November 12, 2002

Mr. J. Alan Price
Site Vice President - Millstone
c/o Mr. D. A. Smith, Manager - Licensing
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, CT 06385

SUBJECT: MILLSTONE POWER STATION UNITS 2 AND 3 - NRC INSPECTION REPORTS 50-336/02-05 AND 50-423/02-05

Dear Mr. Price:

On September 28, 2002, the NRC completed inspections at your Millstone Unit 2 and Unit 3 reactor facilities. The enclosed reports document the inspection findings which were discussed with you and other members of your staff on October 21, 2002.

These inspections examined 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of these inspections, the inspectors identified three Unit 2 issues and two Unit 3 issues of very low safety significance (Green). All of these issues were determined to involve violations of NRC requirements. The inspectors also identified a Unit 2 violation for which the final significance has not yet been determined. This violation will be tracked as a unresolved item pending a final significance determination. Because five of the findings were of very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. In addition, a violation of very low safety significance identified by Dominion is listed in Section 4OA7 of the Unit 3 report. If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of these inspection reports, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Millstone facility.

The NRC has increased security requirements at Millstone Power Station in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC has issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to inspect the licensee's security controls and its compliance with the Order and current security regulations.
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 component of NRC's document system (ADAMS).

ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

***/RA/***

Brian J. McDermott, Chief
Projects Branch 6
Division of Reactor Projects

Docket Nos.: 50-336, 50-423
License Nos.: DPR-65, NPF-49

Enclosures:
(1) NRC Inspection Report 50-336/02-05
Attachment 1: Supplemental Information

(2) NRC Inspection Report 50-423/02-05
Attachment 1: Supplemental Information
Attachment 2: TI 2515/145- Circumferential Cracking of RPV Head Penetration Nozzles Reporting Requirements

<u>cc w/encl:</u>
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**ENCLOSURE 1**

**U.S. NUCLEAR REGULATORY COMMISSION**
**REGION I**

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|---|---|
| Docket No.: | 50-336 |
| License No.: | DPR-65 |
| Report No.: | 50-336/02-05 |
| Licensee: | Dominion Nuclear Connecticut, Inc. |
| Facility: | Millstone Power Station, Unit 2 |
| Location: | P. O. Box 128<br>Waterford, CT  06385 |
| Dates: | June 30, 2002 - September 28, 2002 |
| Inspectors: | S. M. Schneider, Senior Resident Inspector, Unit 2<br>P. C. Cataldo, Resident Inspector, Unit 2<br>S. R. Kennedy, Resident Inspector, Unit 2<br>J. M. Brand, Resident Inspector, Seabrook<br>T. F. Burns, Reactor Inspector, Division of Reactor Safety (DRS)<br>A. L. Burritt, Senior Resident Inspector, Limerick 1and 2<br>L. M. Cheung, Senior Reactor Inspector, DRS<br>G. V. Cranston, Reactor Inspector, DRS<br>G. T. Dentel, Senior Resident Inspector, Seabrook<br>K. M. Jenison, Senior Projects Engineer, Division of Reactor Projects<br>K. A. Mangan, Reactor Inspector, DRS<br>A. C. McMurtray, Senior Resident Inspector, Peach Bottom 2 and 3<br>G. C. Smith, Senior Physical Security Inspector, DRS |
| Approved by: | Brian J. McDermott, Chief<br>Projects Branch 6<br>Division of Reactor Projects |

## SUMMARY OF FINDINGS

IR 05000336-02-05; Dominion Nuclear Connecticut, Inc.; on 06/30-09/28/02; Millstone Power Station; Unit 2. Maintenance Rule Implementation; Personnel Performance During Non-routine Plant Evolutions; Identification and Resolution of Problems.

The inspection was conducted by resident and regional inspectors. The inspectors identified three Green issues, all of which were Non-Cited Violations. In addition, the inspectors identified one violation for which the final significance has not yet been determined. The significance of most findings is indicated by the color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

### A. <u>Inspector Identified Findings</u>

**Cornerstone: Mitigating Systems**

- **Green.** The inspectors identified a non-cited violation (NCV) of technical specification 6.8.1 concerning an inadequate preventive maintenance procedure, which caused a failure of the “C” charging pump high speed coupling and rendered the “C” charging pump incapable of performing its required safety function. Specifically, vendor manual instructions related to grease removal and seal inspections were not translated into the licensee's procedures.

  The finding impacted the Mitigating Systems cornerstone and affected the availability of the "C" charging pump to perform its required safety function. However, this finding was of very low safety significance (Green) based on a Phase 1 Significance Determination Process evaluation because the finding did not represent an actual loss of the charging system's safety function or an actual loss of charging pumps for greater than the technical specification allowed outage time. Because the finding is of very low safety significance and it was captured in the licensee's corrective action program, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (Section 1R12.1)

- **TBD.** The inspectors identified a violation of 10 CFR 50.65(a)(2) concerning a failure to demonstrate that the condition of a component was being effectively controlled through preventive maintenance. A solenoid operated valve in the “A” emergency diesel generator (EDG) ventilation system failed and no preventive maintenance had been specified for the component, contrary to the vendor's recommendations. The failure of the “A” EDG's ventilation exhaust damper rendered the EDG incapable of performing its required safety function.

The finding impacted the Mitigating Systems cornerstone and affected the availability of the "A" EDG. The inspectors evaluated the significance of this finding using the SDP Phase 1 worksheets and the SDP Phase 2 risk-informed inspection notebook (Revision 1) for Millstone Unit 2. Based on the results of the SDP Phase 2 evaluation, a SDP Phase 3 evaluation must be performed. However, the information necessary to complete the SDP Phase 3 evaluation was not available at the conclusion of the inspection period and therefore this issue will be tracked as an unresolved item pending a final significance determination. (Section 1R12.2)

- **Green.** The inspectors identified a Severity Level IV non-cited violation (NCV) of 10 CFR 50.59 involving a procedure change to allow the use of the "A" high pressure safety injection (HPSI) flow path as an alternate charging flow path in Mode 3. The licensee's safety evaluation failed to accurately assess the temperature transients in piping associated with this flow path. The procedure change was developed during a forced shutdown of Unit 2 and the HPSI system piping and nozzle were subjected to thermal transients that were not bounded by the Final Safety Analysis Report (FSAR).

  This finding is associated with the Mitigating Systems cornerstone and it had the potential to impact the NRC's ability to perform its regulatory function. However, because of the potential for the thermal transients to impact the integrity of the HPSI system under subsequent operational conditions, the inspectors evaluated the finding in accordance with Appendix "A" of the Significance Determination Process. The inspectors determined that the impact from thermal cycles in excess of the FSAR analyses was of very low safety significance (Green) because a subsequent licensee analysis showed there would be no actual loss of the system's safety function. Because the finding is of very low safety significance and because the finding was captured in the licensee's corrective action program, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (Section 1R14.1).

- **Green.** The inspectors identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, for inadequate corrective actions to promptly identify and correct welds susceptible to fatigue failure following two weld failures in the chemical and volume control system (CVCS) which occurred in July 1999 and November 2001.

  This finding is associated with the Mitigating Systems cornerstone and it affected the reliability of the charging system. The failure to promptly identify and correct susceptible welds in the CVCS system resulted in two additional weld failures, on like welds, during August 2002. The finding was of very low safety significance (Green) because neither weld failure would have prevented the CVCS discharge header from completing its safety function while the Unit was at power. Because the finding is of very low safety significance and the finding was captured in the licensee's corrective action program, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (Section 4OA2.1).

# Report Details

## SUMMARY OF UNIT 2 STATUS

The Unit operated at essentially 100 percent power for the duration of the inspection period with the exception of the period between August 3, 2002 and August 11, 2002. On August 3, 2002, the unit was shutdown due to a Chemical and Volume Control System (CVCS) charging header leak. After completion of repairs, the unit experienced a reactor trip from 52 percent power during the plant startup on August 7, 2002 due to a feedwater pump discharge check valve failure. After completion of repairs to the discharge check valve, the unit returned to 100 percent power on August 11, 2002 and operated at essentially 100 percent power for the remainder of the inspection period.

## 1. REACTOR SAFETY [R]
**Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity**

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the licensee's preparation for adverse weather relative to the protection of safety-related structures, systems, and components (SSCs) during the current hurricane season. This review also focused on the adequacy of applicable procedures and design features established to protect safety-related service water pumps from the effects of a hurricane. The inspectors reviewed the licensee's updated Final Safety Analysis Report (FSAR) regarding design features of various SSCs, and reviewed the following licensee procedures relative to hurricane preparations and protections:

- AOP 2560, Revision 009-04, "Storms, High Winds and High Tides"
- C OP 200.6, Revision 001-01, "Storms and Other Hazardous Phenomena (Preparation and Recovery)"
- MP 2721C, Revision 7, "Protection and Restoration of Service Water Pump Motor During a PMH"

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 "A" Emergency Diesel Generator

a. Inspection Scope

The inspectors performed a partial system alignment check on the "A" emergency diesel generator (EDG) during maintenance activities on the "B" EDG. The inspectors verified that the "A" EDG was correctly aligned for operation in accordance with OPS Form 2346A-002, "A DG Pre-start Checklist," Revision 019-01 and OPS Form 2613A-2, "DG Valve Alignment Checklist, Facility 1," Revision 015-05.

b. Findings

No findings of significance were identified.

.2 "B" Motor-Driven Auxiliary Feedwater Pump

a. Inspection Scope

The inspectors performed a partial system alignment check on the "B" motor-driven auxiliary feedwater pump (MDAFP) system during an operability run of the "A" MDAFP system. The inspectors verified that the "B" MDAFP was correctly aligned for operation in accordance with Surveillance Procedure (SP) 2610C, Revision 012-05, "AFW System L/U Valve Operability, and Operational Readiness Tests" and OPS Form 2610C-002, Revision 019-05, "Auxiliary Feedwater System Lineup Verification.”

b. Findings

No findings of significance were identified.

.3 "B" Low Pressure Safety Injection

a. Inspection Scope

The inspectors performed a partial system alignment check on the "B" low pressure safety injection (LPSI) system during preventive maintenance activities on the "A" LPSI system. The inspectors verified that the "B" LPSI was correctly aligned for operation in accordance with Surveillance Procedure (SP) 2604M, Revision 009-07, "LPSI System Alignment and Valve Tests, Facility 2", OPS Form 2604M-1, Revision 9, "LPSI System Electrical Alignment Check, Facility 2", and OPS Form 2604M-2, Revision 017-02, "LPSI System Valve Alignment Check, Facility 2."

b. Findings

No findings of significance were identified.

.4 "A" Emergency Diesel Generator and Associated Equipment

a. Inspection Scope

The inspectors performed a partial system alignment check on the "A" EDG and its associated equipment. The inspectors verified that the electrical equipment was correctly aligned for operation in accordance with the appropriate Plant and Instrument Drawing (P&ID) schematics, Surveillance Procedure (SP) alignment checks (including SP 2613A, Revision 020-03, "EDG Valve Alignment Checklist") and Millstone Unit 2 FSAR descriptions.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Plant Inspections

a. Inspection Scope

The inspectors performed walkdowns of the following plant areas to assess licensee control of transient combustibles and ignition sources, the material condition of reactor plant fire protection systems and features, and the material condition and operational status of fire barriers:

- Motor Driven Auxiliary Feed Pump Room - Turbine Building, 1'-6" Elevation (Fire Area T-3)
- Steam Driven Auxiliary Feed Pump Room - Turbine Building, 1'-6" Elevation (Fire Area T-4)
- West Piping Penetration Room - Auxiliary Building, -25'-6" and -5'-0" Elevation (Fire Areas A-8B/C/R-2)
- East 480 Volt Load Center Room - Auxiliary Building, 36'-6" Elevation (Fire Area A-28/R-11)
- East DC Equipment Room - Auxiliary Building, 14' Elevation (Fire Area A-20)
- East Battery Room - Auxiliary Building, 14' Elevation (Fire Area A-22)
- "A" Safeguards Room - Auxiliary Building, -45' Elevation (Fire Area A-8)
- "B" Safeguards Room - Auxiliary Building, -45' Elevation (Fire Area A-3)
- Z1 Switchgear Room - Turbine Building, 31'-6" Elevation (Fire Area T-7)

The inspectors reviewed the following related licensee documents:

- Refer to Attachment 1 for fire protection evaluations reviewed
- Unit 2 Fire Hazards Analysis
- Unit 2 Fire Fighting Strategies
- Unit 2 Combustible Loading Calculations
- Fire Hazards Analysis Boundary Drawings

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Observation

a. Inspection Scope

The inspectors observed plant personnel performance during an unannounced fire brigade drill on July 16, 2002, to evaluate the readiness of station personnel to prevent and fight fires. The drill simulated fighting a fire at Breaker B6244 (computer room air-conditioning unit) in Motor Control Center (MCC) B62, in the Unit 2 Ventilation Equipment Room. The inspectors observed the fire brigade members using protective clothing, turnout gear, and self-contained breathing apparatus and entering the fire area in a controlled manner. The inspectors also observed the fire fighting equipment brought to the fire scene to evaluate whether sufficient equipment was available to effectively control and extinguish the simulated fire. The inspectors evaluated whether the permanent plant fire hose lines were capable of reaching the fire area and whether hose usage was adequately simulated. The inspectors observed the fire fighting directions and communications between fire brigade members. The inspectors evaluated the simulated smoke removal operations to verify that they would be effective. The inspectors verified that the pre-planned drill scenario was followed and observed the post drill critique to evaluate if the drill objectives were satisfied and that any drill weaknesses were discussed.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors evaluated the licensee’s preparation and protection from the effects of external flooding conditions. The inspectors reviewed the FSAR, and various procedures to determine the efficacy and readiness of protection for applicable safety-related structures, systems and components. The inspectors performed a walkdown of the Unit 2 floodgates, verified the adequacy of the floodgates and removable flood planks to perform their design function, reviewed recent licensee inspection results of floodgate inspections, and verified that previously identified deficiencies had been entered into the licensee's corrective action program for resolution. Additionally, the inspectors reviewed the licensee's inspection activities regarding safety-related manholes at Unit 2. The inspectors also reviewed the following licensee procedures relative to flood protection:

- SP 2665, Revision 004-03, "Building Flood Gate Inspections"
- AOP 2560, Revision 009-04, "Storms, High Winds and High Tides"
- C EN 104I, Revision 004, "Condition Monitoring of Structures"
- MP 2721C, Revision 7, "Protection and Restoration of Service Water Pump Motor During a PMH"
- C OP 200.6, Revision 001-01, "Storms and Other Hazardous Phenomena (Preparation and Recovery)"
- SP 2615, Revision 6, "Flood Level Determination"

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed the conduct of licensed operator requalification simulator training exercises on August 14 and 15, 2002. The inspectors observed licensed operator performance relative to the following activities: effective communications, implementation of normal, abnormal and emergency operating procedures, command and control, and technical specification compliance. The inspectors verified that the training evaluators adequately addressed operator performance issues that were identified during the exercise, and that applicable training objectives had been achieved. In addition, the inspectors observed the determination of emergency classifications associated with the exercise (See Sections 1EP6.1 and 1EP6.2).

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

.1 "C" Charging Pump High Speed Coupling Failure

a. Inspection Scope

The inspectors reviewed the licensee's activities following the failure of the high speed coupling on the "C" Charging Pump, specifically, the implementation of the maintenance rule in accordance with 10 CFR 50.65. The review also verified the licensee's evaluation of the event as a maintenance rule functional failure in accordance with MP-24-MR-FAP710, Revision 0, "Maintenance Rule Functional Failures and Evaluations," and NUMARC 93-01, Revision 2, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

b. Findings

The inspectors identified a non-cited violation of Technical Specification (TS) 6.8.1.a. associated with the failure to establish and implement adequate preventive maintenance procedures for the high speed coupling of the "C" Charging Pump. The issue was determined to be of very low safety significance (Green).

Introduction

On April 15, 2002, the "C" Charging Pump high speed coupling failed. The failure resulted from internal gear damage on the motor side coupling half due to lack of grease lubrication. The licensee determined that the lack of lubrication and ultimately, the coupling failure, was caused by inadequate preventive maintenance. Specifically, the licensee identified that maintenance procedures did not include vendor recommended O-ring inspections and steps for proper lubrication of the coupling.

Description

Maintenance Procedure MP 2701F, Revision 012-11, "Lubrication," contains instructions for the cleaning, inspection, and lubrication of couplings. Additionally, MP 2703C10, Revision 003-01, "Charging Pump Speed Reducer Overhaul," contains instructions that include re-assembly and lubrication of couplings for the charging pumps. However, the procedures did not contain vendor recommended steps or guidance to ensure that (1) the grease-retaining O-rings were properly lubricated and of satisfactory condition, (2) complete removal of the old grease was emphasized to preclude premature grease degradation, and (3) coupling guards were inspected to identify grease loss from the couplings that would lead to coupling degradation. For example, while the grease-retaining O-rings (seal rings) inspections and lubrications are identified in the vendor technical manual (VTM) 25203-309-001, "Installation, Operation and Maintenance of Reciprocating Charging Pumps," licensee procedures that implement maintenance activities do not address the physical condition or lubrication of these O-rings prior to installation. Also, while the removal of the old grease is discussed in the VTM, the licensee’s procedures do not explicitly require complete removal of old grease prior to re-packing, which the licensee concluded has led to premature degradation of the grease. Additionally, licensee procedures did not provide guidance regarding the loss of grease from the coupling that precedes eventual coupling failure, as evidenced by the presence of grease on the inner side of coupling guards.

Analysis

The inspectors determined that the failure to establish and implement adequate preventive maintenance procedures for charging pump couplings was more than minor based on the finding impacting the Mitigating Systems Cornerstone, and affecting the availability of the "C" charging pump to perform its required safety function. The inspectors determined this finding was of very low safety significance (Green) based on a Phase 1 SDP evaluation. The finding did not represent an actual loss of the charging system's safety function or an actual loss of charging pumps for greater than the technical specification allowed outage time.

Enforcement

Technical Specification 6.8.1.a. requires, in part, that written procedures be established, implemented, and maintained for the activities described in Appendix “A” of RG 1.33, "Quality Assurance Program Requirements (Operation)." Specifically, Section 9 of RG 1.33, Appendix “A”, "Procedures for Performing Maintenance," details that maintenance that can affect the performance of safety-related equipment should be performed in accordance with written procedures or documented instructions appropriate to the circumstances. The licensee’s failure to adequately establish and implement procedures covering the cleaning, inspection and lubrication of couplings, as evidenced by the coupling failure that occurred on April 15, 2002, is a violation of Technical Specification 6.8.1.a **(NCV 50-336/02-05-01)**. This violation is associated with an inspection finding that is characterized by the significance determination process as having very low safety significance (Green) and is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as CR-02-04484.

.2 "A" Emergency Diesel Generator Exhaust Damper Solenoid Valve Failure

a. Inspection Scope

The inspectors reviewed the licensee’s activities following the failure of the solenoid valve for the "A" EDG ventilation exhaust damper, 2-HV-255A, relative to the implementation of the maintenance rule in accordance with 10 CFR 50.65. The review also verified the licensee’s evaluation of the event as a maintenance rule functional failure in accordance with MP-24-MR-FAP710, Revision 0, "Maintenance Rule Functional Failures and Evaluations," and NUMARC 93-01, Revision 2, "Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

b. Findings

The inspectors identified a violation of 10 CFR 50.65 (a)(2), which involved the failure to perform preventive maintenance that resulted in the subsequent failure of the "A" EDG ventilation exhaust damper solenoid valve. Additional information is necessary for the NRC to make a significance determination for this finding, consequently this issue will be tracked as an unresolved item.

Introduction

On August 14, 2002, the discharge damper for the "A" EDG room ventilation system did not open on demand due to a solenoid operated valve (SOV) failure. The licensee declared the "A" EDG inoperable, entered the appropriate technical specification, replaced the SOV, and restored the "A" EDG to operable status.

Description

During the licensee's investigation of the failure, independent failure analyses identified foreign material located in the internal passages of the SOV that may have contributed to the valve's failure to open. The manufacturer's recommended maintenance includes periodic replacement of specific parts of the solenoid based on a qualified service life of four years. At the time of this failure, the SOV had been in service for approximately eight years and the licensee had not formally evaluated or established a preventive maintenance schedule.

Analysis

The inspectors determined that the failure to perform adequate preventive maintenance for the exhaust damper SOV was more than minor because the finding impacted the Mitigating Systems Cornerstone, and affected the availability of the "A" EDG to perform its required safety function.

The inspectors evaluated the significance of this finding using the SDP Phase 1 worksheets and the SDP Phase 2 risk-informed inspection notebook (Revision 1) for Millstone Unit 2. Based on the results of the Phase 2 evaluation, a Phase 3 evaluation must be performed by an NRC Senior Reactor Analyst. The information necessary to complete the Phase 3 evaluation was not available at the conclusion of the inspection period and therefore this issue will be tracked as an unresolved item pending completion of the SDP Phase 3 evaluation. (**URI 50-336/02-05-02**)

.3 Periodic Evaluation

a. Inspection Scope

The inspector reviewed the periodic evaluation required by 10 CFR 50.65 (a)(3) for Millstone Nuclear Power Station, Units 2 and 3 to verify that structures, systems and components (SSCs) within the scope of the maintenance rule were included in the evaluation and, balancing of reliability and unavailability was given adequate consideration. The inspector reviewed the licensee's most recent periodic evaluation report for Units 2 and 3 which covered the interval October 1999 through August 2001. The inspector verified that the periodic evaluation was completed within the required two year time periods.

The inspector selected the following Unit 2 (a)(1) systems for detailed review:

Reactor Protection System (RPS)
High Pressure Safety Injection (HPSI)
Auxiliary Feedwater (AFW)
Service Water (SW)
Main Steam (MS)
Chemical and Volume Control System (CVCS)

Unit 3 (a)(1) systems selected for detailed review were:

Auxiliary Feedwater (AFW)
Service Water (SW)
Main Steam Isolation Valves (MSIV)
Containment Isolation (CI)

The inspector verified: (1) goals and performance criteria were appropriate, (2) industry operating experience was considered, (3) problem identification and resolution of maintenance rule-related issues were addressed, (4) corrective action plans were effective, and (5) performance was being effectively monitored. The inspector verified that adjustments were made in action plans for SSCs in (a)(1) status as a result of the licensee's review of system performance against established goals. The inspector reviewed documentation for a sample of high safety significant SSCs to verify that the licensee balanced reliability and availability/unavailability and adjusted (a)(1) goals as necessary. The inspector reviewed availability/unavailability tracking and trending data for RPS, HPSI and AFW from Unit 2 and determined that the trends were in the acceptable range and performance criteria had not been exceeded.

The inspector selected a sample of high safety significant SSCs that were in (a)(2) status to verify that the licensee had established appropriate performance criteria (PC). Also, the inspector evaluated whether the licensee examined any SSCs that failed to meet their PC and reviewed those SSCs that exhibited repeated maintenance preventable functional failures for consideration of movement to (a)(1) status.

The inspector reviewed documentation for a sample of systems that the licensee had changed from (a)(1) status to (a)(2) status during the periodic assessment period. The inspector selected RPS and HPSI from Unit 2 and 125 Volt DC from Unit 3 to verify that (a)(1) goals had been met to return the systems to (a)(2) status.

In addition, the inspector verified that the licensee had established and implemented a preventive maintenance program to manage preventive maintenance activities for systems in both (a)(1) and (a)(2) status. A sample of risk significant systems in (a)(1) and (a)(2) status was reviewed to verify the performance of condition monitoring and scheduled maintenance.

b. Findings

No findings of significance were identified.

## 1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors verified the conduct and adequacy of scheduled maintenance risk assessments for plant conditions affected by the conduct of the following scheduled maintenance and testing activities:

- Unit 2 Work Schedule for the week of 7/8/02 - maintenance and testing on the "B" boric acid pump repair.
- Unit 2 Work Schedule for the week of 7/14/02 - maintenance and testing on the "A" motor-driven auxiliary feedwater pump.
- Unit 2 Work Schedule for the week of 7/29/02 - maintenance and testing on the High Pressure Safety Injection (HPSI) Pump.
- Unit 2 Work Schedule for the week of 8/4/02 - maintenance and testing on the containment sump/shutdown cooling heat exchanger outlet valve.
- Unit 2 Work Schedule for the week of 8/11/02 - maintenance and testing on the "A" Low Pressure Safety Injection (LPSI) pump seal cooler relief lift and loss of Reactor Building Closed Cooling Water (RBCCW).
- Unit 2 Work Schedule for the week of 8/11/02 - maintenance and testing on the Charging Pumps Discharge Check Valve IST, Two Pump Test (2601J-1).
- Unit 2 Work Schedule for the week of 8/25/02 - maintenance and testing on the "B" Charging Pump, "A" and "B" Auxiliary Feedwater Pumps, "B" Emergency Diesel Generator, and "B" and "C" HPSI Pumps.

The inspectors compared the results from the licensee's Equipment Out of Service (EOOS) quantitative risk assessment tool for the above plant configurations with the licensee's stated risk. The inspectors also verified that the licensee entered appropriate risk categories and implemented risk management actions. In addition, the inspectors reviewed the following related licensee documents:

- Refer to Attachment 1 for documents reviewed under maintenance risk assessments and emergent work evaluations
- Major Equipment Schedule
- Control Room Operator Log
- NUMARC 93-01, Revision 2, Section 11, "Assessment of Risk From Performance of Maintenance Activities"

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 Reactor Shutdown Due to Chemical and Volume Control System Charging Header Weld Failure

a. Inspection Scope

The inspectors reviewed personnel performance in coping with non-routine evolutions and transients. Specifically the inspectors reviewed personnel response during shutdown of the plant due to a failure of a Chemical and Volume Control System (CVCS) charging header weld. The inspectors reviewed operator logs, plant computer data, and response procedures. The inspectors also reviewed the following related licensee documents:

- Refer to Attachment 1 for documents reviewed under a reactor shutdown due to chemical and volume control system charging header weld failures
- Control Room Operator Log
- Technical Specifications
- FSAR

b. Findings

The inspectors identified a Severity Level IV non-cited violation of 10 CFR 50.59 involving a procedure change for use of the "A" HPSI flow path as an alternate charging flow path in Mode 3. The licensee's safety evaluation failed to recognize that actual temperature transients would not be bounded by the existing analyses for thermal cycles. This finding was determined to be of very low safety significance (Green).

Introduction

On August 3, 2002, the licensee initiated a shutdown of Millstone Unit 2, due to weld failures in the CVCS discharge header. In order to keep the unit in Mode 3 with the CVCS discharge header isolated for repairs, the licensee revised its operating procedures to allow use of the HPSI flow path. The procedure change was evaluated and found to be acceptable by the licensee using a 10 CFR 50.59 screening process.

Description

The inspectors observed the shutdown and reviewed the licensee’s preparation for utilizing an alternate charging path through the "A" high pressure safety injection (HPSI) piping and nozzle. In this alternate charging configuration the charging water would not be reheated through the regenerative heat exchanger as is the case when utilizing the normal flow path. An existing CVCS procedure allowed alternate injection via HPSI but required proceeding directly to Mode 5. Therefore, a procedure change was needed to use the alternate charging flow path in Mode 3 less than (<) 1750 psig. The licensee subsequently generated a 10 CFR 50.59 screen which qualitatively concluded that the alternate charging flow path was bounded by FSAR design considerations for the expected thermal transients. On August 4, 2002, the licensee commenced use of the alternate charging path until repairs of the charging header were completed on

August 5, 2002. The alternate charging flow path was used on four occasions during the CVCS charging header maintenance period.

On August 5, 2002, the inspectors reviewed the 10 CFR 50.59 screen and questioned the technical basis for the thermal cycles associated with the use of the alternate charging flow path. On August 6, 2002, the licensee identified that the alternate charging flow path thermal transients were not bounded. The licensee had concluded on August 3, 2002, that the activity was bounded by existing design analysis since this alternate charging flow path was described in the FSAR. However, the licensee incorrectly concluded that design thermal transients such as emergency safety injection and shutdown cooling operation were more severe due to higher temperature differentials and/or higher flow rates than the thermal transients which would be experienced while injecting in Mode 3 less than 1750 psig. The actual thermal transients experienced in Mode 3 less than 1750 psig had a higher temperature differential and a lower flow rate than the operations referenced in the 10 CFR 50.59 screen. The design basis cyclic transients discussed in the FSAR and used in the fatigue analysis are based on engineering design specifications generated for Millstone Unit 2. These specifications allow 500 injection cycles with a temperature differential of 305 degrees Fahrenheit at 1500 gpm flow.

On August 15, 2002, the licensee generated a technical evaluation which quantitatively assessed the allowed number of thermal cycles given the actual temperature range when utilizing the alternate injection flow path. The design basis temperature range was qualified for 500 cycles at 305 degree temperature differential under a 1500 gpm flow. The licensee concluded that the actual temperature differential of 390 degrees at 44 gpm flow would reduce the allowable number of cycles to 256 cycles. Since only four injections were made using the alternate charging flow path, the piping and nozzle thermal fatigue ASME Code limits were not exceeded.

Analysis

This finding is associated with the Mitigating Systems cornerstone and had the potential to impact the NRC’s ability to perform its regulatory function. In accordance with the NRC’s Enforcement Policy, violations of 10 CFR 50.59 are normally dispositioned outside of the significance determination process based on their impact to the regulatory process. However, because of the potential for the thermal transients to impact the integrity of the HPSI system under subsequent operational conditions, the inspectors evaluated the finding in accordance with the SDP, MC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations.” The inspectors determined that the thermal cycles in excess of the FSAR analyses screened to “Green” in the Phase 1 Mitigating Systems SDP since the finding did not result in an actual loss of the system’s safety function or an actual loss of safety function for a single train for greater than its Technical Specification allowed outage time. The system’s safety function was not impacted because a subsequent quantitative evaluation showed that the piping and nozzle thermal fatigue usage limits were not exceeded. Therefore, the finding is considered to be of very low safety significance (Green).

Enforcement

10 CFR 50.59(a)(6) defines, in part, tests or experiments not described in the FSAR to mean any activity where any structure, system, or component is utilized or controlled

outside of the reference bounds of the design basis as described in the FSAR. The licensee utilizes a 10 CFR 50.59 screening process to determine if changes to the facility or procedures require a 10 CFR 50.59 evaluation which is then utilized to determine if a proposed activity requires NRC approval via license amendment. An annual report, required by 10 CFR 50.59, is submitted to the NRC to describe all changes, tests, and experiments made to the plant that have been reviewed by the 10 CFR 50.59 evaluation process. The failure of the licensee to correctly conclude that the proposed alternate injection path would subject the "A" HPSI piping and nozzle to thermal transients which were outside of the FSAR design basis resulted in the failure to conduct a 10 CFR 50.59 evaluation prior to utilizing this flow path and is considered a violation of 10 CFR 50.59 **(NCV 50-336/02-05-03)**. This Severity Level IV violation is associated with an inspection finding that is characterized by the significance determination process as having very low safety significance (Green), and is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR-02-08161.

.2 Automatic Reactor Trip on Low Steam Generator Water Level

a. Inspection Scope

The inspectors reviewed personnel performance in coping with non-routine evolutions and transients. Specifically the inspectors reviewed personnel response to an automatic reactor trip due to low steam generator water level. The inspectors reviewed operator logs, plant computer data, and response procedures. The inspectors also reviewed the following related licensee documents:

- Refer to Attachment 1 for documents reviewed under an automatic reactor trip on low steam generator level
- Control Room Operator Log
- FSAR

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 Resistance Temperature Detector

a. Inspection Scope

The inspectors reviewed the operability determination associated with the reactor coolant system (RCS) resistance temperature detectors (RTD) following the licensee identification that the RCS RTDs had not been calibrated since March 1999. The inspectors reviewed RTD operability to ensure that operability was justified and that RTDs remained available and no unrecognized increase in risk had occurred. The inspectors reviewed the following related licensee documents:

- CR-02-07563, Lapsed Surveillance on All Four Channels of Delta-T Power, TMLP, LPD and Reactor Cold Leg Temperature Indication
- IC2417L, Revision 06, "Primary Coolant System RTD Data Collection"
- Technical Specifications
- RP-5, Revision 002-04, "Operability Determinations"

b. Findings

No findings of significance were identified.

.2 Relief Valve Lift During Reactor Building Closed Cooling Water Valve Testing

a. Inspection Scope

The inspectors reviewed the operability determination associated with a reactor building closed cooling water (RBCCW) relief valve for a low pressure safety injection (LPSI) seal cooler. The inspectors verified that operability of RBCCW was justified and that LPSI remained available and no unrecognized increase in risk had occurred. The inspectors reviewed the following related licensee documents:

- OD-MP2-018-02, "A 2 gpm Leak from 2-RB-309 was discovered after Performance of SP 2611C on 2-RB-28.1C"
- RP-5, Revision 002-04, "Operability Determinations"

b. Findings

No findings of significance were identified.

.3 Breaker B5215, "A" Enclosure Building Filtration System Fan 25A Supply

a. Inspection Scope

The inspectors reviewed the operability determination associated with Breaker B5215, "A" enclosure building filtration system Fan 25A supply, when the inspectors identified that the breaker post maintenance test acceptance criteria had not been met. The inspectors reviewed the operability determination to ensure that operability was justified and that Breaker B5215 remained available and no unrecognized increase in risk had occurred. The inspectors reviewed the following related licensee documents:

- CR-02-07378, PM Acceptance Criteria was not met when PM on B5215 was done under M2-95-7753
- OD-MP2-015-02, "Breaker B5215 Failed PM Acceptance Criteria"
- FSAR
- RP-5, Revision 002-04, "Operability Determinations"

b. Findings

No findings of significance were identified.

.4 Charging System Operability With Discharge Pulsation Dampener Support Pedestal Gaps

a. Inspection Scope

The inspectors reviewed preliminary operability determination (OD) MP2-023-02, which was initiated following the licensee's identification that vertical gaps existed in the support pedestals of the discharge pulsation dampeners of the "A" and "C" charging pumps. The inspectors evaluated the engineering basis that supported continued operability of the charging system with the existence of gaps in the support pedestals under postulated seismic loads. The inspectors verified that the licensee had entered the issue into its corrective action program for resolution as CR-02-09267.

b. Findings

No findings of significance were identified.

.5 "B" Emergency Diesel Generator Operability Following the Failure to Meet the Minimum Voltage During an Operability Run

a. Inspection Scope

The inspectors evaluated the licensee's response regarding operability following the failure to meet surveillance procedure minimum voltage requirements during an operability surveillance run for the "B" emergency diesel generator (EDG) on July 31, 2002. The inspectors reviewed the licensee's common cause failure mode evaluation that concluded the "A" EDG was operable and not susceptible to a similar failure mode. A successful operability run was subsequently performed on the "B" EDG on August 1, 2002. The inspectors reviewed the following related licensee documents:

- SP 2613L, Revision 001-06, "Diesel Generator Slow Start Operability Test, Facility 2"
- CR-02-07973, During the Performance of SP2613L, the "B" EDG Failed to Achieve the Required Output Voltage
- RP-5, Revision 002-04, "Operability Determinations"

b. Findings

No findings of significance were identified.

.6 Chemical and Volume Control System Operability

a. Inspection Scope

The inspectors reviewed the operability determination associated with the pressure boundary of the charging portion of the chemical and volume control (CVCS) system to ensure that operability was justified and that the CVCS system remained available and no unrecognized increase in risk had occurred. The inspectors also reviewed compensatory measures to ensure that the compensatory measures were in place and were appropriately controlled. The inspectors reviewed the following related licensee documents:

- RECO/OD-MP2-020-02, "Questionable Pressure Boundary Reliability Caused by Pressure Pulsations and Piping Vibration"
- RP-5, Revision 002-04, "Operability Determinations"

b. Findings

No findings of significance were identified.

.7 Feedwater Containment Isolation Check Valve Did Not Go Fully Closed

a. Inspection Scope

The inspectors reviewed the operability determination associated with feedwater containment isolation check valve 2-FW-5B to ensure that operability was justified and that the feedwater containment isolation check valve remained available and no unrecognized increase in risk had occurred. The inspectors reviewed the following related licensee documents:

- FSAR
- RP-5, Revision 002-04, "Operability Determinations"
- CR-02-08583, Minor Documentation Errors Identified with GDC-57 Compliance Review of Valves 2-FW-5A and 5B
- Millstone Unit 2 Pre-Inservice Operating Company Deficiency Report dated 5/9/73
- Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission, Millstone Unit 2, dated 5/10/74

- OD-MP2-017-02, "2-FW-5B Operator did not Fully Stroke Closed"
- M2-EV-02-0033, Revision 0, Evaluation of Partial Operator Stroke of the Main Feedwater CIV Check Valve, Millstone Unit 2

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

.1 "A" Emergency Diesel Generator Electronic Governor Replacement

a. Inspection Scope

The inspectors reviewed the automated work order (AWO) associated with maintenance on the "A" emergency diesel generator (EDG) for replacement of the electric governor. The inspectors verified that the selected post-maintenance tests were appropriate for the maintenance activity that was conducted and adequately demonstrated that the "A" EDG would continue to perform its required safety function. The inspectors reviewed the following related licensee documents:

- AWO M2-02-07247, "A Diesel Load Shearing and Speed Control Device"
- MP-20-WP-GDL40, Revision 001, "Pre- and Post-Maintenance Testing"
- PT 21416G1, Revision 2, "MP2 Diesel Generator Woodward 2301A Bench Test"
- PT 21416H1, Revision 2, "MP2 ‘A’ Diesel Generator (H7A) Woodward 2301A Replacement and Adjustment"

b. Findings

No findings of significance were identified.

.2 "B" Charging Pump Reduction Gear

a. Inspection Scope

The inspectors reviewed the AWO associated with inspection and replacement of the "B" charging pump reduction gear couplings. The inspectors verified that the selected post-maintenance tests adequately demonstrated that the charging pump would continue to perform its required safety functions. The inspectors also verified that the selected post-maintenance tests were appropriate for the maintenance activity that was conducted and that vibration test results and lubricating oil samples were within acceptable criteria. The inspectors also performed a plant walkdown of the charging pump rooms and verified that identified deficiencies were entered into the licensee's corrective action program for resolution. The inspectors reviewed the following related licensee documents:

- AWO M2-02-05539, "'B' Charging Pump Assembly"
- Maintenance Procedure MP 2703C10, Revision 3, "Charging Pump Reducer Overhaul"
- MP-20-WP-GDL40, Revision 001, "Pre- and Post-Maintenance Testing"
- CBM 104, "Vibration Data"

- Charging Pump Lubricating Oil Sample Reports

b. Findings

No findings of significance were identified.

.3 Preventive Maintenance on the Breaker for the Reactor Building Component Cooling Water Area Sump Pump

a. Inspection Scope

The inspectors reviewed the AWO associated with preventive maintenance on a 480V breaker for the reactor building component cooling water (RBCCW) heat exchanger area "A" sump pump completed on August 21, 2002. The inspectors verified that the selected post-maintenance tests were appropriate for the maintenance activity that was conducted. The inspectors interviewed the maintenance technician and the work control senior reactor operator and reviewed the following related licensee documents:

- AWO M2-99-15324, "Starter PM and Cubicle Inspection"
- AWO M2-01-10446, "Overcurrent Test, Contact Resistance and Megger Check"
- MP-20-WP-GDL40, Revision 1, "Pre- and Post-Maintenance Testing"

b. Findings

No findings of significance were identified.

.4 "A" H2 Sensor O-Ring and Flow Switch Replacement

a. Inspection Scope

The inspectors reviewed the AWO associated with maintenance on the "A" H2 monitoring system. The inspectors verified that the selected post-maintenance tests were appropriate for the maintenance activity that was conducted and adequately demonstrated that the "A" H2 monitoring system would continue to perform its required safety function. The inspectors also verified that identified deficiencies were entered into the licensee's corrective action program for resolution. The inspectors reviewed the following related licensee documents:

- AWO M2-02-05940, "Flow Switch is at End of Qualified Life"
- AWO M2-01-13573, "H2 Sensor and Vessel O-Rings are at End of Qualified Life"
- MP-20-WP-GDL40, Revision 001, "Pre- and Post-Maintenance Testing"
- SP-2608H, Revision 006-03, "Leak Test of the Hydrogen Sampling System"
- SP-2403CA, Revision 000-03, "'A' Hydrogen Analyzer Calibration"
- SP-2403CLA, Revision 000, "'A' Hydrogen Analyzer System Functional Test"
- Technical Specifications

b. Findings

No findings of significance were identified.

.5 Weld Repair on Chemical and Volume Control System Discharge Piping

a. Inspection Scope

The inspectors reviewed the AWO associated with maintenance on the weld repair on the charging portion of the chemical and volume control system (CVCS). The inspectors verified that the selected post-maintenance tests were appropriate for the maintenance activity that was conducted and adequately demonstrated that the CVCS system discharge piping would continue to perform its required safety function. The inspectors also verified that identified deficiencies were entered into the licensee's corrective action program for resolution. The inspectors reviewed the following related licensee documents:

- AWO M2-02-10076, "Repair Through Wall Pipe Leak on 2-CCB-6"
- MP-20-WP-GDL40, Revision 001, "Pre- and Post-Maintenance Testing"
- EN 21218, "Post Repair/Replacement Component Leakage Test"
- Technical Specifications

b. Findings

No findings of significance were identified.

.6 "A" Boric Acid System and Transfer Pump

a. Inspection Scope

The inspectors reviewed AWOs associated with maintenance on the "A" boric acid transfer pump. The inspectors verified that the selected post-maintenance tests adequately demonstrated that the subject pump, its supporting equipment and other associated charging system equipment would perform their required safety function. The inspectors also verified that the selected post-maintenance tests were appropriate for the maintenance activity that was conducted and that deficiencies were entered into the licensee's corrective action program for resolution. The inspectors reviewed the following related licensee documents:

- CR-02-07414, Bravo Boric Acid Pump Threaded Fitting Leaks After Replacement
- AWOs M2-01-09226, 00-0359, and 99-10622, "'A' Boric Acid Pump Maintenance"

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors evaluated the licensee's activities related to (1) the reactor trip on August 7, 2002 due to a failed feedwater system check valve, and (2) the repair and recovery from fatigue cracks located in socket welds in the charging discharge header of the Chemical and Volume Control System, which were initially identified on August 3, 2002 and then again on August 27, 2002. The inspectors observed management meetings regarding restart following repair activities, reactivity control during ascension to criticality, and control room activities during the plant shutdown, low power operations, and reactor startups. The inspectors also reviewed licensee response to an off-normal plant configuration, alternate charging injection utilizing HPSI piping, which was implemented due to the location of the weld failure on August 3, 2002. The inspectors also reviewed various operator actions during all three events, some of which are documented in Section 1R14. Refer to Attachment 1 for documents reviewed under refueling and outage activities.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Turbine Driven Auxiliary Feedwater Pump

a. Inspection Scope

The inspectors reviewed licensee performance of surveillance testing of the turbine driven auxiliary feedwater (AFW) pump and structures, systems, and components to ensure these systems are capable of performing their intended safety functions and to ensure related technical specification requirements were met. The following surveillance tests were reviewed as part of this activity:

- SP-2660, Revision 006, "Auxiliary Feedwater Pump Turbine Periodic Testing"
- SP-2610B, Revision 014, "TDAFP Tests"
- SP-2610C, Revision 012, "AFW System Lineup, Valve Operability, and Operational Readiness Tests"
- SP-2610E, Revision 009, "MSIV Closure and Main Steam Valve Operational Readiness Testing"

The inspectors attended test briefs, verified selected prerequisites and precautions, and verified the tests were performed in accordance with the procedural steps. The inspectors also reviewed completed data sheets and verified that TS requirements were met. The inspectors also reviewed the following related licensee documents:

- Technical Specification Surveillance Requirements
- Individual Surveillance Test Procedure Data Forms

b. Findings

No findings of significance were identified.

.2 "B" Emergency Diesel Generator Operability Surveillance

a. Inspection Scope

The inspectors reviewed the licensee's performance of surveillance testing of the "B" EDG on 8/01/02, conducted in accordance with SP 2613L, Revision 001-06, "Diesel Generator Slow Start Operability Test, Facility 2." The inspectors attended the shift brief, verified that selected prerequisites and precautions were met, and verified the test was conducted in accordance with the applicable procedural steps. The inspectors also reviewed the completed surveillance data sheets and verified that the appropriate technical specification and procedure acceptance criteria had been satisfied.

b. Findings

No findings of significance were identified.

.3 "A" Motor Driven Auxiliary Feedwater Pump Operability Test

a. Inspection Scope

The inspectors reviewed licensee performance related to the following surveillance test:

- SP 2610A, Revision 010-04, "Motor Driven Auxiliary Feedwater Pump Operability Test"

The inspectors verified that test results for the operability surveillance were in accordance with the technical specifications, FSAR and the surveillance test procedure acceptance criteria. The inspectors also verified that performance of the test adequately demonstrated equipment operability and that design basis functions were met for the tested portions of the "A" motor driven auxiliary feedwater system and components.

b. Findings

No findings of significance were identified.

.4 Reactor Coolant System Resistance Temperature Detectors

a. Inspection Scope

The inspectors reviewed licensee performance of surveillance testing of reactor coolant system (RCS) resistance temperature detector (RTD) calibration following licensee identification that the RCS RTDs had not been calibrated since March 1999. The inspectors reviewed surveillance calibration data obtained from the plant computer to verify that the RTDs were capable of performing their intended safety functions and to ensure related technical specifications requirements were met. Refer to Attachment 1 for documents reviewed under reactor coolant system resistance temperature detectors.

b. Findings

No findings of significance were identified.

.5 Containment Sump and Shutdown Cooling System

a. Inspection Scope

The inspectors observed the licensee performance of surveillance testing of the containment sump and shutdown cooling system to verify the systems were capable of performing their intended safety function and to ensure related TS requirements were met. Test performance data with TS surveillance requirements and other established performance criteria were compared to current and historical surveillance test data to validate selected system performance parameters. The following related licensee documents were reviewed:

- SP-2604H, Revision 015-02, Containment Sump and Shutdown Cooling System Operability Testing
- SP-2604A, Revision 012-06, High Pressure Safety Injection (HPSI) Pump Operability and Inservice Testing

b. Findings

No findings of significance were identified.

**Emergency Preparedness [EP]**

1EP6 Drill Evaluation

.1 Notification Performance Indicator and Drill Critique

a. Inspection Scope

The inspectors observed a licensee drill, which would be utilized for reporting performance indicator data for notification, to identify any weaknesses or deficiencies in licensee performance. The inspectors observed the licensee drill critique to ensure that the licensee appropriately identified drill deficiencies. The following related licensee documents were reviewed:

- Millstone Emergency Plan
- MP-26-EPI-FAP07, Revision 001-06, "Notifications and Communications"
- Millstone Unit 2 Training Drill CFD 02-026 through 02-09
- Emergency Planning Drill Objectives

b. Findings

No findings of significance were identified.

.2 Requalification Training Emergency Classification

a. Inspection Scope

The inspectors evaluated event classifications that occurred during a licensed operator simulator examination conducted on August 14, 2002. The inspectors verified that the simulator training scenario utilized for the examination was of appropriate scope and that the classifications were evaluated against appropriate criteria, consistent with the following documents:

- NEI 99-02, Revision 1, "Regulatory Assessment Performance Indicator Guidelines"
- MP-26-EPA-GDL01, Revision 0, "Emergency Planning Performance Indicators"

b. Findings

No findings of significance were identified.

## 3. SAFEGUARDS

### Physical Protection [PP]

3PP1 Access Authorization

a. Inspection Scope

An in-office review was conducted of materials related to the effectiveness of Access Authorization self-assessments, selected corrective actions related to reviewed event reports and logged security events, the performance of table top security drills and the functionality of selected access control equipment - including an E-field sensor.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

.1 Safety System Unavailability Performance Indicators

a. Inspection Scope

The inspectors verified the following performance indicators:

- Safety System Unavailability - HPSI
- Safety System Unavailability - Residual Heat Removal

The inspectors reviewed operating logs, maintenance history and surveillance test history for unavailability information for these systems from July 2001 to June 2002. The inspectors also verified the licensee's calculation of critical hours for both units and evaluated applicable safety system equipment unavailability against the performance indicator definition. Refer to Attachment 1 for documents reviewed under performance indicator verifications.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Weld Failures in the Unit 2 Chemical and Volume Control System

a. Inspection Scope

A Problem Identification and Resolution Inspection was performed to review the corrective actions associated with weld failures in the chemical and volume control (CVCS) system charging header that occurred in July 1999 and November 2001. The licensee documented both of these issues in the corrective action program. The inspectors reviewed the corrective actions associated with the failures and other related corrective action documents to ascertain the adequacy of the licensee’s evaluation and corrective actions. The corrective action documents reviewed are listed in Attachment 1. Additionally, the inspectors walked down plant equipment, reviewed plant procedures, interviewed plant personnel and observed the dye penetrant testing on a third CVCS weld that failed on August 3, 2002.

b. Findings

The inspectors identified a non-cited violation of 10 CFR 50, Appendix “B”, Criterion XVI which involved the failure to take adequate corrective actions for a condition adverse to quality to promptly identify and correct welds susceptible to fatigue failures in the charging portion of the CVCS system. The issue was determined to be of very low safety significance (Green).

Introduction

The inspectors reviewed the corrective actions related to condition report CR-01-11536 and CR-99-2053. These CRs document the licensee's investigation and corrective actions related to the failure of socket welds on the CVCS charging pump discharge header in July 1999 and November 2001. In both cases, the licensee determined that vibration induced cyclic stress was the cause of the failure.

Description

As a result of the 1999 failure, the licensee repaired the failed weld and reinforced several other sensitive welds. However, the licensee neither identified all the welds that were susceptible to this type of failure nor prevented the failure mode dynamics. An evaluation of the system, using OM-3 of the ASME code as a screening criterion, was used to identify susceptible welds. The evaluation concluded that half of the welds selected (sixteen in the sample) were susceptible to the failure mode with the current system configuration. The inspectors noted that there was no consideration to expand the sample past the original sixteen welds even though half of the sample welds were found susceptible. Additionally, the inspectors found that initial corrective actions requesting an evaluation of the piping design, vibration analysis, and piping support evaluation were not done. No other corrective actions were performed.

The inspectors found that corrective actions performed as a result of the failure in November 2001 were incomplete. The focus of the investigation was on the cause of the failure and the extent of condition of the problem, although vibration induced stress failures of socket welds were well understood at the site and in the industry. The licensee's extent of condition review determined that forty welds were susceptible to this type of failure. Additionally, corrective actions from a previous CR were incorporated to restore degraded piping support brackets back to their original condition. This deficiency was found prior to the pipe weld failure, however, no evaluation was performed to determine if this action would be beneficial or detrimental to the piping system's failure dynamics. Further, the licensee did not initiate actions to address recommendations submitted as part of a previously completed extent of condition analysis. Subsequent to these corrective actions a vibration induced stress weld failure occurred on August 3, 2002 which required the unit to shutdown to repair.

On August 27, 2002, an additional socket weld in the charging portion of the CVCS system failed. This socket weld is located on pressure transmitter tubing and was able to be isolated using an upstream isolation valve. A reasonable expectation of continued operability was generated which concluded that the CVCS system remained operable but not fully qualified due to the uncertainty of the system's long term pressure boundary reliability. Also, the operational flexibility of the CVCS system was affected in that operating system configurations for charging pump "A" were restricted.

Analysis

The inspectors determined that the failure to take adequate corrective actions in response to CVCS weld failures was more than minor because the failure mechanism reduced the reliability of the system and the systems' ability to perform its safety-related function. This issue is applicable to the mitigating system cornerstone because the CVCS system, a risk-significant, safety-related system, is required to respond to an anticipated transient without scram (ATWS) event. The inspectors evaluated the significance of the finding in accordance with the NRC's Significance Determination Process (SDP) under Manual Chapter 0609, Appendix "A," "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the finding was of very low safety significance because, although the system was isolated on August 3, 2002, as a result of the failure, manual operation could have been used to restore the system if an ATWS condition had occurred while the Unit was at power. Also, on August 27, 2002, the weld failure was isolable and did not affect the ATWS safety function. The issue screened to Green in the Phase I SDP because mitigating system equipment remained operable, there was no loss of safety function, and no technical specification limiting conditions for operation was exceeded.

Enforcement

10 CFR 50, Appendix "B", Criterion XVI requires, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this requirement, after the failure of CVCS welds in July 1999 and November 2001, the licensee failed to take adequate corrective actions to correct this condition adverse to quality. A subsequent weld failure on August 3, 2002 forced the unit to shutdown for repairs, and then another socket weld failure occurred on August 27, 2002, which resulted in restrictions on the use of the CVCS system. The licensee has since designed and installed a clamp on the charging discharge header in an attempt to "detune" the system. Vibration levels have been significantly reduced. The licensee has also established a CVCS Charging Header Investigation Team to investigate and recommend long term solutions for the charging header weld failure issues. The failure of the licensee to take adequate corrective actions for the CVCS system weld failures is considered a violation of 10 CFR 50, Appendix "B", Criterion XVI (**NCV 50-336/02-05-04**). This violation is associated with an inspection finding that is characterized by the significance determination process as having very low safety significance (Green) and is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as CR-02-08251.

.2 Steam Leak in the Turbine Driven Auxiliary Feedwater Pump Steam Trap Gasket

a. Inspection Scope

The inspectors reviewed condition reports (CRs) M2-00-0258, CR-01-08544, CR-01-08552, CR-01-09119, CR-01-10308, and CR-01-10376 regarding issues associated with a steam leak in the turbine driven auxiliary feedwater pump steam trap gasket, due to using improper gasket material. The inspectors verified that the licensee was identifying, evaluating, and correcting problems associated with this issue and that the corrective actions were appropriate. This issue was identified in Inspection Report 50-336/01-07 as a non-cited violation and was selected for follow-up review due to ineffective corrective actions and subsequent recurrence of the steam leak problem.

The inspectors verified that corrective actions were implemented by the licensee and were commensurate with the significance of the issue. The inspectors also reviewed the licensee's actions regarding extent of condition, generic implications, timeliness of corrective action, actions to prevent recurrence, and identification of the root and contributing causes of the problem. The inspectors discussed the human performance issues associated with the steam leak with the licensee.

b. Findings

No findings of significance were identified.

.3 Maintenance Rule Issues

a. Inspection Scope

The inspectors reviewed a sample of corrective action reports shown in Attachment 1 which identified problems related to maintenance rule issues. The inspectors verified that problems with SSCs in the maintenance rule scope were being identified, evaluated, appropriately dispositioned and entered into the corrective action program.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

.1 Maintenance Rule Exit Meeting

The inspectors presented the inspection results to Mr. Denny Hicks, Director, Nuclear Station Safety and Licensing, Millstone Station, and other members of the licensee's staff on July 26, 2002.

.2 Resident Exit Meeting Summary

The inspectors presented the inspection results to Mr. Alan Price and other members of licensee management on October 21, 2002. The inspectors asked the licensee whether any material examined during this inspection should be considered proprietary. No proprietary information was identified.

## ATTACHMENT 1

## SUPPLEMENTAL INFORMATION

a. Key Points of Contact

| | |
|---|---|
| J. A. Price | Site Vice President - Millstone |
| D. Hicks | Director - Nuclear Station Safety and Licensing |
| S. Sarver | Acting Director - Nuclear Station Operations and Maintenance |
| S. Scace | Director - Nuclear Engineering |
| P. Dillon | System Engineer |
| R. McIntosh | Licensing |
| T. Ryan | Maintenance Rule Coordinator, Unit 2 |
| K. Yearwood | Maintenance Rule Coordinator, Unit 3 |

b. List of Items Opened, Closed and Discussed

Opened

| | | |
|---|---|---|
| 50-336/02-05-02 | URI | Licensee’s failure to implement appropriate preventive maintenance to the EDG exhaust damper solenoid valve (1R12.2) |

Opened and Closed

| | | |
|---|---|---|
| 50-336/02-05-01 | NCV | Licensee’s failure to adequately establish, implement, and maintain procedures covering the cleaning, inspection and lubrication of couplings (1R12.1) |
| 50-336/02-05-03 | NCV | Licensee's failure to correctly conclude that the proposed alternate injection path would subject the "A" HPSI piping and nozzle to thermal transients which were outside of the FSAR design basis (1R14.1) |
| 50-336/02-05-04 | NCV | Licensee's failure to take adequate corrective actions to promptly identify and correct CVCS weld susceptibility to fatigue failures (4OA2.1) |

c. Partial List of Documents Reviewed

Fire Protection Evaluations Reviewed

FP-EV-98-001, "Unsealed Penetrations in the Appendix R Fire Barriers Separating Appendix R Areas R-3 and R-12"
FP-EV-98-005, Revision 2, "Partial Suppression and Partial Detection in Appendix R Fire Area R-3"
FP-EV-98-0007, Revision 0, "Fire Protection Evaluation for Partial Suppression in Appendix R Fire Area R-14"

FP-EV-98-0012, Revision 0, "Technical Evaluation for the Lack of Fire Dampers in Ductwork Penetrating the Appendix R Boundaries Between the South LPSI Pump Room and the HPSI Pump Room"
FP-EV-98-0015, Revision 0, "Technical Evaluation for the Non-Fire Rated Water-Tight Door in the Appendix R Wall Separating the Auxiliary Building (-) 45' General Area and the South LPSI Pump Room"
FP-EV-98-0016, Revision 0, "The Existence of an Open Grate Floor in the Appendix R Boundary Separating the North LPSI Pump Room and the Recirculation Valve Access Area"
FP-EV-98-0017, Revision 0, "Technical Evaluation for the Existence of a Removable Concrete Block Wall Section in the Appendix R Wall Separating the South LPSI Pump Room and the HPSI Pump Room"
FP-EV-98-0019, Revision 0, "Technical Evaluation for the Existence of a Removable Concrete Block Wall Section in the Appendix R Wall Separating the Auxiliary Building (-) 45' General Area and the South LPSI Pump Room"
FP-EV-98-0021, Revision 0, "Technical Evaluation for the Lack of Fire Dampers in duct work Penetrating the Appendix R Wall Between the North and South "LPSI" Pump Room"
FP-EV-98-0035, Revision 0, "Technical Evaluation for the Removal of the Fire Barrier Penetration Seal from the Train "B" Containment Sump Recirculation Suction Header Encapsulation Pipe Penetration in the Fire Boundary Wall Separating the Train "A" and Train "B" LPSI Pump Room"
FP-EV-98-0041, Revision 0, "Metal Partition Wall in the Charging Pump Cubicle"

Maintenance Risk Assessments and Emergent Work Evaluation

MP-14-OPS-GDL02, Revision 005, "Operations Standards"
MP-20-MMM, Revision 001, "Work Management"
MP-20-WM-FAP02.1, Revision 005-01, "Conduct of On-Line Maintenance"
MP-20-WM-SAP02, Revision 1, "On-Line Maintenance"
M2-EV-99-0093, Revision 04, "Evaluate Compensatory Measures to Use During Loss of Cooling/Ventilation Systems Supporting Vital Switchgear Room"
CR-02-08302, Unit 2 Unplanned Entry into PRA Condition Orange
M2-EV-02-0029, Revision 0, "HPSI Pump Availability during Surveillance Testing"
SPROC ENG02-001, Maintenance Rule (a)(4) Re-Scoping Project
CR-02-08327, Risk Color Change for FEGs 2350X10 and 2350X20 From Yellow to Orange in EOOS

Reactor Shutdown Due to Chemical and Volume Control System Charging Header Weld Failure

CR-02-08161, Recent Operation of Alternate Charging Not Bounded by Existing Fatigue Analysis
10 CFR 50.59 Screen Form dated 08/03/02, Change to procedures to allow use of an alternate charging path
Engineering Specification for Reactor Coolant Pipe and Fittings
Design Specification for Nuclear Piping Systems
Technical Evaluation for Assessment of Fatigue Usage for Alternate Charging Through the Safety Injection Nozzles, Millstone Unit 2

RAC-12, Revision 003-01, 50.59 Screens and Evaluations
OP-2304A, Revision 019-06, Volume Control Portion of CVCS
OP-2304A, Revision 019-07, Volume Control Portion of CVCS
OP-2205, Revision 013, Plant Shutdown
OP-2206, Revision 010-06, Reactor Shutdown

Automatic Reactor Trip on Low Steam Generator Level

CR-02-08189, Millstone Unit Two Automatic Reactor Trip on Low Steam Generator Level
Event Review Team Report on CR-02-08189
Plant Computer traces
Non-Emergency Report Form No. 2002040, Automatic Reactor Trip due to Low Steam Generator Level
M2-EV-02-0032, Revision 0, Technical Evaluation for 2-FW-1B lost internals on feedwater system, Millstone Unit 2
OP2321, Revision 017, Main Feedwater System
EOP 2525, Revision 20, Standard Post Trip Actions

Steam Leak in the Turbine Driven Auxiliary Feedwater Pump Steam Trap Gasket

| | |
|---|---|
| M2-00-0258 | A Steam Leak in the Terry Turbine Auxiliary Feed Pump Room From Blown Gasket on Steam Trap ST-156; Made Terry Turbine Inoperable. |
| CR-01-08544 | TDAFW Pump Inoperable Due to Steam Leak on Supply Header Trap ST-156. |
| CR-01-08552 | Stock Code Calls for Spiral Wound Gasket, Gasket in Stock is Garlock. |
| CR-01-09119 | During MEPL Review of SI Tank T39D Possible Improper Material Was Noted. |
| CR-01-10308 | Ineffective Corrective Action for Resolving Preferred Steam Trap Gasket Material. |
| CR-01-10376 | A/R Was Set to Complete, But Actions Had Not Been Completed. |

Maintenance Rule

Periodic Assessment of Maintenance Rule Program, October 1999 through August 2001
CR M2-99-3180 Monitor/Minimize Maintenance Rule Risk Significant System Unavailability Time
Maintenance Rule Action Plan (CR M3-00-1455) 125 Volt DC System (3345C)
Maintenance Rule Action Plan (CR M3-98-0323) Service Water System
Maintenance Rule Action Plan (CR M2-00-3099) Main Steam System (2316)
Maintenance Rule Action Plan (ACR 05498) Main Steam Isolation Valves (3316A)
Maintenance Rule Action Plan (CR M3-97-0802) Containment Isolation System (3312A)
Maintenance Rule (a)(1) Evaluation (ACR 6361) Chilled Water System (2330C)
Maintenance Rule Action Plan (ACR 6364) Process and Area Radiation Monitoring (2404)
Maintenance Rule Action Plan (CR M3-98-3011) Radiation Monitoring System (3404)
Corrective Action Reports, Maintenance Rule Activities, 01-07564, 01-11021, 01-10347 and 99-3180 (Identified in Periodic Assessment)
System Health Report - Containment Isolation, 04/10/2002

System Health Report - Service Water, 04/10/2002
Expert Panel Meeting Minutes - December 19, 2001
Expert Panel Meeting Minutes - December 21, 1999
Expert Panel Meeting Minutes - April 16, 2002 (Includes Scoping Tables)
Maintenance Rule Unavailability Monitoring - Service Water Pumps, Unit 3, A, B, C, and D
Maintenance Rule Unavailability Monitoring - 125 Volt DC, Unit 3, Batt/Bus 1, 2, 3, 4, and 5
Maintenance Rule Unavailability Monitoring - Station Blackout Diesel, Unit 3
Maintenance Rule Action Plan (CR M3-00-1520) Station Black Out Diesel Generator
Maintenance Rule Periodic Assessment MP-24-MR-FAP740, REV. 000
Maintenance Rule Goal Setting and Monitoring MP-24-MR-FAP730
Millstone Station Maintenance Rule Program OA 10, Rev. 003
Work Order M39817738 Quarterly CVCS Stroke Testing
Work Order M39920134 AFW Motor Bearing Oil Change
Work Order M30019463 AFW Motor Bearing Oil Sampling and Change
Work Order M29803093 Diesel Room Ventilation, Inspect Casing, Rotating Assembly
Work Order M29805601 Emergency Diesel Generator "A," semi annual elect PM
Work Order M29805615 Emergency Diesel Generator "A," semi annual mechanical PM

Refueling and Outage Activities

OP-2202, Revision 019-08, Reactor Startup
Event Review Team Report, CR-02-08189, Millstone Unit 2 Automatic Reactor Trip on Low Steam Generator Level
Plant Process Computer parameters, and Control Room Shift Logs
Chemistry Form 2859-001, Revision 004, Offsite Doses From Unit 2 Vent Iodine and Particulate Releases
EN 21221, Revision 002-01, Check Valve Examination and Testing
CR-02-08761, Unit 2 Charging Header Leak Downstream of 2-CH-425
AOP 2568, Revision 007, Reactor Coolant System Leak
MP-26-EPA-REF02, Millstone Unit 2 Emergency Action Level (EAL) Technical Basis Document
Forced Outage and Startup Schedules

Reactor Coolant System Resistance Temperature Detectors

IC2417L, Revision 006, "Primary Coolant System RTD Data Collection"
CR-M2-97-2750, Discrepancies found in several TS reviews relating to RPS
CR-02-07563, Lapsed Surveillance on all four channels of "Delta T" Power, TMLP, LPD, and Reactor Cold Leg Temperature Indication
Branch Technical Position HICB-13, Guidance on Cross-Calibrations of Protection System Resistance Temperature Detectors
Technical Specifications
FSAR
Calculation PA-XX-XXX-09776E, Revision 2, "Cold Leg Temperature Loop Accuracy"
Calculation PA-XX-XXX-0964GE, Revision 2, "Hot Leg Temperature Loop Accuracy"

Performance Indicator Verification

MP-16-PI-GDL01, Revision 001, "Maintenance of NRC Performance Indicators"
Licensee HPSI Performance Indicator Data July 2001 - June 2002
System Health Reports
NEI 99-02, Revision 1, "Regulatory Assessment Performance Indicator Guidelines
NRC web site Performance Indicators
PRA Memorandum NE-02-F-50 dated March 19, 2002, "Runout HPSI Flows Modeled in Analysis with Maintenance Rule Unavailability Applicability"
CR-01-09987, Discrepancies Found During NRC Audit of HPSI/RHR Unavailability Hours Over Last 12 Months
Licensee Shutdown Cooling and Containment Spray Performance Indicator Data July 2001 - June 2002

Failure of a Weld in the Unit 2 Charging System 'A' Header

Corrective Action Reports
AR 01004795
M2-99-2053
CR-02-07215
CR-01-11536
CR-02-08076
AR-01004979
CR-01-06692
CR-02-01003
M2-99-1734
CR-01-11536
M2-99-1752
CR-01-06459

Miscellaneous Documents
M2-EV-99-0120 - Acceptance Criteria for Pipe Vibration Monitoring of Charging system Piping
25203-309-001 - Installation, Operation and Maintenance of Reciprocating Charging Pumps
Charging System Health Report Second Quarter 2002
Charging System Maintenance Rule Scope Report
MP2-070-01 - Operability Determination
MTACONFIG-01-253 - Failure Analysis of Pipe Support 413971
NME-WM-99-227 - Fractograph Examination in ¾-CCB-6 of "B" Charging Pump
SP 260011, Rev. 001-06 - Charging Pump Inservice Tests
SP-21162-1, Rev. 3 - Volume Control Portion of the CVCS Leakage Test
Fig 09.02-02 Piping and Instrumentation Diagram - Charging System
84-RPS-341-GM, Rev. 1 - Calculation for Available NPSH at Suction of Charging Pumps
Engineering work request
M2-01-12071
M2-01-0009 ALARA Shield
M2-00-18767 'C' Charging Pump Pipe Weld
M2-00-18764 Welding Reinforcement of 'B' Charging Pump Piping

d. List of Acronyms

| | |
|---|---|
| AFW | auxiliary feedwater |
| A/R | action requests |
| ATWS | anticipated transient without scram |
| AWO | automated work order |
| CI | containment isolation |
| CR | condition report |
| CVCS | chemical and volume control system |
| EDG | emergency diesel generator |
| EOOS | equipment out of service |
| FSAR | Final Safety Analysis Report |
| HPSI | high pressure safety injection |
| LEAC | loss of one division of emergency ac |
| LOOP | loss of offsite power |
| LPSI | low pressure safety injection |
| MCC | motor control center |
| MDAFP | motor-driven auxiliary feedwater pump |
| MS | main steam |
| MSIV | main steam isolation valve |
| OD | operability determination |
| P&ID | piping and instrumentation diagram |
| PC | performance criteria |
| RBCCW | reactor building closed cooling water |
| RPS | reactor protection system |
| RTD | resistance temperature detector |
| SDP | significance determination process |
| SI | safety injection |
| SP | surveillance procedure |
| SSCs | structures, systems and components |
| SW | service water |
| TS | technical specification |
| TDAFW | Turbine Driven Auxiliary Feedwater |
| VTM | vendor technical manual |

**ENCLOSURE 2**

**U.S. NUCLEAR REGULATORY COMMISSION**
**REGION I**

| | |
|---|---|
| Docket No.: | 50-423 |
| License No.: | NPF-49 |
| Report No.: | 50-423/02-05 |
| Licensee: | Dominion Nuclear Connecticut, Inc. |
| Facility: | Millstone Power Station, Unit 3 |
| Location: | P. O. Box 128<br>Waterford, CT  06385 |
| Dates: | June 30, 2002 - September 28, 2002 |
| Inspectors: | A. C. Cerne, Senior Resident Inspector, Unit 3<br>B. E. Sienel, Resident Inspector, Unit 3<br>M. C. Barillas, Reactor Engineer, Division of Reactor Projects (DRP)<br>T. F. Burns, Reactor Inspector, Division of Reactor Safety (DRS)<br>A. L. Burritt, Senior Resident Inspector, Limerick 1 and 2<br>P. C. Cataldo, Resident Inspector, Unit 2<br>L. S. Cheung, Senior Reactor Inspector, DRS<br>J. M. D'Antonio, Operations Engineer, DRS<br>M. S. Ferdas, Reactor Inspector, DRS<br>E. H. Gray, Senior Reactor Inspector, DRS<br>K. M. Jenison, Senior Projects Engineer, DRP<br>A. X. Lohmeier, Reactor Inspector, DRS<br>A. C. McMurtray, Senior Resident Inspector, Peach Bottom 2 and 3<br>T. A. Moslak, Health Physicist (DRS)<br>S. Pindale, Sr. Reactor Inspector, DRS<br>S. M. Schneider, Senior Resident Inspector, Unit 2<br>G. C. Smith, Senior Physical Security Inspector, DRS<br>R. J. Urban, Senior Enforcement Specialist, Region I |
| Approved by: | Brian J. McDermott, Chief<br>Projects Branch 6<br>Division of Reactor Projects |

SUMMARY OF FINDINGS

IR 05000423-02-05; Dominion Nuclear Connecticut, Inc.; on 06/30-09/28/02; Millstone Power Station; Unit 3; Personnel Performance During Non-routine Plant Evolutions, Safety System Design and Performance Capability.

The inspection was conducted by resident and regional inspectors. The inspectors identified two green issues, that were also determined to be Non-Cited Violations. The significance of most findings is indicated by the color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

## A. Inspector Identified Findings

### Cornerstone: Barrier Integrity

- **Green.** The inspectors identified a Non-Cited violation (NCV) of technical specifications (TS) 6.8.1 for an inadequate operating procedure, which resulted in a failure to maintain an isolated reactor coolant system (RCS) loop pressure below its TS required pressure limit.

  The finding impacted the Barrier Integrity Cornerstone and had an actual impact of exposing an isolated RCS loop to a pressure that exceeded a pressure-temperature limit delineated in the TS. The finding was of very low safety significance (Green) because there was no adverse impact on the structural integrity of any RCS components and the requirements of TS were met. Because the finding is of very low safety significance and was captured in the licensee's corrective action program, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (Section 1R14).

### Cornerstone: Mitigating Systems

- **Green.** The inspectors identified a Non-Cited violation (NCV) of 10CFR50, Appendix "B", Criterion III - Design Control, concerning a failure to evaluate the ability of the service water piping to withstand a column separation water hammer. Specifically, the licensee failed to evaluate whether certain portions of the service water return piping from the recirculation spray system were susceptible to transient loads in excess of those described in design basis structural integrity limits.

  The finding impacted the Mitigating Systems Cornerstone and had the potential to reduce the reliability of service water cooling to the recirculation spray system. However, this finding was determined to be of very low safety significance (Green) because a subsequent operability determination concluded that the affected piping system would remain functional under postulated accidents conditions. Because the finding is of very low safety significance and was captured in the licensee's corrective action program, this finding is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (Section 1R21).

## B. Licensee Identified Violations

A violation of very low safety significance, which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking number is listed in Section 4OA7 of this report.

## Report Details

### SUMMARY OF UNIT 3 STATUS

The plant began the inspection period on June 30, 2002, operating at approximately 100 percent power. On August 15, the plant began coastdown operation in preparation for Refueling Outage 8 (3R08). On September 5, with the reactor at approximately 83 percent power, operators performed a manual reactor shutdown, placing the reactor in Mode 5 (Cold Shutdown) on September 7 and Mode 6 (Refueling) on September 11. At the end of the inspection period on September 29, the reactor was in Mode 5 in preparation for restart.

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspector reviewed the licensee's preparation for adverse weather relative to the protection of safety-related systems, structures, and components (SSCs) from weather-related risks identified for the site including tornado, hurricane, high winds, extreme high or low temperatures, and extreme ultimate heat sink conditions. This review included a walkdown of the SSCs in the intake structure including traveling screens, service water pumps, circulating water pumps and area heaters to verify implementation of cold weather features to ensure continued operability during adverse weather. Also, the inspector verified equipment and components located in areas exposed to outside weather were adequately protected from high winds and high wind generated missiles. In addition, the inspector verified that surveillance of the intake SSCs within the structure and current weather conditions were performed at the specified frequency and actions were taken to address deficiencies identified.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

The inspector reviewed the activities and documentation associated with the replacement of Battery 3 and Battery 4, each containing 60 cells, to determine the adequacy of the replacement process. These batteries supply power to Vital dc Buses 301A-2 and 301B-2 which in turn supply power through inverters to 120 Vac Vital Buses VIAC-3 and VIAC-4 upon loss of ac power. At the time of the inspection, the licensee planned to replace the batteries during the September 2002 outage because the equipment was approaching the end of its qualified life (20 years). The batteries and associated equipment were selected for review because they were part of the system that contributes most to the prevention of core damage.

The existing batteries were original equipment, installed during plant construction, and manufactured by GNB. The replacement batteries were also from the same manufacturer, but with a different part number and had higher capacity (Model NCN-11 with 825 AH instead of Model NCX-750 with 750 AH.) The licensee did not consider this activity to be a plant modification, and therefore, design changes were not required. The inspector reviewed the equivalency evaluation completed by Nuclear Logistics Inc., Fort Worth, Texas, to verify that the physical sizes, weight, and material met the one-for-one equivalency replacement criteria. The inspector also reviewed the purchase order (No. 03005026) to verify that the replacement batteries were the same as those being evaluated.

In addition, the inspector conducted a walkdown of both sets of batteries, the seismically qualified battery racks, the battery room ventilation systems, and the interface equipment such as the battery chargers and inverters to observe their physical and material conditions. To assure that the existing batteries were in good operating condition, the inspector reviewed the test records of the past 18-month service tests (performed on February 3, 2001, and February 14, 2001) and 5-year capacity tests (performed on May 2, 1999, and April 19, 1999). The inspector also verified that the test anomalies associated with the service tests (incorrect discharge amperes) were appropriately documented and adequately justified in Condition Report CR 01-07742.

b. Findings

No findings of significance were identified.

## 1R05 Fire Protection

### .1 Emergency Diesel Generators

a. Inspection Scope

The inspector performed walkdowns of Fire Areas EG-3 and EG-4 associated with the "A" and "B" Emergency Diesel Generators (EDG). In addition, delivery system piping and equipment that provide the automatic fire suppression for EDG instrumentation inputs into the plant computer and instrument rack rooms in the control building were also inspected. The inspector confirmed that fire detection and suppression equipment located in these areas was consistent with and met the requirements of the Millstone 3 Fire Protection Evaluation Report (FPER). Numerous pieces of supporting documentation were also reviewed to ensure that compensatory measures were in accordance with the Millstone Unit 3 Technical Requirements Manual (TRM).

b. Findings

No findings of significance were identified.

.2 Circulating and Service Water Pump House

a. Inspection Scope

The inspector performed a walkdown of the following plant areas to observe conditions related to fire protection:

- Circulating and Service Water Pump House-East Service Water Cubicle, 14-foot 6 inch elevation, Fire Area CSW-3,
- Circulating and Service Water Pump House-West Service Water Cubicle, 14-foot 6 inch elevation, Fire Area CSW-4, and
- Circulating and Service Water Pump House, North Floor Area, Sodium Hypochlorite Room and Service Water Valve Access Enclosure, 14 foot 6 inch elevation, Fire Area CSW-1.

These areas were selected for inspection because risk significant systems, structures, and components were located in these areas. The inspector verified the availability and operational status of manual fire fighting equipment in these and adjacent areas. Also, the inspector assessed the licensee’s control of transient combustibles and ignition sources and evaluated the material condition of the areas.

b. Findings

No findings of significance were identified.

.3 MCC and Rod Control Areas

a. Inspection Scope

The inspector conducted walkdowns of Fire Areas AB-5 and 6, the east and west MCC and Rod Control Areas, at elevations 24'-6" and 43'-6" of the Auxiliary Building. The inspector confirmed that fire detection and suppression equipment located in the areas was as specified in the Millstone 3 FPER. The inspector discussed with the cognizant licensee fire protection engineer the fire rating and qualification of the foam material filling the seismic ("shake-space") gap between the containment and the AB-5 and 6 fire zones, evaluated the hose reel equipment available to the fire brigade from Fire Hose House MP3-10 to supply the committed suppression water supply to Area AB-6, and checked the emergency lighting units in these areas. The inspector noted no equipment out of service or degraded components that would require the implementation of compensatory measures (e.g., hourly fire roves) in accordance with the Unit 3 Technical Requirements Manual.

b. Findings

No findings of significance were identified.

.4 Response to Potential Fire

a. Inspection Scope

The inspector also reviewed the on-shift operators and fire brigade response to an electrical ground that tripped a 4.16 kV supply breaker to a non-safety related load center (32G) and created smoke and an unknown fire condition in the normal switchgear room of the Unit 3 service building. The appropriate emergency operating procedure (EOP 3509) was confirmed to be used by the control room operators to direct response actions. The inspector checked the fire brigade response actions and interviewed a member of the fire brigade regarding assigned duties and contingency actions. The inspector also verified the stationing of a continuous fire watch in the area until the 32G load center could be safely accessed for inspection and fire checks. Other appropriate compensatory measures were noted to have been established by the operations unit supervisor, in accordance with the provisions of the Unit 3 Technical Requirements Manual. The inspector confirmed that the over-current ground condition and subsequent load center trip were documented in condition report CR-02-09791 for follow-up corrective actions and analysis.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspector selected the intake structure, which contains two separate safety-related service water pump cubicles, to evaluate its ability to withstand internal and external flooding. The intake structure is below the design basis flood level of +23.8 feet, and is therefore subject to external flooding as well as internal flooding.

Through inspections of the two service water pump cubicles, the inspector confirmed that the system was configured as described in the Final Safety Analysis Report (FSAR) and piping and instrumentation diagrams (P&IDs). The inspector verified that penetrations were properly sealed, watertight doors were adequate, and there was no communication between the two trains of service water such that a flood in one cubicle would render the other train of service water inoperable in the adjoining cubicle. The inspector also verified that the severe weather procedure could be used to reasonably cope with external flooding conditions. The inspector reviewed the following licensee documents:

- AOP 3569, "Severe Weather Conditions," Rev. 14
- 01-ENG-01884M3, "MP3 Service Water Pump Cubicle Internal Flooding Evaluation," Rev. 0
- P(R)1196, "Potential for Cross Cubicle/Building Flooding Via the Equipment and Floor Drainage Systems for ESF, Auxiliary, Fuel, Waste Disposal, Diesel Generator, Service, Control and Intake Buildings"
- P(R)1072, "Service Water Cubicle Flooding Hazards Analysis (Internal Flooding)"

- Millstone Roof Flooding Calculations
- Preventive Maintenance - "Service Water Cubicle Watertight Doors"
- EN-31098, "Annual PM - Roof Inspection of the Circulation and Service Water Pump House," Rev. 3

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspector observed the performance of special procedure (SPROC) 99-3-11, (EN 31169), 3HVQ*ACUS2A Condenser Thermal Performance Test, for the Engineered Safety Features (ESF) Building, "2A" Emergency Air Conditioning Unit. This air conditioning unit is one of two units that cool the containment recirculation pumps and coolers in the ESF building when the containment recirculation pumps are operating during design basis accident conditions. The inspector verified that the recorded test data matched the readings on temporary and permanent plant instruments connected to the air conditioning unit to monitor performance. The inspector reviewed the recorded test data to verify that the heat removal capability met the system design specified in the Millstone 3 Final Safety Analysis Report (FSAR) and vendor heat load calculations. The inspector verified that the heat exchanger performance test methodology and acceptance criteria for this air conditioning unit were consistent with accepted industry practice. The inspector reviewed the test documentation for potential deficiencies which could mask degraded performance and common cause performance problems.

The inspector also reviewed the previous test records associated with this air conditioning unit to assess whether the licensee was meeting their commitments to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

Activities inspected during the Millstone Unit 3, refueling outage number 8 (3R08) included steam generator tube eddy-current testing (ECT), reactor pressure vessel (RPV) closure head penetration visual examination (VT), ultrasonic tests (UT) of pressurizer and main steam piping welds, magnetic particle tests (MT) on FWS-17-FW-80, visual/video examination of the pressurizer heater penetrations of the lower pressurizer head, and radiographic testing (RT) of reactor coolant system (RCS) charging pump system check valves. The objective of the inspection was to verify the effectiveness of the inservice inspection (ISI) program in monitoring RCS and risk significant boundary degradation.

The inspector assessed the effectiveness of the licensee's ECT program, procedures, and inspection activities for monitoring the condition of steam generator tubes. This assessment was based on the rules and regulations of the steam generator examination program for the Unit 3 steam generator examination guidelines, NRC Generic Letters, the Code of Federal Regulations 10CFR50, the Technical Specification for Millstone Unit 3, and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Sections V and XI. Supporting the assessment, were parts of EPRI PWR steam generator examination guidelines, and the Millstone Unit 3 steam generator integrity degradation assessment for 3R08.

To evaluate steam generator tube integrity, the inspector reviewed the licensee's plans regarding steam generator repair criteria, the ECT program scope and procedures, and the previous operating cycle performance. The inspector reviewed the licensee awareness of types of degradation experienced from past site and industry-wide operating experience to identify potential problem areas. The inspector reviewed the types of ECT probes used by the licensee. The inspector observed the attention given to finding foreign materials and evaluation of affected tubes. The 3R08 steam generator outage activities, including ECT scope, were also compared to the appropriate EPRI and NRC guidelines.

The inspector observed data gathering and control of specific tube identification, reviewed a sample of the ECT results of the steam generators "A" and "C" tubes, and observed the work of a resolution analyst and the Independent Qualified Data Analyst (QDA) in evaluating the findings of the primary and secondary analyst teams. The inspector confirmed that tube retention or plugging was performed in accordance with established repair criteria limits.

The licensee's activities performed in response to NRC Bulletins 2001-01 and 2002-02, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," were inspected against the requirements of Temporary Instruction (TI) 2515/145. The description of the inspection scope and results is in Section 4OA5 as specified by the TI.

The pressurizer head to support skirt weld and main steam system pipe to end cap weld MSS-33-FW-1-HM weld conditions in the plant were observed and the UT calibration blocks, UT procedure, calibration process, and inspection results were examined. The UT data sheets for welds ID 03-007-SW-Z, pressurizer longitudinal shell weld, and ID 03-007-SW-D, pressurizer relief nozzle to head weld, were also reviewed. The UT inspection method, acceptance criteria, and documentation for these tests were reviewed.

The radiographs of charging pump system check valves to confirm closure as part of the IST program were reviewed. The adequacy of the disposition and the sample expansion for a MT identified unacceptable indication at the toe of weld FWS-17-FW-80 in the feedwater system were reviewed.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspector observed a simulator exam conducted as part of licensed operator requalification training. The inspector observed operator use of emergency and abnormal operating procedures in response to a failed controlling channel feedwater flow instrument followed by a steam generator tube rupture. The inspector discussed the scenario and training objectives with training personnel and attended the trainees' critique following the scenario.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

.1 Periodic Evaluation

Refer to NRC Inspection Report 50-336/02-05, Millstone Unit 2, Section 1R12.3 for specific details.

.2 Chemical and Volume Control System

a. Inspection Scope

The inspector reviewed licensee actions taken in response to condition report CR-01-11768, which was written to request an evaluation for the chemical and volume control system (CVCS) to be placed in a(1) status. The inspector reviewed the maintenance rule scoping documents for the CVCS and charging pump component cooling (CCE) systems and the licensee's a(1) evaluation and action plan to confirm appropriate goals were set. The inspector also reviewed the corrective actions completed to date in response to the identified equipment problems which led to the a(1) condition. The licensee concluded that the CCE system, currently monitored as part of the CVCS system, should be designated as an a(1) system.

b. Findings

No findings of significance were identified.

## 1R13 Maintenance Risk Assessments and Emergent Work Evaluation

### .1 "A" Residual Heat Removal (RHR) Motor and Pump Rotating Assembly Replacement

a. Inspection Scope

The inspector assessed the maintenance risk associated with the planned corrective maintenance activities involving the replacement of the "A" RHR motor and pump rotating assembly. This maintenance activity involved using a mobile crane to remove engineered safety features (ESF) building roof plugs and also to remove and replace the RHR pump assembly. The inspector reviewed Technical Evaluation M3-EV-02-0019 which addressed the location of the portable crane, the lifting and lay down of the ESF building roof plugs, and lifting of the pump assemblies including a safe load path to the transport staging area. The inspector also reviewed clearance order 3C08- RHS01-0022.

The inspector checked the work area for the "A" RHR pump to verify that adjacent safety-related equipment was not adversely impacted by the pump replacement activities. The inspector reviewed the work planning and execution including compensatory actions taken to address schedule delays as a result of fit up problems with the new pump. The inspector verified that the compensatory action of reinstalling the ESF building roof plugs and testing the secondary leakage collection system was consistent with technical specification requirements. The inspector interviewed cognizant engineering personnel regarding the reason for and resolution of the pump fit up problem as well as the planned post maintenance testing. The inspector utilized the Equipment Out of Service (EOOS) quantitative risk assessment tool to evaluate the risk of the above plant configuration and compared the result to the licensee’s stated risk. The inspector also verified that the schedule delays did not invalidate the risk assessment performed prior the start of maintenance.

b. Findings

No findings of significance were identified.

### .2 Emergent Work

a. Inspection Scope

The inspector reviewed the work planning and corrective maintenance activities for emergent work items associated with the following CRs:

- CR-02-06421 Oil leakage from the inboard end of the MDFWP (3FWS-P1)
- CR-02-07226 Elevated temperature in terry turbine steam supply valve room
- CR-02-07705 "A" EDG auxiliary fuel oil pump loss of power alarm

The inspector also conducted a risk assessment of the Mode 5 maintenance activities conducted on the containment purge exhaust and supply valves. While the Unit 3 technical specifications require these containment isolation valves (CIV) to be operable only in Mode 4 and above, questions arose during the maintenance planning for this valve work as to whether technical specification (TS) 3.3.2 delineated containment isolation instrumentation requirements that prohibited the CIV work in Mode 5. The inspector reviewed a SORC approved licensing position (LR-02-080) on the TS questions and compliance issues related to this maintenance activity. The inspector also reviewed TRM Table 3.3-5 provisions for the instrumentation associated with the containment purge exhaust and supply valves, and evaluated the consistency of these technical requirements with the noted licensing position. The license conditions and applicable surveillance requirements delineated in TS 3.3.2 (Table 3.3-3), TS 3.6.1.7, TS 3.9.4, and TS 3.9.9 were checked to verify that the licensee's understanding of the lack of a technical requirement for a containment purge system isolation function, with the plant in Mode 5, was compatible with the documented regulatory controls and licensing basis.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspector observed operations in the control room during certain plant evolutions that were associated with the 3R08 refueling outage. The inspector checked operator performance for conformance with the planned actions and the steps directed by the following procedures:

- OP 3206 Plant Shutdown
- OP 3208 Plant Cooldown
- OP 3216 Reactor Coolant System Drain (IPTE)
- OP 3250.01 Individual Loop Drain and Fill

Adherence to selected procedural prerequisites and precautions were verified against the plant conditions and system lineups applicable to each evolution. During the performance of the plant evolutions noted above, the inspector assessed control room activities, including licensed operator communications, equipment manipulations using the main control board controls, response to alarms, and compliance with the applicable procedures.

On September 7, 2002, the inspector observed control room activities during the transition to shutdown cooling operations. The inspector discussed the evolution with the reactor operators and confirmed adherence to technical specification requirements and performance in accordance with approved procedures OP 3208; OP 3310A, Residual Heat Removal System; and SP 3601G.2, RCS and Pressurizer Heatup and Cooldown Rate.

For the infrequently performed test and evolution (IPTE), the inspector confirmed cognizance by specified personnel of their IPTE responsibilities. Because of the risk significance (i.e., reduced time to core boiling considerations) of the draindown controls for the reactor coolant system (RCS), the inspector verified the conduct of an operator briefing and discussion of the IPTE termination criteria delineated in OP 3216.

The inspector observed operator actions to suspend the individual "A" loop drain and respond to unexpected plant conditions on September 9, 2002. The inspector reviewed operator logs, plant computer data, and RCS piping and instrumentation diagrams (P&ID), and discussed system configuration details with the cognizant operators to evaluate operator understanding of the plant response to stopping the "A" loop draindown. Subsequently, operations management and engineering personnel were also interviewed regarding the adequacy of the OP 3250.01 procedural controls and engineering calculations were reviewed to evaluate the regulatory aspects and hardware impact of all unexpected plant conditions. In addition, a region-based inspector reviewed Calculation Number 02ENG-01939M3, "Structural Evaluation of the Unit 3 ’A’ Loop Due to an Overpressurization Event," on September 30, 2002. The inspector verified that the assumptions stated in the calculation were appropriate and consistent with the plant conditions at the time of the event, the proper engineering methods were used, and that there were adequate technical bases to support the conclusions contained within the calculation.

b. <u>Findings</u>

The inspector identified a non-cited violation of TS 6.8.1 in that OP 3250.01 was found to provide inadequate instructions for an activity referenced in Appendix A of USNRC Regulatory Guide 1.33, i.e., draining an RCS loop. The issue was determined to be of very low safety significance (Green).

<u>Introduction</u>

On September 9, 2002, during the draining evolution of the "A" RCS loop, using OP 3250.01, the inspector witnessed the control room operators’ response to an alarm indicating "RCP Seal Injection Flow Lo,” coincident with a main control board indication of lowering flow to the number 1 reactor coolant pump (RCP). This condition was determined to be caused by increasing pressure in the "A" RCS loop, which occurred when the operators stopped the "A" loop draindown approximately 30 minutes earlier due to leakage across the isolated boundary. The inspector verified that approximately 10 minutes after receiving the MCB alarm, the operators opened the loop 1 relief line isolation valve (3RCS*V13), which had been closed in accordance with the procedural steps of OP 3250.01, and thereby stopped the loop 1 pressurization.

Description

The inspector reviewed OP 3250.01 and noted that the implementation of the Section 1 instructions for "Draining Reactor Coolant Loop 1" did not provide cold overpressure protection for an isolated loop. Additionally, upon questioning from the inspector, the licensee conducted a bench test on the non-credited, backup overpressure protection device (a relief valve) and found that it did not lift at its rated setpoint. Further problems with the procedural instructions of OP 3250.01 were noted in that: (1) pressure indication on the isolated RCS loop was available to the operators, but was not required to be valved into the system; (2) although the procedure cautioned that the "isolated loop pressure shall not be allowed to exceed 1,000 psia," neither a caution regarding the proper sequence for stopping the draindown, nor a contingency for responding to a loop overpressurization were procedurally delineated for operator consideration after the relief valve isolation valve (3RCS*V13) was closed; and (3) conflicting provisions regarding the position of a valve (3RCS*V50) available for loop 1 overpressure protection, were identified in OP 3250.01 vs. P&ID EM-102A (reference: CR-02-10359).

Analysis

The inspector determined that the pressurization of the isolated RCS loop affected the Barrier Integrity Cornerstone and was more than minor because it subjected the loop to a pressure that exceeded the pressure-temperature limit delineated in TS. Subsequent licensee engineering calculations estimated the peak predicted pressure in loop 1 to have been 1723 psig. The licensee documented this event in CR-02-09226 and performed an evaluation of reportability in accordance with the requirements of 10 CFR 50.72 and 50.73. The inspector also determined that the licensee complied with the applicable TS 3.4.9.1 actions, for pressure/temperature limits of the RCS in Mode 5, by initiating the operator's actions necessary to meet the limitations imposed by TS Figure 3.4-3 and by performing an engineering evaluation of the overpressure effect upon the structural integrity of the affected RCS loop. Since NRC inspection of this engineering evaluation verified no adverse impact on the structural integrity of any RCS components and no residual effects that would prevent heatup of the plant to Mode 4, this finding of an inadequate procedure screens out as having very low safety significance (Green).

Enforcement

The Millstone Unit 3 Safety Technical Specifications Manual, in Section 6.8.1, specifies that written procedures shall be implemented, covering activities referenced in the applicable procedures recommended in Appendix "A" of Regulatory Guide (RG) 1.33, Revision 2, February 1978. RG 1.33, Revision 2, in Appendix "A", lists the need for adequate PWR system procedures providing instructions for (among other typical safety-related activities) draining the reactor coolant system. The inspector determined that the instructions documented in the procedure (OP 3250.01) used to drain an isolated portion of the RCS loop 1 on September 9, 2002, were inadequate in assuring compliance with an OP 3250.01 requirement to "maintain isolated loop pressure less than 1,000 psia at all times." The inspector determined that this procedural inadequacy represented a violation of TS 6.8.1 **(NCV 50-423/02-05-05)**. This violation is associated with an inspection finding that is characterized by the significance determination process as having very low safety significance (Green) and is being treated as a non-cited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. The event leading to this NRC violation is documented in the licensee's corrective action program as CR-02-09226.

1R15 Operability Evaluations

.1 Atmospheric Dump Valve Bypass Valve

a. Inspection Scope

The inspectors reviewed the operability determination associated with the atmospheric dump valve bypass valve to ensure that operability was justified and that the atmospheric dump valve bypass valve remained available and no unrecognized increase in risk had occurred. The inspectors also reviewed compensatory measures to ensure that they were in place and were appropriately controlled. The inspectors reviewed the following related licensee documents:

- CR-02-08666, "Not Able to Meet Performance Criteria of CMEB 9.5.1"
- 56.72 Report 39144, "Unanalyzed Condition Concerning Steam Generator Atmospheric Relief Bypass Valves"
- Non-Emergency Report Form 2002050, "Unanalyzed Condition Involving Steam Generator Atmospheric Dump Valves"
- LER 2002-003-00, "Inadequate Validation of Fire Safe Shutdown Analysis Assumptions"
- SPEC No. 25212-BIP-9.5.1, Section 3.1.8, "Decay Heat Removal Via Main Steam"

b. Findings

No findings of significance were identified.

.2 RHR Heat Exchanger Weld Defect

a. Inspection Scope

The inspector reviewed the licensee's initial operability determination (OD) MP3-011-02, which was initiated following the identification of a surface defect in a weld located on the support skirt for the "A" residual heat removal (RHR) system heat exchanger, 3RHS*E1A. The inspector evaluated the engineering basis that supported the licensee's conclusion that the "A" RHR heat exchanger continued to be operable with a weld defect in the support skirt. The inspector verified that the licensee had entered this issue into the corrective action program for resolution as CR-02-08652.

b. Findings

No findings of significance were identified.

.3 Solid State Protection System

a. Inspection Scope

The inspector witnessed the conduct of the Train "B" solid state protection system (SSPS) operational test, in accordance with surveillance procedure SP 3446B12, on August 9, 2002. During the conduct of this testing, questions arose regarding the as-left operability of the Train "B" reactor trip signals associated with the low shaft speed circuitry for all four reactor coolant pumps (RCP). After troubleshooting and corrective maintenance activities, the inspector confirmed the re-conduct of SP 3446B12 with test results validating the system operability in accordance with the surveillance test criteria. However, since no cause for the initial test failure could be definitively established, the inspector requested the licensee's basis for determining that the problem had been corrected and would not recur during continued plant operation.

On August 15, 2002, the inspector received and reviewed a copy of technical evaluation M3-EV-02-0026, SSPS Train "B" interim surveillance test interval, that documented the licensee's engineering analysis of the noted SSPS logic test failure. The inspector verified the licensee consideration of the history of "termi-point" connection and other logic card wiring problems, as well as the past failures of the Train "B" low shaft speed signal testing. An assessment of the safety significance of the recent problem and an evaluation of the risk associated with conducting the Train "B" SSPS surveillance test at an increased periodicity during power operations were documented in this technical evaluation. The inspector checked that the conclusions reached by the licensee and recorded in M3-EV-0026 were consistent with a determination of operability for the Train "B" SSPS system. The inspector also confirmed that an additional operational test (SP 3446B12) of the Train "B" SSPS was successfully completed with the plant at power and that further inspection and testing on the affected SSPS wiring and logic cards were conducted during 3R08.

b. Findings

No findings of significance were identified.

.4 Operability Determinations

a. Inspection Scope

The following operability determinations (ODs) 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.

- MP3-007-02 Inspection Performed on Fire Water Tank M7-6B Determined Degradation of Internal Tank Coating and Pitting Corrosion
- MP3-010-02 Pipe Wall Loss Found at Three Locations on Train "A" Service Water Piping
- MP3-012-02 SIL Cold Leg Injection Swing Check Valve Inspection Identified a Missing Anti-rotation Pin
- MP3-013-02 Core Mapping Identified a Rod Cluster Control Assembly in Incorrect Fuel Assembly

The fire water tank OD was initiated following the identification of internal coating and corrosion degradation.  The inspector determined that the tank remained available to supply water to the fire water systems on site based on the licensee's determination that the degradation was not sufficient to challenge the structural integrity of the tank.

For the service water piping wall loss, the inspector verified that the remaining wall dimensions exceeded the minimum design wall thickness and design basis stress requirements for the affected service water pipe and that the degraded pipe sections were replaced during 3R08.

For the pin found missing on the low pressure safety injection (SIL) system check valve, the inspector's review of the OD evaluated the licensee's consideration of the pin material and the potential adverse impact caused by the loose part on the operability of downstream safety systems and their components.  The inspector confirmed that the subject swing check valve, 2SIL*V012, was repaired during 3R08 with the complete replacement of the check valve internal disc arm assembly.

OD MP3-013-02 was initiated when a rod cluster control assembly (RCCA) was discovered to be in an incorrect fuel assembly.  The licensee discovered this condition during planned core mapping activities which are performed to verify proper fuel assembly and insert placement after refueling and before restart.  During refueling, the safety function of the RCCAs is to provide negative reactivity to maintain the required shutdown margin.  Based upon the refueling boron concentration maintained during the outage and the fact that no fuel assemblies were found to be in an incorrect location, the inspector determined that shutdown margin requirements were maintained.  The inspector confirmed that the RCCA was moved to its correct core location prior to restart.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

.1 "A" Residual Heat Removal (RHR) Motor and Pump Rotating Assembly Replacement

a. Inspection Scope

The inspector reviewed the post maintenance test (PMT) plan for the "A" RHR pump following replacement of the motor and pump rotating assembly, worked under automated work order (AWO) M3-01-06260.  The inspector reviewed the scope of the work activities, discussed the AWO with the system engineer and maintenance supervisor, and verified that the planned PMT was appropriate to restore the operability of the pump.

The inspector reviewed the following related licensee documents:

- Technical Evaluation M3-EV-02-0022, Assessment of Post-Maintenance Test Plan for 3RHS*P1A 2002 Seal Package Replacement.
- SP 3610A.1,Residual Heat Removal Pump 3RHS*P1A Operational Readiness Test

- NUREG-1482, Guidelines for Inservice Testing at Nuclear Power Plants
- P&ID 25212-26912, Low Pressure Safety Injection

On August 16, 2002, the inspector witnessed the performance of the operational readiness test for the newly installed residual heat removal pump, 3RHS*P1A. The inspector verified that the surveillance criteria for the pump test were met, that acceptable inservice test (IST) data were collected and recorded, and that the pre-established PMT requirements for the pump replacement activities were checked and validated before 3RHS*P1A was declared operable.

b. Findings

No findings of significance were identified.

.2 Interim Replacement of the Engineered Safety Features (ESF) Building Roof Plugs for the "A" RHR Pump Room

a. Inspection Scope

The inspector reviewed the post maintenance testing (PMT) following the interim replacement of the ESF building roof plugs performed in accordance with automated work order (AWO) M3-02-06918. The plugs were removed to support the replacement of the "A" RHR pump but had to be replaced prior to completion of the pump work to meet technical specification requirements. The PMT activity was performed using Surveillance Procedure (SP) 3614I.3, Supplemental Leak Collection and Release System Negative Pressure Verification. The inspectors reviewed PMT activities and results to ensure: 1) the PMT was appropriate for the scope of the maintenance work completed; 2) the acceptance criteria were clear and demonstrated operability of the system; and 3) the PMT was performed satisfactorily in accordance with the SP. The inspector also reviewed related licensee documents including an Engineering Record of Correspondence concerning sealing of U3 ESF Building Roof Plugs.

b. Findings

No findings of significance were identified.

.3 Emergency Diesel Generator and Component Cooling Heat Exchanger

a. Inspection Scope

The inspector reviewed the completed documentation for post maintenance testing (PMT) performed in accordance with the following automated work orders (AWOs):

- M3-01-18053 Replace 3HPV*MOD23B "B" Emergency Diesel Generator Enclosure Ventilation Inlet Hydramotor Assembly
- M3-02-04463 10 Year Preventive Maintenance Overhaul of the Bettis Actuator for the "C" Component Cooling Heat Exchanger Outlet Temperature Control Valve, 3CCP*TV32C

The inspector reviewed the scope of the work activities and verified that the PMTs performed were appropriate to restore the operability of the components and associated systems. The inspector reviewed the PMT plans, acceptance criteria and test results to verify that the acceptance criteria were satisfied. The inspector verified that the tests demonstrated that the components satisfied the applicable design and licensing bases specified in the Millstone 3 FSAR or vendor technical manuals or calculations.

b. Findings

No findings of significance were identified.

.4 Residual Heat Removal (RHR) Pressure Relief

a. Inspection Scope

The inspector reviewed the completed documentation for post maintenance testing performed on the RHR system using SP 3762WD, Residual Heat Removal Pressure Relief Device Setting and Testing. Testing was associated with several automated work orders. In addition, numerous supporting documents were reviewed including surveillance procedures, Millstone Unit 3 license amendments and Millstone Unit 3 design basis documents.

b. Findings

No findings of significance were identified.

.5 "B" Safety Injection System Pump

a. Inspection Scope

The inspector reviewed the completed documentation for post maintenance testing performed on the "B" safety injection system pump under SP 3630E.2, Safety Injection Pump “B” Cooling Pump Operational Readiness Test. The surveillance was conducted in association with automated work order (AWO) M3-02-08182 and CR-02-08207. In addition, numerous supporting documents were reviewed including surveillance procedures, design calculations, ASME code case and other Section XI materials, and Millstone Unit 3 design basis documents.

b. Findings

No findings of significance were identified.

.6 Solid State Protection System (SSPS) Logic

a. Inspection Scope

The inspector reviewed and observed portions of the post maintenance testing performed on the SSPS logic using SP 3446B, SSPS Logic. The testing was associated with worked performed under automated work orders (AWO) M3-02-02054 and M3-02-02928. In addition, numerous supporting documents were reviewed including surveillance procedures, Millstone Unit 3 license amendments and Millstone Unit 3 design basis documents.

b. Findings

No findings of significance were identified.

.7 Residual Heat Removal (RHR) Heat Exchanger Maintenance

a. Inspection Scope

The inspectors reviewed the completed documentation for post maintenance testing performed on the RHR system using SP3610A.1, Residual Heat Removal System Operational Readiness Test. Testing was associated with several AWOs. In addition, numerous supporting documents were reviewed including surveillance procedures, Millstone Unit 3 license amendments and Millstone Unit 3 design basis documents.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspector reviewed the following areas related to the 3R08 refueling outage for conformance to technical specification requirements and approved procedures. Selected activities were verified for each evolution.

- Review of the initial shutdown risk evaluation for the initial outage schedule and implementation of recommendations
- Coastdown operations (OP 3204)
- Plant shutdown (OP 3206)
- Reactor cooldown and transition into shutdown cooling operation

- Shutdown risk management (OP 3260A)
- RCS inventory control
- Second draindown of the RCS to 3 to 4 feet below the reactor vessel flange in preparation for reinstalling the reactor vessel head. This review included attendance at the shift focus brief; observation of portions of the draindown performed in accordance with OP 3216, Reactor Coolant System Drain (IPTE); and an evaluation of the adequacy and use of RCS temperature and level indication available to operators during the draindown
- Control and coordination of activities to minimize shutdown risk
- Shutdown risk evaluations
- Operation of the RCS to maintain pressure, temperature, and level within established ranges
- Operation of the spent fuel pool cooling system, focused on operation while the core was fully offloaded from the reactor vessel to the spent fuel pool
- Refueling operations, including fuel handling, inventory, control, and accounting in the reactor core and spent fuel pool
- Core mapping activities (CR-02-10011 follow-up)
- Multiple rod drop testing (Mode 5)
- Steam generator ("A" and "C") eddy current testing and results review
- Reactor pressure vessel head inspection (VT) activities (NRC BU 2002-01)
- Main steam isolation valve ("B" and "D") maintenance activities
- Nuclear Oversight surveillance and audit activities for 3R08
- Containment closeout walkdown on September 27, prior to Mode 4. Referenced SP 3612A.1, Containment Inspections
- Mode change verification (OP 3201) and management reviews

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability

a. Inspection Scope

The inspectors evaluated the licensee’s response to Unresolved Item 50-423/01-011-01 (See Section 4OA5 of this report). The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), calculation 01-ENG-01880C3 (Rev. 0), “Service Water System RSS Heat Exchanger Fluid Transient Operability Assessment”; Operability Determination MP3-065-01, and discussed the issue with engineering and licensing personnel.

b. Findings

Unit 3 Recirculation Spray System Potential Water Hammer Scenario

Introduction

The licensee failed to identify and evaluate a design deficiency related to the ability of the service water piping that cools the recirculation spray system (RSS) heat exchangers to withstand the effects of a column separation water hammer. This finding was determined to be of very low safety significance (Green) because this deficiency would not have rendered the service water/RSS heat exchangers incapable of performing their intended safety function. The finding was determined to be a non-cited violation of 10 CFR 50, Appendix “B”, Criterion III (Design Control).

Description

Unresolved Item 50-423/01-011-01 identified that the licensee did not thoroughly and conservatively analyze the RSS heat exchanger service water return piping for a potential water hammer event as a result of a postulated service water system flow transient. Specifically, under loss of service water flow conditions during a design basis accident, flashing could begin at the point where the service water temperature rises to the saturation temperature corresponding to the system pressure in the RSS coolers, creating column separation (a large steam filled void separating two or more saturated water volumes in the system). The subsequent resumption of service water flow could cause the upstream water volume to be accelerated toward the downstream water volume, resulting in a potential pressure wave (water hammer) that would propagate through the service water system. In response to this concern, the licensee 1) initiated condition reports 01-08655 and 01-09051; 2) completed calculation 01-ENG-01880C3 (Rev. 0), “Service Water System RSS Heat Exchanger Fluid Transient Operability Assessment”; and 3) completed Operability Determination MP3-065-01 to evaluate and reconcile the piping system loads resulting from a water column rejoin fluid transient event.

The licensee’s subsequent analyses determined that the service water system (including piping, supports, nozzles, expansion joints) were operable for several postulated column rejoin transient event variations. While the licensee’s analysis demonstrated operability, it identified certain components (e.g., RSS cooler service water discharge nozzles and some pipe supports/expansion joints) that did not fully meet design basis structural integrity limits. Accordingly, the licensee plans to implement a modification in refueling outage RFO9 (Fall 2004) to restore the system to full qualification. This action was being tracked via corrective action operability determination assignment (CAOD) 01006360-04.

Analysis

This was a performance issue since this oversight could have been reasonably within the licensee's ability to foresee and correct, and which should have been prevented. In particular, the licensee evaluated piping sections that could experience unacceptable water hammer loads during a review of design calculation NP(B)-271-FA, Rev. 4, "Water Hammer Analysis for Service Water System Due to Pump Re-Start Following LOOP/LOCA and LOOP Events." However, this evaluation failed to recognize and analyze the RSS heat exchanger service water return piping as being susceptible. This issue affects the mitigating system cornerstone and was more than minor since it could have affected the capability of the service water/RSS heat exchangers to remove heat during postulated accidents. Additionally, the finding was more than minor because it was similar to an issue described in NRC Inspection Manual Chapter 0612, Appendix E, Section 3.a, where a calculation error was significant enough that a modification was necessary to correctly and completely resolve the deficiency.

This issue was determined to be of very low safety significance (Green) based on a subsequent operability determination that concluded that the affected piping system would remain functional under the postulated accidents. The issue screened to Green in the At-Power Reactor Safety Significance Determination Process (SDP) Phase 1 because the design deficiency was confirmed not to result in a loss of function.

Enforcement

10 CFR 50, Appendix "B", Criterion III, (Design Control) requires in part, that the design basis for safety-related equipment be correctly translated into specifications, drawings, procedures and instructions. Contrary to this requirement, design calculation NP(B)-271-FA, which analyzed the service water system for water hammer events, failed to analyze the service water-cooled RSS heat exchangers and identify the column separation water hammer design deficiency. However, because of the very low safety significance of this issue, and because it was entered into the licensee's corrective action program in condition reports 01-08655 and 01-09051, the issue was treated as a non-cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy, issued on May 1, 2000 (65 FR 25368). **(NCV 50-423/02-05-06)**

1R22 Surveillance Testing

a. Inspection Scope

The inspector reviewed licensee performance related to the following surveillance tests:

- C SP 750 Battery Weekly and Quarterly Surveillance
- SP 3604A.5* Chemical and Volume Control System Valve Operability Test
- SP 3604A.6* Charging/SI Pump Inoperability Verification
- SP 3610B.4* Accumulator Check Valve Stroke Test
- SP 3612B.1* Containment Manual Isolation Valves (Outside Containment) Valve Position Verification
- SP 3614I.3 Supplemental Leak Collection and Release System (SLCRS)

- SP 3623.2 Turbine Overspeed Protection System Test
- SP 3626.13 Service Water Heat Exchangers Fouling Determination
- SP 3646A.2 Emergency Diesel Generator “B” Monthly Tests
- SP 3646A.17* Train “A” ESF with LOP Test (IPTE)
- SP 37126 Main Steam Code Safety Valve Surveillance Testing (IPTE)

(Note: Asterisk (*) on SP number denotes test conducted during the refueling outage, 3R08)

The inspector observed the quarterly surveillance of battery 301A-1 per procedure C SP 750. The inspector verified the battery cells’ electrolyte level, temperature, individual cell voltages and specific gravity were within the acceptance criteria of TS 4.8.2.1.b.1, 4.8.2.1.b.2 and 4.8.2.1.b.3. Portions of SP 3604A.5 were observed in the control room and discussed with operators to verify proper performance of the test. In addition, the inspector verified selected precautions and prerequisites were maintained throughout test performance.

The inspector verified the conduct of SP 3604A.6 at the required procedural periodicity to confirm compliance with TS 3.4.9.3 for cold overpressure protection of the reactor coolant system during Mode 5 conditions prior to plant entry into Mode 6 for refueling activities during 3R08.

The inspector noted that surveillance SP 3610B.4 had been accepted by operations as satisfactory although the licensee had initiated CR-02-09715 to document that data for the "B" accumulator testing was outside the acceptance band. The inspector subsequently reviewed the procedure and completed data sheets in order to confirm that the safety-related check valves were operable in accordance with procedure acceptance criteria and technical specifications’ requirements.

The inspector observed and reviewed portions of SP 3614I.3, which tests a risk significant structure, system, and component. In addition, the inspector discussed SLCRS performance data, historical alignment problems, and CRs associated with SLCRS operability with responsible site personnel. These activities were performed to ensure that this system was capable of performing its intended safety functions and to ensure related TS requirements were met. Test performance data, TS surveillance requirements, and other established performance criteria were compared to current and historical surveillance test data to validate selected system performance parameters. In addition, test performance data were compared to design basis calculations, FSAR Chapter 15, post accident equipment sequencing assumptions, pre-operational test data sheets and SLCRS system Inservice testing results.

For the performance of the infrequently performed test and evolution (IPTE), involving the engineered safety features (ESF) with loss of power (LOP) testing, the inspector observed the operations shift briefing and complete conduct of the test through the data collection and into the commencement of system restoration.

The inspector observed the performance of the main steam valve testing on four of the eight main steam safety valves (MSSVs) scheduled for in service testing prior to 3R08. During plant power reduction to take the unit off-line for 3R08, the plant was maintained just below 50% power to conduct the MSSV testing. The inspector verified hydroset testing devices were properly used in accordance with the procedure, the MSSVs "as found" lift set pressures met TS 4.7.1.1 requirements and MSSVs adjustments were controlled to ensure "as left" lift set pressures remained within acceptance criteria.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspector reviewed Temporary Modification (TM) 3-02-007, installing a seal on the service water (SWP) enclosure tube in the control building, changing the control room habitability boundary to allow repairs to SWP piping while the TS 3.7.7 and 3.7.8 requirements for the control room envelope were being maintained. The inspector reviewed the design details for the installation of the temporary equipment and inspected the as-built configuration at elevation 64'-6" in the control building. The inspector also evaluated both the technical evaluation and the 10 CFR 50.59 screen form supporting this TM.

The inspector noted that a 1996 bypass jumper (3-96-020) was referenced as part of the technical evaluation, including the seismic review and qualification of the modification details. The Millstone Unit 3 FSAR Section 6.4.2.5, addressing the design of the control room envelope, was discussed with the cognizant design and SWP system engineers to assess the proper consideration of accident analysis and seismic qualification requirements. Subsequently, the inspector examined an additional seismic review performed by licensee engineering to confirm the adequate seismic capacity of this TM, in accordance with the Unit 3 design requirements and FSAR details.

The required SWP piping repairs were completed during the conduct of 3R08 activities at the plant. The inspector verified that TM 3-02-007 was removed from service prior to the startup from 3R08.

b. Findings

No findings of significance were identified.

**Emergency Preparedness [EP]**

1EP6 Drill Evaluation

a. Inspection Scope

The inspector observed a Unit 3-based simulator exam. The licensee had preselected the drill classification results to be included in the EP drill performance indicator (PI). The inspector reviewed the licensee's Emergency Planning Services Department Instruction 18, Administration of NRC Performance Indicators, and discussed the performance results with the instructor to confirm correct implementation of the PI program. The drill evaluation form was also reviewed to verify proper documentation of results, which included one successful classification.

b. Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

**Occupational Radiation Safety [OS]**

2OS2 ALARA Planning and Controls

a. Inspection Scope

During the period September 23 - 26, 2002, the inspector conducted the following activities to evaluate the effectiveness of administrative, operational, and engineering controls to limit personnel exposure for tasks conducted during the Unit 3 refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and the licensee's procedures.

- The inspector reviewed pertinent information regarding cumulative exposure history, current exposure trends, and work-in-progress reviews in order to assess the licensee's effectiveness in establishing exposure goals and keeping actual personnel exposure as low as is reasonable achievable (ALARA) when performing outage work activities. Also reviewed were the results of the licensee's efforts to reduce plant source terms through system flushing, component decontamination, temporary shielding installation, and shut down chemistry controls.

- Independent radiation surveys were performed in the radiological controlled areas (RCA) of the Unit 3 containment and auxiliary buildings to confirm the accuracy of posted survey results and assess the adequacy of radiation work permits and associated controls. Technical Specification Locked High Radiation Areas (TSLHRA) were selected in these buildings and verified to be properly secured and posted during tours.

- The inspector reviewed the exposure controls specified in ALARA Reviews (AR) for all work activities whose actual (or projected) cumulative exposure exceeded 5 person-rem. Work activities that were reviewed included Reactor Disassembly/Reassembly (AR 3-02-01), Steam Generator Eddy Current Testing (AR 3-02-02), Steam Generator Sludge Lancing/Upper Bundle Flush (AR 3-02-03), Valve and MOV maintenance (AR 3-02-11), Installation/Removal of Scaffolding (AR 3-02-13), and Reactor Head Inspection (AR 3-02-20). Work-In-Progress ALARA Reviews and ALARA Council meeting minutes were reviewed to assess the licensee's methods of forecasting dose estimates

- Jobs-in-progress having radiological significance were observed. The inspector reviewed the associated exposure controls specified in radiation work permits (RWP) and observed pre-job briefings for testing of a loose parts monitor instrument, located in the reactor vessel annulus area (RWP 323-2); drain down/decontamination of the north saddle of the reactor cavity (RWP 305-2); removal of staging from containment (RWP 331): and scaffolding removal from the auxiliary building (RWP 231). For these tasks, the inspector interviewed selected workers on their knowledge of the relevant radiation work permit, electronic dosimetry set points, and job-site radiological conditions.

- The inspector attended post-job ALARA debriefings for the maintenance on AOV-8149 A/B/C, maintenance of SIH-003, steam generator sludge lancing/foreign object search and retrieval (FOSAR), and scaffolding installation/removal to assess the effectiveness of the ALARA controls established for these tasks.

- The inspector reviewed various records regarding monitoring of radiological conditions in the RCA, including Radiation Protection Technician log books, personnel contamination reports, air sample results, contamination survey records, and instrument daily source check data to assess the effectiveness of radiological control measures.

b. Findings

No findings of significance were identified.

## 3. SAFEGUARDS

### Physical Protection [PP]

3PP1 Access Authorization

a. Inspection Scope

An in-office review was conducted of materials related to the effectiveness of Access Authorization self-assessments, selected corrective actions related to reviewed event reports and logged security events, the performance of table top security drills and the functionality of selected access control equipment - including an E-field sensor.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES [OA]

### 4OA1 Performance Indicator Verification

.1 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed implementation of the licensee's Occupational Exposure Control Effectiveness Performance Indicator (PI) Program. Specifically, the inspector reviewed Condition Reports, and associated documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned personnel exposures since the last inspection against the criteria specified in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2, to verify that all occurrences that met the NEI criteria were identified and reported as Performance Indicators.

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams and Scrams With Loss of Normal Heat Removal

a. Inspection Scope

The purpose of this inspection was to confirm the information presented in the licensee's June 2002 Unplanned Scrams per 7000 Hours Critical and Scrams With a Loss of Normal Heat Removal performance indicators was complete and accurate. The inspector reviewed selected operator logs, plant process computer data, and licensee monthly operating reports for the period July 1, 2001, through June 31, 2002. This time frame was selected as the last confirmation of this PI was performed for data through June 30, 2001.

b. Findings

No findings of significance were identified.

.3 High Pressure Safety Injection System Unavailability

a. Inspection Scope

The purpose of this inspection was to confirm that the information presented in the licensee's March 2002 Safety System Unavailability PI for the high pressure safety injection (HPSI) system was complete and accurate. The inspector reviewed licensee event reports submitted from January 2001 through March 2002; verified selected operator log entries for equipment out of service; reviewed guidance provided to licensees in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 2; and reviewed licensee Technical Evaluation M3-EV-00-0029, Revision 1, issued in September 2001, which describes the monitoring requirements and PI calculation method for the HPSI system.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 FIN 50-423/01-09-015: Licensee Requalification Exam Results

a. Inspection Scope

Inspection Report 50-423/01-009 identified a Green finding due to two of ten crews failing their licensed operator requalification examinations. The inspector reviewed the following attributes of the facility response to these failures:

i) Complete and accurate identification of the problem.
ii) Consideration of extent of condition
iii) Identification and completion of appropriate corrective actions.

b. Findings

The crew performance problem was identified in a timely fashion. The failures occurred due to the operators in both crews taking 1.5 to 2.0 minutes longer to isolate a ruptured steam generator (SG) than the credited time in the FSAR of 30.5 minutes. These failures were noted as they occurred during a licensed operator requalification examination. Steam Generator Tube Rupture (SGTR) simulator training had been conducted in 2000 during which seven of nine crews performed satisfactorily, and in 2001 in which all crews performed satisfactorily. The inspector did not consider this to indicate a recurrent problem which should have been identified earlier.

The facility performed an appropriate extent of condition review. In addition to the review of prior SGTR scenario performance, the facility evaluated other crews in SGTR scenarios as part of ongoing requalification training. This was an appropriate level of evaluation.

The facility took appropriate corrective actions. The training department determined that the cause of the failures was a combination of two factors: the crews did not feel pressured to meet the FSAR assumed times because in their actual scenarios a ruptured SG overfill (basis for the 30.5 minute criterion) was not approached, and the crews thought that discussions concerning altering the scenario critical tasks had meant that the critical task they failed was no longer applicable. The facility conducted remedial training for both crews to show them where they could save time in SGTR response, and the crews bettered their times by 10 minutes in reexamination. Lessons learned have been incorporated into 2002 requalification training. The facility was also continuing to evaluate modification of this critical task.

.2 Maintenance Rule Issues

Refer to NRC Inspection Report 50-336/02-05, Millstone Unit 2, Section 4OA2.3 for specific details.

.3 Inservice Inspection Issues

a. Inspection Scope

The inspector reviewed a sample of corrective action reports shown in Attachment 1, which identified problems related to inservice inspection issues. The inspector verified that problems were being identified, evaluated, appropriately dispositioned, and entered into the corrective action program.

b. Findings

No findings of significance were identified.

.4 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed Radiological Protection Department self-assessments, management observations, and programmatic internal appraisals relating to the implementation of operational, engineering, and administrative controls for performing work in radiologically controlled areas. The inspector also reviewed eleven (11) condition reports, relating to ALARA planning and controls, initiated between March 2002 and September 2002, to evaluate Dominion Nuclear’s threshold for identifying, evaluating, and resolving problems in implementing the ALARA program. This review was conducted against the criteria contained in 10 CFR 20, Technical Specifications, and the licensee’s procedures.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

.1 (Closed) Licensee Event Report (LER) 50-423/01-04-00: Failure of Neutron Flux Accident Monitoring Instrumentation. Around August 24, 2001, the licensee identified that the Channel 1 Gamma Metrics Neutron Flux Monitor was inoperable for more than 12 days, exceeding the seven day allowed outage time of Technical Specification 3.3.3.6, Action Statement a. The Gamma Metrics Neutron Flux Monitor Channels may be used post-accident to verify that the reactor is subcritical and remains subcritical. The Channel 1 Gamma Metrics Neutron Flux Monitor was restored to operable status on August 24, 2001. While the Channel 1 Gamma Metrics Neutron Flux Monitor was inoperable, other sufficient indications were available in the control room for the plant operators to ensure that the reactor remained subcritical following a design basis event.

The initial corrective action for this issue involved daily monitoring of the Gamma Metrics Neutron Flux Monitoring Channels to ensure that the equipment remained operable. The final corrective actions included establishment of a deviation alarm in the plant computer between Channels 1and 2 of the Gamma Metrics Monitor, to alert plant reactor operators of a channel deviation and possible instrument failure.

The issue was entered into the licensee's corrective action program as Condition Report (CR) 01-08843. Based on the on-site review of this LER, the inspector determined that this issue was a licensee-identified minor violation and no findings of significance were identified.

.2 (Closed) LER 50-423/2002-001-00: Control Room Emergency Ventilation System Surveillance Failure. The licensee identified that the ventilation system flow measurements obtained to satisfy the monthly TS surveillance requirement for the Control Room Emergency Ventilation System (CREVS) had been corrected twice for the effects of local pressure. A review of historical surveillance data identified several cases where the recorded system flows were outside the required range, which left Train "A" in an inoperable, but available condition that exceeded the TS allowed outage time. The licensee reported this event as a condition prohibited by the plant's TS in accordance with 10 CFR 50.73 (a)(2)(i)(B). The inspector determined that this condition had a credible impact on safety because if left uncorrected, the finding could have become a more significant safety concern by masking degraded operation of the CREVS. However, because the CREVS was shown to be capable of performing its accident mitigation function during subsequent surveillance tests when it produced acceptable flow rates and there were no actual consequences, this issue was of very low safety significance (Green). See Section 4OA7 of this report. This LER is closed.

.3 (Closed) LER 50-423/2002-003: Inadequate Validation of Fire Safe Shutdown Analysis Assumptions. On April 26, 2002, the licensee identified that one of the assumptions used in their post fire safe shutdown analysis was not adequately validated. In a letter to the NRC dated July 1, 1985, the licensee had assumed 15 minutes was available to isolate the power operated relief valve (PORV) following a fire in the control room, cable spreading room, or instrument rack room, that could result in control room evacuation, without causing an uncontrollable condition in the reactor coolant system pressure boundary. The actions provided for isolating the PORVs in emergency operating procedure, EOP 3509.1, Control Room, Cable Spreading Area or Instrument Rack Room Fire, were developed based on the assumption that 15 minutes were available. A recent analysis by the licensee indicated that if a PORV was inadvertently opened by a fire-induced hot short in the control circuit during the fire, which could cause the operators to abandon the control room, the reactor coolant system could quickly (about 1.5 minutes, according to a memo from M. Kai dated June 3, 2002) depressurize to saturation conditions, making the system difficult to control. This situation represented an unanalyzed condition.

The immediate compensatory actions implemented by the licensee included increased surveillance to verify the operability of the fire detection and suppression systems, and to confirm control of transient combustibles and ignition sources in the affected areas. The licensee also revised EOP 3509.1 to require the operators to close Train "A" PORV block valve (3RCS*MV8000A) and Train "B" PORV block valve (3RCS*MV8000B) early before evacuating the control room.

The inspector reviewed the revised EOP (Revision 006) and verified that the required actions for closing both PORV block valves were stated in step 2 of EOP 3509.1. The inspector also verified with the licensee that the PORV block valve could be reopened, when needed, from outside of the control room, as stated in the EOP. This LER is closed.

The event described in the LER involved the hot short issue which is the subject of an industry initiative. The issue remains unresolved pending generic resolution of guidance for evaluating fire induced circuit failure. (**URI 50-423/02-05-07**)

4OA5 Other Activities

.1 TI 2515/145 - Circumferential Cracking of RPV Head Penetration Nozzles

a. Inspection Scope

The inspector reviewed the licensee’s activities to detect circumferential cracking of RPV head penetration nozzles in response to NRC Bulletins 2001-01 and 2002-02 as required by Temporary Instruction (TI) 2515/145. This included interviews with analyst personnel, reviews of qualification records and procedures, and observations of selected video tape records of the reactor vessel closure head visual examination. The inspector independently viewed a sample set of 91 out of the total 316 views of 79 penetrations and related reactor head surface examined by the plant staff. In accordance with TI 2515/145, the inspector verified that deficiencies and discrepancies associated with the RCS structures and the examination process, if identified, would be placed in the licensee’s corrective action process. The specific reporting requirements of TI 2515/145 are documented in Attachment 2.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item 50-423/01-011-01: The licensee did not thoroughly and conservatively analyze the recirculation spray system (RSS) heat exchanger service water return piping for a potential water pressure wave (hammer event) as a result of a postulated service water system flow transient. In response to this concern, the licensee 1) initiated condition reports 01-08655 and 01-09051; 2) completed calculation 01-ENG-01880C3 (Rev. 0), “Service Water System RSS Heat Exchanger Fluid Transient Operability Assessment”; and 3) completed Operability Determination MP3-065-01 to evaluate and reconcile the piping system loads resulting from a water column rejoin fluid transient event. The licensee’s analyses determined that the service water system (including piping, supports, nozzles, expansion joints) was operable for several postulated column rejoin transient event variations.

The inspectors reviewed the results of the calculation and operability determination, including assumptions, methodology, and conclusions. The inspectors also discussed specific portions of the calculation and operability determination with the licensee. The inspectors determined that the licensee’s completed and planned responses to this item were acceptable, and accordingly, this item is closed. However, because this issue was determined to be a violation of NRC requirements, this item was reviewed and evaluated as a finding (non-cited violation) as described in Section 1R21 of this report.

.3 (Closed) Unresolved Item 50-423/01-011-02: This item questioned whether the individual system design basis needs to consider the most severe transient to which the system could be exposed, as well as the UFSAR Chapter 15 accidents. Specifically, although the design basis specified a concurrent loss of coolant accident (LOCA) and loss of off-site power (LOOP), an NRC inspection team questioned whether the licensee must evaluate the consequences of a particular scenario where a LOOP occurs subsequent to a LOCA.

The inspectors reviewed the licensee’s UFSAR, NRC Information Notice (IN) 93-17, “Safety Systems Response to Loss of Coolant and Loss of Offsite Power,” Rev. 1, and discussed this issue with licensee and NRC headquarters staff. Based upon this review, the inspectors determined that the licensee was not required to perform additional analysis. IN 93-17 discusses certain scenarios in which a LOCA with a delayed LOOP may occur (where the LOOP was not the result of a failure separate from the LOCA). However, those scenarios were not relevant to this item. The IN further stated that the NRC was considering generic actions to determine whether all licensees should be required to demonstrate the capability to respond to a LOCA followed by a LOOP. To date, no action has been promulgated. Accordingly, the inspectors determined that no further analysis by the licensee was required. Additionally, the inspectors did not identify that there was safety significance to warrant a backfit analysis in this instance. No violations of NRC requirements were identified and this unresolved item was closed.

.4 (Closed) Unresolved Item 50-423/01-011-03: The licensee may not have properly evaluated the consequences of an analyzed safety grade cold shutdown scenario (Boration Phase). In particular, the licensee did not evaluate the postulated mitigative capability of containment heat removal systems which may be required to ensure containment integrity in the analyzed scenario.

The inspectors reviewed the Millstone 3 UFSAR and Engineering Evaluation M3-EV-970233, both of which discussed the safety grade cold shutdown design basis event for Millstone 3. The inspectors reviewed the limiting initiating event determination, equipment availability analysis, single failure assumptions, and mitigative system capabilities. Based upon this review, the inspectors determined the licensee's analysis was consistent with the guidelines provided in Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System." The inspectors also determined that the licensee's analysis provided a capability to shutdown and cooldown the plant with only safety grade equipment without causing a release of radioactive materials to the environment. The inspectors determined the licensee's analysis is acceptable. No violations of NRC requirements were identified and this unresolved item was closed.

4OA6 Meetings, including Exit

.1 Maintenance Rule Exit Meeting

The inspectors presented the inspection results to Mr. Denny Hicks, Director, Nuclear Station Safety and Licensing, Millstone Station, and other members of the licensee's staff on July 26, 2002.

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

.2 Operator Licensing Exit Meeting Summary

The inspector presented the inspection results to a member of licensee management in a telephone exit for Operator Licensing examination report 50-423/02-301 on August 26, 2002.

.3 Inservice Inspection Exit Meeting Summary

The inspector presented the inspection results to Mr. Stephen Scace, and other members of the licensee staff, at the conclusion of the inspection on September 18, 2002. The licensee acknowledged the conclusions and observations presented.

The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. Some proprietary items were reviewed during the inspection but no proprietary information is presented in this report.

.4 ALARA Planning and Controls Exit Meeting

The inspectors presented the inspection results to Mr. Denny Hicks, Director, Nuclear Station Safety and Licensing, Millstone Station, and other members of the licensee's staff on September 26, 2002.

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

.5 Resident Exit Meeting Summary

The inspectors presented the inspection results to Mr. Price and other members of licensee management on October 21, 2002. The inspectors asked the licensee whether any material examined during this inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-identified Violations

The following violation of very low safety significance (green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCV.

a. Technical Specification 3.7.7 requires both trains of the Control Room Emergency Ventilation System (CREVS) to be operable in all modes. Restoration of an inoperable ventilation train when the plant is operating in Modes 1 - 4 is required within seven days, or a plant shutdown to Mode 3 is required within the next six hours, and Mode 5 the following 30 hours. Train "A" of the CREVS was inoperable due to historical surveillance data being outside the acceptance criteria for time periods equal to the monthly surveillance interval, which is greater than the allowed outage time. Because the CREVS was shown to be capable of performing its accident mitigation function during subsequent surveillance tests, this violation is not more than of very low significance, and is being treated as a non-cited violation. The licensee entered this issue into the corrective action program as CR-02-00577.

## ATTACHMENT 1

## SUPPLEMENTAL INFORMATION

a. Key Points of Contact

Dominion Nuclear Connecticut, Inc.

| | |
|---|---|
| J. A. Price | Site Vice President - Millstone |
| D. Hicks | Director -Nuclear Station Safety and Licensing |
| C. Maxson | Manager - Nuclear Engineering |
| S. Sarver | Acting Director - Nuclear Station Operations and Maintenance |
| S. Scace | Director - Nuclear Engineering |
| D. Smith | Manager - Licensing |

b. List of Items Opened, Closed and Discussed

Opened

| | | |
|---|---|---|
| 50-423/02-05-07 | URI | Resolution of Hot Shorts Issues (4OA3.3) |

Opened and Closed

| | | |
|---|---|---|
| 50-423/02-05-05 | NCV | Inadequate Procedure to Drain an Isolated Reactor Coolant System Loop (1R14) |
| 50-423/02-05-06 | NCV | Design Control NCV. Failure to identify and evaluate a design deficiency related to the ability of the recirculation spray system (RSS) to withstand the effects of a column separation pressure wave (water hammer). (1R21) |

Closed

| | | |
|---|---|---|
| 50-423/01-04-00 | LER | Failure of Neutron Flux Accident Monitoring Instrumentation (4OA3.1) |
| 50-423/02-01-00 | LER | Control Room Emergency Ventilation System Surveillance Failure (4OA3.2) |
| 50-423/02-03-00 | LER | Inadequate Validation of Fire Safe Shutdown Analysis Assumptions (4OA3.3) |
| 50-423/01-011-01 | URI | Service Water System may not withstand column separation water hammer at RSS heat exchanger outlet |
| 50-423/01-011-02 | URI | Service Water System design may not account for most severe transient to which the system could be exposed |
| 50-423/01-011-03 | URI | Acceptability of Safety Grade Cold Shutdown methodology consequences |

Discussed

50-423/01-09-01 FIN Licensee requalification exam results (4OA2.1)

c. Partial List of Documents Reviewed

Licensed Operator Requalification Examinations

| | |
|---|---|
| CR-01-12048 | "Potentially Outdated Criteria Contained in Simulator Exam Scenario Critical Task" |
| LORTSE25 | Scenario guide Used for Annual Exam for the failed crews |
| NTP-147 Att 3 | Counseling Records for the failed crews |
| LORTSE6, 19,32,38 | Scenario guides for retake examinations |
| LessonC02303L | SGTR procedure lesson plan‘ |
| S02303L, S02301L | SGTR training scenarios |
| LORTSE60 | SGTR evaluation scenario |

Inservice Inspection

Steam Generator Eddy Current Testing

M3-EV-02-0008, Rev. 0, “Millstone Unit 3 Steam Generator Integrity Degradation Assessment”
U3-24-SIP-REF01, Rev. 001, “MS Unit 3 Steam Generator Eddy Current Data Analysis Reference Manual”
ASME Boiler and Pressure Vessel Code Sections V and XI.
NRC Generic Letters 95-03 (Circumferential Cracking of Steam Generator Tubes,

General Inservice Inspection

UT procedure MP-PDI-UT-1, Rev. 0, PDI Generic Procedure (Rev. C) for UT of ferritic pipe welds
UT procedure MP-UT-8, Rev. 1, for the pressurizer lower head to skirt weld UT
UT data sheets dated 9/17/02 for pressurizer lower head to skirt weld UT
UT data for package 308-01-040, reference ID:MSS-33-FW-1-HM
UT data for package 308-01-007, reference ID:03-007-SW-D
UT data for package 308-01-013, reference ID:03-007-SW-Z
CR 02-09730 on pressurizer heater leak indications
CR 02-09555 on unreadable ISI iso
CR 02-09705 on 2 calibration blocks with same identification
CR 02-09753
CR 02-09260 MT indication

CRDMs

Drawing 35R05002 MS3 Inspection MAP for CRDMs
Procedure 83-0043, Rev. 0, "Reactor Vessel Head Remote Visual Inspection for MS U3"
CR 02-09547
Letter B18735, MS U3 to NRC dated 9/11/02 on response to NRC Bulletin 2002-02.
EPRI Report 1006899, Rev. 1 - on PWR Rx Head Penetration Visual Examination for leakage
MS U3 procedure MP-VE-11, Rev. 000-01 - for visual examination of Rx head penetrations

ALARA Planning and Controls

Procedures:

| | |
|---|---|
| RPM 1.1.1, Rev. 6 | Health Physics Organization and Responsibilities of Key Radiological Personnel |
| RPM 1.3.8, Rev. 7 | Criteria for Dosimetry Issue |
| RPM 1.3.14, Rev. 5 | Personnel Dose Calculations and Assessments |
| RPM 1.4.1, Rev. 6 | ALARA Reviews and Reports |
| RPM 1.4.2, Rev. 1 | ALARA Engineering Controls |
| RPM 1.5.1, Rev. 8 | Routine Survey Frequency |
| RPM 1.5.2, Rev. 4 | High Radiation Area Key Control |
| RPM 1.5.5, Rev. 4 | Guidelines for Performance of Radiological Surveys |
| RPM 1.5.6, Rev. 3 | Survey Documentation and Disposition |
| RPM 2.1.1, Rev. 4 | Issuance and Control of RWPs |
| RPM 2.1.2, Rev. 1 | ALARA Interface with the RWP Process |
| RPM 5.2.2, Rev. 9 | Basic Radiation Worker Responsibilities |
| RPM 5.2.3, Rev. 3 | ALARA Program and Policy |
| RPM 5.2.6, Rev. 4 | Guidelines for Radiological Controls of Radiography |
| RPM 2.10.2, Rev. 8 | Air Sampling Counting and Analysis |
| RPM 2.11.1, Rev. 8 | Survey and Decontamination of Personnel and Clothing |

ALARA Reviews:

| | |
|---|---|
| AR 3-02-01 | Reactor Disassembly and Reassembly |
| AR 3-02-02 | Steam Generator Eddy Current Testing and Foreign Object Search and Retrieval |
| AR 3-02-03 | Steam Generator Sludge Lancing/Upper Bundle Flush/FOSAR |
| AR 3-02-11 | Valve Repairs and MOV maintenance |
| AR 3-02-13 | Installation and Removal of Scaffolding |
| AR 3-02-20 | Reactor Head Inspection |

ATTACHMENT 1 - SUPPLEMENTAL INFORMATION (Cont.)

Departmental Self-Assessments:

MP-SA-02-001 Radioactive Material Control
MP-SA-02-061 Dosimetry Laboratory Operations
MP-SA-02-028 Exposure Control

Condition Reports:

02-10175, 02-10115, 02-09836, 02-09832, 02-09718, 02-09570, 02-09514, 02-09188, 02-08660, 02-08631, 02-07549

Personnel Contamination Reports:

Level 2 Contaminations: M3-02-01, M3-02-02, M3-02-03, M3-02-04, M3-02-05, M3-02-06
Log book for Level 1 Skin and Clothing Contaminations

Management Observations:

02-5849, 02-5839, 02-5820, 02-5808, 02-5777, 02-5749, 02-5689, 02-5688, 02-5687, 02-5653, 02-5524, 02-5520, 02-5501, 02-5490, 02-5356

Other:

3R08 Outage ALARA Guide

Documents Reviewed

UFSAR

Calculation 01-ENG-01880C3 (Rev. 0), "Service Water System RSS Heat Exchanger Fluid Transient Operability Assessment"

Operability Determination MP3-065-01

Engineering Evaluation M3-EV-970233 (Safety Grade Cold Shutdown Design Basis Event for Millstone 3)

NRC Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System"

d. List of Acronyms

| | |
|---|---|
| 3R08 | refueling outage number 8 |
| ALARA | as low as is reasonably achievable |
| AR | ALARA Reviews |
| ASME | American Society of Mechanical Engineers |
| AWO | automated work order |
| CCE | charging pump cooling |
| CFR | Code of Federal Regulations |
| CIV | containment isolation valves |
| CR | condition report |
| CRDM | control rod drive mechanism |
| CREVS | control room emergency ventilation system |
| CVCS | chemical and volume control system |
| ECT | eddy current testing |
| EDG | emergency diesel generator |
| EOOS | equipment out of service |
| EOP | emergency operating procedure |
| ESF | engineered safety feature |
| EPRI | Electric Power Research Institute |
| FIN | finding |
| FOSAR | foreign object search and retrieval |
| FPER | fire protection evaluation report |
| FSAR | final safety analysis report |
| GL | generic letter |
| HPSI | high pressure safety injection |
| IPTE | infrequently performed test and evolution |
| ISI | inservice inspection |
| IST | inservice test |
| LER | licensee event report |
| LOP | loss of power |
| LOCA | Loss of Coolant Accident |
| LOOP | Loss of Offsite Power |
| MCB | main control board |
| MSSV | main steam safety valve |
| MT | magnetic particle test |
| NCV | non-cited violation |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| NEI | Nuclear Energy Institute |
| NRR | Nuclear Reactor Regulation |
| OD | operability determination |
| OP | operating procedure |
| P&ID | piping and instrumentation diagram |
| PI | performance indicator |
| PMT | post maintenance testing |
| PORV | power operated relief valve |
| PWR | pressurized water reactor |
| PWSCC | primary water stress corrosion cracking |

| | |
|---|---|
| QDA | qualified data analyst |
| RCA | radiological controlled areas |
| RCCA | rod cluster control assembly |
| RCP | reactor coolant pump |
| RCS | reactor coolant system |
| RG | Regulatory Guide |
| RHR | residual heat removal |
| RPV | reactor pressure vessel |
| RSS | recirculation spray system |
| RT | radiographic testing |
| RWP | radiation work permits |
| SDP | significance determination process |
| SG | steam generator |
| SGTR | steam generator tube rupture |
| SIL | low pressure safety injection |
| SLCRS | supplemental leak collection and release system |
| SORC | site operations review committee |
| SP | surveillance procedure |
| SPROC | special procedure |
| SSPS | solid state protection system |
| SWP | service water |
| TI | temporary instruction |
| TM | temporary modification |
| TRM | technical requirements manual |
| TS | technical specification |
| TSLHRA | technical specification locked high radiation areas |
| URI | unresolved item |
| UT | ultrasonic testing |
| VT | visual examination |

## Attachment 2

## TI 2515/145 - Circumferential Cracking of RPV Head Penetration Nozzles Reporting Requirements

a.1. The examination was performed by qualified and knowledgeable personnel using effective video imaging and optical equipment. The visual examination was done as a VT-2 type examination with evaluation by personnel qualified to the VT II or VT III level with specific training that included review of the EPRI Report 1006296, Revision 1 that provides visual examiners with information and guidance to detect leakage.

a.2. The visual examination was in accordance with approved and adequate procedures.

a.3. The examination was adequate to identify, disposition and resolve deficiencies.

a.4. The examination performed was capable of identifying the PWSCC phenomenon described in the Bulletin.

b. The general condition of the Reactor Vessel (RV) head was mostly clean bare metal with some localized staining and minor debris. The video taped inspection showed no boron deposits that were considered to result from leakage through the CRDMs.

c. Small boron deposits, as described in Bulletin 2001-01, could be identified and characterized by the visual examination technique used. None were found during this visual inspection.

d. No material deficiencies associated with concerns in NRC Bulletin 2001-01 or Bulletin 2002-02 were found.

e. The as low as is reasonably achievable (ALARA) radiation exposure controls for the visual examination process was effective with a completed job dose of 0.822 person rem, which was below the project estimate.

TI 2515/145, Section 04.04 c, requires that inspectors report lower-level issues concerning data collection and analysis, and issues deemed to be significant to the phenomenon described in Bulletin 2001-01. No lower-level issues were identified by the inspector.