January 30, 2004

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Dominion Resources
5000 Dominion Boulevard
Glenn Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION UNIT 2 AND UNIT 3 - NRC INTEGRATED INSPECTION REPORT 05000336/2003010 AND 05000423/2003010

Dear Mr. Christian:

On December 31, 2003, the U.S. Nuclear Regulatory Commission (NRC) completed inspections at your Millstone Power Station Unit 2 and Unit 3. The enclosed integrated inspection report documents the inspection findings, which were discussed on January 21, 2004, with Mr. J. Alan Price and other members of your staff.

The 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.

This report documents two NRC-identified findings of very low safety significance (Green). Both of these issues were determined to involve violations of NRC requirements. However, because of their 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. If you contest these non-cited violations, you should provide a response within 30 days of the date of these inspection reports, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Millstone Power Station.

Since the terrorist attacks on September 11, 2001, NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial power nuclear power plants during calendar year 2002 and the remaining inspection activities for Millstone Power Station were completed in calendar year 2003. The NRC will continue to monitor overall safeguards and security controls at Millstone.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response (if any) 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 Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

James M. Trapp, Chief
Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000336/2003010 and 05000423/2003010
w/Attachment: Supplemental Information

cc w/encl:
J. A. Price, Site Vice President - Millstone
C. L. Funderburk, Director, Nuclear Licensing and Operations Support
D. W. Dodson, Acting Manager - Licensing
L. M. Cuoco, Senior Counsel
V. Juliano, Waterford Library
S. Comley, We The People
J. Buckingham, Department of Public Utility Control
E. Wilds, Director, State of Connecticut SLO Designee
First Selectmen, Town of Waterford
D. Katz, Citizens Awareness Network (CAN)
R. Bassilakis, CAN
J. M. Block, Attorney, CAN
G. Winslow, Citizens Regulatory Commission (CRC)
J. Markowicz, Co-Chair, NEAC
E. Woollacott, Co-Chair, NEAC
R. Shadis, New England Coalition Staff
W. Meinert, Massachusetts Municipal Wholesale Electric Company
C. Brinkman, Manager, Washington Nuclear Operations

<u>Distribution w/encl</u>: **(VIA E-MAIL)**:
H. Miller, ORA
J. Wiggins, ORA
T. Madden, OCA
J. Jolicoeur, OEDO
R. Pulsifer, PM, NRR
G. Wunder, Backup PM, NRR
V. Nerses, PM, NRR
R. Architzel, NRR
S. Schneider, SRI - Millstone Unit 2 and Unit 3
J. Trapp, RI
K. Jenison, RI
D. Screnci, PAO
Region I Docket Room (with concurrences)

DOCUMENT NAME: G:\BRANCH 6\Millstone\ML2003010
After declaring this document "An Official Agency Record" it **<u>will</u>** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP		RI/DRP		RI/DRP	
NAME	SSchneider/*KMJ for*		KJenison/*KMJ*		JTrapp/*JMT*			
DATE	01/30/04		01/30/04		01/30/04		02/ /04	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	05000336, 05000423
License No.:	DPR-65, NPF-49
Report No.:	05000336/2003010 and 05000423/2003010
Licensee:	Dominion Nuclear Connecticut, Inc.
Facility:	Millstone Power Station, Unit 2 and Unit 3
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	September 28, 2003 - December 31, 2003
Inspectors:	S. M. Schneider, Senior Resident Inspector S. R. Kennedy, Resident Inspector K. A. Mangan, Resident Inspector C. M. Long, Reactor Engineer S. T. Barr, Operations Engineer, Division of Reactor Safety (DRS) B. A. Bickett, Reactor Inspector, DRS A. X. Lohmeier, Reactor Inspector, DRS K. M. Jenison, Senior Project Engineer, DRP N. T. McNamara, Emergency Preparedness Inspector, DRS M. C. Modes, Senior Reactor Inspector, Division of Reactor Safety (DRS) T. A. Moslak, Health Physicist, DRS
Approved by:	James M. Trapp, Chief Projects Branch 6 Division of Reactor Projects

Enclosure

CONTENTS

SUMMARY OF FINDINGS

IR 05000336/2003010, 05000423/2003010; 09/28/2003 - 12/31/2003; Millstone Power Station, Unit 2 and Unit 3; Other Activities.

The report covered a three-month period of inspection by resident inspectors and three announced inspections by regional inspectors. Two (Green) non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 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. NRC-Identified and Self-Revealing Findings

Unit 2

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III (Design Control), for the failure to take measures to assure that the design basis information was correctly translated into procedures for the installation of temporary cooling when normal cooling is lost to the West 480 Volt AC switchgear room and to verify that the design information was correct. Specifically, on June 4, 2003, the licensee installed temporary ventilation cooling equipment for the switchgear room in accordance with procedures that were inconsistent with design assumptions. Subsequently, certain of the design assumptions were determined to be incorrect. The licensee has appropriately revised the inadequate procedures.

  The finding is more than minor because the failure to provide the appropriate direction for establishing temporary cooling to the affected vital switchgear room resulted in inadequate room cooling, which if left uncorrected, could have resulted in exceeding the design temperature limit of the safety related and risk significant electrical equipment in the room. The finding is associated with the equipment performance attribute of the initiating events and mitigating systems cornerstones, and the containment SSC and barrier performance attribute of the barrier integrity cornerstone. Since more than one cornerstone was affected, a Reactor Safety Significance Determination Process Phase 2 analysis was performed. The analysis resulted in a finding of very low safety significance (Green) because the improper installation of the compensatory measures did not result in an actual loss of the supported 480 Volt AC System or electro hydraulic control functions. This finding is related to the licensee's Problem Identification and Resolution process. (Section 4OA5.1)

- Green.   This NCV was inadvertently not included in the summary of finding section of NRC Inspection Report 05000336/2003004, dated November 10, 2003. As a result, this finding is included in this report and the NCV will be documented under this report number.

  The inspectors identified a non-cited violation for failure to comply with 10 CFR 50, Appendix B, Criterion III, Design Control for two design changes, which adversely affected the charging system and for which post modification testing was not conducted to ensure the charging system could fulfill its design function under anticipated conditions.

  This finding is more than minor because it is associated with the equipment performance attribute of the mitigating systems cornerstone objective. Specifically, the charging system was not capable of providing adequate high pressure injection to the reactor coolant system following an initiating event that resulted in the simultaneous auto-start of the two standby charging pumps. A Phase 3 Significance Determination Process evaluation determined that this performance deficiency was of very low safety significance (Green). This finding is related to the cross cutting area of Problem Identification and Resolution process. (Section 4OA5.2)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

Unit 2 began this inspection period on September 28, 2003, operating at approximately 100 percent power. On October 11, 2003, the unit conducted a plant shutdown in preparation for Refueling Outage 15 (2R15). On November 26, 2003, after completion of the refueling outage, the reactor achieved criticality and was placed on-line. The generator was briefly taken off line on November 27, 2003, for main turbine overspeed testing and then reconnected to the electrical power grid. From November 27-30, 2003, the reactor was manually tripped twice and the turbine-generator was tripped several times due to high vibration on newly installed low pressure turbine rotors. The generator was last connected to the electrical power grid on November 30, 2003 and returned to 100 percent power on December 6, 2003. The unit remained at essentially 100 percent power for the remainder of the inspection period.

Unit 3 operated at essentially 100 percent power for the duration of the inspection period except on November 29, 2003 when power was reduced to 75 percent because of condenser fouling. The fouling was due to high winds at the intake structure. Power was restored to 100 percent on November 30, 2003.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 3 Samples)

a. Inspection Scope

The inspectors performed three samples of cold weather preparations. The inspectors reviewed Dominion's preparations for adverse weather and its impact on the protection of safety-related systems, structures, and components. The inspection was intended to ensure that the indicated equipment, its instrumentation, and its supporting structures were configured in accordance with Dominion procedures and that adequate controls were in place to ensure functionality of the systems. The inspectors reviewed licensee procedures and walked down the system. Documents reviewed during the inspection are listed in the Attachment.

Unit 2

- Refueling water storage tank
- Emergency diesel generator rooms heating and fuel oil system heat tracing

Unit 3

- Fire water storage tank and piping system

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdowns (71111.04Q - 6 Samples)

The inspectors performed six partial system walkdowns during this inspection period. The inspectors reviewed the documents listed in the Attachment to determine the correct system alignment. The inspectors conducted a walkdown of each system to verify that the critical portions of selected systems, such as valve positions, switches, and breakers, were correctly aligned in accordance with these procedures and to identify any discrepancies that may have had an effect on operability. Other licensee documents reviewed during the inspection are listed in the Attachment. The following systems were reviewed based on their risk significance for the given plant configuration:

Unit 2

- Partial alignment of "B" emergency diesel generator (EDG) while "A" EDG was inoperable (10/27/2003)
- Partial alignment of Shutdown Cooling during Mid-loop operations (11/4/2003)
- Partial alignment of containment spray (CS), Facility 2 during Facility 1 maintenance (12/12/2003)

Unit 3

- Partial alignment of "B" recirculation spray system train during maintenance on "A" train (10/22/2003)
- Partial alignment of "B" motor driven auxiliary feedwater (MDAFW) train during "A" MDAFW maintenance (11/16/2003)
- Partial alignment of station blackout diesel lineup during maintenance on "B" EDG (11/24/2003)

Complete System Walkdown (71111.04S - 1 Sample)

The inspectors completed a detailed review of the alignment and condition of the Unit 2 125VDC Vital DC system. The inspectors conducted a walkdown of the system to verify that the critical portions, such as valve positions, switches, and breakers, were correctly aligned in accordance with procedures and any discrepancies that may have had an effect on operability. The inspectors used the licensee procedures and other documents listed below to verify proper system alignment:

- OP 2345C, Revision 18-00, 125 Volt DC Station Battery System
- SP 2619C-001, Revision 012-03, Control Room Weekly Checks

- Drawing No. 25203-30024, Unit 2 125VDC Emergency and 120VAC Vital System

The inspectors also conducted a review of outstanding maintenance work orders to verify that the deficiencies did not significantly affect the 125 VDC Vital System function. The inspectors discussed system health with the system engineer and reviewed the condition report (CR) database to verify that equipment alignment problems were being identified and appropriately resolved.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

1. Quarterly Sample Review (71111.05Q - 12 Samples)

a. Inspection Scope

The inspectors performed walkdowns of 12 fire protection areas during the inspection period. The inspectors reviewed the licensee's fire protection program to determine the required fire protection design features, fire area boundaries, and combustible loading requirements for selected areas. The inspectors walked down those areas to assess the licensee's control of transient combustible material and ignition sources. In addition, the inspectors evaluated the material condition and operational status of fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The inspectors then compared the existing conditions of the inspected fire protection areas to the fire protection program requirements to ensure all program requirements were being met. Documents reviewed during the inspection are listed in the Attachment. The fire protection areas reviewed included:

Unit 2

- Motor Driven Auxiliary Feedwater Pump Cubicle - Turbine Building, 1' 6" Elevation (Fire Area T-3)
- Turbine Driven Auxiliary Feedwater Pump Cubicle - Turbine Building, 1"-6" Elevation (Fire Area T-4)
- West Battery Room - Auxiliary Building, 14'-6" Elevation (Fire Area A-23)
- B51 Motor Control Center Cubicle and General Area - Auxiliary Building, 14'-6" Elevation (Fire Area A-12A)
- Turbine Building, 14'-6" Elevation (Fire Area T-1, Zone A)
- Turbine Building, 14'-6" Elevation (Fire Area T-2)

Unit 3

- Engineered Safety Features Building - Fire Area 1
- Engineered Safety Features Building - Fire Area 2

- Engineered Safety Features Building - Fire Area 3
- Engineered Safety Features Building - Fire Area 4
- Engineered Safety Features Building - Fire Area 5
- Engineered Safety Features Building - Fire Area 6

b. Findings

No findings of significance were identified.

2. Annual Fire Drill Observation (71111.05A - 2 Samples)

a. Inspection Scope

Unit 2

The inspectors observed the annual Unit 2 fire brigade drill with offsite assistance on December 9, 2003, to evaluate the readiness of station personnel to respond to and fight fires. Twenty-four members of the Town of Waterford fire companies participated in the drill. The drill demonstrated response to a fire in the Unit 2 "B" emergency diesel generator room as simulated in the licensee's Fire Simulator. The inspectors observed fire brigade members regarding their use of protective clothing and appropriate turnout gear, including self-contained breathing apparatus, and their approach and methods in the combat of the fire, as well as their interaction with the offsite fire chiefs and responders. The inspectors observed implementation of the fire fighting strategies by the fire brigade and offsite response units, as well as the communications between participants throughout the drill. The inspectors reviewed the pre planned drill scenario objectives, determined whether drill scenario objectives were met and observed the post drill critique to verify that the licensee identified, discussed, and entered adverse conditions into the corrective action program. Documents reviewed during the inspection are listed in the Attachment.

Unit 3

The inspectors observed plant personnel performance during a fire brigade drill on December 8, 2003 to evaluate the readiness of station personnel to prevent and fight fires. The drill simulated a fire in the Unit 3 Auxiliary Boiler 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 reviewed the preplanned drill scenario objectives, determined whether drill scenario objectives were met and observed the post drill critique to verify that the licensee identified, discussed, and entered adverse conditions into the corrective action program. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 2 Samples)

a. Inspection Scope

The inspectors reviewed two samples of flood protection measures. These samples included Unit 2 East DC switchgear room and portions of the Unit 3 control building switchgear and motor control center areas. These reviews were conducted to evaluate the licensee's protection of the enclosed safety-related systems from internal flooding conditions. The inspectors performed a walkdown of the area, reviewed the Final Safety Analysis Report, as well as numerous licensing and design basis documents, including flooding calculations, to ensure as-found conditions in the inspected areas remained consistent with those indicated in the design basis documentation.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08P - 5 Samples)

a. Inspection Scope

The inspectors reviewed selected inservice inspection (ISI) and steam generator (SG) inspection activities. The inspectors also reviewed the utilization of mechanical clamping devices to repair two leaking pressurizer heater penetrations. The inspectors reviewed the following non-destructive test procedures for compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI, Appendix VIII as modified by 10 CFR 50.55a(a)(xv):

- NU-PDI-2, Rev. 1, "PDI Procedure for the Manual Ultrasonic Examination of Austenitic Piping Welds."
- MP-PDI-UT-1, Rev. 000, "PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds."
- MP-PDI-UT-2, Rev. 000, "PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds."

The inspectors verified that the technicians' individual qualifications satisfied the requirements of 10 CFR 50.55a(a)(xiv) and interviewed vendor technicians who performed the ultrasonic examinations (UT) of various reactor components to assess their level of knowledge. The inspectors reviewed a number of UT data packages for conformance with the requirements of the ASME Code, Section XI (a.k.a. Code). The inspectors accompanied the qualified ultrasonic test technicians and observed the examination of the pressurizer bottom head-to-vessel girth weld in conformance with ASME Section XI, Appendix VIII requirements.

The inspectors reviewed the "Steam Generator Eddy Current Data Analysis Reference Manual," the "Steam Generator Integrity Degradation Assessment" M2-EV-03-0040 Revision 4, and the "Steam Generator Condition Monitoring and Operational Assessment for Refueling Outage 14," to determine whether the SG tube inspection program documents were consistent with the Electric Power Research Institute (EPRI) "Pressurized Water Reactor Steam Generator Examination Guidelines."

The inspectors reviewed the planned activities for in-situ pressure testing of SG tubing, observed eddy current testing and interviewed technicians regarding the techniques and probes used during the inspections. The inspectors discussed the location and disposition of loose parts identified through ultrasonic testing and reviewed the criteria for plugging of SG tubes. The inspectors observed the data analysis and verification system used to assure the integrity of data analyzed by the remotely located primary data analysts. The inspectors reviewed the steam generator bobbin inspection results and the disposition of recordable findings.

The inspectors reviewed two pressurizer heater penetration sleeve leaks discovered by visual inspection during the previous outage and the subsequent discovery, during the current outage, of two additional heater penetration sleeve leaks. Through interviews the inspectors determined the previous leaks had been repaired using a pressure retaining clamp. The inspectors determined the licensee had received approval from the Office of Nuclear Reactor Regulation for the use of this non-ASME code repair.

The licensee had not removed a heater assembly or performed any examinations during the previous outage to determine the root cause for the leaks. However, the inspectors interviewed engineering personnel and determined that the licensee had concluded that the flaws were axial in orientation based on engineering judgement and industry operating experience. During this inspection, preliminary industry data became available which indicated the potential for circumferential flaws in these heater sleeves. The inspectors discussed this information with the licensee and the licensee subsequently elected to remove the four leaking pressurizer heater sleeves for examination. The licensee performed UT of the leaking pressurizer heater sleeves and confirmed that the flaws were axial in nature. The inspectors reviewed the results of the UT examinations to confirm that the licensee properly evaluated this condition. The heater penetrations were subsequently repaired by installation of mechanical clamping devices.

b. Findings

Introduction. The licensee identified reactor pressure boundary leakage at the pressurizer heater penetration sleeve. Similar leakage had been identified at two other sleeves during the previous outage.

Description. Technical Specification 3.4.6.2 states that reactor coolant system leakage shall be limited to no pressure boundary leakage in Modes 1 through 4. Contrary to this requirement, on October 11, 2003 while shutdown, the licensee identified two pressurizer heater penetration sleeve leaks as indicated by small boron deposits at the weld. The boric acid deposits indicated that the sleeve welds had been leaking during the previous operating cycle. The penetration sleeves are part of the reactor pressure boundary. The leaks were discovered during scheduled visual examination of the pressurizer heater penetration area. An Unresolved Item (URI) has been opened to determine if the licensee's corrective actions taken during a previous outage were adequate (discussed in IR 5000336/2002006 and 5000336/2003002) and to establish the significance of this current issue. The issue will be tracked as URI 05000336/2003010-05, Reactor Coolant System Pressurizer Pressure Boundary Leakage.

1R11 Licensed Operator Requalification Program (71111.11)

1. Licensed Operator Requalification Program Quarterly Inspection (71111.11Q - 2 Samples)

a. Inspection Scope

The inspectors observed the conduct of licensed operator simulator training for Unit 2 on November 13, 2003 and for Unit 3 on November 21, 2003. The inspectors evaluated the ability of each operations crew to mitigate the consequences of the failures presented in the accident scenarios and observed if the licensee's evaluators adequately address operator performance deficiencies that were identified during the exercise related to the training objectives. Additionally, the inspectors evaluated the use of formal communications, response to alarms, use of procedures and teamwork, and oversight provided by the shift supervisor. The inspectors also observed the evaluations of emergency plan action levels and notifications. Finally, the inspectors also reviewed the simulator physical fidelity as compared to the actual Unit control room. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

2. Licensed Operator Requalification Program Bi-Annual Inspection (71111.11B - 1 sample)

a. Inspection Scope

On December 22, 2003, the inspectors conducted an in-office review of licensee annual operating test results for 2003. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)." For both units, the inspectors verified that:

- Crew failure rate was less than 20%. (Crew failure rate was 0% at both units.)
- Individual failure rate on the dynamic simulator test was less than or equal to 20%. (Individual failure rate was 0% at both units.)
- Individual failure rate on the walkthrough test was less than or equal to 20%. (Individual failure rate was 0% at Unit 2 and 2% at Unit 3.)
- Overall pass rate among individuals for all portions of the exam was greater than or equal to 75%. (Overall pass rate was 99%.)

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12Q - 7 Samples)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of seven degraded conditions, involving safety-related structures, systems and/or components (SSC) for maintenance effectiveness during this inspection period. The inspectors reviewed licensee implementation of the Maintenance Rule (MR), 10 CFR 50.65, and verified that the conditions associated with the referenced CRs were appropriately evaluated against applicable MR functional failure criteria in accordance with associated scoping documents and procedures. The inspectors also discussed these issues with the system engineer and maintenance rule coordinator to verify that they were appropriately tracked against each system's performance criteria and that the systems were appropriately classified in accordance with MR implementation guidance. Documents reviewed during the inspection are listed in the Attachment. The following conditions were reviewed:

Unit 2

- #2 Steam Generator Main Steam Isolation Valve (2-MS-64B) Failed to Stroke Open (CR-03-11381)
- Pressurizer Power Operated Relief Valve Pilot Valve Leak (CR-03-11745)
- “A” EDG Emergency Stop Pushbutton Failure (CR-03-10518)
- “B” Charging Pump 3-Way Valve Installed Backwards (CR-03-09340)

Unit 3

- 3 HVR*FN13A "A" Charging Pump Cubicle Supply Fan High Vibrations (CR-03-11664)
- RMS-34 Out Of Service (CR-03-06005)
- Radiation Consul Unavailability Due To Lockout (CR-03-10523)

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13 - 8 Samples)

a. Inspection Scope

The inspectors reviewed eight maintenance risk assessments during the inspection period. The inspectors utilized the Equipment Out of Service (EOOS) quantitative risk assessment tool to evaluate the risk of the plant configurations and compared the result to the licensee's stated risk. The inspectors also used shutdown risk safety assessments for qualitative reviews of shutdown and refueling outage activities. The inspectors verified that the licensee entered appropriate risk categories and implemented risk management actions as necessary. The inspectors reviewed these activities against the requirements of 10 CFR 50.65(a)(4) and the guidance contained in NUMARC, 93-01, "NEI Industry Guidance for Maintaining the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 91-06, "Industry Guidance for Shutdown Operations," and licensee outage risk assessment guidance. Documents reviewed during the inspection are listed in the Attachment. The inspectors verified the conduct and adequacy of risk assessments for plant conditions affected by the conduct of the following scheduled maintenance and testing activities:

Unit 2

- Refueling outage 2R15 Shutdown Safety Assessment on 10/16/2003
- Refueling outage 2R15 Shutdown Safety Assessment on 10/18/2003
- Refueling outage 2R15 Shutdown Safety Assessment on 10/26/2003
- Refueling outage 2R15 Shutdown Safety Assessment on 11/04/2003
- Refueling outage 2R15 Shutdown Safety Assessment on 11/17/2003

Unit 3

- Work Schedule on 11/10/03 - maintenance and testing of safety injection pump and air-conditioning units during high wind conditions
- Work Schedule on 11/16/03 - maintenance and testing of station blackout diesel and auxiliary feed pump
- Work Schedule on 12/15/03 - risk associated with station blackout diesel, service water header, and auxiliary feedwater outages

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions (71111.14 - 2 Samples)

a. Inspection Scope

The inspectors reviewed two events which demonstrated personnel performance in coping with the non routine evolutions and transients identified below. The inspectors observed operations in the control room, reviewed applicable operating and alarm response procedures and TSs, plant process computer indications, and control room shift logs to evaluate the adequacy of the licensee response to these events. The inspectors also verified the events were entered into the corrective action program to resolve identified adverse conditions. Documents reviewed during the inspection are listed in the Attachment. The inspectors also reviewed the licensee response to a loss of shutdown cooling event which was assessed under Inspection Procedure 71153, Event Followup (Section 4OA3.1). This review is not included as a sample in this section.

Unit 2

- On October 14, 2003, operators responded to a loss of shutdown cooling and declared an Unusual Event when an uncontrolled reactor coolant system temperature, increase greater than 10 degrees, occurred. This event was reviewed under Inspection Procedure 71153, Event Followup, (Section 4OA3), and is not included as a sample under this inspection procedure.

- On October 26, 2003, during performance of the Safety Injection Actuation System (SIAS) pushbutton portion of the Facility 1 Loss of Normal Power (LNP) test, the "A" EDG unexpectedly shed electrical load, when the SIAS pushbutton was manually actuated. The operators responded by terminating the LNP test and attempted to trip the diesel using the emergency trip pushbuttons on the control board with no response. As the operators were manually reducing load prior to opening the EDG output breaker, the breaker tripped on reverse power. The diesel was then manually tripped locally using the fuel racks. The licensee determined that the cause of the "A" EDG unexpected load shedding and the failure of the emergency pushbutton to operate was caused by a faulty alarm reset switch. The alarm reset switch was replaced and EDG was returned to service on October 28, 2003 and the licensee evaluated the "B" EDG alarm reset switch for extent of condition.

- On November 27, 2003, while at approximately 30 percent reactor power, Operations personnel manually tripped the reactor in response to high and increasing turbine vibration. When turbine vibration increased above the high alarm set point, operators manually tripped the reactor as required by procedures and entered the Standard Post Trip Actions and the Reactor Trip Recovery procedures. The licensee determined that the high turbine vibration was caused by rubbing between the turbine casing and the new monoblock rotors which were installed during the 2R15 outage. The licensee had anticipated high turbine vibration with new turbine monoblock rotors based on published industry operating experience which reported that several sites had experienced rubbing at the monoblock and vibrations during startup following installation. The licensee experienced a total of six turbine trips and two manual reactor trips due to rubbing induced high vibration on the turbine. Following the final manual turbine trip on November 30, 2003, the turbine vibration stabilized within acceptable limits.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 7 Samples)

a. Inspection Scope

The inspectors reviewed seven operability determinations associated with degraded or non-conforming conditions to ensure that operability was justified and that mitigating systems or those affecting barrier integrity 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 licensee performance to ensure all related technical TS and Final Safety Analysis Report (FSAR) requirements were met. Documents reviewed during the inspection are listed in the attachment. The inspectors reviewed the following degraded or non-conforming conditions:

Unit 2

- Service Water Leak at Flanged Connection to "A" Emergency Diesel Generator (CR-03-10524)
- Pressurizer Power Operated Relief Valve Operability due to Pilot Valve Leak (CR-03-11745)
- Steam Generator #1 Automatic Dump Valve Leak (CR-03-12103)
- Containment Sump Operability (NUREG 6806)

Unit 3

- Evaluation of "C" Service Water (SW) Pump Coupling Degradation (CR-03-09578)
- Evaluation of Degraded SW Motor-Operated Valve (MOV) Couplings (CR-03-09656)
- Reactor Coolant Pump Underspeed Reactor Trip Signal (CR-03-10884)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16 - 4 Samples)

a. Inspection Scope

Selected Operator Workarounds (2 samples).

The inspectors reviewed two risk significant operator workarounds during the inspection period. The inspectors evaluated each condition to determine if it should have been classified as an operator workaround and if there was any effect on human reliability in responding to an initiating event. Documents reviewed during the inspection are listed in the Attachment.

Unit 2

- Operator Workaround #520, Leakage from #1 Safety Injection Tank (SIT) requires periodic filling to maintain TS operability

Unit 3

- Operator Response for Electric Fire Pump operation related to inoperable minimum flow line

Cumulative Effects of Operator Workarounds (2 samples).

The inspectors reviewed the current listing of active operator workarounds for Millstone Unit 2 and Unit 3. The inspectors reviewed applicable site procedures, performance indicator (PI) data and COP 200.9, Operational Performance Status. The review was conducted to verify that Millstone procedures and practices provided the necessary guidance to plant personnel, that the cumulative effects of the known operator workarounds were addressed and that the overall impact on the affected systems was assessed by the licensee. The inspectors independently assessed the cumulative impact of known operator workarounds to determine if it adversely affected the ability of plant operators to implement emergency procedures, respond to plant transients, or perform normal functions within the expectations of the established Dominion risk models. In support of this assessment, the inspectors reviewed various condition reports (CRs) regarding operator workarounds, and verified that workarounds were being identified, tracked, and resolved in the licensee's corrective action program.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A - 2 Samples)

a. Inspection Scope

The inspectors reviewed one sample of a permanent plant modification on each unit. The two samples included a Unit 2, 125 Volt DC battery charger modification (M2-02007) and a Unit 3, incipient fire detection system modification (M3-01008) in the cable spreading room. The inspectors performed a walkdown of each area and reviewed the applicable Final Safety Analysis Report sections, licensing and design basis documents, calculations, construction and modification specifications and standards, implementing procedures, licensee inspection and closeout procedures, contractor and vendor documentation, seismic, IEEE and other national standards and the Dominion 50.59 process. These reviews were conducted to ensure (1) the modified components, structures and systems remained consistent with the assumptions indicated in the design basis documents, (2) that system availability, reliability, functional capability and safety function were maintained, and (3) no unrecognized conditions that significantly affected risk were introduced into the plant as a result of the modifications.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 7 Samples)

a. Inspection Scope

The inspectors reviewed seven post maintenance test (PMT) activities during this inspection period. The inspectors reviewed these activities to determine whether the PMT adequately demonstrated that the safety-related function of the equipment was satisfied given the scope of the work specified. In addition, the inspectors evaluated the applicable test acceptance criteria to verify consistency with the associated design and licensing bases, as well as TS requirements. In addition, the inspectors verified that conditions adverse to quality were entered into the corrective action program for resolution. Documents reviewed during the inspection are listed in the Attachment. The following maintenance activities and their post maintenance tests were evaluated:

Unit 2

- "A" Service Water Dump Discharge Check Valve Inspection (M2-03-12472)
- Feedwater Check and Containment Isolation Valve (2-FW-5B) Inspection and Actuater Troubleshooting and Assembly (M2-01-06626)
- “B” Charging Pump 3-Way Valve Installation (M2-03-08089)
- “C” High Pressure Safety Injection 5 Year Coupling Inspection (M2-00-14079)

Unit 3

- Turbine Driven Auxiliary Feedwater Pump following Control Valve Maintenance (M3-03-08402)
- "B" Service Water Strainer Repair (M3-02-07623)
- Emergency Diesel Generator Fuel Oil Transfer Pump Repair (M3-03-11819)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20 - 1 Sample)

a. Inspection Scope

The licensee conducted a Unit 2 refueling outage from October 11, 2003 through November 25, 2003. The inspectors evaluated the outage plan and outage activities to confirm that the licensee had appropriately considered risk, had developed risk reduction and plant configuration control methods, had considered mitigation strategies in the event of losses of safety functions, and had adhered to license and TS requirements. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes, reactor vessel draindowns, mid-loop operations, vacuum fill, shutdown cooling operation, fuel handling and other maintenance, testing, and outage related activities. Additionally, the inspectors conducted a walkdown of the containment, prior to its final closeout, to ensure no loose material or debris was present which could be transported to the containment sump. The inspectors also observed startup, heatup and power ascension activities following the outage. Documents reviewed during the inspection are listed in the Attachment. The inspectors also verified that conditions adverse to quality were entered into the corrective action program for resolution. Some of the specific activities observed included:

- New fuel receipt inspection
- Fuel handling, core loading, and fuel element assembly tracking
- Core Alterations
- Reactor cooling system pressure, level, and temperature instruments operability
- Decay heat removal system monitoring (Shutdown Cooling System)
- Mid-loop and reduced inventory operations
- Tagout and tagout clearance control
- Vacuum fill operations
- Reactor Startup and Power Operations
- Low Power Physics Testing

b. Findings

Introduction. The inspectors identified an issue concerning reactor vessel level disagreements while draining the refuel pool and reactor vessel. This issue is unresolved pending completion of inspection followup activities and completion of a safety significance determination, as appropriate.

Description. On November 15, 2003, the RCS mid-loop wide range level indication failed to come on scale during the draindown of the refuel pool to the refueling water storage tank (RWST). With the pressurizer empty and while waiting for the RCS mid-loop wide range level indication to come on scale, the operators were monitoring refuel pool level by remote camera and mass balance calculations. Operators, on the previous shift, had released the watchstander at the refueling pool assigned to monitor pool level so the remote camera monitor and mass balance calculations were being used as independent inventory checks during the draindown.

When the wide range level indicator did not come on scale at 99 inches above hot leg centerline as expected, the operators responded by continuing to drain down toward the desired level. Operators drained the reactor vessel to a level of approximately 96 inches above hot leg centerline (as determined by mass balance calculations) and then stopped due to disagreement between the remote camera indication (the remote camera indicated approximately 125 inches above hot leg centerline) and the mass balance calculations. An operator was sent into containment to determine the actual level of the refuel pool and reported that the refuel pool level was at 96 inches which agreed with the mass balance calculation. The operators investigated and determined that the remote camera indication was being read incorrectly and that the locally observed refueling pool level agreed with the mass balance calculation. Operators adjusted the remote camera picture and verified its indication now agreed with the mass balance calculation and with locally observed refuel pool level. With a spotter now inside containment to monitor refuel pool level and with the mass balance calculation agreeing with the remote camera indication, the operators continued the draindown to 86 inches above hot leg centerline.

When the local refuel pool level spotter was released on the previous shift, this draindown evolution was essentially conducted with only one reliable means of inventory tracking, specifically, the mass balance calculations. Additionally, operators did not investigate why the wide range level indicator did not come on scale as expected around 99 inches above hot leg centerline nor did they investigate this problem when other inventory indicators were subsequently shown to be in disagreement. The licensee determined that the wide range level indication did not come on scale as expected because the instrument had previously been isolated to support unrelated testing and was not restored to service following this testing. Once returned to service, the instrument agreed with the visual indications and the mass balance. This issue is being tracked as URI 05000336/2003010-01: Conducted reactor vessel drain down without sufficient level indication.

1R22 Surveillance Testing (71111.22 - 9 Samples)

a. Inspection Scope

The inspectors reviewed nine surveillance activities, including two inservice testing (IST) activities, to determine whether the testing adequately demonstrated the ability of the equipment to perform its intended safety-related function. The inspectors attended pre-job briefs, verified that selected prerequisites and precautions were met and that the tests were performed in accordance with the procedural steps. Additionally, the inspectors evaluated the applicable test acceptance criteria to verify consistency with associated design basis, licensing bases and TS requirements, and that the applicable acceptance criteria were satisfied. The inspectors also verified that conditions adverse to quality were entered into the corrective action program for resolution. The following surveillance activities were evaluated:

Unit 2

- Main Steam Safety Valve Test (SP-2730B)

- Auxiliary Feedwater Cross-Connect (2-FW-44) Valve Stroke and Timing IST (SP2610C)
- Containment Spray System Alignment Check and Valve Tests, Facility 1 (SP 2606)
- Integrated Test of Facility 1 Engineered Safeguards Features Components (SP 2613G)
- Low Power Physics Test (EN 21004K)

Unit 3

- Protection Set 1 Quarterly Operational Test (SP 3443A21-001)
- RSS Valve Stroke Time Test - Train B (SP 3606.9)
- AFW Isolation Valve Stroke Testing (SP 2610C)
- Security Diesel and Auto Transfer Switch to Technical Support Center (OP 2396)

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 2 Samples)

a. Inspection Scope

The inspectors reviewed two temporary modifications to verify that the temporary modification did not affect the safety function of important safety systems. The inspectors reviewed each temporary modification and its associated 10 CFR 50.59 screening against the Final Safety Analysis Report (FSAR) and TS to ensure the modification did not affect system operability or availability. Documents reviewed during the inspection are listed in the Attachment.

Unit 2

- Use of Inflatable Pipe Plugs for Maintaining Containment Closure

Unit 3

- Isolation of CO2 Fire Suppression System to Cable Spreading Room

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04 - 1 Sample)

a. Inspection Scope

A regional in-office review was conducted of licensee-submitted revisions to the emergency plan, implementing procedures and emergency action levels which were received by the NRC during the period of September 2003 through December 2003. A thorough review was conducted of plan aspects related to the risk significant planning standards (RSPS), such as classifications, notifications and protective action recommendations. A cursory review was conducted for non-RSPS portions. These changes were reviewed against 10 CFR 50.47(b) and the requirements of Appendix E and they are subject to future inspections to ensure that the combination of these changes continue to meet NRC regulations. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 4, and the applicable requirements in 10 CFR 50.54(q) were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 - 2 Samples)

a. Inspection Scope

The inspectors observed the conduct of licensed operator simulator training for Unit 2 on November 13, 2003 and December 9, 2003. The inspectors evaluated each Operations crew activities related to evaluating the scenario and making proper classification and notification determinations. Additionally, the inspectors assessed the ability of the licensee's evaluators to adequately address operator performance deficiencies identified during the exercise. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS2 ALARA Planning and Controls (71121.02 - 15 Samples)

a. Inspection Scope

During the period November 3 - 7, 2003, the inspectors conducted the following activities to verify that the licensee was properly implementing operational, engineering, and administrative controls to maintain personnel exposure as low as is reasonably achievable (ALARA) for tasks conducted during the Unit 2 refueling outage. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and the licensee's procedures. This inspection activity represents the completion of fifteen (15) samples relative to this inspection area and the biennial inspection requirement.

Radiological Work Planning

- The inspectors reviewed pertinent information regarding cumulative exposure history, current exposure trends, and ongoing activities to assess current performance and exposure challenges. The inspectors determined the plant's 3-year rolling collective average exposure.

- The inspectors reviewed the refueling outage work scheduled during the inspection period and the associated work activity exposure estimates. Scheduled work reviewed included pressurizer heater removal, pressurizer mechanical nozzle sealing assembly (MNSA) installation, and reactor head penetration repairs.

- The inspectors reviewed procedures associated with maintaining worker dose ALARA and with estimating and tracking work activity specific exposures.

- The inspectors reviewed the 2R15 Daily Dose Summary Reports, detailing the worker estimated and actual exposures, through November 7, 2003, for jobs performed during the refueling outage.

- The inspectors evaluated the exposure mitigation requirements, specified in ALARA Reviews (AR), and compared actual worker cumulative exposure to estimated dose for tasks associated with these work activities. Jobs reviewed included Reactor Vessel Disassembly/Reassembly (AR 2-03-01), Reactor Head Penetration/Bare Metal Visual Inspection/Head Stand Work & Head Replacement Measurements (AR 2-03-20/21), Chemical & Volume Control System (CVCS) Improvements (AR 2-03-33), and Pressurizer Heater Inspection/Removal/Replacement (AR 2-03-29).

- The inspectors evaluated the departmental interfaces between radiation protection, operations, maintenance crafts, and engineering to identify missing ALARA program elements and interface problems. The evaluation was accomplished by interviewing the ALARA Coordinator, attending an ALARA Council meeting (on 11/06/03), reviewing ALARA Council Meeting minutes (dated 10/22/03 & 10/30/03); reviewing outage related Nuclear Oversight Observation Reports; and attending pre-job briefings for jobs in progress (pressurizer heater removal and MNSA clamp installation).

- The inspectors compared the person-hour estimates provided by maintenance planning and other work groups with actual work activity time requirements and evaluated the accuracy of these time estimates. Specific work activities evaluated included the Reactor Vessel Head Penetration Repairs, C-Reactor Coolant Pump (C-RCP) Motor Replacement, and Reactor Refueling Activities.

- The inspectors determined if work activity planning included the use of temporary shielding, system flushes, and operational considerations to further minimize worker exposure. The inspectors reviewed the recently installed permanent shielding for the shutdown cooling system piping and letdown piping, and temporary shielding installed for reactor head penetration repairs and C-RCP motor replacement. The inspectors also reviewed reactor coolant chemistry data taken subsequent to the peroxide flush.

- The inspectors reviewed the revised ALARA Review for the reactor head penetration inspections/repairs to determine if revised dose projections were properly justified and that lessons learned from the activity were being addressed.

Verification of Dose Estimates and Exposure Tracking Systems

- The inspectors reviewed the assumptions and basis for the current annual collective exposure estimate and the refueling outage dose projection. The inspectors reviewed six (6) personnel contamination event reports, whole body counting data, and related calculations for internal dose and shallow dose estimates for selected personnel.

- The inspectors reviewed the licensee's method for adjusting exposure estimates, and re-planning work, when emergent work was encountered. The inspectors reviewed the Work-In-Progress (WIP) ALARA Reviews to determine if actual exposure was correlating with forecasted estimates. WIP reviews evaluated included reactor disassembly (WIP 2-03-01), steam generator (S/G) eddy current testing (WIP 2-03-02-03), S/G nozzle dam installation (WIP 2-02-02), S/G secondary side sludge lancing (2-03-03), S/G foreign object retrieval (WIP 2-03-03-02), refueling (WIP 2-03-04-02), and under-head inspection activities (WIP 2-03-05).

- The inspectors reviewed the licensee's exposure tracking system to determine whether the level of dose tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support the control of collective exposures. Included in this review were the Radiation Work Permits (RWP) for CVCS improvements (RWP-250), pressurizer heater replacement (RWP-310), and reactor head penetration repairs (RWP-353).

Job Site Inspection and ALARA Control

- The inspectors observed maintenance and operational activities being performed for pressurizer heater replacement, reactor head penetration repairs, and CVCS weld inspections to verify that radiological controls, such as required surveys, job coverage, and contamination controls were implemented; personnel dosimetry was properly worn; and that workers were knowledgeable of work area radiological conditions.

- The inspectors reviewed the exposures of individuals in selected work groups, including mechanical maintenance, contractors, and radiation protection to determine if supervisory efforts were being made to equalize doses among the workers. The inspectors also interviewed the Radiation Protection Manager and a contractor supervisor regarding the equalization of dose for reactor head repairs.

Source Term Reduction and Control

- The inspectors reviewed the current status and historical trends of the plant's source terms. Through interviews with the Chemistry Supervisor and the ALARA Coordinator, the inspectors evaluated the licensee's source term control strategy. Specific strategies being employed by the licensee included post shutdown peroxide flushes of reactor coolant piping, and installation of permanent shielding on shutdown cooling piping and letdown piping.

Radiation Worker Performance

- The inspectors observed radiation worker and radiation protection technician performance during pressurizer heater replacement, reactor head penetration repairs, and CVCS weld inspections. The inspectors determined whether the individuals were aware of current radiological conditions, access controls, and that the skill level was sufficient with respect to the radiological hazards and the work involved. The inspectors also observed worker training regarding the chamfering of reactor head penetration welds.

- The inspectors attended the pre-job briefings for exposure significant tasks performed during the inspection period to determine the adequacy and accuracy of information provided to workers. Pre-job briefings attended included pressurizer heater replacement and MNSA clamp installation.

- The inspectors reviewed condition reports, related to radiation worker and radiation protection technician errors, and personnel contamination reports (PCR) to determine if an observable pattern traceable to a similar cause was evident.

Declared Pregnant Workers

- The inspectors reviewed the radiological control records for one (1) declared pregnant worker performing outage related activities during the inspection period.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151 - 6 Samples)

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators (PIs) listed below to verify the accuracy of the PI data reported during that period. The PI definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify the basis for reporting each data element.

Unit 2 Reactor Safety Cornerstones (2 Samples)

- Auxiliary Feedwater System
- Safety System Functional Failures

Unit 3 Reactor Safety Cornerstones (2 Samples)

- Reactor Coolant System Specific Activity
- Reactor Coolant System Leak Rate

The inspectors reviewed licensee event reports, monthly operating reports, plant process shift logs, condition reports and NRC inspection reports to identify safety system equipment failures and unavailability that occurred from the 4th quarter of 2002, through the 3rd quarter 2003. Additionally, the inspectors observed a primary sample to verify the adequacy of the procedure and ensure that activity levels in the primary system were being accurately determined. The inspectors also reviewed daily logs to determine daily leakage rates from the primary system. Documents reviewed during the inspection are listed in the Attachment. The inspectors compared this information with the licensee's data reported to the NRC for the performance indicators (PI) listed above to verify that PI reporting and proximity to PI thresholds published on the NRC website were accurate.

Occupational Exposure Control Effectiveness (1 Sample)

The inspectors reviewed implementation of the licensee's Occupational Exposure Control Effectiveness Performance Indicator (PI) Program for Unit 2 and Unit 3. Specifically, the inspectors reviewed recent Condition Reports (CR), and associated documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures 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. This inspection activity represents the completion of one (1) sample relative to this inspection area; completing the annual inspection requirement.

RETS/ODCM Radiological Effluent Occurrences (1 Sample)

The inspectors reviewed a listing of relevant effluent release reports for the period October 1, 2002 through October 1, 2003, for issues related to the public radiation safety performance indicator, which measures radiological effluent release occurrences per site that exceed 1.5 mrem/quarter whole body or 5.0 mrem/quarter organ dose for liquid effluents; 5 mrads/quarter gamma air dose, 10 mrad/quarter beta air dose, and 7.5 mrads/quarter for organ dose for gaseous effluents. This inspection activity represents the completion of one (1) sample relative to this inspection area; completing the annual inspection requirement for Unit 2 and Unit 3.

The inspectors reviewed the following documents to ensure the licensee met all requirements of the performance indicator from the fourth quarter 2002 to the fourth quarter 2003 (4 quarters):

- monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases;
- quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and
- dose assessment procedures.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

1. Daily Review of Problem Identification and Resolution (PI&R)

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for followup, the inspectors performed a daily screening of items entered into the licensee's corrective action program. This review was accomplished by reviewing summary lists of each condition report, attending screening meetings, and accessing the licensee's computerized database.

b. Findings

No findings of significance were identified.

2. ALARA Planning and Controls

a. Inspection Scope

The inspectors reviewed Nuclear Oversight Observation Reports, a Nuclear Oversight Audit (MP-03-A14) and fifteen (15) Supervisory Work Observation reports relating to the implementation of physical, engineering, and administrative controls for performing work in radiologically controlled areas. The inspectors also reviewed sixteen (16) Condition Reports, relating to maintaining personnel exposure ALARA, to evaluate the threshold for identifying, evaluating, and resolving problems in implementing the ALARA program. This review was conducted against the criteria contained in 10 CFR 20, TS, and the licensee's procedures.

b. Findings

No findings of significance were identified.

3. Cross-References to PI&R Findings Documented Elsewhere

- Section 4OA5.1 describes a finding for the failure to take measures to ensure that procedures were effective to implement compensatory cooling to the West 480 VAC switchgear room in June 2003. This issue is attributable to the PI&R cross-cutting area because the licensee failed to identify the improper installation of compensatory measures and failed to evaluate the effectiveness of the installation once actions beyond the procedural guidance were necessary to maintain acceptable 480 VAC room temperature.

- Section 4OA5.2 discusses a finding for the failure to adequately control two design changes to the charging system to ensure the charging system could fulfill its design function under anticipated conditions. As a result, when the charging system was called upon to perform its design function following a complicated reactor trip on March 7, 2003, the system failed and an alternate injection path had to be established to support plant cooldown.

4OA3 Event Followup (71153 - 4 Samples)

1. Loss Of Shutdown Cooling During Unit 2 Refueling Outage

a. Inspection Scope

The inspectors observed control room personnel actions following a loss of shutdown cooling on October 14, 2003. The inspectors arrived in the control room within two hours of the onset of the event and observed the followup actions by the operators, monitored plant conditions, and were briefed by the Shift Manager who was in charge in the control room during the event. As part of the followup to this event, the inspectors reviewed control room logs, discussed the event response with operators, reviewed response procedures, and attended Event Review Team (ERT) and Site Operations Review Committee (SORC) meetings which discussed the event, an apparent cause, operator response, and corrective actions. Documents reviewed and considered for evaluation of the licensee response to this event are listed in the Attachment.

b. Findings

Introduction. An issue was identified related to the implementation of vendor technical manual instructions into a procedure used to parallel AC sources which supply vital shutdown cooling loads. This issue is unresolved pending completion of follow-up inspections regarding plant configuration and completion of the significance determination evaluation.

Description. On October 14, 2003, a loss of shutdown cooling occurred with the plant in a cold shutdown condition. Operators were preparing to drain to mid-loop when the power supply to the shutdown cooling system valves failed. While performing OP-2345B, "120 Volt Vital Instrument AC System," a power failure caused the shutdown cooling heat exchanger throttle valve to fail closed and the shutdown cooling heat exchanger bypass valve to fail open. This bypassed the shutdown cooling heat exchanger which resulted in a reactor coolant system temperature increase from approximately 101°F (degrees Fahrenheit) to 115°F. Due to this uncontrolled reactor coolant system temperature increase of greater than 10°F operators declared an Unusual Event in accordance with their Emergency Action Level tables. NRC inspectors responded to the site to observe the licensee's actions. Power was restored to the subject valves, within approximately 15 minutes and operators restored shutdown cooling and stabilized reactor coolant system temperature.

The licensee established an Event Review Team to investigate event and to determine an apparent root cause. The team determined that the procedure, OP-2345B, used by operators to restore the vital 120 volt AC inverter to service following completion of preventive maintenance on the system, was in error. Specifically, the portion of the procedure used to parallel the two AC sources, that could provide power to the vital shutdown cooling loads, did not prevent a switch lineup that allowed both power supplies to be paralleled without synchronous protection. When the power supplies were paralleled, they were out of phase which resulted in the failure of both power supplies and the repositioning of shutdown cooling components to bypass the shutdown cooling heat exchanger. Further investigation by the licensee determined that this procedure was developed in October 1992 from a vendor technical manual which described the operation of the system. This version of the vendor technical manual allowed the improper switch lineup. A later vendor technical manual revision had been issued in August of 1992 which corrected this error and was available for use by the licensee in February of 1993. However, the licensee failed to implement the technical requirements of the new revision at that time and the vulnerability to this particular switch lineup has existed since that time. The failure to implement the revised requirements resulted in the loss of both power supplies to the vital shutdown cooling equipment.
(URI 05000336/2003010-02 - Loss of Shutdown Cooling when AC supply sources were paralleled out of phase)

2. (Discussed) LER 05000336/2003-005-00, Loss Of Shutdown Cooling During Refueling Outage

On October 14, 2003 with Unit 2 in Mode 5, a loss of shutdown cooling occurred when power was lost to a vital 120 VAC electrical panel. Upon loss of power to this panel, a common shutdown cooling heat exchanger throttle valve failed to close and a common shutdown cooling heat exchanger bypass valve failed to open. This resulted in bypassing flow around both trains of shutdown cooling heat exchangers which resulted in the loss of control of reactor coolant temperature. When more than 10°F increase in reactor coolant temperature occurred, the licensee declared an Unusual Event. Power was restored to the 120 VAC electrical panel within 15 minutes, the system lineup was restored, and control of reactor coolant temperature was regained. The inspectors responded to the plant and observed licensee followup actions in the control room. More detailed inspection results associated with the event discussed in this LER are included in Section 4OA3.1. The licensee documented this loss of shutdown cooling event in CR-03-09838. This LER remains open.

3. (Closed) LER 05000336/2003003-01, The Charging System Did Not Perform Its Design Function In Response To Falling Pressurizer Level

On March 7, 2003 with Unit 2 in Mode 1 at approximately 100%, an automatic reactor trip occurred while performing Reactor Protection System matrix logic and trip path relay testing. The reactor trip resulted in a reactor coolant system cooldown, and a drop in pressurizer level followed by the auto-starting of both standby charging pumps in addition to the one in operation. The combined flow of the three pumps fell and became erratic, varying from a high of 50 gpm to a low of 0 gpm. Because of this unexpected response, plant operators declared all three charging pumps inoperable and entered the Action Statement for TS 3.0.3. Pressurizer level was restored using a charging pump aligned to the alternate charging flow path through the high-pressure safety injection line, and a plant cooldown was commenced. The LER concluded that the direct cause of this event was that the charging pump discharge pulsation dampeners were inadequately designed to prevent pressure spikes caused by simultaneous pump starts. During Refueling Outage-15, improved discharge pulsation dampeners were designed and installed. More detailed inspection results associated with the event discussed in this LER are included in Inspection Report 05000336/2003006 dated June 30, 2003 and in Section 4OA5.2 of this report. The licensee documented the failed equipment in CR-03-03359. This LER is closed.

4. (Discussed) LER 05000336/2003004-00, Reactor Coolant System (RCS) Pressure Boundary Leakage Event

On October 11, 2003, while Millstone Unit 2 was in a refueling outage, the licensee identified that very small leakage through two pressurizer heater sleeve penetrations had occurred while the unit was in operation. The leakage was similar to that previously identified by the licensee and discussed in NRC Inspection Report 05000336/2002002, dated May 12, 2003. Further inspection of pressurize leaks at Millstone Unit 2 will be conducted under URI 05000336/2003010-05, Reactor Coolant System Pressurizer Pressure Boundary Leakage (section 71111.08).

4OA5 Other Activities

1. (Closed) URI 050000336/2003003-01; West 480 Volt AC Switchgear Ventilation Lineup

Introduction. A Green non-cited violation (NCV) was identified for the failure to take measures to assure that the design basis was correctly translated into the procedure for the installation of temporary cooling to the vital West 480 Volt AC (W480VAC) switchgear room and for the failure to ensure design calculation assumptions for air flows in the room were correct.

Description. On June 4, 2003, the licensee identified a service water (SW) leak on the cooling supply line to the safety-related cooler in the W480VAC switchgear room. The licensee isolated the cooler to affect repairs and entered Technical Requirements Manual (TRM) Limiting Condition of Operation (LCO) 3/4.8.2.1B.b, Onsite Power AC Distribution Systems Electrical Switchgear Room Ventilation. The TRM LCO action requires the immediate implementation of switchgear ventilation system compensatory actions in accordance with Operating Procedure (OP) 2315D - "Vital Electric Switchgear Room Cooling" or enter the applicable TS action statement for the associated switchgear. The OP 2315D compensatory actions for the W480VAC room cooling consists of creating an air flow path through the room by opening supply and exhaust doors and, if required, installing two Appendix R air blowers and flexible ducting.

Following implementation of these compensatory cooling measures on June 4, 2003, room temperature increased from approximately 70°F to 92°F within a few hours. Room temperature continued to increase, however more gradually, over the next two days until it reached approximately 102°F 46 hours after the initial establishment of the temporary room cooling. At this point, operators identified that the switchgear room temperature could not be maintained below the licensee's design temperature limit of 104°F with the temporary cooling as installed. Additional measures were taken to lower the room temperature that included installing suction and discharge ducting, and lowering the room temperature via a non-vital air-conditioning system that serviced the hallway from which the blowers were drawing air. The W480VAC switchgear room temperature was stabilized as a result of these efforts, however, the licensee did not perform an operability evaluation to verify that the additional actions required to maintain room temperature below 104°F would be effective for all design basis accidents.

On June 10, 2003, the inspectors reviewed the installation of the compensatory measures and compared them to the licensee's design calculations and technical evaluation, developed in February 1999 and October 2000, to verify that the design basis for the compensatory actions was sufficient to ensure operability of the vital W480VAC switchgear during all design basis events. The inspectors determined that the flow path evaluated in the design basis calculation had not been translated to OP 2315D (Revision 11) which was used to install the temporary cooling to the W480VAC switchgear room on June 4, 2003. Specifically, the design calculation required that a door on the opposite side of the switchgear room, opposite the Appendix R fan location, be opened to provide a flow path through the room. Revision 11 of OP 2315D did not instruct the operators to open this "exhaust" door. The inspectors also determined that a later revision to the procedure was issued during the time the compensatory measures were in place. This revision would have directed this door to be opened, however, it was not implemented and as a result the operators did not recognize the need to open the exhaust door. Normal room cooling to the W480VAC switchgear room was restored on June 14, 2003.

On October 8, 2003, operators secured normal W480VAC switchgear room cooling and established the OP 2315D (Revision 12) compensatory actions to support minor corrective maintenance on the normal room cooling heat exchanger. OP 2315D (Revision 12) had been revised to include steps to open the "exhaust" door across the room from the Appendix R fans supply location. When the compensatory actions had been instituted, design engineers took the opportunity to perform a walk-down of the compensatory measures installation. During this walk-down, the design engineers discovered that almost no flow was exhausting through the open "exhaust" door. The design basis calculation had assumed approximately 7000CFM of air flow would exhaust through this location. The design engineers informed operators that the compensatory measures appeared to be ineffective. Operations declared OP 2315D compensatory actions unavailable for the purpose of retaining operability of the W480VAC system, secured the maintenance on the normal room cooling heat exchanger, and restored normal cooling to the room. Subsequent testing of the ventilation lineup determined that a supply fan in the east cable vault (part of the discharge flow path) was required to be secured to achieve the design bases flow rates through the room. The inspectors found that OP 2315D and supporting design documents did not address the requirement to secure this fan.

Analysis. The performance deficiency associated with this issue is that the licensee failed to ensure procedures were developed that assured the design basis assumptions and requirements were satisfied. Specifically, a design assumption to open a door as a ventilation exhaust path was not included in the procedure and the design assumption for air flow through the exhaust path was determined to be incorrect. Traditional enforcement does not apply to this issue because there were no actual safety consequences, impacts on the NRC's ability to perform its regulatory function, or willful aspects to the violation.

The finding is more than minor because, if left uncorrected, one division of the 480 Volt vital AC system(480VACS) may not be available to respond to design basis events due to increasing room temperatures exceeding the switchgear design temperature limits and subsequent failure of the switchgear. Therefore, the equipment performance attribute of the mitigating systems cornerstone and the objective of ensuring the availability of systems that respond to initiating events to prevent undesirable circumstances was affected, since the 480VACS provides vital power to a number of safety-related systems designed to mitigate design basis events. The finding also affects the containment SSC and barrier performance attribute of the barrier integrity cornerstone and the objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events because, hydrogen recombiners and containment air recirculation fans are also powered from the 480VACS. Finally, the finding impacts the equipment performance attribute of the initiating events cornerstone objective of limiting the likelihood of those events that upset plant stability since temperature sensitive electro hydraulic control (EHC) equipment is located in this room. The failure of this equipment would result in a turbine trip.

Manual Chapter 0609, Appendix A - SDP was used to determine the risk associated with this finding. Phase 1 of the Appendix requires that a Phase 2 analysis be performed because three cornerstones were affected. The entry into the tables associated with the Phase 2 analysis assumes that the 480VACS is inoperable. However, this finding concerns the support room cooling system for the 480VACS, therefore, an evaluation of the impact on this system due to the degraded cooling system was performed. The design basis assumption for room temperature to ensure operability of the 480VACS is 104°F or less. A licensee analysis of the switchgear determined that the system would function for room temperatures up to 122°F during design basis events. Since the actual room temperature did not exceed 104°F, it is reasonable to conclude that although the compensatory measures for the 480VACS room cooling system were inoperable, the 480VACS remained operable. This evaluation also applies to the EHC equipment located in this room. Since both trains of the 480VACS and the EHC equipment remained operable, there is no entry condition for evaluating this in the Phase 2 Tables. Therefore, the safety significance of this issue is very low (Green).

This issue is attributable to the PI&R cross-cutting area because the licensee failed to identify the improper installation of compensatory measures and failed to evaluate the effectiveness of the installation once actions beyond the procedural guidance were necessary to maintain acceptable 480 VAC room temperature.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, requires, in part, that measures shall be established to assure the design basis is correctly translated into specifications, procedures and instructions. Contrary to this requirement, on June 4, 2003 the design calculation used to establish the required flow path was not correctly translated into the licensee's procedure which implemented compensatory actions for loss of normal W480VAC room cooling. Also, a design basis assumption of air flow through an exhaust door in the flow path established by this design calculation was determined to be incorrect following subsequent testing, however, because the issues have been entered in the licensees corrective actions program (CR-03-05497, 03-6337, 03-08825, 03-09043, 03-09052, 09-09218, 03-09580, 03-09745) this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000336/2003010-03).

2. (Closed) URI 05000336/2003006-01; Failure to Perform Adequate Post-Modification Test of Design Changes to the Charging System

Introduction. A Green non-cited violation (NCV) was identified for the failure to comply with 10 CFR 50, Appendix B, Criterion III, Design Control, for two design changes which adversely affected the charging system and for which post modification testing was not specified, or performed, to ensure the charging system could fulfill its design function under anticipated conditions. The NCV was documented and discussed in Inspection Report 05000336/2003004 dated November 10, 2003, but the NCV was inadvertently left out of the summary findings and the inspection report cover letter. In addition, the number assigned to the NCV was incorrect. As a result the enforcement portion of the finding discussion is reproduced in this inspection report and the below indicated number is assigned to the NCV.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, requires, in part, that measures shall be established to assure the design basis is correctly translated into specifications, procedures and instructions. Contrary to this requirement, on March 7, 2003 the licensee's failure to implement adequate design control or to perform post modification testing contributed to the failure of the charging system. However, because the issue was determined to be of very low safety significance and has been entered into the corrective action program (CR-03-03359), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000336/2003010-04)

3. Reactor Containment Sump Blockage (TI 2515/153)

a. Inspection Scope

The inspectors completed the inspection activities associated with TI 2515/153, which included an evaluation of Dominion's response to NRC Bulletin, 2003-003, Containment Sump. The inspectors performed a walkdown of the containment sump area to assess the as-built condition of the sump screen and other components, the architecture of the sump grating and supports and the condition of the surfaces and components internal to the containment sump screen. In support of the inspection, the Unit 2 Final Safety Analysis Report, CR 03-10695, Dominion compensatory measures related to Bulletin 2003-003, and the data in NRC NUREG/CR6808, Knowledge Bases for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling and Sump Performance, were reviewed. In addition, several design basis documents, engineering calculations, parametric and statistical analyses, drawings, supporting contractor information, the Dominion response to Bulletin 2003-003 and other generic NRC publications were reviewed.

b. Findings

No findings of significance were identified.

4. RPV Head and Nozzle Inspection Program (TI 2515/150)

a. Inspection Scope

The inspectors reviewed the implementation of the reactor pressure vessel head and nozzle inspection program. The head inspections were implemented consistent with NRC Order EA-03-009 issued February 11, 2003, based on the susceptibility ranking of the plant. The inspectors confirmed that the licensee appropriately requested relief from the requirements stipulated in the Order as required. The inspectors reviewed the results of the visual and non-visual NDE activities implemented or planned, including volumetric examination, surface examination and the bare metal visual examination. The specialist inspectors were unable to observe the bare metal visual inspection activities since these activities were not performed while the inspectors were onsite. The licensee has agreed to supply a videotape of this inspection for subsequent review. The licensee has reported that no indications of through wall leakage were observed during this inspection.

The licensee identified eleven indications during the inspection that required resolution. The licensee, inspectors, and NRR had discussions regarding the planned repair actions for these indications. The indications were either removed by grinding when possible or by replacement of a portion of the penetration nozzle. The responses by the licensee to the questions directed by TI 2515/150 are contained in Attachment B of this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Inservice Inspection Activities Report Exit Meeting Summary

The inspectors presented the inspection results to Mr. Skip Jordon, Director, Nuclear Engineering, and other members of the licensee management at the conclusion of the inspection on October 24, 2003. The licensee acknowledged the conclusions presented.

Occupational Radiation Safety Report Exit Meeting Summary

On November 7, 2003, the inspectors presented the Unit 2 and Unit 3 inspection results to licensee management and other staff who acknowledged the findings.

Integrated Report Exit Meeting Summary

On January 21, 2004, the resident inspectors presented the overall inspection results to Mr. J. Alan Price and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

S. Alligood, Technician, Radiation Protection
T. Armagno, Supervisor, Health Physics
V. Ballestrini, Effluent Chemist
P. Calandra, ALARA Coordinator
M. Coty, Supervisor, Unit 2 Licensed Operator Requalification Training
G. D'Auria, Supervisor, Chemistry Program
T. Degata, Technician, Radiation Protection
D. Delcore, Shift Supervisor, Health Physics, Unit 2
D. Dodson, Supervisor, Station Licensing
J. Doroski, Health Physicist
R. Fuller, ISI Level III
R. Griffin, Manager, Radiological Protection & Chemistry
S. James, Project Lead, Pressurizer Heater Replacement
A. Johnson, Supervisor, Radiation Protection Support (Technical)
A. Jordan, Director, Nuclear Engineering
T. Kulterman, Supervisor, Unit 3 Licensed Operator Requalification Training
E. Laine, (Acting) Manager, Radiological Protection & Chemistry
J. Langworthy, Technician, Radiation Protection
P. Parulis, Manager, Nuclear Oversight
A. Price, Site Vice President - Millstone
M. Roche, Technician, Chemistry Effluent
D. Regan, Supervisor, Radiation Protection Support (ALARA)
J. Roddy, Engineering Support
S. Sarver, Director, Nuclear Station Operations & Maintenance
S. Scace, Director, Nuclear Station Safety and Licensing
R. Schonenburg, Materials Engineer
R. Sosin, Health Physicist
M. Wynn, Health Physicist

NRC Personnel

S. T. Barr, Operations Engineer, DRS
B. A. Bickett, Reactor Inspector, DRS
S. R. Kennedy, Resident Inspector
A. X. Lohmeier, Reactor Inspector, DRS
C. M. Long, Reactor Engineer
K. M. Jenison, Senior Project Engineer, DRP
K. A. Mangan, Resident Inspector
N. T. McNamara, Emergency Preparedness Inspector, DRS

M. C. Modes, Senior Reactor Inspector
T. A. Moslak, Health Physicist, DRS
S. M. Schneider, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000336/2003010-01	URI	Conducted reactor vessel draindown without sufficient level indication (1R20)
05000336/2003010-02	URI	Loss of Shutdown cooling when AC supply sources were paralleled out of phase (4OA3.1)
05000336/2003010-05	URI	Reactor Coolant System Pressurizer Pressure Boundary Leakage (71111.08)

Opened and Closed

05000336/2003010-03	NCV	West 480 VAC Switchgear Room Compensatory Cooling. (4OA5.1)
05000336/2003010-04	NCV	Failure to perform adequate post modification test of design changes to the charging system (4OA5.2)

Closed

050000336/2003003-01	URI	West 480 Volt AC Switchgear Ventilation Lineup (4OA5.1)
05000336/2003003-01	LER	The Charging System Did Not Perform Its Design Function In Response To Falling Pressurizer Level (4OA3.3)
05000336/2003006-01	URI	Failure to perform adequate post modification test of design changes to the charging system (4OA5.2)

Discussed

05000336/2003004-00	LER	Reactor Coolant System (RCS) Pressure Boundary Leakage Event
05000336/2003005-00	LER	Loss Of Shutdown Cooling During Refueling Outage (4OA3.2)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection (71111.01)

SP 3670.1-001,Revision 022-09, Mode 1-4 Daily and Shiftly Control Room Rounds

Section 1R04: Equipment Alignment (71111.04)

DWG. 12179-EM-112C-34, Low Pressure Safety Injection/Containment Recirculation
DWG. 12179-EM-130B-36, Feed Water System
DWG. 12179-EM-158D, Station Blackout Diesel Air Start System
OP 2346A, Revision 024-06, Emergency Diesel Generators
OP3346D-003 Revision 4, SBO Diesel Lube Oil
OP 3346D-005 Revision 1, SBO Diesel Instrument Alignment
OP 3346D-006, Revision 5, SBO Diesel MCC Electric Alignment
SP 2613B-002, Revision 016-04, DG Valve Alignment Checklist, Facility 2
SP 2606D, Revision 009-06, CS System Alignment Check and Valve Tests, Facility 2

Section 1R05: Fire Protection (71111.05A)

WC-7, Revision 004-01, Fire Protection Programs
MP-24-FPP-PRG, Revision 002-02, Fire Protection Program
CR-03-12561, Annual Offsite Assistance Fire Drill
Millstone Unit 2 Fire Fighting Strategies
Millstone Unit 2 Fire Hazards Analysis
Unit 2 Offsite Assistance Drill Scenario
Fire Brigade Drill Assessment Data Sheet
Millstone Unit 3 Nuclear Power Fire Fighting Strategy
FPI 50-001, Revision 008, Fire Brigade Drill Assessment Data Sheet
CR-03-12654, Drill play feedback

Section 1R05: Fire Protection (71111.05Q)

Millstone Unit 2 Fire Hazards Analysis
Millstone Unit 2 Fire Hazard Analysis Boundary Drawing
MP-24-FPP-PRG, Revision 002-02, Fire Protection Program
WC-7, Revision 004-01, Fire Protection Program
Millstone Nuclear Power Station-Unit 3, Fire Protection Evaluation Reports, December 2001
MNPS-3-FSAR, Revision 16, Fire Protection Evaluation Report

Section 1R06: Flood Protection Measures (71111.06)

W2-517-01070RE, Revision 0, Millstone Unit 2 Internal Flooding Evaluation

Section 1R08: Inservice Inspection Activities (71111.08)

CRDMEDY-04058M2, Revision 0 (1/28/02)
M2-EV-03-0040, Revision 0 (9/02/2003): Millstone 2 Steam Generator Integrity Degradation Assessment
M2-EV-02-0010, Revision 0 (03/19/2002): Millstone 2 Steam Generator Condition Monitoring and Operational Assessment Refueling Outage 14
MP-PDI-GEN-1, Revision 000 (02/11/02): General Requirements for the Implementation of PDI Ultrasonic Procedures
MP-PDI-UT-2, Revision 000-01 (02/20/02): PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds
MP-PDI-UT-1, Revision 000 (02/11/02): PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds
Protocol PDI-UT-1, Revision 6 (03/01/2001) - Tables 1 & 2
Protocol PDI-UT-2, Revision 8 (01/21/2002) - Tables 1 & 2
Data Package 215-01-038 M2-03-02307
Data Package 215-01-051 HSI-CF-E-050
Data Package 215-01-049 HSI-CF-366
WCAP-15813-P, Structural Integrity Evaluation of RVV Upper Head Penetrations 8/2003
CRDM 13, RPVH Penetration Ultrasonic Examination indication Sheet 10/21/02
CRDM 22, RPVH Penetration Ultrasonic Examination indication Sheet 10/21/02
CRDM 26, RPVH Penetration Ultrasonic Examination indication Sheet 10/21/02
CRDM 57, RPVH Penetration Ultrasonic Examination indication Sheet 10/21/02
CRDM 13, RPVH Weld Profile
CRDM 22, RPVH Weld Profile
CRDM 26, RPVH Weld Profile
CREDM 57, RPVH Weld Profile
CR-03-10261, CEDM 57 Indications Indicative of PWSCC Requiring inspection
CR-03-10259, CEDM 13 Indications Indicative of PWSCC Requiring inspection
2R15 Rx Head Inspection Turnover Log - Last 24 hours 10/21/03
WO M2 03 08876, Visual exam of RF Bottom during 2R15 11/08/03
MS Unit 2 RVH 2R15 Inspection Summary - Nozzles 13, 17, 22, 26, 31, 37, 42, 46, 47, 57, 60, and 68

Section 1R11: Licensed Operator Requalification (71111.11Q)

EOP 3509.1, Control Room Cable Spreading Area, or Instrument Rack Room Fire
DCR # M3-00012 Revision 0, Millstone Unit 3 Repowering of MP1 Fire Pump, Stack and Auxiliaries
NTP 1, Revision. 002-03, Attachment, Simulator Crew or Individual Evaluation Form

Section 1R12: Maintenance Rule Implementation (71111.12Q)

Millstone Unit 2 Maintenance Rule Scoping Report
Unit 2 FSAR Chapter 10, Steam and Conversion System
Unit 2 Technical Requirements Manual Section 3.4.6.3, Containment Isolation Valves
Millstone Unit 3 Maintenance Rule Scoping Tables
Unit 2 Emergency Diesel Generator System Engineer System Health Report

CR-03-11381, 2-MS-64B Failed to Stroke Open
CR-03-10518, Unexpected Instrument Response and Loss of EDG Control During LNP Test
CR-03-11745, Pressurizer PORV, 2-RC-402, Has Developed a Leak in the Pilot Valve Portion Of The Pipe
CR-03-09340, Gaseous Release Occurred due to 2-Ch-994 Installed Backwards
CR-02-11107, Numerous 3RMS*RE41 Detector Failure Alarms
CR-03-10523, Unplanned LCO Entry - Rad Monitor Computer Off Line
CR-03-09413, RMS Host "A" Locked Up
CR-03-08559, U3 Rad Monitor Display System Failure
CR-02-13247, New CR To Document Additional Corrective Actions For MP3 RMS System Not Meeting Maintenance Rule Performance Criteria
CR-02-00628, CAT 1E Radiation Monitor Loop 6 Went Off Line
CR-02-00345, Radiation Monitoring System Host "B" Is Offline
CR-02-11954, Loop 1 Rad Monitors Became Unreachable, Unplanned LCO's
CR-03-05702, RMS Aydin Console Selected To Display Data From Backup Computer
CR-03-05709, RMS Computer "B" Not Responding To Keyboard Input From Control Room RMS Aydin Console
CR-03-06005, RMS 34 Detector Failure
CR-02-07143, RMS 34 Failed As Found Calibration
CR-03-11664, 3HVR*FN13A Inoperable - Unplanned Technical Specification LCO Entry
CR-03-12288, MP3 RMS Exceeds Maintenance Rule Performance Criteria
MP-26-EPI-FAP06, Revision 000-04, Classification and PARS

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

Shutdown Safety Assessment Checklists dated 10/16/2003, 10/18/2003, 10/26/2003, 11/04/2003, 11/17/2003
OP 2264, Revision 009-03, Conduct of Outages
NUMARC 93-01, Industry Guidelines for Monitoring Effectiveness of Maintenance at Nuclear Power Plants
NUMARC 91-06, Industry Guidelines for Shutdown Operations

Section 1R14: Operator Performance During Non-Routine Evolutions and Events (71111.14)

SP 2613G, Revision 010-02, Integrated Test of Facility 1 Components (IPTE)
OP 2346A, Revision 024-06, Emergency Diesel Generators
CR-03-10518, Unexpected Instrument Response and Loss of EDG Control during LNP Test
CR-03-10593, ERT Recommendation is to Replace the "Alarm Reset" Push Button on the "B" EDG
CR-03-12035, Plant was Tripped due to High Turbine Vibration
OP 2203, Revision 016-06 Plant Startup
OP 2323A, Revision 021-07, Turbine
OP 2329, Revision 20, Condenser Air Removal System

Section 1R15: Operability Evaluations (71111.15)

CR-03-09656, 3SWP*MOV24B Coupling May be Showing Signs of Degradation
CR-03-09578, 3SWP*P1C Coupling and Bolting are Corroded
CR-03-10884, RCP Underspeed Logic Failed Semi-Automatic Test in SSPS Train B OP Test
CR-03-09341, A SW Strainer Blowdown MOV is not Functioning Properly
CR-03-00916, Valve SWP*MOV 24A Leaks by Seat
CR-03-10524, Service Water Leak at Flanged Connection on "A" Service Water Supply to Emergency Diesel Generators
CR-03-11745, Pressurizer PORV, 2-RC-402, Has Developed a Leak in the Pilot Valve Portion of the Pipe
CR-03-12103, Control Room Reported Water Dripping from Bottom of Insulation on 2-MS-190A
DWG# 25212-39001, Primary Coolant System Trip Signals
SP 3446B12, Revision 11-03, SSPS OP Test Train B
OD MP2-058-03, Revision 0, Leak at Service Water Flanged Connection to EDGs
RP-5, Revision 003, Operability Determinations
Drw # 25203-30001, Revision 20, Main Single Line Diagram
Calc 96020-1367-M2 Revision 1, Insulation Debris Transport and Head Loss
NUREG 6762, GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance
NUREG 6806, Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance
Letter Dated Dec 10, 1999 From Northeast Utilities to Continuum Dynamics
Reg Guide 1.82, Revision 2, Water Sources for Long-Term Recirculation Cooling Following a Loss of Coolant
Serial # - 03-368, Millstone Unit 2 Sixty Day Response to NRC Bulletin 2003-01
Docket No 50-336-B16918, Millstone Unit 2 Ninety Day Response to Generic Letter 97-04
Docket No 50-336-B17019, Millstone Unit 2 Supplemental Ninety Day Response to Generic Letter 97-04
Docket No 50-336-B17523, Millstone Unit 2 Ninety Day Response to Generic Letter 98-04
Calc 98-122, Revision 2, Containment Spray Flow Analysis
Calc 97-122, Revision 2, ECCS Flow Analysis for Millstone Unit 2
DWG # 25203-11173, Containment Floor Plan Elevation 22'6"
Calc # 07077-US(B)-003, MP2 Minimum Sump Water Level Following a Loss of Coolant Accident (LOCA)
Calc # 07077-US(B)-002, Maximum Containment Water Level During LOCA

Section 1R16: Operator Workarounds (71111.16)

COP 200.9, Revision 002, "Operational Performance Status"
Unit 2 Operations Key Performance Indicator Report
OP 3341D, Revision 014-01, Attachment 76, Points Monitored in Zone Panel 11

Section 1R17: Permanent Plant Modifications (71111.17A)

DCR M2-02007, Installation of Permanent Jumpers in MP2 125VDC Switchgear D01, D02, and

D03
DCR No M3-01008, Installation of Incipient Fire Detection System in MP3 Cable Spreading Room
DM3-00-0018-02, Cable Spreading Area - Installation of Incipient Fire Detection System (Electrical/I&C Portions)

Section 1R19: Post Maintenance Testing (71111.19)

AWO M2-03-12472, "A" Service Water Cooling Pump Outlet Check Valve Maintenance
AWO M2-01-06626, #2 Steam Generator Main Feed Supply Air Assist Check Valve Assembly
AWO M2-00-14079, “C” High Pressure Safety Injection Pump 5 Year Coupling Inspection
AWO M2-03-08089, “B” Charging Pump Pulsation Dampener and 3-Way Valve Installation
AWO M3-02-17606, Auxiliary Feedwater Pump 3FWA*P2 Turbine Speed Calibration
AWO M3-03-08402, Auxiliary Feedwater Pump (Turbine Driven) Control Valve
AWO M3-03-11819, Removal/Overhaul/Install Fuel Oil Transfer Pump
AWO M3-02-07623, PM, 4 year - Strainer Overhaul
SP 3622.3-001, Revision 013-01, TDAFW Pump Operational Readiness Test
SP 2610C-5, AFW Remote Position Indication IST
SP 21236, Revision 001-01, Disassembly and Stroke Testing of Check Valves in the In-Service Test Program
SP 2604B, Revision 013-08, HPSI Pump Operability and Inservice Testing, Facility 2
CEN 110, Revision 000-01, Post Repair/Replacement Leakage Test
MP-20-WP-GDL40, Revision 002, Pre-and Post Maintenance Testing
Work Order M3 03 04437, Terry Turbine Trip Throttle Valve
Work Order M3 02 17606, Auxiliary Feedwater Pump 3FWA*P2 Turbine Speed
SP 3622.3-001, TDAFW Operation Readiness Test

Section 1R20: Refueling and Outage Activities (71111.20)

OP 2206, Revision 010-06, Reactor Shutdown
OP 2205, Revision 013, Plant Shutdown
OP 2207, Revision 025-00, Plant Cooldown
EN 21008, Revision 012-04, Refueling Worklist Administrative Control
OP 2209A, Revision 024-05, Refueling Operations
OP 2307, Revision 012-08, Low Pressure Safety Injection System
OP 2301E, Revision 021-09, Draining the RCS (IPTE)
OP 2301E, Revision 022, Draining the RCS (IPTE)
OP 2218, Revision 007-09, Reduced Inventory Operations
SPROC OPS0302-03, Revision 000-01, Vacuum Fill of the Reactor Coolant System
OP 2301D, Rev 026-02, Filling and Venting the RCS
OP 2202, Revision 020-02, Reactor Plant Startup IPTE
OP 2201, Revision 029-02, Plant Heatup
EN 21004, Revision 002-02, Reactivity Computer Setup
EN 21004K, Cycle 16, Low Power Physics Test
OP 2307, Revision 012-08, Low Pressure Safety Injection System
CR-03-11600, Reactor Cavity Draindown Without Wide Range Level Instruments in Operation.

Control Room Logs
2R15 Shutdown Risk Review Team Pre-Outage Report

Section 1R22: Surveillance Testing (71111.22)

SP 2610C, Revision 012-07, AFW System Lineup, Valve Operability, and Operational Readiness Tests
SP 2730B, Revision 12, Main Steam Safety Valve Testing (IPTE)
SP 2606, Containment Spray System Alignment Check and Valve Tests, Facility 1
SP 3666.1, Revision 005-03, Technical Support Center Ventilation Test
SP 3443A21-001,Revision 19-01, Protection Set Cabinet I Operational Test Data Sheet
SP 3606.9, Revision 009-01, Recirculation Spray Valve Operability - Train B
SP 2613G, Revision 010-02, Integrated Test of Facility 1 Components (IPTE)
OP 3315E, Revision 3, Technical Support Center Ventilation
OP 2396, Revision 007-06, Security System Emergency Diesel Generator
Millstone Unit 2 Final Safety Analysis Report, Sections 4.3 and 10.3
Millstone Unit 2 Final Safety Analysis Report, Section 10.4; Dwg. 10.04-02, Sheet 03
Millstone Unit 2 Technical Specification 3.7.1.1/4.7.1.1 and 3.4.7.1
Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 275 DNC, Inc., Millstone Unit 2
Commitment Record No. RCR-00203
OPS Form 2610C-014, 2-FW-44 Valve Stroke and Timing IST
OPS Form 3666.2-1, Revision 4, Technical Support Center Emergency Power Test
OPS Form 2654E-1, Revision 3, Change 2, Monthly Run of Security System Emergency Diesel Generator
Millstone Unit 2 Technical Specifications
Millstone Unit 3 Technical Specifications
DCM Form 5-1A, Revision 7, Change 1, Security Diesel Generator Loading Calculation
MP 2722B, Revision 003-02, EOF Diesel Generator and Protective Services System Emergency Diesel Generator Load Run
EN 21004K, Revision 000, Cycle 16, Low Power Physics Test
EN 21004F, Revision 003-04, Control Rod Worth Measurements

Section 1R23: Temporary Plant Modifications (71111.23)

TM M2-03-005, Use of Inflatable Pipe Plugs for Maintaining Containment Closure
TM M3-01-035, Disable CO2 Fire Suppression System in Cable Spreading Area

Section 1EP6: Drill Evaluation (71114.06)

Millstone Emergency Plan, Revision 29, Change 5
MP-26-EPI-FAP06, Revision 000-04, “Classification and PARS”
MP-26-EPI-FAP07, Revision 003-01, “Notifications and Communications”

Section 2OS2: ALARA Planning and Controls (71121.02)

RPM 1.1.1, Revision 7, Health Physics Organization and Responsibilities of Key Radiological Personnel
RPM 1.3.8, Revision 8, Criteria for Dosimetry Issue
RPM 1.3.12, Revision 7, Internal Monitoring Program
RPM 1.3.13, Revision 6, Bioassay Sampling and Analysis
RPM 1.3.14, Revision 6, Personnel Dose Calculations and Assessments
RPM 1.3.15, Revision 1, DAC-hour Tracking
RPM 1.4.1, Revision 7, ALARA Reviews and Reports
RPM 1.4.2, Revision 1, ALARA Engineering Controls
RPM 1.4.4, Revision 2, Temporary Shielding
RPM 1.5.1, Revision 8, Routine Survey Frequency
RPM 1.5.2, Revision 4, High Radiation Area Key Control
RPM 1.5.5, Revision 4, Guidelines for Performance of Radiological Surveys
RPM 1.5.6, Revision 3, Survey Documentation and Disposition
RPM 2.1.1, Revision 5, Issuance and Control of RWPs
RPM 2.1.2, Revision 2, ALARA Interface with the RWP Process
RPM 2.2.6, Revision 9, Continuous Air Monitors
RPM 2.4.1, Revision 3, Posting of Radiological Control Areas
RPM 2.10.2, Revision 10, Air Sample Counting and Analysis
RPM 2.11.1, Revision 8, Survey and Decontamination of Personnel and Clothing
RPM 5.2.2, Revision 10, Basic Radiation Worker Responsibilities
RPM 5.2.3, Revision 3, ALARA Program and Policy

Effluent Performance Indicator Verification (71151)
SP 2857, Revision 4, Liquid Effluent Doses From Unit 2
SP 2858, Revision 4, Offsite Dose Noble Gases From Unit 2
SP 2859, Revision 4, Offsite Dose - Iodine and Particulate Releases
SP 2861, Revision 5, Liquid and Gaseous Effluent Dose Projection
SP 3865, Revision 5, Liquid Effluent Dose
SP 3867, Revision 7, Offsite Dose - Noble Gas from Unit 3
SP 3868, Revision 5, Offsite Dose - Iodine and Particulate Releases from Unit 3
SP 3878, Revision 6, Unit 3 Monthly Liquid and Gaseous Effluent Dose Projection
SP 822/2822/3822, Revision 9, Calculation of Cumulative Annual Station Offsite Doses

ALARA Reviews
AR 2-03-01, Reactor Vessel Disassembly/Reassembly
AR 2-03-20, Reactor Head Penetration/Bare Metal Visual Inspection/Head Stand Work & Head Replacement Measurements
AR 2-03-33, Chemical & Volume Control System (CVCS) Improvements
AR 2-03-29, Pressurizer Heater Inspection/Removal/Replacement

Nuclear Oversight Specialist Reports
Dated October 11, 12, 13, 15, 18, 21, & 28, 2003

Nuclear Oversight Audit
Audit MP-03-A14, Radiation Protection & Chemistry

Condition Reports
03-11159, 03-10933, 03-10407, 03-10355, 03-10024, 03-09873,03-09375, 03-09361, 03-09340, 03-08603, 03-07927, 03-07754, 03-07600, 03-07504, 03-07434, 03-07272

Radiation Safety Committee Meeting Minutes:
Meeting conducted on 10/22/03 and 10/30/03

Effluent Release Reports:
Noble Gas Releases from Unit 3, December 2002 through September 2003
Iodine and Particulate Releases from Unit 3, December 2002 through September 2003
Liquid Effluent Doses from Unit 3, January 2003 through September 2003
Noble Gas Releases from Unit 2, September 2002 through September 2003
Iodine and Particulate Releases from Unit 2, September 2002 through September 2003
Liquid Effluent Doses from Unit 2, September 2002 through September 2003

Work Observation Reports:
03-6881, 03-6978, 03-7105, 03-7133, 03-7134, 03-7166, 03-7173, 03-7179, 03-7200, 03-7290, 03-7295, 03-7419, 03-7471, 03-7566, 03-7580

Section 4OA1: Performance Indicator Verification (71151)

SP 3670.1-001, Revision 022-09, Mode 1-4 Daily and Shiftly Control Room Rounds
CP 3807F Revision 004-05, Operation of The Reactor Plant Sample Sink
Operator Logs
Auxiliary Feedwater Safety System Unavailability PI Data Submittals
Auxiliary Feedwater System Health Report dated 10/08/2003
LER 2003-003, The Charging System Did Not Perform Its Design Function in Response to Falling Pressurizer Level
LER 2002-004-00, Reactor Shutdown Due to Entry into Technical Specification 3.0.3
Safety System Functional Failures PI Data Submittals

Section 4OA3: Event Followup (71153)

Unit 2 Emergency Action Levels
LERs 2003-003-00 and 2003-003-01, The Charging System Did Not Perform Its Design Function In Response To Falling Pressurizer Level
LER 2003-005-00, Loss of Shutdown Cooling During Refueling Outage
Loss of Shutdown Cooling Event Notification #402
CR-03-0988, NRC Reportable Loss of Shutdown Cooling. Loss of VA-10 caused 2-SI-657 to Close
2R15 Shutdown Risk Review Team Pre-Outage Report dated October 2, 2003
Event Review Team Report on Millstone Unit 2 Loss of Shutdown Cooling Event, CR-03-09838
Event Notification 40245
Millstone Unit 2 Loss of Shutdown Cooling Unusual Event
MP2 High Risk Evolution Plan and Evaluation for 2R15
AOP 2572, Revision 009, Loss of Shutdown Cooling

AOP 2504A, Revision 003-03, Loss of Non-Vital Instrument Panel VR-11
AOP 2504C, Revision 003-04, Loss of 120 VAC Vital Instrument Panel VA-10
OP-2345B, Revision 016-05, 120 Volt Vital Instrument AC System
Unit 2 Operator Logs
LER 05000336/2003004-00, Reactor Coolant System (RCS) Pressure Boundary Leakage Event

Section 4OA5: Other

CR-03-05497, Revise Compensatory Actions for Loss of Cooling To West 480 Volt Switchgear Room Cooling Systems to Provide Better Cooling
CR-03-06337, Supplemental Design Engineering Items Requiring Resolution - Pertaining to MP2 Comp Actions for Loss of Cooling to West 480 Volt Switchgear Room
CR-03-08825, OP 2315D Compensatory Measures for The West 480 Volt Switchgear Room are Not Fully Qualified
CR-03-09043, West 480 VAC Switchgear Room Compensatory Measures for Turbine Building HELB
CR-03-09052, CR-03-06337 Inappropriate Classified As Level N
CR-03-09218, OD No MP2-057-03, Rev 0, “West 480 VAC Switchgear Compensatory Ventilation Measures and HELB Mitigation Safety Function”
CR-03-09432, TR/AWO Required to Perform Flow Testing for the West 480 volt Room Compensatory Measures
CR-03-09580, Compensatory Measures Established for MP2 West 480 VAC per OP2315D DO Not Provide Expected Flowrate
CR-03-09745, Operability of the East and West DC Switchgear Room Ventilation Compensatory Measures May be Called into Question
CR-03-12177, DC and 480 Switchgear Room Compensatory Ventilation Fans, Electric Power Supply Seismic Qualification Status
MP2-057-03 Operability Determination for CR-03-06337
MP2-060-03 Operability Determination for CR-03-09580
MP2-062-03 Operability Determination for CR-03-12177
M2-EV-99-0093, Revision 4 - MP2 Technical Evaluation Evaluating Compensatory Measures to use During Loss of Cooling/Ventilation Systems Supporting Vital Switchgear Rooms
Night Order 9-15-03-01, Compensatory Measures for West 480 Volt Switchgear
SFP 14-002, Appendix “R” Ventilation Fan Speed Check Data Sheet
Millstone Plant Unit 2 Specific Plant Area/Zone Separation Analysis Safe shutdown Equipment Success Path Report
NE-02-F-139 - MP2-West 480 Volt Switchgear Room Loss of Ventilation Evaluation to Support Reportability Evaluation
NUCENG-03-98 - MP2 - West 480 VAC Switchgear Room Compensatory Ventilation System, Scoping Calculation for Air Flow Rate
TRMCR 99-2-18, Technical Requirements Manual Change for Switchgear Ventilation
S2-EV-99-0090 - Safety Evaluation for TRM 2-00-17 Switchgear Ventilation
Letter from Super Vacuum Manufacturing Company, INC to Stone and Webster (Millstone Nuclear) - dated October 16, 1998
M2-EV-03-0046, Technical Evaluation for Assessment of Vital Switchgear Compensatory

Measures Operability Given October 2003 Ventilation System Ventilation Flow Test Results and Conditions Identified in CR-03-09574, CR03-09580, CR-03-09043, and CR-03-06337
Work Order M2 99 02537, Vital Electric Switchgear Cooling System - Miscellaneous Items
OP 2315D, Revision 11, Vital Electric Switchgear Room Cooling Systems
OP 2315D, Revision 12, Vital Electric Switchgear Room Cooling Systems
M2-EV-99-0031 - Appendix R Fire Safe Shutdown Component Temperature Ratings Vital Switchgear Rooms
92-FFP-934ES, Revision 3, MP2 West 480 Volt AC Load Center Heat Gains
97-HVAC-02036M2, Revision 1, West 480 Volt Switchgear Room Loss of Ventilation, F-51

LIST OF ACRONYMS

AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
AWO	automated work order
CR	condition report
CSR	cable spreading room
CVCS	chemical and volume control system
EDG	emergency diesel generator
EOOS	equipment out of service
ERT	event review team
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
ISI	inservice inspection
IST	in-service testing
LCO	limiting condition of operation
LOCA	loss of coolant accident
MNSA	mechanical nozzle sealing assembly
MOV	motor-operated valve
MR	maintenance rule
OP	operating procedure
P&ID	piping and instrumentation diagram
PCR	personnel contamination report
PI	performance indicator
PWSCC	primary water stress corrosion cracking
RCA	radiologically controlled area
RCS	reactor coolant system
RSPS	risk significant planning standards
RWST	refueling water storage tank
SDP	significance determination process
SG	steam generator
SORC	site operations review committee
SP	surveillance procedure
SSC	structures, systems and components
SW	service water

TRM	technical requirements manual
TS	technical specification
UT	ultrasonic test
WIP	work-in-progress

TI 2515/150 Revision 2
Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles Reporting Requirements

a.1. Was the examination performed by qualified and knowledgeable personnel?

The ultrasonic test (UT) examinations were performed by qualified and knowledgeable personnel using calibrated equipment. The procedures were demonstrated to be capable of identifying control rod drive mechanism (CRDM) degradation. In addition, Level II and Level III examiners received training in this type of inspection that included a review of industry experiences, lessons learned, inspection results and procedure requirements.

a.2. Was the examination performed in accordance with approved procedures?

The examinations were performed in accordance with approved procedures.

a.3. Was the examination able to identify, disposition, and resolve deficiencies?

The examination was adequate to identify, disposition and resolve deficiencies. A detailed systematic visual examination by quadrants was made of each penetration. The UT documentation included computer-based data storage for further review during future examinations.

a.4. Was the examination capable of identifying the PWSCC phenomenon described in Order EA-03-009?

The examination performed was capable of identifying the Primary Water Stress Corrosion Cracking (PWSCC) phenomenon described in the Order. The examination was adequate to identify, disposition and resolve deficiencies. The examinations were complimentary to each other and designed to provide a full outside head surface and CRDM weld volumetric examination.

b. What was the condition of the reactor vessel head?

The licensee reported that the video taped inspection showed no boron deposits that were considered a result of leakage through the CRDM to head welds, or the CRDMs.

c. Could small boron deposits, as described in the Bulletin 2001-01, be identified and characterized?

Small boron deposits, as described in Bulletin 2001-01, could have been identified and characterized by the visual examination and chemical analysis of any residue.

d. What material deficiencies were identified that required repair?

The “J” weld connecting the CRDM to the shell had indications at several penetration locations.

e. What, if any, significant items could impede effective examination?

The licensee reported that they were unable to fully remove the head insulation around some CRDM penetrations. The licensee reported that they used a remote visual examination device to inspection these nozzles.

f. The basis for the temperature used in the susceptibility ranking calculation.

The basis for determination of the temperature used in the susceptibility ranking is described in calculation CRDMEDY - 04058M2, The calculation determined the reactor head temperature based on the reference plant operating temperature during full power operations.

g. During non-visual examinations, was the disposition of indications consistent with the guidance provided in Appendix D of this TI? If not, was more restrictive flaw evaluation guidance used?

Disposition of non-visual indications was available in the procedures for the UT and PT examinations and conformed with the Guidance provided in Appendix D of the TI.

h. Did procedures exist to identify potential boric acid leaks from pressure retaining components above the RPV head?

Procedures exist to identify potential boric acid leaks through careful bare metal examination and chemical analysis.

i. Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure retaining components above the RPV head?

There were no boric acid leaks reported by the licensee.