August 31, 2000

Mr. S. E. Scace, Director
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's MILLSTONE INSPECTION REPORT NOS. 05000336/2000-009 AND 05000423/2000-009

Dear Mr. Scace:

On August 12, 2000, the NRC completed inspections at your Millstone Units 2 & 3 reactor facilities. The enclosed reports present the results of these inspections. The results were discussed on August 30, 2000, with Messrs. M. Brothers and R. Necci and other members of your staff.

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

The NRC identified four Unit 2 issues, three of which were evaluated under the risk significance determination process and were determined to be of very low safety significance (green). The other issue was determined to have no color. These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection reports. Of the four issues, three were determined to involve violations of NRC requirements, but because of their very low safety significance the violations are not cited. If you contest these noncited violations, you should provide a response within 30 days of the date of these inspection reports, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington 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 available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosures: (1) NRC Inspection Report 05000336/2000-009
(2) NRC Inspection Report 05000423/2000-009

cc w/encl:
B. D. Kenyon, President and Chief Executive Officer - NNECO
R. P. Necci, Vice President - Nuclear Technical Services
L. J. Olivier, Senior Vice President and Chief Nuclear Officer - Millstone
M. H. Brothers, Vice President - Nuclear Operations
F. C. Rothen, Vice President - Nuclear Work Services
J. T. Carlin, Vice President - Human Services - Nuclear
G. D. Hicks, Director - Nuclear Training Services
C. J. Schwarz, Station Director
D. A. Smith, Manager - Regulatory Affairs
L. M. Cuoco, Senior Nuclear Counsel
J. R. Egan, Esquire
N. Burton, Esquire
V. Juliano, Waterford Library
J. Buckingham, Department of Public Utility Control
State of Connecticut SLO Designee
First Selectmen, Town of Waterford
D. Katz, Citizens Awareness Network (CAN)
T. Concannon, Co-Chair, NEAC
R. Bassilakis, CAN
J. M. Block, Attorney, CAN
G. Winslow, Citizens Regulatory Commission (CRC)
E. Woollacott, Co-Chair, NEAC

Distribution w/encl: **<VIA E-MAIL>**:
E. Adensam, NRR (RIDSNRRDIPMLPDI)
J. Clifford, NRR
V. Nerses, PM, NRR
T. Madden, OCA
D. Thatcher, NRR
J. Shea, OEDO
J. Zimmerman, PM, NRR
Region I Docket Room (with concurrences)
S. Jones, SRI - Millstone
H. Miller, RA, RI
J. Wiggins, DRA, RI
J. Linville, DRP
R. Summers, DRP
K. Jenison, RI
R. Junod, DRP
D. Screnci, PAO
M. Oprendek, DRP

DOCUMENT NAME: G:\BRANCH6\MILL2000-009.WPD
After declaring this document "An Official Agency Record" it **will/will not** be released to the Public. **To receive a copy of this document, indicate in the box:** "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP	
NAME	SJones/JL for		JLinville/JL	
DATE	08/31/00		08/31/00	

OFFICIAL RECORD COPY

ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	05000336
License No.:	DPR-65
Report No.:	05000336/2000-009
Licensee:	Northeast Nuclear Energy Company
Facility:	Millstone Nuclear Power Station, Unit 2
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	July 2, 2000 - August 12, 2000
Inspectors:	P. C. Cataldo, Resident Inspector, Unit 2 S. R. Jones, Senior Resident Inspector, Unit 2 M.S. Ferdas, Reactor Inspector, Division of Reactor Safety K. M. Jenison, Senior Projects Engineer, Division of Reactor Projects
Approved by:	James C. Linville, Chief Projects Branch 6 Division of Reactor Projects Region I

SUMMARY OF FINDINGS

IR 05000336/2000-009; on 07/02-08/12/00; Millstone Nuclear Power Station; Unit 2. Maintenance Rule Implementation, Maintenance Risk Assessment and Emergent Work Evaluation, Operator Work-Arounds, Surveillance Testing, Cross-cutting Issues.

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

Cornerstone: Initiating Events

- **Green.** The NRC found that the licensee failed to establish appropriate performance goals and monitor system performance against those goals after the plant-level performance criterion for unplanned scrams was exceeded and significant unplanned capability loss was accrued due to ineffective corrective and preventive maintenance of the control rod drive system. Since exceeding the plant level performance criterion in February 2000, the plant has experienced additional control rod drive problems including dropped control rods on May 30, 2000, that forced a reactor shutdown from Operational Mode 2, "Startup." Based on the increased initiating event frequency related to the degraded performance of the control rod drive system in maintaining commanded rod position, the Significance Determination Process classifies this condition as one of very low safety significance. This violation of paragraph (a)(2) of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," is being treated as a non-cited violation. (Section 1R12.1)

Cornerstone: Mitigating Systems

- **Green.** When operators removed safety-related switchgear cooling systems from service, they failed to recognize that compensatory measures were required to ensure operability of the associated switchgear for certain design basis conditions, as specified in Section 11 of the Unit 2 Technical Requirements Manual. As a result, the licensee failed to take appropriate action as required by Unit 2 Technical Specifications 3.8.2.1 and 3.8.2.3, for an inoperable vital 480 volt load center and an inoperable train of vital DC switchgear respectively. This technical specification violation is being treated as a non-cited violation. The loss of switchgear cooling events were evaluated using the NRCs Significant Determination Process and, based on the short exposure time and the availability of the redundant train, the condition was found to be of very low safety significance. (Section 1R13.1)

- **Green.** Following surveillance testing and operation of the "B" emergency diesel generator (EDG) on July 5, 2000, 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. Because no actual loss of safety function occurred, the condition was evaluated through the Significance Determination Process as a condition of very low safety significance. This condition is identical to a previous violation associated with the failure to restore the automatic voltage regulator to its required position on July 7, 1999, but the licensee had not implemented corrective actions associated with that violation. This failure to implement timely corrective actions for a condition adverse to quality, as required by Criterion XVI, “Corrective Action,” of 10 CFR Part 50, Appendix B, is being treated as a non-cited violation. (Section 1R22.1)

Cross-Cutting Issues: Human Performance

- **No Color.** The NRC identified the following three examples where plant design changes were not translated into appropriate specifications and procedures due to inadequate performance of design change reviews: (1) following the implementation of a reactor protection system (RPS) wiring modifications, four technical specification (TS) surveillance procedures affected by the modification were not appropriately revised; (2) following replacement of the turbine-driven auxiliary feedwater pump (TDAFP) impeller, non-conservative technical specification and surveillance procedure acceptance criteria were not revised to be consistent with the resulting changes in pump performance; (3) following calculation of revised RPS trip setpoint and allowable values, a non-conservative technical specification allowable value was not revised. Because these conditions were administrative in nature and did not affect the operability of the systems, these design control violations were individually classified as violations of minor significance and were not subject to formal enforcement action. (Section 4OA4)

TABLE OF CONTENTS

Table of Contents (cont'd)

Report Details

SUMMARY OF UNIT 2 STATUS

The plant was at 100% power throughout the inspection period with the exception of a short time period on July 29, 2000, for performance of monthly turbine control valve testing.

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

1R04 Equipment Alignments

a. Inspection Scope

Inspectors performed the following partial system alignment checks:

Following the performance of surveillance tests affecting containment spray (CS) system valves, the inspector verified that valves associated with the "A" containment spray train were properly aligned using procedure SP 2606C, "CS System Alignment, Operability, and Operational Readiness Tests, Facility 1," and system piping and instrumentation diagram 25203-26015.

During on-going maintenance work on the "C" service water strainer, the inspector verified that the "A" train service water system was correctly aligned in accordance with SP 2612C-001, "Service Water, Facility 1," and system piping and instrumentation diagram 25203-26008.

b. Findings

There were no findings identified during this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspector performed a walkdown of the Unit 2 main control room to observe the overall material condition and operational status of fire detection and suppression equipment, and the licensee's control of transient combustible material and ignition sources within fire zone A25. In addition, the inspector evaluated the condition of fire barriers and penetration seals, and the operational status of the fixed suppression halon system in the Unit 1 control room that is credited in the Fire Protection Program to prevent the spread of fire in both control rooms.

b. Findings

There were no findings identified during this inspection.

1R07 Heat Sink Performance

a. Inspection Scope

The inspector verified that the licensee's processes and programs were adequate to ensure proper performance of the 'A' and 'B' emergency diesel generator (EDG) air coolers, jacket water coolers, and lube oil coolers, and the vital AC switchgear room coolers.

The inspector reviewed the following documents to verify that thermal performance tests of the "B" emergency diesel generator and a vital switchgear cooler were conducted in accordance with accepted industry practices and the acceptance criteria were consistent with design basis values:

Procedure EN 21228	Thermal Performance Test for the 'B' Emergency Diesel Heat Exchangers
Procedure EN 21234	Thermal Performance Test of Unit 2 Vital AC Switchgear Cooler X-183
Calculation 98-119	MP2 EDG Heat Exchanger Thermal Performance Test Analysis
Calculation 97-211	Analysis of X-183 Thermal Performance Test Results

The inspector verified that the test properly reflected design basis assumptions and that test instrument inaccuracies were considered.

The inspection/cleaning methods and frequencies for the EDG coolers and vital AC switchgear room coolers were reviewed with the system engineer to ensure that they were consistent with expected degradation. The inspector reviewed inspection/cleaning records associated with the following work orders from January 2000 to June 2000 to verify that the results were recorded, evaluated and dispositioned such that the final heat exchanger condition was acceptable:

M2-99-03185	A Emergency Diesel Semi-Annual & Quarterly PM
M2-99-09363	A Emergency Diesel Generator Assembly 18 Month Surveillance
M2-99-06591	B Emergency Diesel Semi-Annual & Quarterly PM
M2-99-05006	B Emergency Diesel Quarterly & Semi-Annual PM
M2-99-01320	B Emergency Diesel Generator Assembly 18 Month Surveillance
M2-99-09660	West 480V Unit Load Center Room Cooling Coil ECT/Hydrolasing PM
M2-99-09658	West 480V Unit Load Center Room Cooling Coil ECT/Hydrolasing PM
M2-99-12344	6.9 & 4.16 KV Switchgear Room Cooling Coil ECT/ Hydrolasing
M2-98-09606	6.9 & 4.16 KV Switchgear Room Cooling Coil ECT/ Hydrolasing

The following condition reports (CRs) related to the extent of bio-fouling, debris fouling, and chemical control were reviewed to verify that the licensee adequately identified and resolved problems:

M2-99-1828	Abnormally High Amount Fine Grass Marle Hard White Calcium Based Buildup on Service Water Outlet End Tubesheet
M2-99-2441	Quarterly Inspection of A RBCCW Hx (X18A), Found Piece of Rubber Gasket Material w/ Mussels Attached & Buildup of Marle On Inlet & Outlet
M2-99-2499	During Quarterly Inspection of B RBCCW Heat Exchanger X18B significant Buildup of Marle Calcium Carbonate Deposit was Present Closed to CR M2-99-2441
M2-99-2574	Quarterly Inspection of C RBCCW Heat Exchanger X18C Gasket Calcium Fouling Calcium Carbonate Buildup Investigation Being Done Per CR M2-99-2441
M2-99-3115	Significant Calcium Carbonate Found on Outlet End of X18A A RBCCW Hx 6 Mussels Found Attached to Tube Sheet
M2-99-3293	Quarterly Inspection of X18C
M2-00-0886	Mussels found in B RBCCW Heat Exchanger During Quarterly PM
M2-00-1116	Maintenance Must Plug Heat Exchanger Tube Due to 85% OD Wall Loss on Diesel Water Jacket Heat Exchanger Tube
M2-00-1073	Plastocor Coating Damage on “A” RBCCW (X18A) Water Outlet Flange

The inspector also walked down portions of the service water system (SWS) to assess material condition and to verify that the chemical treatment for corrosion control and biotic fouling for the SWS was properly established to ensure required heat exchanger performance was maintained.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On July 7, 2000, the inspector observed the conduct of a licensed operator requalification simulator examination. The inspector assessed licensed operator performance in areas such as: communications, implementation of normal and emergency procedures, command and control, technical specification compliance, and implementation of emergency plan actions. In addition, the inspector evaluated simulator fidelity compared with the actual control room, as well as the evaluator’s critique of the examination. The inspector also verified that the evaluator’s addressed operator performance issues that were identified during the test, and that examination objectives had been achieved.

b. Findings

There were no findings during this inspection.

1R12 Maintenance Rule Implementation

.1 (Closed) LER 50-336/2000-003-00: Control Rod Drive System Problems

a. Inspection Scope

The inspector reviewed maintenance rule implementation and corrective actions associated with longstanding repeated occurrences of dropped control rods against the licensee's maintenance rule implementing procedures. The inspector also reviewed Licensee Event Report 50-336/2000-003-00, "Manual Reactor Trip After Two Control Element Assemblies Unexpectedly Drop into the Core."

b. Findings

In 1999, control rod 65 experienced the following problems: failing to withdraw on command during testing in April, a blown fuse during startup activities in May, and an unrecoverable drop in September. The September event resulted in a forced shutdown, during which the licensee identified damaged insulation on leads supplying the upper gripper coil for control rod 65. The licensee implemented a temporary repair to sleeve the leads without a full view of the extent of the damage. Subsequently, control rod 65 slipped during startup activities in January 2000 and dropped completely into the core on February 11, 2000. Minutes after control rod 65 dropped, control rod 63 dropped for unrelated reasons. The control room operators manually tripped the reactor as required by procedure in response to the two control rods that were completely inserted into the core.

During the outage that followed the reactor trip, the licensee replaced the coil and associated leads for control rod 65. Vendor inspection of the removed coil revealed that the corrective maintenance performed in September 1999 was ineffective in that the installed sleeves did not cover areas of damaged insulation on the leads. Also during the outage, the licensee determined that control rod 63 dropped due to a failure in that control rod's power switching circuit. Subsequent control rod testing prior to restart revealed numerous control rods with problems related to the power switching circuits. As a result of this testing, the licensee replaced many components in the control rod drive power switching circuits, such as potentiometers, fuses and power reed relays. In LER 50-336/2000-003-00, the licensee documented that deficiencies in the preventive maintenance program for the control rod drive system contributed to the reactor trip on February 11, 2000.

The licensee had designated the control rod drive system as a maintenance rule (a)(2) system. For the manual withdrawal and insertion function of the control rod drive system, which included dropped or slipped control rods, the licensee used plant level criteria to demonstrate that performance of the system was being effectively controlled by preventive maintenance. The plant level performance criteria in Engineering Department Instruction (EDI) 30720, "Maintenance Rule Plant Level Monitoring," were established at less than 3 unplanned scrams per 24 month rolling period and less than 6 percent unplanned capability loss factor per 24 month rolling period. The EDI specified that systems that contributed significantly to exceeding the criteria were to be evaluated for goal setting under paragraph (a)(1) of the maintenance rule, 10 CFR 50.65.

The licensee documented that the plant level criteria for unplanned scrams was exceeded by the reactor trip on February 11, 2000, in Condition Report (CR) M2-00-0391. In addition, the licensee determined that 13.1 percent of potential generation capability between May 9, 1999, (i.e., the date the plant restarted from an extended shutdown of over three years in length) and February 29, 2000, was lost due to unplanned events, and over half of that lost generation resulted from control rod drive problems. Based on exceeding the plant level criteria for reactor scrams, the licensee's maintenance rule expert panel evaluated the control rod drive system for goal setting on June 28, 2000. Because the licensee considered the problems with control rod 65 to have resulted from a manufacturing defect, the licensee did not establish system-level performance goals for the control rod drive system and maintained the classification of the system as (a)(2).

The inspector concluded that the February 11, 2000, reactor trip was maintenance preventable in that: (1) the licensee had identified the cause of repeated problems with control rod 65 in September 1999 and, therefore, could have fully corrected the problem at the time by performing effective maintenance; and (2) the cause of the second dropped control rod was related to an ineffective preventive maintenance program. The additional dropped rods experienced during restart from refueling on May 30, 2000, caused by inadequate voltage supplied to the power switching circuits, further indicated that the licensee had not developed effective preventive maintenance measures to ensure the control rods would perform their intended function by that time.

The inspector evaluated this condition using the Significance Determination Process. Based on the increased initiating event frequency related to the degraded performance of the control rod drive system in maintaining commanded rod position, the Significance Determination Process classifies this condition as one of very low safety significance (Green).

Paragraph (a)(1) of 10 CFR 50.65 requires, in part, that the performance or condition of systems shall be monitored against established goals, to provide reasonable assurance that the systems are capable of performing their intended functions.

Paragraph (a)(2) of 10 CFR 50.65 requires, in part, that monitoring as specified in paragraph (a)(1) is not required where it has been demonstrated that the performance or condition of a system is being effectively controlled through the performance of appropriate preventive maintenance such that the system remains capable of performing its intended function.

Contrary to the above, as of June 28, 2000, the licensee had not demonstrated that the performance of the control rod drive system was being effectively controlled through the performance of appropriate preventive maintenance such that the system remained capable of performing its intended function, and the licensee had not implemented monitoring of the system against licensee-established goals as required by paragraph (a)(1) of 10 CFR 50.65. This violation of paragraph (a)(2) of 10 CFR 50.65 is being treated as a Non-Cited Violation **(NCV 05000336/2000-009-01)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). This violation is in the licensee's corrective action program as CR M2-00-0391. In response to this NRC identified condition, the licensee has modified a corrective action assignment associated with CR M2-00-0391 to reevaluate the control rod drive system for system-level performance monitoring under paragraph (a)(1) of 10 CFR 50.65 at the next expert panel meeting.

.2 Charging Pump Functional Failures

a. Inspection Scope

The inspectors reviewed maintenance rule implementation associated with the following condition reports:

M2-00-1699 "B" Charging Pump Cycled On and Off Rapidly After Reactor Trip
M2-00-1840 "B" Charging Pump Discharge Relief Valve Failed Open
M2-00-1933 "C" Charging Pump Low Lube Oil Pressure

The inspector verified that the conditions were correctly classified with respect to maintenance preventable functional failures based on Engineering Department Instruction 30710, "Maintenance Rule Functional Failures."

The inspector also reviewed the Maintenance Rule Action Plan for the Chemical and Volume Control System, Revision 2, dated July 31, 2000. The inspector verified that the performance criteria for the boron injection function of less than 3 functional failures per 24 months for the system and less than 2 functional failures per 24 months per train were consistent with the failure-to-run frequency and the failure-to-start probability used in the licensee's risk assessment model for the charging pumps. Since, in addition to exceeding the unavailability performance criterion, the functional failure performance criteria had also been exceeded for the boron injection function, the inspector verified that appropriate goals were documented in the action plan to monitor the effectiveness of corrective actions.

b. Findings

There were no findings identified.

.3 Service Water Strainer Shaft Key Failures

a. Inspection Scope

The inspector reviewed the maintenance rule implementation associated with failures of the "A" service water strainer shaft key due to driftwood jamming the strainer on two occasions in October 1999 and July 2000. The inspector verified that the licensee's classification of these events as degraded conditions rather than functional failures was consistent with Engineering Department Instruction 30710, "Maintenance Rule Functional Failures," in that the licensee placed the spare "B" service water pump and strainer in operation in the "A" train prior to the "A" stainer differential pressure exceeding its analyzed differential pressure for full service water flow.

b. Findings

There were no findings identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

.1 (Closed) LER 50-336/2000-011-00: Inoperable Vital Switchgear Cooling

a. Inspection Scope

The inspector reviewed work controls implemented for the performance of design work on vital switchgear enclosure B61 cooler unit AC-4. In addition, the inspector reviewed the events related to inoperable switchgear cooling and associated technical specification issues detailed in LER 50-336/2000-011-00, “Technical Specification Action Statements Exceeded for Electrical Busses When Vital Switchgear Ventilation Equipment Taken Out of Service.”

b. Findings

The licensee removed the load center B61 enclosure cooler AC-4 from service for wiring modifications and temperature switch adjustments without implementing compensatory cooling measures or entering associated technical specification (TS) action statements, as required by a recent revision to Section 11 of the Technical Requirements Manual (TRM). The compensatory measures were required to ensure operability of the vital load center B61 for certain design basis conditions. However, the load center enclosure did not actually exceed its design temperature throughout the approximately 87 hours AC-4 was out of service. In the event conditions occurred that would cause the enclosure temperature to increase, the load center would not have become inoperable for several hours.

The inspector reviewed condition report (CR) M2-00-2009, the automated work order for the wiring modifications M2-00-09778, the subsequent root cause investigation report, and Licensee Event Report (LER) 50-336/2000-011-00, “Technical Specification Action Statements Exceeded for Electrical Busses When Vital Switchgear Ventilation Equipment Taken Out of Service.” The licensee’s investigation identified that the root cause was a lack of organizational awareness of vital switchgear room operability requirements, and the inspector found that corrective actions identified by the licensee were appropriate.

A Region I Senior Reactor Analyst (SRA) evaluated the short term loss of cooling to the B61 enclosure under the NRC’s Significance Determination Process (SDP). The evaluation consisted of a bounding calculation with the B61 switchgear and associated components it supplies considered inoperable without operator recovery for an exposure time of 3.5 days utilizing the NRCs Revision 3 standardized plant analysis risk (SPAR) model for Millstone Unit 2, and concluded that the increase in core damage frequency (CDF) was 4.90E-7 per reactor year. In addition, the licensee utilized their Equipment Out Of Service (EOOS) model that resulted in a comparable CDF increase of 1.65E-7 per reactor year. Based on the preceding risk factors, the NRC concluded that the condition was of very low safety significance (Green).

A second event similar to the B61 event was also reported via LER 2000-011. The licensee identified that vital chiller X169B, which supports the train “B” 125-volt DC switchgear room cooling function, was taken out of service for maintenance on July 3, 2000, and that the required compensatory measures had not been implemented during the 8 hour period the chiller was out of service. Because the “B” DC switchgear room cooling fan is powered from B61 and was assumed to have been inoperable for the analysis of the time period the B61 enclosure cooler was out of service, the safety significance of the time period vital chiller X169B was out of service would be bounded by the B61 enclosure cooler case, and therefore be of very low safety significance.

Unit 2 TS 3.8.2.1, states in part, that with A.C. Bus 22F inoperable, restore the inoperable bus and/or associated load center to OPERABLE status within 8 hours or be in COLD SHUTDOWN within the next 36 hours. TS 3.8.2.3 states in part, that with one 125-volt D.C. bus inoperable, restore the inoperable bus to OPERABLE status within 2 hours or be in COLD SHUTDOWN within the next 36 hours. Therefore, the licensee's failure to identify the inoperable switchgear cooling systems and the effect on their respective 480-volt AC and125-volt DC busses, and subsequent failure to comply with the appropriate TS action statements is a violation of technical specifications. This violation is being treated as a Non-Cited Violation **(NCV 05000336/2000-009-02)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), in that the associated condition was of very low safety significance and was entered in the licensee's corrective action program.

.2 Testing of the Spare High Pressure Safety Injection Pump

a. Inspection Scope

The inspector reviewed work controls implemented to manage risk when surveillance testing of the "B" high pressure safety injection (HPSI) pump, which can be aligned to either the "A" train or the "B" train, was scheduled to occur during testing of the "A" emergency diesel generator (EDG). The inspector verified that the licensee managed risk to an acceptable level in accordance with their on-line maintenance procedures by ensuring the "B" HPSI pump was aligned to the "A" train during "A" EDG testing.

b. Findings

There were no findings identified.

.3 Concurrent Work on the “C” Service and Closed Cooling Water Pumps

a. Inspection Scope

The inspector reviewed work controls implemented to manage risk when the “C” service water pump and the “C” reactor building closed cooling water pump were removed from service concurrently for preventive maintenance on August 7, 2000. The inspector verified that the licensee managed risk to an acceptable level in accordance with their on-line maintenance procedures.

b. Findings

There were no findings identified.

1R15 Operability Evaluations

.1 Inadequate Bearing Oil Identified In the “C” High Pressure Safety Injection Pump

a. Inspection Scope

The inspector evaluated the adequacy of the licensee’s basis for future operability of the “C” high pressure safety injection (HPSI) pump following identification of inadequate oil in the outboard bearing housing during a routine operability surveillance test. In addition, the inspector evaluated the licensee’s extent of condition investigation following the discovery that a potential contributing cause was an incorrect adjustment of a bearing housing oil bubbler assembly, a bubbler that exists in several other safety related and non-safety related pumps throughout the plant. The inspector verified that the licensee had an adequate basis for continued operability of the affected pumps in that:

(1) pump vibration measurements and bearing oil samples indicate acceptable condition of the “C” HPSI pump bearings.

(2) inspection of similarly configured pumps verified that adequate oil was being supplied to the bearings.

b. Findings

There were no findings during this inspection.

.2 Thermal Margin/Low Pressure Reactor Trip

a. Inspection Scope

The inspector reviewed Operability Determination MP2-027-00, which addressed continued operability of the reactor protection system thermal margin/low pressure trip although the setpoint for the “A” channel has drifted in the non-conservative direction several times. The inspector verified that the licensee had an adequate basis for the continued operability of the thermal margin/low pressure trip in that:

(1) The licensee has identified the cause as increased resistance across contacts in the flow-dependent setpoint selector switch, which can be readily corrected by cycling the switch.

(2) The condition is infrequent such that three unaffected channels remain operable and provide adequate redundancy to complete the reactor trip function.

(3) The licensee has implemented a temporary plant process computer alarm to provide early indication of non-conservative changes in the setpoint.

b. Findings

There were no findings identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed the licensee's lists of operator work-arounds, control room panel deficiencies, tagouts greater than 90 days in age, and alternate plant configurations. The inspector discussed transient operations with several licensed operators and reviewed recent condition reports involving plant transient response to identify other conditions that could affect the operator's ability to effectively respond to transients. The inspector also evaluated the cumulative effects of the identified conditions on the ability of operators to respond to transients.

b. Findings

There were no findings identified.

1R19 Post Maintenance Testing

.1 Service Water System Testing

a. Inspection Scope

The inspector reviewed the post-maintenance and surveillance tests associated with automated work order M2-99-08124, "'C' Service Water Pump Discharge Strainer Assembly," to verify that the applicable tests adequately demonstrated operability of the service water system.

b. Findings

There were no findings identified during this inspection.

.2 "B" Auxiliary Feedwater Pump Testing

a. Inspection Scope

The inspector reviewed the post-maintenance and surveillance tests associated with automated work order M2-99-08679, "'B' Auxiliary Feedwater Pump Motor," to verify that the applicable tests adequately demonstrated operability of the "B" Auxiliary Feedwater Pump.

b. Findings

There were no findings identified during this inspection.

.3 "C" Reactor Building Closed Cooling Water Pump Testing

a. Inspection Scope

The inspector verified that post-maintenance testing associated with automated work order M2-00-12912, "'C' Reactor Building Closed Cooling Water Pump Mechanical Preventive Maintenance," in combination with a surveillance test performed by operators adequately demonstrated operability of the pump.

b. Findings

There were no findings identified.

1R22 Surveillance Testing

.1 Improper Setting of "B" Diesel Generator Voltage Regulator

a. Inspection Scope

The inspector reviewed an event where the "B" emergency diesel generator output voltage was low when it was started for surveillance testing on August 2, 2000.

b. Findings

Following surveillance testing and operation of the "B" emergency diesel generator (EDG) on July 5, 2000, 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. Because no actual loss of safety function occurred, the condition was evaluated through the Significance Determination Process as a condition of very low safety significance (Green).

After a slow start of the "B" emergency diesel generator (EDG) on August 2, 2000, 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 approximately 4050 volts on the control panel gage. The surveillance procedure specified a minimum voltage of 4100 volts. Technical Specification surveillance requirement 4.8.1.1.2.a.2 specifies that the “B” EDG output voltage reach 97 percent of its design voltage of 4160 volts (approximately 4036 volts). Because the voltage was high enough to pick-up the “ready to load” relay, which was at the Technical Specification voltage setpoint, and that relay permitted automatic loading of the EDG, the “B” EDG remained operable. The licensee document the low voltage condition in condition report M2-00-2195.

Step 4.1.72 of surveillance procedure SP 2613L specifies that the operator readjust the automatic voltage regulator to maintain 4160 volts after separating from the grid but prior to securing the “B” EDG. However, plant process computer data from July 5, 2000, indicates that the “B” EDG output voltage was not readjusted during a surveillance test on that date following separation from the grid, which was at about 4050 volts. The failure to readjust voltage to its normal value after operation connected to the grid is a concern because, particularly on days of heavy electrical demand, the grid voltage could be low enough that the EDG voltage during a subsequent automatic start would be too low to allow for automatic loading of the EDG.

This failure to readjust the voltage prior to shutdown is identical to a previous violation that occurred on July 7, 1999, which was documented in NRC Inspection Report 50-336/99-12 as NCV 50-336/99-12-02. However, the licensee had not implemented corrective actions from that previous violation, which were associated with condition reports M2-99-2179 and M2-99-2380. The failure to implement timely corrective actions is a violation of Criterion XVI, “Corrective Action,” of 10 CFR Part 50, Appendix B. This violation is being treated as a Non-Cited Violation **(NCV 05000336/2000-009-03)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368), in that the associated condition was of very low safety significance and was entered in the licensee’s corrective action program.

.2 Turbine-Driven Auxiliary Feedwater Pump Testing

a. Inspection Scope

On June 28, 2000, the inspector observed the preparation for, and the conduct of, the turbine driven auxiliary feedwater pump (TDAFP) operability test performed in accordance with SP 2610B, “TDAFP Tests.” The inspection activities included a review of test results to ensure licensee compliance with applicable acceptance criteria, technical specifications and equipment design bases, and verification that operators performed actions in accordance with applicable procedures. In addition, the inspector verified that equipment operability was adequately determined. The inspector also observed the pre-job brief to verify that operators were cognizant of test conditions and the overall impact on the plant.

b. Findings

There were no findings identified during this inspection.

.3 Reactor Protection System Bistable Testing

a. Inspection Scope

On June 29, 2000, the inspector observed the performance of surveillance procedure SP 2401GA, “RPS Channel “A” Bistable Trip Test.” The inspector reviewed test results to verify compliance with acceptance criteria and associated technical specifications, and verified the bases of various acceptance criteria and associated setpoints.

b. Findings

There were no findings identified during this inspection.

.4 Reactor Protection System Logic Testing

a. Inspection Scope

The inspector observed the performance of the AB logic matrix functional test portion of surveillance procedure SP 2401D, “RPS Matrix Logic and Trip Path Relay Test.” The inspector verified that the surveillance test was performed in the correct sequence specified in the surveillance procedure.

The Inspector reviewed procedure SP 2401D and the completed data sheet for the surveillance performed on July 28, 2000. The inspector verified that the test data was complete and satisfied the specified acceptance criteria. The inspector also verified that the specified test frequency and acceptance criteria were adequate to demonstrate operability of the reactor protection system logic and reactor trip breakers, as required by Technical Specification 4.3.1.1.1.

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

Emergency AC Power System Unavailability

a. Inspection Scope

The inspector reviewed shift manager logs, system engineer unavailability data, and various corrective action program records (condition reports) to determine the accuracy and completeness of the performance indicator (PI) data. Specifically the inspector verified the licensee’s PI data relative to the reported values submitted to the NRC. The inspector also verified the PI data collection methodology was consistent with both licensee and industry guidance.

b. Findings

There were no findings identified during this inspection.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

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

b. Findings

The inspector found that the licensee did not implement timely corrective actions in that, one year after an operator failed to restore an emergency diesel generator automatic voltage regulator to its correct position after operational testing in parallel with the grid, the identical condition recurred and the licensee had not yet implemented corrective actions from the first occurrence (Section 1R22.1).

4OA4 Cross-cutting Issues

.1 Human Performance Issues And Design Control

a. Inspection Scope

The inspector reviewed human performance issues related to the implementation of the licensee's design control process.

b. Findings

The NRC identified the following three examples where plant design changes were not translated into appropriate specifications and procedures due to inadequate performance of design change reviews: (1) following the implementation of a reactor protection system (RPS) wiring modifications, four technical specification (TS) surveillance procedures affected by the modification were not appropriately revised; (2) following replacement of the turbine-driven auxiliary feedwater pump (TDAFP) impeller, non-conservative technical specification and surveillance procedure acceptance criteria were not revised to be consistent with the resulting changes in pump performance; (3) following calculation of revised RPS trip setpoint and allowable values, a non-conservative technical specification allowable value was not revised.

Prior to the performance of RPS bistable testing (Section 1R22.3), the licensee identified that a recently performed plant modification (Design Change DM2-03-0564-99) that rerouted various RPS cables was implemented without appropriate revisions to four TS surveillance procedures. With the reactor at power and reactor coolant pumps running, the "Low Flow Bistable Trip Test" section of the surveillance test surveillance test could not have been completed as written. The necessary procedure revisions would have addressed the removal and relocation of certain terminal board connections that are utilized for successful completion of the surveillance test. The NRC has determined that this failure to revise affected surveillance procedures during the design control process is a violation of minor significance, and as such, is not subject to formal enforcement action. The licensee has entered this issue into the corrective action program as condition report (CR) M2-00-1880.

During review of acceptance criteria associated with the TDAFP operability test (Section 1R22.2), the inspector identified that an impeller replacement (Design Change DM2-00-0898-99) conducted in May 2000 changed pump performance such that it was inconsistent with the basis of the acceptance value specified in TS 4.7.1.2.a.2.b. Specifically, the licensee had changed the TS and surveillance procedure (SP) differential pressure acceptance criterion based on a specific pattern of pump degradation that only affected performance at low flow. After impeller replacement, pump performance returned to original specifications. Uniform degradation at all flow ranges from the original specified performance would allow the pump to meet the TS acceptance criterion without being able to develop required differential pressure at higher flow rates. The NRC has determined that this failure to revise acceptance criteria based on changes to equipment performance is a violation of minor significance, and as such, is not subject to formal enforcement action. The licensee has entered this issue into their corrective action program as CR M2-00-2185.

Design basis calculation, 92-030-1202E2, Revision 2, which became effective in February 1999, was performed to determine bounding values for the RPS low flow trip setpoint, allowable value, and as-found and as-left acceptance criteria. The inspector identified that the licensee’s implementation of this design basis calculation had failed to address the applicable TS Trip Setpoints and Allowable Values identified in TS Table 2.2-1, “Reactor Protection Instrumentation Trip Setpoint Limits.” The current RPS low flow trip allowable value specified in TS Table 2.2-1 is 90.9%, which is less conservative than the design basis calculation value of 91.1%. The NRC has determined that this failure to translate changes in calculated values into technical specifications is a violation of minor significance, and as such, is not subject to formal enforcement action. The licensee has entered this issue into their corrective action program as CR M2-00-2270.

.2 Human Performance Problems

a. Inspection Scope

The inspector reviewed human performance related to operator control of equipment configuration.

b. Findings

The inspector found that operators failed to restore an emergency diesel generator voltage regulator to its correct position following operational testing, as specified in the associated surveillance procedure (Section 1R22.1). Operators also failed to implement necessary compensatory measures to ensure operability of vital electrical switchgear, as specified by the Unit 2 Technical Requirements Manual, when the associated safety-related cooling systems were removed from service (Section 1R13.1).

4OA5 Other

.1 (Closed) LER 50-336/2000-004-00: On February 14, 2000, the unit failed to assure that channel checks for the wide range logarithmic neutron flux monitors were conducted every 12 hours while the unit was in mode 5. Because of a personnel error, the required surveillance periodicity for the monitors was exceeded by approximately three hours. The surveillance was adequately performed on February 15, 2000. Corrective measures were addressed in condition report (CR) M2-00-0398, which included general operator training upgrades and specific remediation measures. Because of successful surveillance results prior to and after this event and the short duration of the untested condition, the failure to perform the TS required surveillance, within the required periodicity, was considered a violation of minor significance and not subject to formal enforcement action.

.2 (Closed) LER 50-336/2000-005-00: On February 16, 2000, the licensee failed to ensure containment integrity was maintained by verifying the position of certain locked or sealed valves located inside containment prior to the transition from Mode 5, “Cold Shutdown,” to Mode 4, “Hot Shutdown. Technical Specification (TS) Surveillance Requirement (SR) 4.6.1.1.a excludes the routine verification of certain locked closed valves located inside the containment while in modes 1 through 4. However, the TS SR requires that these previously excluded valves be verified closed prior to entering Mode 4 from Mode 5, if they had not been verified closed within the previous ninety-two days. The surveillance was adequately performed on February 22, 2000, after the unit reentered Mode 5. Corrective measures were addressed in condition report (CR) M2-00-0465, which included revisions to procedure OP 2201, “Plant Heatup.” Because of successful surveillance results after this event and the short duration of the untested condition, the failure to perform the TS required surveillance prior to entering Mode 4 on February 16, 2000, was considered a violation of minor significance and not subject to formal enforcement action.

.3 (Closed) LER 50-336/2000-006-00: On March 7, 2000, the unit identified a historical failure to test four out of twenty-six installed remote fire detector panel supervisory circuits, in accordance with technical specification (TS) 4.3.3.7.2, between March 1978 and November 1995. Although the affected supervisory circuits were not appropriately tested from November 1995 to March 2000, the requirement to test the circuits was removed from the TS in November 1995, by license amendment 191. Each of the affected panels was adequately tested in March 2000. The licensee determined that fire protection defense in depth was maintained for the affected zones during the historical period and that there was no safety significance associated with the missed surveillances. Corrective measures were addressed in condition report (CR) M2-00-0583, which upgraded surveillance test procedures. Because of successful surveillance results and the maintenance of fire protection program defense in depth, the failure to perform the required TS surveillance was considered a violation of minor significance and not subject to formal enforcement action.

4OA6 Meetings, including Exit

.1 Resident Inspector Exit Meeting

The inspectors presented the inspection results to the Vice President of Nuclear Operations and the Vice President of Nuclear Technical Services and other members of licensee management at the conclusion of the inspection. The licensee acknowledged the findings presented.

ITEMS OPENED AND CLOSED

Opened and Closed During this Inspection

NCV 05000336/2000-009-01	licensee failed to initiate performance monitoring of the control rod drive system against established goals (1R12.1)
NCV 05000336/2000-009-02	licensee's failure to identify the inoperable switchgear cooling systems and the effect on their respective 480-volt AC and125-volt DC busses, and subsequent failure to enter and comply with the appropriate TS action statements (1R13.1)
NCV 05000336/2000-009-03	failure to implement timely corrective actions (1R22.1)

Previous Items Closed

50-336/2000-003-00	LER	Manual Reactor Trip After Two Control Element Assemblies Unexpectedly Drop into the Core (1R12.1)
50-336/2000-004-00	LER	Failure to Perform Channel Check for Wide Range Logarithmic Neutron Flux Monitor (4OA5)
50-336/2000-005-00	LER	Failure to Assure Containment Integrity for Certain Locked or Sealed Valves Located Inside Containment Prior to Entering Mode 4 from Cold Shutdown (4OA5)
50-336/2000-006-00	LER	Historical Condition: Remote Fire Detector Panel Supervisory Circuits not Tested in Accordance with Technical Specification Requirements (4OA5)
50-336/2000-011-00	LER	Technical Specification Action Statements Exceeded for Electrical Busses When Vital Switchgear Ventilation Equipment Taken Out of Service (1R13.2)

LIST OF ACRONYMS USED

CDF	core damage frequency
CR	condition report
CS	containment spray
ECT	eddy current testing
EDG	emergency diesel generator
EDI	Engineering Department Instruction
EOOS	equipment out of service
HPSI	high pressure safety injection
LER	licensee event report
PI	performance indicator
RBCCW	reactor building closed cooling water
RCP	reactor coolant pump
RPS	reactor protection system
SDP	significance determination process
SPAR	standardized plant analysis risk
SR	surveillance requirements
SRA	senior reactor analyst
SWS	service water system
TDAFP	turbine driven auxiliary feedwater pump
TRM	Technical Requirements Manual
TS	technical specification

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

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

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

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

Reactor Safety

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

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

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

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

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

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	05000423
License No.:	NPF-49
Report No.:	05000423/2000-009
Licensee:	Northeast Nuclear Energy Company
Facility:	Millstone Nuclear Power Station, Unit 3
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	July 2, 2000 - August 12, 2000
Inspectors:	A. C. Cerne, Senior Resident Inspector, Unit 3 B. E. Sienel, Resident Inspector, Unit 3
Approved by:	James C. Linville, Chief Projects Branch 6 Division of Reactor Projects Region I

SUMMARY OF FINDINGS

IR 05000423/2000-009; on 07/02-08/12/00; Millstone Nuclear Power Station; Unit 3.

The inspection was conducted by resident inspectors. There were no findings identified during this inspection.

TABLE OF CONTENTS

Report Details

SUMMARY OF UNIT 3 STATUS

The plant operated at approximately 100 percent power throughout the inspection period.

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

1R04 Equipment Alignments

a. Inspection Scope

During planned maintenance on the “B” train of the motor driven auxiliary feedwater system, the inspector verified the correct alignment of the “A” train equipment. In addition, the inspector verified the “B” train was restored properly following maintenance. The inspector performed the partial walkdowns by comparing actual equipment alignment to approved licensee piping and instrumentation diagrams to confirm correct system lineup.

Subsequent to the restoration of the “B” train of motor driven auxiliary feedwater equipment (MDAFW), the inspector conducted a complete walkdown of the turbine driven auxiliary feedwater (TDAFW) system outside containment, including the main steam supply system to the terry-turbine, the demineralized and condensate water suction supply valves to the TDAFW pump, and the discharge piping and in-line components to the point where the MDAFW and TDAFW lines merge to penetrate containment. This system was selected for the first complete system walkdown, in accordance with the new reactor oversight program, based upon its highest ranking as a contributor to core damage frequency with a single component out of service.

In addition to verifying the correct alignment of major TDAFW components, the inspector confirmed that the identified equipment tags and trouble reports did not represent conditions that could adversely affect system functionality. The inspector also evaluated overall equipment conditions, including the pipe support layout and the potential for unacceptable non-safety component impact, based upon the criteria specified in the licensee’s seismic interaction program. Selected pipe support installation details and load calculations were spot-checked.

b. Findings

There were no findings identified and documented during this inspection of equipment alignment.

1R05 Fire Protection

a. Inspection Scope

The inspector performed walkdowns of the “A” motor driven auxiliary feedwater (MDAFW) pump room (Fire Area ESF-7) and the “B” MDAFW pump room and area housing engineered safety features (ESF) building air conditioning equipment (collectively Fire Area ESF-9). The inspector confirmed that fire detection and suppression equipment located in the area was as specified in the Millstone 3 Fire Protection Evaluation Report. The inspector also verified that appropriate compensatory measures (i.e. hourly fire roves) were implemented, in accordance with the Unit 3 Technical Requirements Manual, where degraded or out of service equipment was identified.

b. Findings

There were no findings identified and documented during this inspection.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope

During a review of the licensee’s online risk profile for planned maintenance on the “B” train of MDAFW and associated ventilation, the inspector noted the online risk was yellow. Observing that the risk profile changed within the yellow band during the planned work duration, the inspector discussed the bases for these changes with the online risk evaluator to confirm the evaluation was well founded.

During this inspection period, the remote position indication was lost for containment isolation valve, 3SSR*CTV19D, in the steam generator blowdown sample system. Operations personnel declared the valve inoperable and isolated the line in accordance with technical specification (TS) 3.6.3.c by closing an upstream manual valve. Subsequently, the other three blowdown sample lines were also isolated to control the observed leakage, alternate steam generator blowdown sampling provisions were implemented, and work activities were initiated to plan and stage for the replacement of the CTV19D valve.

Based upon not only the TS containment isolation and leakage requirements for continued plant operation, but also the TS provision for radioactive monitoring of the steam generator blowdown effluent pathways, the inspector evaluated the controls in place and the planned maintenance work to replace the CTV19D valve. Condition reports were generated by the licensee to adequately document the temporary valve alignments and observed work control issues. The inspector also examined the field conditions of the isolated valves, checked tagging controls and discussed planned work activities with the responsible maintenance supervisor.

Subsequent to the replacement of 3SSR*CTV19D, and continuing to the end of this inspection period, the inspector confirmed further efforts by the licensee to assess additional steam generator blowdown sample line problems, continue the required effluent monitoring, and maintain the affected containment penetrations in the isolated condition required by the plant TS.

b. Findings

There were no findings identified and documented during this inspection.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 TS 3.0.3 Entry Due to Both Trains of Containment Hydrogen Recombiners Inoperable

a. Inspection Scope

The inspector reviewed operator actions taken in response to both trains of containment hydrogen recombiners declared inoperable due to a radiation monitor failure on the “B” train while the “A” train was inoperable for planned maintenance (licensee CR M3-00-1860). With both trains of hydrogen recombiners inoperable and no technical specification action statement covering this condition, operators entered TS 3.0.3 which requires a plant shutdown within the following six hours if the equipment cannot be returned to operable in an hour. The inspector discussed the plant condition with the assistant operations manager that evening, and reviewed operator logs and technical specification (TS) 3.6.4.2 and 3.0.3 requirements the following day. Licensee personnel were able to restore the “B” recombiner to service without operations personnel reducing reactor power.

b. Findings

No findings were identified.

.2 Chilled Water Chiller Trip

a. Inspection Scope

During routine operations the “A” chilled water (CDS) chiller tripped on high temperature. The inspector noted that control room logs and CR M3-00-1780 documented the use of shift manager dispensation to deviate from the applicable operating procedure.

Normally two of the three chillers are in service, one per train. These chillers supply cooled water to many components, including the containment air recirculation coolers inside containment, which in turn cool containment air and assist in pressure regulation. Reactor plant closed cooling water (RPCCW) is used to cool the CDS chillers. Typically “A” train RPCCW is aligned to cool “A” train CDS and likewise with the “B” train.

In order to realign the CDS system in accordance with operating procedure OP 3330C, Reactor Plant Chilled Water System, operators need to secure the remaining chiller before realigning the RPCCW system and CDS system to another train and starting two CDS chillers. In order to prevent this, and the resultant increase in containment pressure, operators used guidance provided in procedure DC 4, Procedural Compliance, to realign RPCCW without securing the operating CDS chiller.

The inspector discussed the event with the assistant operations manager, reviewed operator logs, piping and instrumentation diagrams, CR M3-00-1780, and procedures OP 3330C and DC 4 to confirm that the operators' actions were acceptable. In addition, the inspector noted that a procedure change had been submitted prior to the event to allow the identified equipment alignment and was implemented following the chiller trip this inspection period.

b. Findings

There were no findings identified during this inspection of personnel performance.

1R15 Operability Evaluations

a. Inspection Scope

The following operability determinations were reviewed. The inspector verified that the engineering justification for operability was sound, any compensatory actions required were in place, and all applicable technical specifications and technical requirements manual actions were met.

- MP3-003-99 Toxic Chemical Analysis for Control Room and Auxiliary Shutdown Panel Habitability Need Analysis
- MP3-084-98 Hydrogen and Nitrogen Gas Accumulation in Boric Acid Gravity Feed Lines
- MP3-024-98 Dose Calculations Crediting the Supplementary Leak Collection and Release System May Not Have Adequately Addressed Bypass Leakage (Revision No. 1)
- MP3-031-99 Leakby of 3RSS*MOV20B during Recirculation Spray System Testing

The inspector also spot-checked relevant system operating and surveillance procedures and the latest surveillance test results to confirm consistency with the data and analyses documented in the applicable ODs. For MP3-024-98 (Rev. 1), the inspector verified OD revision (reference: proposed license amendment request, PLAR 3-98-5) and for MP3-031-99, the inspector confirmed adequate OD closure, in accordance with the controls delineated in station procedure, RP 5 (Rev. 2) for Operability Determinations.

In assessing the supporting documentation for operability determination (OD) MP3-084-98, the inspector reviewed technical evaluation M3-EV-98-0126, Gas Accumulation in Gravity Feed Boration Piping; special procedure (SPROC) EN 98-3-17, Monitoring of Gas Accumulation in the MP3 Gravity Feed Boration Lines; and Calculation 98-ENG-01598M3, Allowable Gas Volume in Gravity Boration Lines.

b. Findings

No findings were identified during this inspection of operability evaluations.

1R16 Operator Work-Arounds

a. Inspection Scope

During a review of operability determination (OD) MP3-084-98, documented above, the inspector observed that the compensatory actions for the OD constituted an operator work-around. The inspector confirmed that the actions were identified as Operator Work-Around 98-016 and an action plan is in place to eliminate the problem. The inspector also evaluated the operators' ability to implement abnormal and emergency operating procedures with this condition.

b. Findings

There were no findings identified during this inspection.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspector reviewed the completed documentation for post maintenance testing (PMT) performed on the "A" emergency diesel generator (EDG) starting air shutdown solenoid, worked under automated work order (AWO) M3-99-12521, and on one of the two "A" EDG starting air header solenoid valves, worked under AWO M3-99-12514. The inspector reviewed the scope of the work activities and verified that the PMTs planned were appropriate to restore the operability of the affected components. Following the completion of system maintenance, the inspector discussed the maintenance with the system engineer and reviewed completed work orders and tests to verify acceptable system restoration.

Following the completion of maintenance activities and PMT of the "B" train quench spray pump, 3QSS*P3B, the inspector conducted a field walkdown of the affected QSS components, verifying system restoration from the testing configuration to the operable engineered safety features lineup. Pump PMT flow data were verified to be consistent with the in-service testing (IST) requirements for the differential head pressure at the reference test flow. The inspector also discussed with the system engineer the impact of a locked-throttled drain valve and a leaking recirculation-test valve on the margin available in the surveillance test criteria to establish adequate QSS forward flow following a post-accident, containment depressurization actuation signal.

b. Findings

No findings were identified during this inspection of post maintenance testing.

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed licensee performance related to the following surveillance tests. The emergency diesel generator, recirculation spray, and motor driven auxiliary feedwater systems all represent significant contributors to the prevention of core damage in design-basis accident scenarios. The main steam isolation valve (MSIV) testing is considered by the licensee to be a high risk to generation as full closure of one of the valves during testing would cause a reactor trip.

- SP 3448E31 Train A - Diesel Sequencer Actuation Logic Test
- SP 3606.2 Containment Recirculation Pump 3RSS*P1B Operational Readiness Test
- SP 3622.2 Auxiliary Feed Pump 3FWA*P1B Operational Readiness Test
- SP 3712AA Main Steam Isolation Valve Partial Stroke Test

The emergency diesel and MSIV testing was observed in the control room to confirm performance of the tests in accordance with approved procedures. The completed data sheets were reviewed for all tests to verify the equipment met procedural acceptance criteria and was operable consistent with technical specification requirements.

The surveillance test data for pump 3RSS*P1B was evaluated using new criteria for the acceptable IST range for both pump flow and differential pressure. This change was required to accommodate the digital volt readings taken from a differential pressure transmitter utilized in place of the controlotron previously used for surveillance flow measurements. The inspector reviewed the pump test data evaluation form that revised the IST reference values, thus establishing the new acceptance criteria for the pump test measurements.

b. Findings

No findings were identified during this inspection of surveillances.

4OA6 Meetings, including Exit

.1 Resident Inspector Exit Meeting

The inspectors presented the inspection results to the Vice President of Nuclear Operations and the Vice President of Nuclear Technical Services and other members of licensee management at the conclusion of the inspection. The licensee acknowledged the findings presented.

.2 Public Meeting to Discuss New NRC Reactor Oversight Process

On July 20, 2000, the NRC held a joint, public meeting with the Nuclear Energy Advisory Council (NEAC) at Waterford Town Hall. At this meeting the NRC explained the new NRC Reactor Oversight Process and how this process is being used at the Millstone Nuclear Power Station.

ITEMS OPENED AND CLOSED

Opened and Closed During this Inspection

None

Previous Items Closed

None

LIST OF ACRONYMS USED

AWO	automated work order
CDS	chilled water
EDG	emergency diesel generator
ESF	engineered safety feature
IST	inservice testing
MDAFW	motor driven auxiliary feedwater
MSIV	main steam isolation valve
OD	operability determination
PMT	post maintenance testing
RPCCW	reactor plant closed cooling water
SPROC	special procedure
TDAFW	turbine driven auxiliary feedwater
TS	technical specification