were found to be inadequate in that they did not correct the indication problem and did not determine that the flow condition would have resulted in a post-accident condition outside the design of the unit. However, the corrective actions documented in the LER 50-336/96-43-00 and corrective action process were determined to be adequate. Because this instrument calibration also affected the American Society of Mechanical Engineers Section XI chill water pump flow testing required by TS 4.0.5, a sample of additional Section XI testing was reviewed. No additional problem examples or generic issues were identified in the subsequent reviews.

The inspector determined that the root cause of the condition described in the LER was a failure to establish and implement formal controls over the testing, calibration and regulation of safety-related flow instrumentation. Failing to adequately implement design controls to ensure that chilled water flow in the vital switchgear room coolers was correctly translated into specifications, drawings and procedures, is a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. This concern was entered as Condition Report M2-96-0928.

c. Conclusions

The licensee reported on January 30, 1997, that the actual vital chilled water flow to the Unit 2 east and west DC switchgear room coolers was lower than the indicated flow and that the reduced flow would have resulted in room temperatures exceeding FSAR post-accident design temperatures. This condition resulted from a failure to adequately implement design controls to ensure that required chilled water flow in the vital switchgear room coolers was correctly translated into specifications, drawings and procedures. This violation of 10 CFR 50, Appendix B, Criterion III, Design Control, is being treated as a **Non-Cited Violation (NCV 50-336/99-12-03)**. The licensee's corrective actions were found acceptable. LER 50-336/96-43-00 is **closed**.

E8.2 (Closed) LER 50-336/97-18-00; Instrument Loop Components do not Meet Regulatory Guide 1.97 Requirements

This license event report (LER) documented that on November 6, 1996, with Unit 2 in Mode 6, the licensee identified that selected instrument loop components did not meet the requirements of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The review prompting this discovery was partly in response to an NRC deviation in Inspection Report 50-336/94-201. In a letter dated January 30, 1995, the licensee committed to perform an engineering review of Regulatory Guide 1.97 commitments and installations.

The inspector conducted an on-site review of the LER, Unit 2 Technical Specifications, associated Unit 2 operating and surveillance procedures, and the corrective actions documented in the licensee's corrective action process. The licensee's corrective actions were found to be adequate. In addition, a limited industry material history data review was performed and no problems were identified. Because the technical discoveries were directly related to a previous NRC deviation and the corrective actions were adequate, no additional NRC deviation is being issued. LER 50-336/97-18-00 is **closed**.

E8.3 (Closed) LER 50-336/97-28-00, -01, & -02; Electrical Equipment Qualification Program Deficiencies

The inspector conducted in-office and on-site reviews of Licensee Event Reports (LER) 50-336/97-28-00, -01 & -02, and LER 50-336/96-19-00, which address various examples of historical electrical equipment qualification (EQ) problems that were identified by the licensee during an extended shutdown from February 1996 to May 1999. LER 50-366/96-19-00 initially reported the EQ program deficiencies and addressed several specific equipment problems. The inspector reviewed the LERs, Unit 2 technical specifications (TS), the Unit 2 EQ program, the Unit 2 Technical Requirements Manual (TRM) and associated corrective actions. A sample of corrective actions showed that the issues were properly tracked and resolved, including those that required procedure changes, modifications, TS change requests and/or updates to the Unit 2 TRM.

The licensee identified numerous equipment problems and implemented appropriate corrective actions during their extensive reviews of the Unit 2 design basis. These reviews were conducted in response to a number of NRC violations involving the Unit 2 design basis. One NRC Escalated Enforcement Item (EEI) 50-336/96-06-12 addressed the EQ program and a specific EQ concern regarding solenoid operated valves. Unit 2 Significant Items List (SIL) No. 19.5 also addressed EQ program deficiencies and additional specific equipment problems. The inspector found that LERs 50-336/97-28-00, -01, & -02 provided additional examples of EQ program deficiencies that the licensee identified in response to EEI 50-336/96-06-12 and SIL 19.5. Therefore, the additional equipment problems reported in LERs 50-336/97-28-00, -01, & -02 are considered to be dispositioned from an enforcement perspective. LERs 50-336/97-28-00, -01, & -02, are **closed**.

E8.4 (Closed) LER 50-336/97-29-02; Piping Stress Analyses

This licensee event report (LER) update amended the original LER that reported an inappropriate application of Section III of the American Society of Mechanical Engineers Code in the calculation of pipe support loads for main steam and feedwater lines located outside containment. The update extends the licensee's review process of piping stress analyses used for the main feedwater piping. Following reanalysis, the licensee determined that the main feedwater lines outside containment required piping support modifications. These modifications were completed and the current analysis includes a consideration of the modifications.

The inspector conducted an on-site review of the LER, associated corrective actions documented in the licensee's corrective action process, supporting codes and standards, and a sample of licensee calculations. The licensee's corrective actions were found to be adequate. LER50-336/97-29-00 & 01 were closed in NRC Inspection Report 50-336/99-05 and Non-Cited Violation 50-336/99-05-09 was issued for inadequate design control associated with the main steam and feedwater piping. No additional violations of NRC requirements were identified during a review of this LER update. LER update 50-336/97-29-02 is **closed**.

E8.5 (Closed) URI 50-336/97-84-02; Fire Protection Program Controls

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in Unresolved Item (URI) 50-336/97-84-02. The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

Four issues composing this URI were reviewed to evaluate the licensee's corrective actions.

Issue 1 - A previously used silicon sealant installation procedure did not specifically implement the Technical Requirements Manual (TRM) requirement for a visual inspection of at least 10% of the total number of penetration seals. In spite of the lack of specific guidance, neither the licensee nor the NRC identified deficient penetration seals or fire assemblies related to this problem.

Issue 2 - A technical basis was not established for damper track inspections.

Issue 3 - Resolution of licensee identified deficiencies documented in Adverse Condition Report (ACR) M2-96-0588.

Issue 4 - Implementation of corrective actions for hot short conditions, identified by the licensee in response to NRC Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire."

The inspector conducted an on-site review of the URI, Unit 2 TRM, Unit 2 Technical Specifications, associated corrective actions documented in the licensee's corrective action process, supporting design change documentation (including portions of Design Change Requests M2-97040, M2-97055, M2-98067, and M2-98089), procedure changes (including procedure SP2618G, "Fire Damper Operability Verification," and procedure SFP 17, "Fire Protection Seal Inspection"), and a sample of licensee calculations.

Issues 3 and 4 of URI 50-336/97-84-02 were determined to have been identified and adequately corrected by the licensee. No violations of NRC requirements were identified.

Issues 1 and 2 constitute two examples of a failure to establish, implement and maintain procedures in accordance with Unit 2 TS 6.8.1. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. The corrective actions affected by the licensee were found to be adequate. This issue was entered as Condition Report M2-98-0443.

c. Conclusions

The NRC performed a follow up inspection of a 1997 URI and identified two instances where the licensee failed to establish and implement adequate fire protection program procedures at Unit 2. One instance was a failure to establish an adequate procedure for damper track inspections and a second instance was a failure to provide a method for the inspection of fire rated assemblies, fire barriers, and fire penetration seals. These two procedural inadequacies constitute a violation of TS 6.8.1 and are being treated as a **Non-Cited Violation (NCV 50-336/99-12-04)**. Corrective actions were found to be adequate. URI 50-336/97-84-02 is **closed.**

E8.6 (Closed) LER 50-336/98-04-00, 98-04-01 and 98-04-02; Auxiliary Feedwater Pump Performance Degraded

a. Inspection Scope (37550, 92903)

The inspector conducted an on-site review of the licensee event report (LER), Unit 2 Technical Specifications, associated corrective actions documented in the licensee's corrective action process, the Unit 2 Final Safety Analysis Report (FSAR), supporting codes and standards, a sample of licensee procurement standards and calculations, selected testing and acceptance documentation, and selected manufacturing documentation.

b. Observations and Findings

On March 16, 1998, while reviewing "A" auxiliary feedwater (AFW) pump post-modification test data, the licensee identified that the pump performance was degraded. Comparison of test data to the vendor's pump curve showed only minor degradation at low flow rates, however, degradation was significantly more pronounced at the credited accident analysis flow rate. The degraded equipment condition resulted from an error in the manufacturing of a replacement impeller, installed in 1995. Following the 1995 modification of the "A" AFW pump impeller, the pump was tested in accordance with and met Unit 2 In-service Testing and Technical Specification (TS) testing requirements. However, the testing performed was a single point/low flow test which was adequate to ensure that the "A" AFW pump would meet FSAR design flow requirements.

The licensee's corrective actions, which included reshaping the impellers, adequately retesting the impeller and pump, and reporting the event to the NRC, were found to be adequate. Failing to adequately establish and implement adequate testing to ensure that a safety-related pump was capable of operating within the requirements of its design is a violation of 10 CFR 50, Appendix B, Criterion XI, Test Control. This Severity Level IV violation is being treated as a Non-Cited Violation consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the issue being entered into their corrective action program. This issue was entered as Condition Report M2-98-0714.

c. Conclusions

The licensee identified in 1998 that the "A" AFW pump was not capable of meeting its design basis flow rate following a modification performed on its impeller. The licensee's corrective actions were found acceptable. This violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, is being treated as a **Non-Cited Violation (NCV 50-336/99-12-05)**. LERs 50-336/98-04-00, 01, & 02 are **closed**.

E8.7 (Closed) IFI 50-336/98-05-04; Steam Generator Blowdown Flow Effects on Primary and Secondary Leakage Calculations

This IFI was written to review the licensee's corrective actions to address steam generator blowdown flow rate estimation errors and their impact on the technical specification (TS) required primary-to-secondary leak rate calculations.

The inspector conducted in-office and on-site reviews of the IFI, Unit 2 TS, Unit 2 operating procedures, selected engineering evaluations (including Technical Evaluation TE-M2-EV-99-0101), and associated corrective actions documented in the licensee's corrective action process. A limited industry experience data review was performed and no issues of a generic nature were identified pertaining to steam generator blowdown flow rate calculations. The corrective actions were implemented by the licensee, in part, as a response to Non-Cited Violation 50-336/95-05-03. The NCV was written to document Unit 2 exceeding its licensed maximum power limit, as a result of steam generator blowdown flow calculation errors. Because the technical discoveries were directly related to a previous NRC violation and the corrective actions were adequate, the issue is considered to be dispositioned from an enforcement perspective. IFI 50-336/98-05-04 is **closed**.

E8.8 (Closed) URI 50-336/98-05-09; Adequacy of Penetration Seal Repairs

In 1998 an unresolved item (URI) was written to review the adequacy of the licensee's corrective actions established to resolve the operability of approximately eighty silicon foam fire seals. In 1984, the licensee documented that certain penetration seal pours did not receive an in-process verification by Quality Control (QC). Both the maintenance and QC organizations administratively closed automated work order (AWO) M2-84-06200, noted that the independent in-process QC verification of the pours had not occurred, and accepted the as-left condition of the seals. As a result, approximately eighty silicone foam, penetration seals, were installed in the floor barrier separating the Unit 2 control room and cable spreading room, without receiving an in-process QC, independent verification of the seal pour depth. The licensee's 1984 corrective actions, for these discrepancies included procedural reviews and revisions, but did not include any physical testing, gaging or dimensional reexaminations to ensure seal operability.

In 1998 an URI was written to review the adequacy of the licensee's corrective actions to resolve the operability of the seals. Subsequent to the URI, the licensee's corrective actions included intrusive testing of eight seals for depth and a document review. No inadequate seals were identified by the licensee during the intrusive tests or the document review.

The inspector conducted an on-site review of the URI, applicable Unit 2 procedures, data from the penetration seals, and associated corrective actions documented in the licensee's corrective action process (including Condition Report M2-98-3811). The inspector determined that the procedural requirements for independent, in-process QC verification were not sufficiently clear in 1984, to ensure consistent implementation. Considering the clarity problem, the inspector determined that the 1984 QC procedural requirements were met. Aside from questions concerning the in-process QC verifications, the 1984 fire barrier penetration seal installation procedure was determined to be adequate in NRC Inspection Report 336/99-03 and during this inspection. The 1984 AWO was determined to be adequately signed and accepted by QC and the Unit 2 maintenance organization, and the in-process verification discrepancies were determined to have been identified and documented by QC.

Because the URI was written to ensure the operability of the penetration seals, the inspector verified that no inadequate seals were identified by the licensee during the 1998 performance of intrusive tests and document reviews. In addition, the NRC observed portions of the intrusive penetration tests and concluded in NRC Inspection Report 336/99-03 that the sampled seals had been adequately poured. Because the licensee identified, appropriately documented and administratively accepted the as-left condition of the eighty seals and no seals were identified to be defective in the 1998 intrusive tests, no violation of NRC requirements was identified. URI 50-336/98-05-09 is **closed**.

E8.9 (Closed) LER 50-336/98-06-00; Reactor Building Closed Cooling Water System Outside Design Basis upon Loss of Service Water

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this licensee event report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On March 27, 1998, with Unit 2 in Mode 6, the licensee postulated that a post-accident loss of a service water pump without operator action to trip the associated reactor building closed cooling water (RBCCW) pump may cause RBCCW system temperatures to exceed the design basis values established in the Unit 2 Final Safety Analysis Report (FSAR). Elevated RBCCW temperatures could result in elevated engineered safety feature room and auxiliary building temperatures, ultimately impacting the unaffected train.

The inspector conducted an on-site review of the LER, Unit 2 Technical Specifications, Unit 2 FSAR, associated Unit 2 operating, abnormal and emergency procedures, and corrective actions documented in the licensee's corrective action process. The corrective actions documented in the licensee's current LER and corrective action process were determined to be adequate.

The root cause of the condition described in the LER was a failure to establish the impact of a specific single failure on the design basis of the unit. The Unit 2 FSAR, Section 9.4.1.2, states that a single failure in either subsystem of RBCCW will not affect the functional capability of the other subsystem. Failing to adequately implement design controls to ensure RBCCW design criteria was correctly translated into specifications, drawings and procedures, is a violation of 10 CFR 50, Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. This concern was entered as Condition Report M2-98-0852.

c. Conclusions

The licensee identified in 1998 that a postulated loss of a Unit 2 service water pump without operator action to trip the associated RBCCW pump could cause RBCCW system temperatures to exceed the design basis values established in the Unit 2 FSAR. The resulting reduction in room cooling capability could have impacted equipment operability in the other train. This violation of 10 CFR 50, Appendix B, Criterion III, Design Control, is being treated as a **Non-Cited Violation (NCV 50-336/99-12-06)**. The licensee's corrective actions were found acceptable. LER 50-336/98-06-00 is **closed**.

E8.10 (Closed) LER 50-336/98-22-00, 50-336/98-22-01, and 50-336/98-22-02: Auxiliary Feedwater DC Power Supply Failure Not Considered In Safety Analysis

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this licensee event report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On September 10, 1998, with Unit 2 in Mode 6, the licensee determined that the auxiliary feedwater (AFW) system did not meet the revised loss of normal feedwater (LONF) safety analysis for the most limiting single failure. The loss of the Train 2, 125 volt DC bus could have resulted in less AFW flow to the steam generators than was credited in the final safety analysis report (FSAR) for a LONF. The FSAR states that all of the reactor operating conditions allowed by the plant technical specifications (TS) were examined to ensure that bounding sub-events were identified for each Standard Review Plan event category. It further states that the engineered safety feature (ESF) systems are required to function in the event of an assumed worst single failure of an active component.

The inspector conducted an on-site review of the LER, Unit 2 TS, Unit 2 FSAR, associated Unit 2 operating, abnormal and emergency procedures, selected AFW system modifications and corrective actions documented in the licensee's corrective action process. The corrective actions documented in the licensee's current LER and corrective action process were determined to be adequate including a modification to the Train 2 AFW pump DC power supply.

The root cause of the condition described in the LER was a failure to establish the impact of a specific single failure on the design basis of the unit. The failure to adequately implement design controls to ensure AFW design criteria was correctly translated into specifications, drawings and procedures, is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. The corrective actions affected by the licensee were found to be adequate. This issue was entered as Condition Reports M2-98-2706 and M2-98-2494.

c. Conclusions

The licensee reported in 1998 that the AFW system did not meet the revised LONF safety analysis for the most limiting single failure. The postulated most limiting single failure could have resulted in less AFW flow to the steam generators than was credited in the FSAR for a LONF event. This violation of 10 CFR 50 Appendix B, Criterion III, Design Control, is being treated as a **Non-Cited Violation (NCV 50-336/99-12-07)**. The licensee's corrective actions were found acceptable. LERs 50-336/98-22-00, 01, & 02 are **closed**.

E8.11 (Closed) IFI 50-336/98-219-13; Enhancements to Address Post-Accident Pump Seal Leakage within the "B" Engineered Safety Features Pump Room

a. Inspection Scope (92903)

The inspector reviewed the licensee's response to Inspector Follow-up Item (IFI) 50-336/98-219-13. This IFI was opened to track review of the licensee's assessment of design or procedural enhancements to reduce the vulnerability to a failure of the emergency core cooling system (ECCS) minimum flow isolation valves as a result of flooding of the "B" engineered safety features (ESF) pump room. During the post-accident recirculation phase, these isolation valves must close to preserve the water inventory necessary for post-accident cooling. The flooding would be caused by leakage from a postulated pump seal failure. The inspector conducted in-office and on-site reviews.

b. Observations and Findings

The Millstone Unit 2 design includes two separate, water-tight ESF pump rooms, each of which contains one train of ECCS equipment. Motor-operated valves with operating switches in the control room are available to isolate each train of ECCS equipment from potential sources of water. Each ESF pump room also contains a sump with a high water level switch that initiates an alarm in the control room. Prior to restarting Unit 2 in 1999, the licensee replaced the level switches with seismic Category 1 level switches which were qualified to withstand the postulated harsh post-accident environment in the ESF pump rooms. The alarm circuit is powered from a reliable source having an on-site, backup power source. Prior to restarting Unit 2 in 1999, the licensee also revised their emergency operating procedures to address alarms indicating high water level in the ESF pump rooms by directing operators to pump-down and, if necessary, isolate the affected ESF pump room.

The two ECCS trains share a common line for recirculation of minimum flow to the refueling water storage tank. This common line is located in the "B" ESF pump room approximately 3 feet above the floor, and it contains two air-operated valves in series. The valves fail open on loss of electric power or air to the valves. The valves are required to be open during the injection phase and closed during the recirculation phase of a design bases accident. If flooding were to occur in the "B" ESF pump room to the extent that electric power to the valves was interrupted, the valves would fail open. In the recirculation phase, this would result in a gradual loss of the reactor coolant inventory from the containment sump and a transfer of this water to the refueling water storage tank, which is vented to atmosphere.

The concern identified in IFI 50-336/98-219-13 involved the potential increased consequences associated with failure of the minimum flow recirculation isolation valves due to flooding in the "B" ESF pump room. The solenoids for these valves are located 5 feet above the floor. The licensee calculated that, for the maximum credible pump seal leakage of 3 gallons per minute, the water level would reach the solenoid valves after 14 days without operator action to isolate the leakage. Both the high pressure safety injection and containment spray pumps would be flooded several days prior to the water level reaching the solenoid valves. The failure of these pumps would provide additional indication of potential flooding in the room. After reviewing this information and the design modifications and procedural changes previously implemented, the licensee determined that the minimum flow recirculation isolation valves were adequately protected from postulated flooding from pump seal leakage. The inspector found that the licensee's determination had an adequate basis.

c. Conclusion

The licensee made an acceptable determination that the minimum flow recirculation isolation valves were adequately protected from postulated flooding from pump seal leakage. In addition, the current design is in compliance with their licensing and design basis with regard to passive failures during the post-accident recirculation phase. Therefore, Inspector Followup Item 50-336/98-219-13 is **closed**.

REPORT DETAILS

Summary of Unit 3 Status

Unit 3 began the inspection period operating at 100 percent power. On October 15, 1999, operators reduced power to approximately 90% to perform a condenser thermal backwash. Following the successful completion of the backwash, power was restored to 100% on October 17. Operators again reduced power to approximately 90% on November 2 to backwash the "A" condenser bay during a storm. Operators subsequently restored power to 100% on November 3, where it remained through the end of the inspection period on November 22.

On November 18, production, maintenance, operations, and other selected groups of plant workers voted on whether to accept representation by the International Brotherhood of Electrical Workers (IBEW) Union. Approximately 467 employees were eligible to vote. The union was rejected by a simple majority.

On November 8 and 9, Mr. Hubert Miller, NRC Regional Administrator for Region I, visited the Millstone site. See Section X3 for further details on his visit.

U3.I Operations

U3 O2 Operational Status of Facilities and Equipment

O2.1 Review of Operational Activities and Equipment Control

a. Inspection Scope (71707, 92901)

The inspectors performed control room and field inspection-tours and reviewed the control of operational activities, including configuration management, the repair of control board components, and the adequacy of procedural guidance for equipment operation in the field. The inspector also discussed chemistry controls, cold weather preparations and operations staff training activities with the unit operations management and support staffs. Operational assessments, conducted by the Nuclear Oversight organization, were also reviewed and discussed with the cognizant quality assurance personnel.

b. Observations and Findings

(1) During the conduct of plant inspection-tours, the inspector routinely checked the position of various safety system valves, comparing the field configuration with that documented in the system piping and instrumentation drawings. The inspector also reviewed operations procedure OP 3260B (Revision 3, Change 12), noting, and applying during the field inspections, the documented criteria for locking valves and other components in a specified position. Where questions arose, the inspector reviewed the Locked Component Checklist (OPS Form 3260B-1) and discussed the observed conditions (e.g., drain valves) with the responsible system engineer.

(2) The inspector assessed the controls established for the on-line replacement of all control rod bank step counter indicator batteries. The inspector noted that a special procedure, SPROC OPS99-3-01, was developed for performance of this activity with the unit in Mode 1 at full power, due to recent failures of two of the subject rod position indicator batteries. The control room operators declared each demand position indicator inoperable, as work was being performed, and appropriately took actions delineated in the applicable section of technical specification (TS) 3.1.3.2. The inspector also reviewed a licensing position, documented by the regulatory affairs staff, supporting the battery replacement work for shutdown banks C, D, and E, where only one group indicator exists in each bank. The inspector determined that this activity was adequately controlled by the SPROC and performed in accordance with TS requirements, with valid justification for voluntary TS entry to conduct the work.

(3) As documented and previously discussed in Inspection Report 50-423/99-09, the licensee's material performance group issued a memorandum to the Unit 3 operations manager temporarily raising the dissolved oxygen limit (for Action Level 1) at the condensate pump discharge (CPD) point from 10 ppb to 12 ppb. During the current inspection period, the inspector requested and reviewed the safety evaluation screening form for the required revision to the chemistry procedure, CP 3802B, effecting this change. Over the course of this inspection period, licensee efforts to reduce CPD dissolved oxygen levels were found to be making a positive impact. In conjunction with the increased condenser efficiency caused by the cooler, seasonal bay temperatures, the licensee's efforts obviated the need for continued implementation of the elevated Action Level 1 limit. On October 22, 1999, as documented in licensee memorandum ME-MP-99-288, the normal Action Level 1 limit of 10 ppb for CPD dissolved oxygen was reinstated.

(4) During the conduct of plant inspection-tours, the inspector checked the configuration of certain system/train alignments; e.g., the service water valve alignment for the two operating reactor plant component cooling (CCP) trains, with the spare CCP train isolated. In the auxiliary building motor control center areas, the inspector noted that certain train "A" 480 volt load center equipment (e.g. the spent fuel pool cooling pump, 3SFC*P1A; and the fuel building exhaust fan, 3HVR*FN10A) was capable of being started and run with a key-locked, local control switch, while the redundant train "B" equipment (e.g., 3SFC*P1B, 3HVR*FN10B) was not designed with such a local control capability. Upon discussion with a cognizant engineer in the licensee's fire protection group, the inspector learned that the unit design provided that train "A" components be relied upon for equipment operation in the event that a control room evacuation was required. However, a review of the applicable emergency operating procedures (e.g., EOP 3509 series for a fire emergency) identified no particular procedure describing when or how such local, manual operation of these train "A" components would be controlled. The inspector questioned whether, without such procedural direction, the plant operators would have sufficient guidance and support (e.g., the correct keys) to conduct local operation of the subject equipment, when required. The licensee initiated condition report CR M3-99-3750 to resolve this issue by identifying all the equipment that could be operated in this manner and by providing the necessary procedural guidance to meet the operations department standards.

During another inspection-tour of the plant, the inspector examined the status of two heat tracing panels in the fuel building, as part of an assessment of the operations department cold weather preparedness activities. The two selected panels (3HTS-PNLF1 and F2) provide heat tracing circuitry for all the required safety-related lines requiring freeze protection at Unit 3. The initial inspection revealed an inconsistency in the calibration tags for the two circuits, but this was corrected by the licensee after it was determined that both sets of these circuit controls had been calibrated in September 1999, and the equipment tags were so annotated. The inspector identified another inconsistency in that operations procedure OP 3352 for Heat Tracing specifies that the freeze protection circuitry be in operation when the outside air is less than 35°F, while at least one supported system procedure (i.e., OP 3308 for the High Pressure Safety Injection system) requires the heat tracing to be energized when the outside air temperature is less than 40°F. The inspector discussed this inconsistency with the Unit 3 heat tracing system engineer, who confirmed that the actual system setpoints were conservative, but agreed that the procedural inconsistency should be corrected and other procedures checked for the same problem.

(5) The inspector attended a portion of training entitled "Improving Performance with Qualification, Validation and Verification Techniques," provided to all six operations shift crews in separate weekly sessions. This training, provided by a contractor and observed by the inspector on October 19, 1999, was based upon the contractor's program for techniques in highly effective human error reduction habits. The inspector noted that the operations initiative to conduct such training reflected a good understanding of the causal relationship between personnel errors and reportable events at Unit 3 since the beginning of this year.

The inspector also reviewed selected Nuclear Oversight surveillance reports providing operations department assessments in September and November 1999 and met with Oversight managers on October 27, 1999, for a briefing on the Nuclear Oversight Verification Plan (NOVP) results for nuclear operations and other routinely assessed areas. The inspector noted that the Nuclear Oversight department continued to provide an ongoing assessment of the operational focus of Unit 3 activities.

c. <u>Conclusions</u>

The licensee provided good control of routine Unit 3 testing, corrective maintenance activities, and other corrective actions that had the potential to adversely impact operations. Some procedural omissions and inconsistencies affecting operator responses to certain plant conditions were identified and deemed appropriate for NRC inspector followup because of the potentially broader implications for other plant areas. Nuclear Oversight continued to monitor performance in the operations area with an assessment of the operational focus of plant activities.

U3 O8 Miscellaneous Operations Issues (92700)

O8.1 (Closed) LER 50-423/98-11-00; Failure to Adequately Test the Turbine Driven Auxiliary Feedwater Pump

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this Licensee Event Report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On February 18, 1998, with Unit 3 in Mode 5, the licensee identified that testing of the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) had not fully met specific aspects of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI In-service Testing requirements. No surveillance procedure could be located that documented the performance of a required biennial position indication verification test of certain TDAFWP solenoid operated valves. In addition, the TDAFWP had been tested outside an accepted two percent band for rotational speed.

The inspector conducted an on-site review of the LER, Unit 3 technical specifications (TS), associated Unit 3 operating and surveillance procedures, and corrective actions documented in the licensee's corrective action process (including condition reports [CRs] M3-98-0802 and M3-98-1002). In addition, a limited review of additional ASME Section XI pump testing activities and data was performed. The limited review of additional ASME Section XI testing required by TS 4.0.5 identified no additional problems or generic issues. The corrective actions documented in the licensee's current LER and corrective action process were determined to be adequate. The issue was appropriately documented, trended, and tracked in accordance with the licensee's corrective action process.

The inspector determined that the root cause of the condition described in the LER was a failure to establish and implement testing required by ASME Section XI, as directed in Millstone Unit 3 TS 4.0.5. Failing to adequately establish and implement testing in accordance with TS 4.0.5 is a violation of NRC requirements. This Severity Level IV violation is being treated as a **Non-Cited Violation (NCV 50-423/99-12-08)**, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. The corrective actions affected by the licensee were found to be adequate, including the documentation, tracking and trending of conditions in accordance with the licensee's corrective action process.

c. Conclusions

The licensee reported on February 18, 1998, that testing of the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) had not fully met specific American Society of Mechanical Engineers (ASME) Section XI in-service testing requirements. This condition resulted from a failure to perform biennial position indication verification tests of certain TDAFWP solenoid operated valves and TDAFWP performance testing outside an accepted two percent band for rotational speed. This violation of Millstone Unit 3 Technical Specification 4.0.5 is being treated as a **Non-Cited Violation (NCV 50-423/99-12-08)**. The licensee's corrective actions were acceptable. Licensee Event Report 50-423/98-11-00 is **closed**.

O8.2 (Closed) LER 50-423/98-14-00; Failure to Provide Weep Holes in Conduits and Junction Boxes

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this Licensee Event Report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On February 20, 1998, with Unit 3 in Mode 5, the licensee identified historical failures to provide weep holes and other modifications in some safety related conduit and junction boxes located in the containment and auxiliary buildings. This condition could have degraded the capability of the plant to mitigate a Final Safety Analysis Report (FSAR) design basis Loss of Coolant Accident (LOCA).

The inspector conducted an on-site review of the LER, Unit 3 FSAR, associated Unit 3 maintenance and surveillance procedures, a sample of design change notices and records, and corrective actions documented in the licensee's corrective action process. The CRs reviewed included M3-98-1010 and M3-97-2235. The corrective actions documented in the licensee's current LER and CR corrective action process were determined to be adequate. The inspector reviewed a sample of the current corrective actions which included the installation of weep holes, repair of Raychem splices, installation of T-drains, installation of ASCO solenoid elbow configuration, verification of name plate data, and engineering analysis. The individual issues were appropriately documented, trended, and tracked in accordance with the licensee's corrective action process.

Past licensee response to NRC Information Notice (IN) 89-63, Submergence of Electrical Circuits Located Above Flood Level, was reviewed by the inspector. A licensee memo (Scace/Marino, dated January 12, 1990) described the licensee's initial actions taken in response to the IN, which were limited to inspections performed on a random number of Unit 1 instruments and their ancillary parts inside and outside the drywell. These inspections identified numerous problems which were documented and deferred, due to schedule constraints, to the next available shutdown of sufficient duration. The findings documented in the January 12, 1990, utility memo caused additional review to be performed on Unit 3. The Unit 3 review findings are documented in numerous corrective action documents and a utility memo (Scace/Marino dated May 7, 1990). The utility memo documents the licensee's decision to accept-as-is the Unit 3 configuration and institute maintenance worker training. The utility memo documents the Unit 3 response to NRC IN 89-63 as NOA 10057 and concludes that:

> Inside containment terminal blocks are not allowed and a Raychem splice is the only approved method of cable termination due to its submergence withstand qualification. For equipment configurations with Raychem splices and cable entrance seals, weep holes are not deemed necessary since the cable terminations and end use equipment are sealed to preclude moisture intrusion, or have submergence withstand (sic) qualification.....since there is an inherent danger of damage to the enclosed cable when drilling weep holes in existing raceway (sic), the proposed NOA action is not recommended at this time. This provides the Millstone 3 response to NOA 10057.

Because the licensee subsequently identified, documented and reported numerous instances where conditions similar to those described in the NRC IN potentially affected the ability of Unit 3 safety related equipment to perform within the design basis described in the FSAR, the inspector determined that the licensee's corrective actions in 1990 were incomplete.

The inspector found that; LER 50-423/98-14-00 documented numerous conditions that existed between 1990 and 1997; the conditions were similar to those described in NRC IN 89-63; the conditions were considered significant by the licensee, because their Environment Equipment Qualification (EEQ) process did not analyze for condensation effects on specific components; and the licensee's analysis identified the need for hardware modifications. As a result, the inspector determined that the root cause of the condition described in the LER was a failure to establish and implement adequate 10 CFR 50.49 analysis to ensure that the Unit 3 FSAR design bases were maintained. Failing to ensure that design criteria are correctly translated into specifications, drawings and procedures, is a violation of 10 CFR 50 Appendix B, Criterion III, Design Control. This Severity Level IV violation is being treated as a **Non-Cited Violation (NCV 50-423/99-12-09)**, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. The current corrective actions affected by the licensee were found to be adequate, including the documentation, tracking and trending of conditions in accordance with the licensee's corrective action process.

c. Conclusions

On February 20, 1998, the licensee reported that they had identified historical failures to provide weep holes and other modifications to some safety related conduit and junction boxes located within the containment and auxiliary buildings. Without these modifications the equipment was potentially degraded and unable to meet the post Loss of Coolant Accident (LOCA) design basis described in the Unit 3 Final Safety Analysis Report (FSAR). This condition resulted from a failure to establish and implement an adequate 10 CFR 50.49 analysis to ensure that the Unit 3 FSAR design basis was maintained. This violation of 10 CFR 50 Appendix B, Criterion III, Design Control, is being treated as a **Non-Cited Violation (NCV 50-423/99-12-09)**. The licensee's current corrective actions were acceptable. Licensee Event Report 50-423/98-14-00 is **closed.**

O8.3 (Closed) LER 50-423/98-21-00; Manual Valve and Check Valve Tests Not Included in the In-Service Test Program

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this Licensee Event Report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On March 18, 1998, with Unit 3 in Mode 5, the licensee identified that some manual valves with an active safety function and several check valves were not being adequately tested in accordance with American Society of Mechanical Engineers (ASME) Section XI.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and surveillance procedures, design change documents (including M3-97093 and M3-00-0690-98) and corrective actions documented in the licensee's corrective action process (including CR M3-98-1471 and ACR 97-0191). In addition, a limited review of additional surveillance activities associated with Unit 3 TS 4.0.5 was performed. The corrective actions documented in the licensee's LER and corrective action process were determined to be adequate. The issue was appropriately documented, trended, and tracked in accordance with the licensee's corrective action process. A sample review of additional ASME Section XI testing documentation identified no problems or generic issues. A sample review of historical valve performance and maintenance indicated that the subject valves had been tested, exercised and maintained at a moderate level of performance, and that no history existed of routine or generic valve failures. The inspector concluded that it was likely that the sampled valves would have performed near their design basis performance requirements.

The inspector determined that the root cause of the condition described in the LER was a failure to adequately implement design controls to ensure that design basis data were correctly translated into specifications, drawings and testing procedures required by Unit 3 TS 4.0.5. Failure to implement the requirements of TS 4.0.5 is a violation of NRC requirements. This Severity Level IV violation is being treated as a **Non-Cited Violation (NCV 50-423/99-12-10)**, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. The corrective actions affected by the licensee were found to be adequate, including the documentation, tracking and trending of conditions in accordance with the licensee's corrective action process.

c. Conclusions

On March 18, 1998, the licensee reported that they had identified certain manual valves with an active safety function and several check valves that were historically not adequately tested in accordance with American Society of Mechanical Engineers (ASME) Section XI. ASME Section XI testing requirements are implemented through Unit 3 Technical Specification (TS) 4.0.5, to ensure that equipment performance criteria assumed in the Final Safety Analysis Report are met. This violation of TS 4.0.5 is being treated as a Non-Cited Violation **(NCV 50-423/99-12-10)**. The licensee's corrective actions were acceptable. Licensee Event Report 50-423/98-21-00 is **closed**.

O8.4 (Closed) LER 50-423/98-23-00; Auto Start of Motor Driven Auxiliary Feedwater (MDAFW) Pump and the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) Started on Steam Generator "C" Low-Low Water Level

This LER documented that on April 11, 1998, with Unit 3 in Mode 4, the licensee experienced an Engineered Safety Features (ESF) actuation while the unit was being heated up. The ESF actuation entailed the automatic start of the MDAFW pump and TDAFWPs on receiving an actual low-low steam generator level signal. After reviewing the event, the NRC issued violation 50-423/97-83-03 for a failure to establish adequate startup operating procedures, including steam generator water level control.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating procedures (including OP 3201, Plant Heatup) and the corrective actions documented in the licensee's CR corrective action process (including CR M3-98-1904). The licensee's corrective actions as documented in a licensee letter (Bowling/NRC, B17327), dated June 13, 1998, were found to be adequate. In addition, a limited industry operating history data review was performed and no additional problems were identified. Because the technical issues were adequately responded to by the licensee, and the event was addressed by a previous NRC violation, no additional violations of NRC requirements were identified. This LER is **closed**.

O8.5 (Closed) LER 50-423/98-41-00 and 98-41-01; Failure to Enter Action Statement for Inoperable Steam Generator Water Level Channel in Accordance with Technical Specifications

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this Licensee Event Report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On October 13, 1998, with Unit 3 in Mode 1, the licensee identified that one steam generator water level instrument channel reading was significantly lower than the 3 remaining instrument channels. The licensee reported that it had operated for a period of approximately three days with an inoperable Engineered Safety Function (ESF) channel not in the tripped condition required by Unit 3 TS 3.3.2. The cause of the condition was a failure of a P-14 bistable input power supply.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and surveillance procedures, and corrective actions documented in the licensee's CR corrective action process (including CRs M3-98-4466, 4471, and 4715). In addition, a limited review of additional computer point calibration and maintenance activities was performed (including work order M3-98-15543).

The inspector found that the corrective actions documented in the licensee's LER and corrective action process were adequate. The issue was appropriately documented, trended, and tracked in accordance with the licensee's corrective action process. Because this instrument failure indicated a potential for other instrument failures to affect the operability of safety related equipment, a sample of additional computer point inputs was performed and no additional problems or generic issues were identified.

Operating with a failed steam generator level channel without placing it in the tripped condition is a violation of Unit 3 TS 3.3.2. This Severity Level IV violation is being treated as a **Non-Cited Violation (NCV 50-423/99-12-11)**, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. The corrective actions affected by the licensee were found to be adequate, including the documentation, tracking and trending of conditions in accordance with the licensee's corrective action process.

c. Conclusions

The licensee reported on October 13, 1998, that it had operated for a period of approximately three days with an inoperable Engineered Safety Function (ESF) channel (steam generator level) not in the tripped condition as required by Unit 3 Technical Specification (TS) 3.3.2. The cause of the condition was a failure of a P-14 bistable input power supply. Upon discovery, the failure was corrected in an adequate manner. Failing to place the inoperable ESF instrument channel in a tripped condition is a violation of Unit 3 TS 3.3.2 and is being treated as a **Non-Cited Violation (NCV 50-423/99-12-11)**. The licensee's corrective actions were acceptable. Licensee Event Reports 50-423/98-41-00 and 50-423/98-41-01 are **closed**.

O8.6 (Closed) LER 50-423/98-42-00: Loose Parts Monitoring System Operational Test

a. Inspection Scope (37550, 92903)

The inspector performed on-site and in-office reviews of the actions taken by the licensee to address the issues identified in this Licensee Event Report (LER). The reviews included inspection of the licensee's corrective actions and supporting references and discussions with licensee personnel.

b. Observations and Findings

On October 20, 1998, with Unit 3 in Mode 1, the licensee determined that (1) the monthly Loose Parts Monitoring (LPM) system monthly operational surveillance did not verify the range and accuracy of the LPM alarm setpoints and (2) the LPM channel calibration surveillance did not verify setpoints, within the range and accuracy required by Unit 3 Technical Specifications.

The inspector conducted an on-site review of the LER, Unit 3 TS, associated Unit 3 operating and surveillance procedures (including SP 3451, Loose Parts Monitor) and corrective actions documented in the licensee's CR corrective action process (including CR M3-98-4503). In addition, a limited review of additional setpoint channel calibration surveillance activities was performed.

The corrective actions documented in the licensee's LER and corrective action process were determined to be adequate. The issue was appropriately documented, trended, and tracked in accordance with the licensee's corrective action process. A sample of additional setpoint channel calibration surveillance activities identified no additional problems associated with establishing the appropriate calibration range or accuracy. The failure to establish adequate monthly operational and channel calibration surveillances is a violation of Unit 3 TS 3.3.3.8 and is being treated as a **Non-Cited Violation (NCV 50-423/99-12-12)**, consistent with Section VII.B.1.a of the NRC Enforcement Policy, which permits closure of most Severity Level IV violations based on the licensee entering the issue into their corrective action program. The corrective actions affected by the licensee were found to be adequate, including the documentation, tracking and trending of conditions in accordance with the licensee's corrective action process.

c. Conclusions

The licensee reported on October 20, 1998, that historical surveillance testing of the Unit 3 Loose Parts Monitoring (LPM) system had not met Technical Specification (TS) requirements for range and accuracy. This violation of TS 3.3.3.8 is being treated as a **Non-Cited Violation (NCV 50-423/99-12-12)**. The licensee's corrective actions were found to be acceptable. Licensee Event Report 50-423/98-42-00 is **closed**.

U3.II Maintenance

U3 M1 Conduct of Maintenance

M1.1 Surveillance Observations

a. Inspection Scope (61726)

The inspectors observed portions of the following surveillance activities, discussed the conduct of work and controls with operations, instrumentation and controls and condition based maintenance personnel, and reviewed selected test results.

- SP 3646A.2 Emergency Diesel Generator B Operability Test
- SP 3606.1 Containment Recirculation Pump 3RSS*P1A Operational Readiness Test
- SP 3610A.2 Residual Heat Removal (RHR) Pump 3RHS*P1B Operational Readiness Test

b. Observations and Findings

On October 5 and October 29, 1999, the inspectors observed the pre-job briefs for SP 3646A.2 and SP 3606.1, respectively. The briefs were attended by the required personnel and included appropriate discussions of surveillance sequence, expected results, communication, and contingency actions associated with the planned tests.

The inspectors observed the containment recirculation pump surveillance both from the control room and locally at the pump cubicle. In the Unit 3 control room, the operators performed the applicable actions satisfactorily and no issues were identified. The inspectors verified that appropriate instrumentation was installed locally for performance of the surveillance. Due to recent issues concerning the adequacy of flow instrumentation during surveillance tests on the auxiliary feedwater system, the licensee decided to use a later model controllotron flow instrument for this test. The instrumentation was properly calibrated and operated well during the surveillance. Appropriate acceptance criteria were met and the test was completed satisfactorily.

The inspector witnessed portions of the monthly testing of the train "B" emergency diesel generator (EDG), in accordance with surveillance procedure SP 3646A.2. The inspector discussed the pre-test equipment checks with the assigned plant equipment operators, examined the "B" EDG and associated support components, and evaluated system restoration to its post-test configuration, as delineated on OPS Form 3646A.2-1. Procedural controls were adequately established to support the conduct of another surveillance procedure, SP 3226.13, checking the efficiency of the "B" EDG engine air cooler and jacket water heat exchangers. After the test, the inspector examined selected portions of the "A" EDG equipment configuration and confirmed consistency with the train "B" post-test component lineup.

On October 27, the inspector observed the setup and calibration activities for flow instrumentation used in the RHR pump surveillance test delineated in SP 3610A.2. The inspector verified proper calibration and physical condition of the maintenance and testing equipment and discussed the use of the temporary instrumentation, as specified in the surveillance procedure, with the responsible system engineer. The inspector subsequently reviewed the associated data form, OPS Form 3610A.2-1, and verified the acceptance criteria were met and consistent with the surveillance requirements and observed field conditions.

c. Conclusions

Observed Unit 3 surveillances were performed in a controlled manner in accordance with approved procedures.

U3.III Engineering

U3 E2 Engineering Support of Facilities and Equipment

E2.1 Review of Engineering Controls and Design Changes

a. Inspection Scope (37551, 92903)

During the conduct of plant inspection-tours and review of operational test activities, the inspector noted certain field configurations that appeared to have been changed from the original plant design. The inspector reviewed the current, relevant design documents for the noted revisions and discussed the adequacy of the affected operational, identification/marking, and procedural controls with the cognizant system engineers.

b. Observations and Findings

The inspector reviewed design change record (DCR) M3-96065, Revision 3, and an associated design change notice (DCN) DM3-00-0817-98. The resulting modifications involved the addition non-pilot operated safety-related solenoid valves to the control systems for four valves in the residual heat removal (RHS) system. Because of the non-safety design of the instrument air supplies, and thus the flow control valve positioners for these valves, direct acting solenoid valves were required in the instrument air line to each valve actuator to enure that the venting of air would place the RHS valves in their fail-safe positions.

The inspector examined solenoid valve 3RHS*SOV607 installed on the actuator for the train "B" RHS flow control valve and confirmed the ASCO model number (L206-381-6RF) specified in the DCR. The inspector also verified other design modification details, including the implementation of required documentation and procedure revisions discussed in the DCN. Evidence of the appropriate safety evaluation screening, instrument setpoint considerations, and design qualification data were available in the DCR/DCN document package.

The inspector also reviewed DCN DM3-P-154-90, associated with plant design change record PDCR MP3-90-003, authorizing the removal of snubbers on some train "B" RHS piping. The material condition of some low pressure safety injection (SIL) valves installed as drain connections in the piping connected to the RHS pump suction lines was discussed with the RHS system engineer, who confirmed the proper material type in accordance with the design. Additionally, the inspector verified the configuration of selected quench spray (QSS) piping, valves, and flow instrumentation that were no longer in procedural use by the operators for returning QSS test flow to the refueling water storage tank (RWST). The field configuration matched the details on the piping and instrumentation drawing (P&ID EM 115A), except for the position of a non-safety-related valve, illustrated as open but actually closed, which was the required position in accordance with the QSS/RWST valve lineup. The responsible system engineer was apprized of this minor drawing discrepancy.

The inspector identified some labeling discrepancies for equipment markings at the entrances to the train "A" and "B" RHS cubicles. Typical label problems included wrong train color codes, identification of safety-related equipment with non-safety markings, and inconsistencies between redundant equipment in the two trains. Upon discussing these deficiencies with licensee personnel, they confirmed the mis-labeled component tags and initiated corrective measures in accordance with station procedure OA 9, "System and Component Labeling". Similarly, in reviewing some design changes relative to the QSS system, the inspector identified and apprized the system engineer of a safety-related valve (3QSS*V44) incorrectly annotated as a non-safety component in surveillance procedure SP 3609.1, Revision 8. Such discrepancies constitute issues of minor safety significance for which the licensee initiated appropriate corrective action.

c. <u>Conclusions</u>

Inspection-tours of Unit 3 areas housing safety-related equipment verified that the components checked were in conformance with current design criteria and that equipment modifications had been adequately controlled in accordance with the design change program. Some equipment labeling and component identification deficiencies were identified, but these items were of minor safety significance and were appropriately placed by the licensee in the corrective action program for resolution.

U3 E8 Miscellaneous Engineering Issues

E8.1 <u>(Closed) URI 50-423/99-09-13; Reactor Plant Aerated Drains (DAS) System Pump Failure</u>

The train "A" recirculation spray system (RSS) cubicle sump pump, 3DAS*P15A, failed its monthly performance test on September 23, 1999. Personnel at the scene reported that it did not sound as though the pump was turning. Subsequent disassembly and inspection of the pump and air-motor internals revealed that the vanes, fabricated of Hylum material, in the air-motor had swelled due to absorption of oil and/or moisture. The subsequent elongation of the vanes apparently caused the air-motor to seize. Since this represented a potential for a common mode failure of both trains of the safety-related pumps, the train "B" RSS cubicle sump pump was removed and disassembled, after restoring train "A" capability with an operable spare pump. Inspection of the air-motor internals revealed that the Hylum vanes in 3DAS*P15B had also swelled, though the elongated vanes were still shorter than the inside dimension of the air motor housing. The pumps had been installed in the plant since May 1999. The vanes of both pumps were then reworked to a closer tolerance to original specification length (+0, -.002 inches), and the two pumps reinstalled in the plant in a series schedule that maintained one train operable at all times. Plans were made to run the pumps monthly and remove, disassemble, and inspect one pump's air-motor on a staggered-month basis.

3DAS*P15A and B were successfully tested on October 20, 1999. The train "A" pump was removed and replaced with a reworked spare pump. The removed pump was disassembled and air-motor vane dimensions checked. Measurements showed that the vanes had expanded approximately .001 inch, and remained within tolerance for a new vane. Performance testing of the newly installed pump resulted in calculated flow slightly exceeding specification for a new pump. The inspector witnessed the post-run layup and maintenance activities on October 22, 1999. Both pumps passed their surveillance tests on November 19, 1999. Blade length measurements taken on the pump removed indicated that blade length had stabilized.

Technical Evaluation M3-EV-99-0098 was issued October 27, 1999, to provide justification for continued operability, document the review of the condition, and provide criteria for the monthly inspections of the pumps. The technical evaluation includes data from several sets of Hylum blades which were exposed to an oily environment for over a year. It assumed that these blades had reached equilibrium swelling. Using the qualification vendor blade dimension specifications, acceptable lengths both for new "dry" blades and for blades which had been in service were developed. The corrective actions recommended include additional testing to quantify blade elongation rates, documentation of installed blade lengths and motor liner lengths for any pumps procured in the future, and bimonthly disassembly and inspection of the installed pumps to evaluate blade elongation in service.

The licensee completed a Substantial Safety Hazard Evaluation (No. 99-03) on November 10, 1999, to determine if the elongation of the Hylum blades was reportable under Part 21 of the regulations. Although the licensee conservatively concluded that the condition constituted a "deviation," there was no indication that it would create a loss of a safety function to the extent that a major reduction in the protection of the public health and safety would result.

Based on the stabilization of the measured blade lengths, the inspector determined that the licensee had taken appropriate actions to resolve the current problems with the DAS safety-related, air-driven sump pumps. The inspector determined that the blade elongation problem had been masked by prior difficulties associated with inadequate preventive maintenance, corrosion, and debris in the air motors, which had been previously reviewed by the NRC, as documented in Inspection Report 50-423/99-07.

The licensee's root cause investigation for these prior DAS sump pump failures (Revision 1, dated June 8, 1999), conducted as follow-up to condition report CR M3-99-1803, documented inadequate qualification testing requirements. This investigation report also references a Nuclear Oversight CR (M3-98-03167) initiated in 1998, which documents concerns regarding the qualification testing of the pumps. Included as part of the investigation report was a National Technical Systems Test Report (61282-00N), dated May 27, 1999, which provided evidence that the Hylum blade growth was a measured phenomenon, although not recognized at that time as a potential failure mechanism. This information suggests that the qualification testing of these DAS pumps was still inadequate, as of June 1999, in not considering continued Hylum blade growth as a common mode failure mechanism for the installed pumps.

The inspector determined that the licensee had failed to assure that the qualification testing for use of the subject air-driven sump pumps in a safety-related application met all design control requirements; representing a violation of 10 CFR 50, Appendix B, Criterion III. However, based upon the license's subsequent corrective actions, promptly and deliberately implemented after the pump failure identified on September 23, 1999, and consistent with Section VII.B.1.a of the NRC Enforcement Policy, this Severity Level IV violation is being treated as a Non-Cited Violation **(NCV 50-423/99-12-13).** In evaluating the effectiveness of the licensee corrective actions, the inspector also concluded that trimming the blades to lengths provided in Technical Evaluation M3-EV-99-0098, improved preventive maintenance, and the provision of filter-dryer skids for the air system constituted reasonable measures for ensuring continued pump operability.

The **unresolved item** and the resulting NCV, discussed above, are hereby **closed.**

V. Management Meetings

X1 Exit Meeting Summary

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

X3 Management Meeting Summary

On November 8 and 9, 1999, Mr. Hubert Miller, NRC Regional Administrator for Region I, and Mr. James Linville, Director of the Millstone Inspection Directorate, visited the site. During the visit, they toured Units 1, 2, and 3, interviewed selected management and staff personnel, and attended a portion of the Millstone Leadership Meeting.

INSPECTION PROCEDURES USED

IP 37550	Engineering
IP 37551	Onsite Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observations
IP 71707	Plant Operations
IP 92700	Onsite follow-up of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92901	Follow-up - Operations
IP 92903	Follow-up - Engineering
IP 93702	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-336/99-12-01	NCV	Failure to adequately implement the operating procedure to maintain the axial power distribution value within the specified limit, as required by Technical Specification 6.8.1
50-336/99-12-02	NCV	Failure to adequately implement the surveillance procedure to adjust "B" EDG output voltage
50-336/99-12-03	NCV	Inadequate design controls to ensure that required chilled water flow in the vital switchgear room coolers (related to LER 50-336/96-43-00)
50-336/99-12-04	NCV	Failure to establish an adequate procedure for damper track inspections and for the inspection of fire rated assemblies, fire barriers, and fire penetration seals (related to URI 50-336/97-84-02)
50-336/99-12-05	NCV	Failure to adequately test to ensure that AFW pump was capable of operating within the requirements of its design (related to LER 50-336/98-04-00, 01, & 02)
50-336/99-12-06	NCV	Inadequate design controls regarding RBCCW design criteria upon a loss of service water (related to LER 50-336/98-06-00)
50-336/99-12-07	NCV	Inadequate design controls regarding an AFW single failure vulnerability on loss of a DC bus (related to LER 50-336/98-22-00, 01, & 02)
50-423/99-12-08	NCV	Failure to adequately establish and implement testing in accordance with TS 4.0.5 (related to LER 50-423/98-11-00)
50-423/99-12-09	NCV	Failure to ensure that design criteria are correctly translated into specifications, drawings and procedures (related to LER 50-423/98-14-00)
50-423/99-12-10	NCV	Failure to adequately implement design controls to ensure that design basis data were correctly translated into specifications, drawings and testing procedures required by Unit 3 TS 4.0.5 (related to LER 50-423/98-21-00)
50-423/99-12-11	NCV	Operating with a failed steam generator level channel without placing it in the tripped condition (related to LER 50-423/98-41-00 & 01)
50-423/99-12-12	NCV	Failure to establish adequate monthly operational and channel calibration surveillances (related to LER 50-423/98-42-00)
50-423/99-12-13	NCV	Failure to assure that the qualification testing for use of the subject air-driven sump pumps in a safety-related application met all design control requirements

Closed

The NCVs opened above are closed.

50-336/97-84-02	URI	Fire Protection Program Controls

50-336/98-05-04	IFI	Steam Generator Blowdown Flow Effects on Primary and Secondary Leakage Calculations
50-336/98-05-09	URI	Adequacy of Penetration Seal Repairs
50-336/98-219-13	IFI	Enhancements to Address Post-Accident Pump Seal Leakage within the "B" Engineered Safety Features Pump Room
50-423/99-09-13	URI	Reactor Plant Aerated Drains (DAS) System Pump Failure

The following LERs were also closed during this inspection:

LER 50-336/96-43-00	Chilled Water Flow in the Vital Switchgear Room Coolers
LER 50-336/97-18-00	Instrument Loop Components do not Meet Regulatory Guide 1.97 Requirements
LER 50-336/97-20-02 & 03	Boron Injection Flowpaths
LER 50-336/97-28-00, 01, & 02	Electrical Equipment Qualification Program Deficiencies
LER 50-336/97-29-02	Piping Stress Analyses
LER 50-336/98-04-00, 01, & 02	Auxiliary Feedwater Pump Performance Degraded
LER 50-336/98-06-00	Reactor Building Closed Cooling Water System Outside Design Basis upon Loss of Service Water
LER 50-336/98-22-00, 01, & 02	Auxiliary Feedwater DC Power Supply Failure Not considered In Safety Analysis
LER 50-423/98-11-00	Failure to Adequately Test the Turbine Driven Auxiliary Feedwater Pump
LER 50-423/98-14-00	Failure to Provide Weep Holes in Conduits and Junction Boxes
LER 50-423/98-21-00	Manual Valve and Check Valve Tests Not Included in the In-Service Test Program
LER 50-423/98-23-00	Auto Start of Motor Driven Auxiliary Feedwater (MDAFW) Pump and the Turbine Driven Auxiliary Feedwater (TDAFW) Pump Started on Steam Generator "C" Low-Low Water Level
LER 50-423/98-41-00 & 01	Failure to Enter Action Statement for Inoperable Steam Generator Water Level Channel in Accordance with Technical Specifications
LER 50-423/98-42-00	Loose Parts Monitoring System Operational Test

LIST OF ACRONYMS USED

ACR	adverse condition report
AFW	auxiliary feedwater
ASI	axial shape index
ASME	American Society of Mechanical Engineers
AWO	automated work orders
CCP	reactor plant component cooling
CFR	Code of Federal Regulations
CPD	condensate pump discharge
CR	condition report
DAS	reactor plant aerated drains
DCN	design change notice
DCR	design change record
ECCS	emergency core cooling system
EDG	emergency diesel generator
EEI	escalated enforcement item
EEQ	electrical equipment qualification
EOP	emergency operating procedure
EQ	equipment qualification
ESF	engineered safety feature
FSAR	Final Safety Analysis Report
HTS	heat tracing system
HVR	reactor plant ventilation
IBEW	International Brotherhood of Electrical Workers
IFI	inspector follow-up item
IN	information notice
LER	licensee event report
LOCA	loss of coolant accident
LONF	loss of normal feedwater
LPM	loose parts monitoring
MDAFW	motor-driven auxiliary feedwater
NCV	non-cited violation
NOVP	nuclear oversight verification plan
NRC	Nuclear Regulatory Commission
P&ID	piping and instrumentation drawing
PDCR	plant design change record
ppb	parts per billion
QC	quality control
QSS	quench spray system
RBCCW	reactor building closed cooling water
RCS	reactor coolant system
RHR	residual heat removal
RHS	residual heat removal
RSS	recirculation spray system
RWST	refueling water storage tank
SDC	shutdown cooling
SFC	spent fuel pool cooling

SIL	significant items list
SPROC	special procedure
TDAFWP	turbine driven auxiliary feedwater pump
TRM	technical requirements manual
TS	technical specification
URI	unresolved item