April 24, 2002

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

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

Dear Mr. Price:

On March 30, 2002, the NRC completed inspections at your Millstone Units 2 & 3 reactor facilities. The enclosed reports document the inspection findings which were discussed on April 10, 2002 with Messrs. D. Hicks and S. Scace and other members of your staff.

These inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

No findings of significance were identified.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). On February 25, 2002, the NRC issued an Order to all nuclear power plant licensees, requiring them to take certain additional interim compensatory measures to address the generalized high-level threat environment. With the issuance of the Order, we will evaluate Dominion Nuclear Connecticut, Inc.'s compliance with these interim requirements.

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

Sincerely,

/RA/

Anthony McMurtray, Acting Chief
Projects Branch 6
Division of Reactor Projects

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

Enclosures:
(1) NRC Inspection Report 50-336/02-02
Attachment 1: Supplemental Information
Attachment 2: TI 2515/145 Reporting Requirements

(2) NRC Inspection Report 50-423/02-02
Attachment 1: Supplemental Information

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ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	50-336
License No.:	DPR-65
Report No.:	50-336/02-02
Licensee:	Dominion Nuclear Connecticut, Inc.
Facility:	Millstone Power Station, Unit 2
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	February 10, 2002 - March 30, 2002
Inspectors:	S. M. Schneider, Senior Resident Inspector, Unit 2 P. C. Cataldo, Resident Inspector, Unit 2 B. E. Sienel, Resident Inspector, Unit 3 J. M. Brand, Resident Inspector, Seabrook G. T. Dental, Senior Resident Inspector, Seabrook T. F. Burns, Reactor Inspector, Division of Reactor Safety (DRS) K. M. Jenison, Senior Project Engineer, Division of Reactor Projects (DRP) M. C. Modes, Senior Reactor Inspector, DRS T. A. Moslak, Health Physicist, DRS
Approved by:	Anthony McMurtray, Acting Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000336-02-02; on 02/10-03/30/02; Dominion Nuclear Connecticut, Inc., Millstone Power Station; Unit 2; Resident Inspection.

The inspection was conducted by resident and regional inspectors. No findings of significance were identified. The significance of most findings is indicated by the color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “no color” or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/reactors/operating/oversight.html

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Violations

No licensee violations were identified.

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Report Details

Summary of Unit 2 Status

At the beginning of the inspection period on February 10, 2002, the plant was in coastdown operation at approximately 80 percent power in preparation for Refueling Outage 14 (2R14). On February 15, with the reactor at 78% power, operators performed a manual reactor shut down, placing the reactor in Mode 5 (Cold Shutdown) on February 17 and Mode 6 (Refueling) on February 21.

Following the completion of 2R14 work, including steam generator tube inspections and reactor pressure vessel head penetration examinations and repairs; operators placed the plant in Mode 3 (Hot Standby) on March 29, where it remained at the end of the inspection period.

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

1R04 Equipment Alignment

.1 Partial Equipment Alignments

a. Inspection Scope

The inspectors conducted partial system walkdowns of the "B" train of service water and the "B" train of reactor building closed cooling water (RBCCW) following the systems' return to service after maintenance. The inspectors verified that the "B" service water system was correctly aligned for operation in accordance with surveillance procedure (SP) 2612D, "Service Water System Lineup and Valve Tests, Facility 2," and system piping and instrumentation diagram (P&ID) 25203-20009, Sheet 2. The correct alignment of the "B" RBCCW system was verified by comparing the actual system alignment, with consideration for plant status at the time, with SP 2611D, "RBCCW System Alignment and Valve Tests, Facility 2," and system P&ID 25203-26022.

b. Findings

No findings of significance were identified.

.2 Complete Equipment Alignment

a. Inspection Scope

Prior to the commencement of refueling activities, the inspectors performed a full system walkdown of the accessible portions of the spent fuel pool (SFP) cooling and purification system. The inspector verified the SFP system was correctly aligned in accordance with OP 2305, "SFP Cooling and Purification System Line-up Verification," and system P&ID 25203-26023. The inspectors also performed a control room walkdown and reviewed the Shutdown Safety Assessment Checklist, Attachment 2, dated March 24, 2002, to verify that all support systems credited in the safety assessment were functional.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed walkdowns of the following plant areas to observe conditions related to fire protection:

- RBCCW Pump and Heat Exchanger Area, Auxiliary Building, -25 Foot Elevation (Fire Area A-1, Zone B)
- Auxiliary Building General Area (housing spent fuel pool cooling pumps and heat exchangers), -5 Foot Elevation (Fire Area A-1, Zone G)
- Low Pressure Safety Injection (LPSI) Pump Room, Auxiliary Building, -45 Foot Elevation (Fire Area A-3)
- High Pressure Safety Injection (HPSI) Pump Room, Auxiliary Building, -45 Foot Elevation (Fire Area A-4)
- Charging Pump Room, Auxiliary Building, -25 Foot Elevation (Fire Area A-6, Zone A)
- Containment Spray and HPSI/LPSI Pump Room, Auxiliary Building, -45 Foot Elevation (Fire Area A-8, Zone A)

The inspectors verified that the fire detection and suppression equipment located in these zones was as specified in the Millstone Unit 2 Fire Hazards Analysis. During the walkdowns, the inspector examined equipment (e.g., emergency lighting units, fire extinguishers) for evidence of degraded or inoperable conditions and assessed the transient combustible materials stored in the fire area.

The inspectors confirmed that the compensatory measures in place for degraded equipment were in compliance with the Millstone 2 Technical Requirements Manual. The inspectors also reviewed condition reports CR-02-01852, CR-02-01897, and CR-02-02602, and examined the cumulative impact and risk significance of several inoperable fire dampers to evaluate the overall impact to the fire protection system.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

.1 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles

The licensee’s examination activities, performed in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," were inspected to the requirements established in Temporary Instruction (TI) 2515/145. In addition, the inspectors utilized guidance regarding examination requirements and test methods provided in a February 21, 2002, NRC letter to Mr. J. Price of Dominion Nuclear Connecticut, Inc.. The details of the inspection scope and results are in Section 4OA5, as specified by the TI.

.2 Inservice Inspection

a. Inspection Scope

The inspectors reviewed the following, in order to determine compliance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Appendix VIII 1995 Edition, 1996 Addenda for procedure qualification and Section XI, 1989 Edition for program requirements:

- Ultrasonic Test (UT) Data Package and a Liquid Penetrant Test Data Package for the Safety Injection Nozzle Safe-End of Steam Generator (SG) #1

- UT Data Package and a Magnetic Particle Test Data Package for the Replacement SG Nozzle

- UT Data Package SG-1-FW-IR-1, 214-01-026 for the Feed Water Nozzle Inside Radius Examination.

In order to verify the inclusion of the Safety Injection System, in response to NRC taking exception, in 10 CFR 50.55a(a)(1)(xi), to portions of some Safety Injection System components being excluded by the ASME Code, the inspectors reviewed:

- UT Data Package SI-CF-A-062, 214-01-034, an augmented inspection of a 10" valve-to-pipe weld in the Safety Injection System

- UT Data Package SI-CF-A-64, 214-01-035, an augmented inspection of a 10" valve-to-pipe weld in the Safety Injection System

- UT Data Package SI-CF-X-39, 214-01-038, an augmented inspection of a 10" elbow-to-reducer in the Safety Injection System.

The inspectors also reviewed the following supporting Millstone procedures for compliance with ASME 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 the technician personnel qualifications in accordance with 10 CFR 50.55a(a)(xiv). The inspectors interviewed vendor UT technicians who performed the ultrasonic examination of the nozzle inner radius.

The inspectors reviewed the non-code repair of Service Water Line 2"-HUD-130, which is the service water supply to a vital switchgear cooling coil, required to perform a cooling function during and following a seismic event. The inspectors reviewed Operability Determination (OD) MP2-029-00, generated as a consequence of condition report M2-00-2265, for the 1/4" diameter pin-hole leak in this service water line that was temporarily repaired using a rubber patch and pipe clamp. The inspectors also reviewed calculation 00CP-02995-M2, generated to assure continued structural integrity of this pipe. The inspectors reviewed the licensee's submittals, Nos. B18219 and B18262, which requested relief from ASME requirements for this service water pipe, in conformance with NRC Generic Letter 90-05. The inspectors reviewed Work Order Number M2-00-15166 which implemented the replacement of the leaking pipe on May 1, 2001, which included an expansion of the repair, after disassembly, because other piping was found to be corroded.

The inspectors reviewed Millstone Unit 2 Steam Generator Eddy Current Data Analysis Reference Manual U2-24-SIP-REF01, Rev. 001. The inspectors determined through this review that steam generator tube inspections were performed within the guidance given in the Electric Power Research Institute (EPRI) Pressurized Water Reactor SG Examination Guidelines, Rev. 5 (TR-107569-V1R5). The inspectors also reviewed document 51-5008209-01, which assures that eddy current techniques used at Millstone Unit 2, during 2R14, were applicable to anticipated or existent degradation and are in conformance with the guidance of Appendix H of the EPRI Pressurized Water Reactor SG Examination Guidelines, Rev. 5. In the case of techniques, supported by EPRI Examination Technique Specification Sheets (ETSS), that did not have specific qualification data directly applicable to the anticipated degradation or generator configuration at Millstone, the inspectors reviewed the licensee's qualification of the technique by extension. The inspectors discussed the principle of extension with the Millstone Unit 2 Eddy Current Level III to determine what electromagnetic theory, geometrical similarities, and/or similar flaw characteristics were used to apply the technique in the absence of a physical demonstration of its ability to detect or size the degradation. The inspectors specifically reviewed the extensions of the following:

- ETSS 96001.1 for thinning at Tube Support Plates (TSPs) and Top of Tube Sheet (TTS)

- ETSS 96004.3 for Wear at TSPs, Anti-Vibration Bar, vertical and diagonal straps, Fan Bar, Lattice Bar, and Tube-to-Tube wear

- ETSS 96008.1 for Axial Outside Diameter Stress Corrosion Cracking (ODSCC) at non-dented egg crates, lattice bars, fan bars, and/or sludge pile region

- ETSS 20409.1 Plus-point for Axial ODSCC at TSPs and freespan areas and circumferential ODSCC at expansion transitions.

The inspectors reviewed the potential risks of wear at the support bar-to-tube contact, wear at the top of tube sheet caused by loose parts, pitting in the area of the secondary side sludge/deposits, plug installation deficiencies at the plug tube interface, and erosion-corrosion in the moisture separators and other internal components. The potential risks were identified in M2-EV-00-0028, Rev. 0, "Millstone Unit 2 SG Condition Monitoring and Operational Assessment Refueling Outage 13." The inspectors compared the risks with the integrity degradation assessment, MS-EV-01-017, Rev. 0, performed prior to the current outage, on September 11, 2001, to determine if the risks were adequately identified. The inspectors compared the SG examination techniques, scope of inspection, and scope of expansion plans against previous outage data to ascertain if the techniques were appropriate and the scope of the inspection was sufficient to capture the anticipated degradation. The inspectors also determined, from the review, if the examinations were done in conformance with U2-24-SIP-REF01, Rev. 001, "Unit 2 Steam Generator Eddy Current Data Analysis Reference Manual." The review of U2-24-SIP-REF01 was implemented to ascertain whether the most appropriate data analysis practices were used, and data was analyzed in a manner consistent with the EPRI SG Guidelines.

The inspectors reviewed the status of loose parts in the SGs to determine if the licensee's response to a loose parts monitor alarm received before the current outage commenced was appropriate. The inspectors reviewed FTI-00-1353, which evaluated the potential damage that could be caused by a wedged hex piece, three pieces of weld wire, and one piece of gasket material left in the generators after a comprehensive loose parts removal program. The inspectors discussed with the licensee the visual examination of the steam generator including remote camera inspection through the dryer and through the handholds along the top of the tube bundle performed by the licensee as a consequence of the loose part alarm.

The inspectors reviewed Annual Report B18321 (2/28/01), containing a list of every imperfection in the steam generator per the technical specification definition of imperfection. The inspectors reviewed the eddy current report for Refueling Outage 13.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed licensee actions taken in response to the following condition reports (CRs) with respect to the maintenance rule.

- CR-01-10969 West DC Switchgear Area Vital Chiller (X169B) Failed to Start
- CR-01-12424 Safety Injection Tank Pressure Decline
- CR-01-12450 Reactor Coolant Pump (RCP) Lower Seal Temperature
- CR-01-11868 Positive Displacement Pump Would Not Start
- CR-02-01281 Alternate Plant Configuration for Pressurizer Backup Heaters
- CR-02-01285 Degraded Screen Wash Pump Fasteners

For each CR identified above, the inspectors reviewed the applicable system's maintenance rule scoping document, corrective actions taken in response to the equipment problem, and maintenance rule functional failure determination. The inspectors confirmed that the licensee appropriately tracked the occurrences against the systems' performance criteria, both for functional failures and unavailability time.

In addition, the system health reports for the reactor coolant system (RCS) and chilled water system were reviewed. Outstanding maintenance activities, preventive maintenance and performance trending information was also reviewed for the RCS. The inspectors reviewed the chilled water system's a(1) action plan and discussed the equipment failure with the system engineer and maintenance rule coordinator to confirm proper unavailability tracking for the system.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

During the outage, the licensee confirmed RCS leakage into the RBCCW system through the thermal barrier heat exchanger on the "C" reactor coolant pump (RCP). The licensee could have performed the replacement of the heat exchanger either with the core fully offloaded or at a reduced reactor coolant system water level, i.e., reduced inventory. The inspectors discussed the licensee's decision to perform this work at reduced inventory with the licensee risk analyst to confirm proper consideration was given to this decision from a shutdown risk perspective.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

During backshift hours on February 17, the inspectors observed control room activities during the transition into shutdown cooling operations and entry into operational Mode 5 (Cold Shutdown - reactor coolant temperature ≤ 200°F). The inspectors discussed the evolution with the reactor operators and confirmed adherence to approved procedures and technical specification requirements, which included operator control and monitoring of required cooldown rates.

During a control board walkdown, performed on March 30, the inspectors noted an annunciator lit for high RCP seal pressure on the "A" RCP. Operating Procedure (OP) 2301C, "RCP Operation," was reviewed to determine the actions required for an RCP seal failure. The inspectors discussed this condition with operators and reviewed operations logs to confirm that appropriate actions were taken for the failed RCP seal. The licensee documented the "A" RCP middle seal differential pressure degradation in CR-02-03847.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The following operability determinations (ODs) were reviewed.

- MP2-003-02 Engineered Safeguards Actuation System (ESAS) Auto Testing System Failure Alarm Generated
- MP2-004-02 Suspected SG #1 Secondary Side Impact Recorded by Loose Parts Monitor (LPM)
- MP2-007-02 "A" Emergency Diesel Generator (EDG) Breaker Tripped Open During Surveillance

The inspectors discussed the ESAS OD with the responsible system engineer to verify that the engineering justification for operability of the system was sound and no compensatory actions were required. The inspectors verified that this issue was entered into the licensee's corrective action program as CR-02-00700. The licensee successfully replaced the failing power supply, which caused the testing system failure, during 2R14.

OD MP2-004-002 was initiated following the receipt of suspected SG #1 secondary side impact events recorded by the Loose Parts Monitor System on February 8, 2002. The inspectors reviewed the OD to determine the acceptability of the licensee's conclusion that the #1 SG was operable, but degraded, following the receipt of elevated noise signatures from specific transducers located on the #1 SG. The inspectors reviewed the adequacy of the licensee's immediate corrective actions following the discovery, which included increased monitoring of SG chemistry parameters and increased monitoring of specific radiation monitors that would indicate primary-to-secondary leakage. The inspectors also attended the licensee's site operations review committee (SORC) meeting during which the safety significance of the recorded impact events and the adequacy of immediate actions taken were evaluated. The inspectors verified that the licensee had entered this issue into the corrective action program for resolution as CR-02-01159.

The inspectors reviewed the emergency diesel generator OD and the adequacy of the licensee's immediate corrective actions. This OD was written following the actuation of the reverse power trip relay which caused the EDG breaker to trip open. The inspectors also attended the licensee's SORC meeting, during which the adequacy of the troubleshooting activities and basis for operability were discussed. The inspectors discussed the issue with the responsible system engineer to verify that the engineering justification for operability of the system was sound and no compensatory actions were required. The inspectors determined that the EDG remained operable, even though the licensee was unable to determine the cause of the breaker trip, since the reverse power trip relay is bypassed during starting and running of the EDG under emergency conditions. The inspectors verified that this issue was entered into the licensee's corrective action program as CR-02-03709.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

A selection of existing and closed operator workarounds was inspected and discussed with control room operators. For those operator workarounds that required operability evaluations, risk assessments, and/or reasonable expectation of continued operability (RECO) evaluations, the inspectors reviewed the accompanying documentation. This documentation was reviewed to ensure the system reliability, system availability, mitigating system function (if appropriate) and potential for mis-operation were considered. For those operator workarounds that were established to account for system degradation, licensee corrective actions and the difficulty of operator compensatory actions were reviewed.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the completed documentation for post-maintenance testing (PMT) performed for automated work order (AWO) M2-01-07706, "Fabricate and Install a new Orifice Plate (FO-6039)." The inspectors reviewed the testing requirements completed for the new welds that were part of the work order, and evaluated their applicability to the requirements specified in the design specifications and in ANSI B31.1. In addition, the inspectors reviewed the results from the special procedure, SPROC ENG01-2-09, "Facility 1 RBCCW Simulated LNP Response Test for DCR M2-01005," and interviewed system and design engineers. These reviews and discussions with engineers were performed to verify that the design change reduced the overall pressure of the system and did not impact operability.

The inspectors also reviewed the programmatic requirements of Millstone procedure, MP20-WP-GDL40, "Pre- and Post-Maintenance Testing." The completed and in-process documentation for PMTs performed under the following AWOs was evaluated:

- M2-92-15697 "C" Reactor Coolant Pump (RCP) Replacement
- M2-01-04047 Steam Generator Manway
- M2-00-00087 17515 through 17518 Pressurizer Solenoid Operated Vent Valves

The inspectors reviewed the scope of the work activities and verified that the PMTs were appropriate to restore the operability of the associated systems.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed the following areas related to the 2R14 refueling outage for conformance to technical specification requirements and approved procedures. Selected activities were verified for each evolution.

- Transition into shutdown cooling operations and initial entry into operational Mode 5
- Initial draindown of the RCS to approximately the centerline of the hot leg piping in preparation for refueling outage activities. The review included an evaluation of the adequacy of RCS temperature and level indications available to operators during the draindown.
- Control and coordination of activities to minimize risk while in a reduced RCS inventory condition, which included contingencies established to ensure containment closure timeliness requirements would be met.
- Shutdown risk evaluations
- In progress outage risk assessment and control
- Mitigation and/or response strategies for losses of key safety functions
- Technical specification compliance with respect to removing equipment from service
- Control of configuration changes and emergent work
- Control, validation and verification of tags and temporary modifications
- Operation of the RCS to maintain pressure, level, and temperature within established ranges
- Operation of site electrical systems
- Operation of the SFP cooling system
- Reactivity Control, including boric acid inventory and SFP temperature
- Resolution of damaged upper tie plate of fuel assembly T57 (CR-02-02081)
- Refueling operations, including fuel handling, inventory, control and accounting
- Core fuel load verification
- Preparation for entry into Mode 5 at the end of 2R14
- Containment closeout walkdown prior to Mode 4
- Overall mode change management. Inspector activities included attendance at various mode change meetings, SORC meetings, outage planning meetings, and risk evaluation meetings; observation of control room activities; and review of open operability determinations and condition report action items prior to mode changes and completion status of various surveillances required for mode changes.
- Estimated critical position determination prior to Mode 2 entry

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed licensee performance related to the following surveillance tests.

- IC 2420F Control Element Drive System (CEDS) Coil Current Trace Test
- SP 2604B HPSI Pump Operability and Inservice Testing Facility 2
- SP 2605D Containment Leak Test, Type C (for Penetration 85 only)
- SP 2613G Integrated Test of Facility 1 Components (IPTE)

The inspectors observed inservice testing of the "C" HPSI pump in the control room. The inspectors verified that the test results met the surveillance procedure acceptance criteria and were consistent with past test results. The inspectors also verified that performance of the test adequately demonstrated equipment operability and capability of the "C" HPSI pump to perform its intended safety function.

The inspectors attended an infrequently performed test or evolution (IPTE) brief for the loss of normal power testing, SP 2613G, and independently verified that select prerequisites for the testing were met prior to test performance. The inspectors confirmed that the test was performed in accordance with approved procedures and that appropriate actions were taken when discrepancies in the procedure were identified.

The completed data sheets were reviewed for all tests to verify the equipment met procedural acceptance criteria and was operable consistent with technical specification requirements. The inspectors discussed CR-02-03868, which documented out of tolerance values identified during the CEDS testing, with the responsible system engineer, to confirm that the use-as-is disposition was justified.

b. <u>Findings</u>

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed and observed portions of Temporary Modification (TM) 2-02-10: “Closure Plate,” which provided for the removal of the "C" RCP from the RCS and its replacement with a blank flange. The inspectors discussed this condition with the cognizant system engineer, control room operators and the maintenance technicians responsible for the placement of the flood control device in addition to observing manipulation of the flange in the plant. The inspectors examined the 10 CFR 50.59 screening associated with the TM and discussed with operations management the likelihood that the modification could be put into place within the required one hour period (prescribed in the TM). The inspectors confirmed that the TM was processed in accordance with site procedure WC 10, "Temporary Modifications." The inspectors also verified that the TM provided an adequate assessment of the complementary alarms, procedural provisions and radiological impact associated with the exigent placement of the flood control device. No concerns with regard to the design basis, technical specification compliance or core heat removal were identified.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupational Radiation Safety [OS]

2OS2 ALARA Planning and Controls

a. Inspection Scope

During the period March 11 - 15, 2002, the inspectors conducted the following activities to evaluate the effectiveness of administrative, operational, and engineering controls to limit personnel exposure 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.

- The inspectors reviewed pertinent information regarding cumulative exposure history, current exposure trends, and ongoing activities. This information was used to assess the licensee’s effectiveness in establishing exposure goals and in keeping actual personnel exposure as low as is reasonably achievable (ALARA) when performing outage work activities. The inspectors also reviewed the results of the licensee’s efforts to reduce plant source terms through system flushing, component decontamination, temporary shielding installation, and shut down chemistry controls.

- The inspectors reviewed the exposure controls specified in ALARA Reviews (AR) for all work activities whose actual (or projected) cumulative exposure exceeded five person-rem. Work activities that were reviewed included:

- Reactor Disassembly/Reassembly (AR 2-02-01)
- Steam Generator Eddy Current Testing (AR 2-02-02)
- Valve Repairs (AR 2-02-11)
- Installation/Removal of Staging (AR 2-02-13)
- "C" Reactor Coolant Pump rotating assembly replacement (AR 2-02-17)
- Reactor Head Penetration Inspections/Repairs (AR 2-02-20)
- Pressurizer Penetration Inspection/Repair (AR 2-02-22).

Work-In-Progress ALARA Reviews and ALARA Council meeting minutes were reviewed to assess the licensee's response in addressing exposure challenges caused by changes in outage work scope and emergent work.

- Jobs-in-progress having radiological significance were observed. The inspectors reviewed the associated exposure controls, observed mock-up training and attended pre-job briefings for the under head manual ultrasonic examination of reactor head penetrations. For this task, the inspectors interviewed selected workers on their knowledge of the relevant radiation work permit, electronic dosimetry set points, and job-site radiological conditions. Interviews were also conducted with workers performing the "C" RCP rotating assembly replacement, pressurizer penetration inspection/repairs, and the boroscopic inspection of the reactor head.

- Independent radiation surveys were performed in the radiological controlled areas (RCA) of the Unit 2 containment building, auxiliary building, fuel handling building, and radwaste processing/storage areas. These independent surveys were used to confirm posted survey results and assess the adequacy of radiation work permits and associated controls. Keys to technical specification Locked High Radiation Areas were inventoried and these areas were verified to be properly secured and posted during tours.

- The effectiveness of various management controls were evaluated by reviewing the actions associated with the following:

  - A departmental self-assessment (MP-SA-02-001), regarding training of radiation protection technicians
  - Nuclear Oversight Field Observation reports
  - Radiation Protection Supervisory observations.

- The inspectors reviewed the following Condition Reports (CRs) relating to the control of personnel exposure and work activities to determine if the issues were identified in a timely manner and that appropriate actions were taken to evaluate and resolve the discrepancies. The regulatory and safety significance of each issue was also evaluated. Included in this review were CR's 02-2792, 02-2574, 02-2480, 02-2317, 02-2309, 02-2283, 02-1878, 02-1769, 02-1038, and 02-825.

Additionally, in evaluating the effectiveness of the licensee's problem identification and resolution program, the inspector attended daily outage status planning meetings, radiation protection department shift turnover meetings, and reviewed control point daily logs.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed implementation of the licensee's Occupational Exposure Control Effectiveness Performance Indicator (PI) Program. Specifically, the inspectors reviewed CRs and associated documents, for occurrences involving locked high radiation areas, very high radiation areas, and unplanned personnel exposures. This inspection reviewed the PIs 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.

b. Findings

No findings of significance were identified.

4OA5 Other

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

a. Inspection Scope

The inspectors reviewed the licensee's activities to detect "leak paths" and/or cracking of reactor pressure vessel (RPV) head penetration (head vent, in core instrumentation (ICI), and control element drive mechanism (CEDM)) nozzles in response to NRC Bulletin 2001-01. The licensee utilized an ultrasonic test (UT) technique to examine the integrity of the interference fit of the penetration tube with the head (to confirm the absence of a leak path) and to confirm the absence of flaws in the tube base material.

This inspection included interviews with data acquisition and analysis personnel, review of personnel qualification records, test procedures and procedure qualification and calibration records. The inspectors selected a sample of CEDM and ICI penetrations to conduct a real-time review of test data acquisition and analysis. The inspectors verified that anomalies, deficiencies and discrepancies identified during the examination process were recorded and placed in the licensee's corrective action process.

b. Findings

No findings of significance were identified.

The specific reporting requirements of TI 2515/145 are documented in Attachment 2.

.2 (Closed) LER 50-336/2001-007: Movement of Heavy Loads. On October 22, 2001, Unit 2 personnel identified an historical issue concerning the movement of heavy loads over a pathway that included a safety related pipe gallery enclosed in a trench. This trench is located below the cask wash down point and the railroad access bay floor. Because the cask crane was not "single failure proof," the licensee performed an analysis and determined that, although the probability was low, a dropped load of sufficient mass could have caused the cask pit floor to fail, resulting in a postulated loss of safety function.

The inspectors verified that the processes controlling such heavy loads were amended to account for this potential. The inspectors coordinated with the Region I, Senior Risk Analyst (SRA) to determine the significance and potential for this postulated event, using the NRC Significance Determination Process. The inspectors' on-site review and the evaluation of the Region I SRA identified no findings of significance.

4OA6 Meetings, including Exit

.1 Resident Exit Meeting Summary

The inspectors presented the inspection results to Messrs. D. Hicks and S. Scace and other members of licensee management on April 10, 2002. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any material examined during this inspection should be considered proprietary. A Framatome Engineering Information Record, 51-5008373-01, used during the Inservice Inspection, was marked proprietary. This document was returned to the licensee after the inspection was performed. No other proprietary information was identified.

.2 Site Management Visit and Other Public Meetings

Millstone Annual Assessment Meeting

In the afternoon of March 21, 2002, Mr. A. Randolph Blough, Director, Division of Reactor Projects (DRP), and Mr. Curtis Cowgill, Branch Chief, DRP, Branch 6, conducted the Millstone annual assessment meeting with Mr. J. Alan Price and other members of the licensee management.

Nuclear Energy Advisory Council (NEAC) Meeting

In the evening of March 21, 2002, Mr. A. Randolph Blough and Mr. Curtis Cowgill conducted a meeting with Mr. J. C. Markowicz and Mr. E. Woollacott and other members of NEAC. This meeting was held to discuss the annual assessment of the Millstone facilities and other items of interest.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. List of Items Opened, Closed and Discussed

Closed

50-336/2001-007	LER	Movement of Heavy Loads (4OA5.2)

b. Partial List of Documents Reviewed

Circumferential Cracking of RPV Head Penetration Nozzles (1R08.1)

Letter, Docket 50-423, B18466, Dominion Nuclear to NRC, Response to NRC Bulletin 2001-01
Letter, Docket 50-336, B18557, Dominion Nuclear to NRC, Supplemental Response to NRC Bulletin 2001-01
Letter, Docket 50-336, B18580, Dominion Nuclear to NRC, Supplemental Response to NRC Bulletin 2001-01 (Additional)
ERC 25203-ER-02-0005, Reactor Vessel Head Penetration Inspection Plan for Ultrasonic Examination of Interference Fit
VPROC NDE02-010 Rev. 000, Remote Ultrasonic Examination of Reactor Vessel Vent Line Penetrations
VPROC NDE02-001 Rev. 000, Remote Ultrasonic Examination of Control Rod Drive Mechanism (CRDM) Nozzles
Condition Report (CR) 02-02225, Vessel Head Penetration Inspections - UT Data Controls
CR 02-02077, Multiple Linear Indications Detected in CEDM Nozzle #21
CR 02-02014, Inspection Technique for RVHP ICI Nozzles
CR 02-01998, Supernut for drive screw on deployment device #1
CR 02-0227, Documentation of UT equipment calibration on RVHP Inspection
NRC Bulletin 2001-001, Circumferential Cracking of Reactor Head Penetration Nozzles
CEDM Nozzle Ultrasonic Examination Data Sheet for Nozzle #21

Inservice Inspection Activities(1R08.2)

Work Order Number M2 00 15166
Operability Determination MP2-029-00
Condition Report CR M2-00-2265
Calculation 00CP-02995-M2
Document 51-5008209-01
U2-24-SIP-REF01, Rev. 001, “Millstone Unit 2 Steam Generator Eddy Current Data Analysis Reference Manual”
M2-EV-00-0028, Rev. 0, “Millstone Unit 2 Steam Generator Condition Monitoring and Operational Assessment Refueling Outage 13"
ME-EV-00-0017, Rev. 0

U2-24-SIP-REF01, Rev. 001, “Unit 2 Steam Generator Eddy Current Data Analysis Reference Manual”
FTI-00-1353
Framatome Engineering Information Record 51-5008373-01 (Proprietary)
Annual Report B18321 (2/28/01)

UT Data Package
BSI-C-3000, 214-01-007
P-6-C-1-B, 214-01-019
SG-1-FW-IR-1, 214-01-026
SI-CF-A-062, 214-01-034
SI-CF-A-64, 214-01-035
SI-CF-X-39, 214-01-038

Test Data Package

Liquid Penetrant	BSI-C-3000, 214-02-022
Magnetic Particle	P-6-C-1-B, 214-03-007

Millstone Procedures
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”

ALARA Planning and Controls (2OS2)

Procedures:

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

<u>ALARA Reviews:</u>
AR 2-02-01, Reactor Disassembly and Reassembly
AR 2-02-02, Steam Generator Eddy Current Testing & Foreign Object Search & Retrieval
AR 2-02-04, Refueling Activities
AR 2-02-11, Valve Repairs & MOV maintenance
AR 2-02-13, Installation & Removal of Staging
AR 2-02-17, C-Reactor Coolant Pump Rotating Assembly Replacement
AR 2-02-20, Reactor Head Penetration Inspections & Repair
AR 2-02-22, Pressurizer Heater Penetration Inspection & MNSA clamp installation

<u>Departmental Self-Assessments:</u>
MP-SA-02-001, Training and Professional Development for Health Physics Personnel

<u>Radiation Protection/Waste Services Department Work Observations:</u>
02-1447, Containment Walkdown
02-1406, Observation of personnel entering MS-2 Containment
02-1405, Decontamination Under Reactor Head
02-1386, Air Sample Counting
02-1384, Determination of contamination control protocol to address Cobalt-58
02-1383, Mock-up training MS-2 Reactor Head inspection
02-1336, Framatone equipment removal from LSA containers
02-1306, Observations of containment personnel hatch control point activities
02-1285, Reactor Head Initial survey and decontamination
02-1284, Radiography
02-1283, Containment tour/Reactor head work/Refueling
02-1281, RP technician work practices
02-1280. Letdown heat exchanger work/Fuel Shuffle
02-1269, HP Control Point Worker Briefing
02-1268, ALARA Briefing for Containment Entry
02-1248, Initial Containment Entry Briefing following plant shutdown

<u>QA/QC Field Observations</u>:
FOQC-02-006, Installation of the S/G #1 Hot and Cold Leg Nozzle Dams
FOQC-02-003, Spent Fuel Pool Foreign Material Exclusion Log
MPS-OP-02-001-02, Housekeeping and Material Condition of the MRRF

<u>Other:</u>
2R14 Outage ALARA Guide
Nuclear Oversight Weekly Report, NO-02-0043

c. List of Acronyms Used

2R14	Unit 2 refueling outage 14
ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
AR	ALARA reviews
AWO	automated work order
CEDM	control element drive mechanism
CEDS	control element drive system
CRs	condition reports
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
ESAS	engineered safeguards actuation system
ETSS	examination technique specifications sheets
HPSI	high pressure safety injection
ICI	in core instrumentation
IPTE	infrequently performed test or evolution
LPSI	low pressure safety injection
LPM	loose parts monitor
OD	operability determination
ODSCC	outside diameter stress corrosion cracking
OP	operating procedure
P&ID	piping and instrumentation diagram
PIs	performance indicators
PMT	post-maintenance testing
RBCCW	reactor building closed cooling water
RCA	radiological controlled areas
RCP	reactor coolant pump
RCS	reactor coolant system
RECO	reasonable expectation of continued operability
RPV	reactor pressure vessel
SFP	spent fuel pool cooling and purification
SG	steam generator
SORC	site operations review committee
SP	surveillance procedure
SRA	senior risk analyst
TI	temporary instruction
TM	temporary modification
TSPs	tube support plates
TTS	top of tube sheet
UT	ultrasonic test

ATTACHMENT 2

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

a.1. The examination was performed by qualified and knowledgeable personnel. Data acquisition personnel were trained on a mock up using the actual equipment and tooling to be used in the field. Also, acquisition personnel received a pre-job briefing conducted by the Level II or Level III examiner in the setup and operation of the data acquisition system. Personnel performing calibration and data analysis were qualified to a minimum of Level III in ultrasonic examination. In addition, personnel had documented training in the Accusonex analysis software and had a valid PDQS for an automated UT procedure using Accusonex software.

a.2. The ultrasonic examination procedures were adequate and were reviewed and approved by the licensee Level III examiner.

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

a.4. The examination performed was capable of identifying the primary water stress corrosion cracking phenomena described in the bulletin.

b. The general condition of the reactor vessel head could not be assessed due to the presence of tightly adherent contoured insulation throughout the penetration locations. This area was not visible by eye or with the aid of remote video devices. This area is surrounded by the CEDM cooling shroud which also effectively obstructs access to the vessel head at the location of the penetrations.

c. A visual examination for boron deposits at the penetration locations could not be accomplished.

d. Six indications were identified in CEDM penetration number 21. The indications were sized, orientation and location determined, and were characterized as linear (cracks). Five of the indications were in the axial direction and one was circumferential. None of the cracks were through wall. The indications were recorded and entered into the licensee's corrective action program. Indication removal and repair of this penetration was planned.

e. The ALARA radiation exposure controls were effective in minimizing personnel exposure during the remote ultrasonic testing of the penetrations from underneath the vessel head. Also, a significant effort was made to reduce exposure to personnel involved in activities in close proximity to the vessel head on the refuel floor. The presence of the CEDM cooling shroud and the tightly adherent contoured insulation present significant obstructions to access of the outside diameter of the head. These obstructions will impede effective examinations of the outside surfaces of the head in the future. Also, the obstructions will present a significant challenge to radiation exposure controls.

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	50-423
License No.:	NPF-49
Report No.:	50-423/02-02
Licensee:	Dominion Nuclear Connecticut, Inc.
Facility:	Millstone Power Station, Unit 3
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	February 10, 2002 - March 30, 2002
Inspectors:	A. C. Cerne, Senior Resident Inspector, Unit 3 B. E. Sienel, Resident Inspector, Unit 3 G. T. Dental, Senior Resident Inspector, Seabrook K. M. Jenison, Senior Project Engineer, DRP T. A. Moslak, Health Physicist, Division of Reactor Safety (DRS)
Approved by:	Anthony McMurtray, Acting Chief Projects Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000423-02-02; on 02/10-03/30/02; Dominion Nuclear Connecticut, Inc., Millstone Power Station; Unit 3; Resident Inspection.

The inspection was conducted by resident and regional inspectors. No findings of significance were identified. The significance of most findings is indicated by the color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “no color” or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/reactors/operating/oversight.html.

A. Inspector Identified Findings

No findings of significance were identified.

B. Licensee Identified Violations

No licensee violations were identified.

TABLE OF CONTENTS

Report Details

Summary of Unit 3 Status

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

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

1R04 Equipment Alignment

.1 Partial Walkdowns of Risk-Significant Systems

a. Inspection Scope

The inspectors conducted partial system walkdowns of the train "A" emergency diesel generator (EDG) combustion air intake and exhaust system and the train "B" EDG building ventilation system. The applicable sections of the Final Safety Analysis report (FSAR) and related system descriptions for both of these systems were reviewed, as were the relevant operating and surveillance procedures and specific engineering procedures (EN 31089 & EN 31096). These reviews were performed to verify ventilation flow measurements and performance monitoring. In addition to verifying the proper status (e.g., fans), position (e.g., dampers), and condition of the active system equipment, the inspectors examined several equipment supports and other passive system components (e.g., penetrations) for structural integrity and evidence of long term degradation.

The inspectors discussed the impact of the observed area conditions (e.g., maintenance scaffolds, gas cylinder storage) with the cognizant system engineers. The inspectors examined the control room panel indications and reviewed some applicable alarm response procedures with the operators on shift, to confirm system operations in accordance with the design criteria and established operating limits. The inspectors also reviewed the licensee’s corrective actions for two condition reports (CRs) 02-02553 and 02-03898, involving documented information that was misleading with regard to operation of both of the inspected EDG support systems. The inspectors confirmed that neither the identified CRs, nor any in-process work activities, adversely affected system operability. As appropriate, the area and equipment environmental qualification criteria were reviewed. The applicable requirements of the technical specifications, with respect to both temperature monitoring and controls, were verified to be maintaining the system equipment within their operable qualification envelopes.

b. Findings

No findings of significance were identified.

.2 Complete Risk-Significant System Walkdown

a. Inspection Scope

The inspectors conducted a complete walkdown of the portion of the underdrain system (SRW) designed to prevent groundwater leakage from adversely affecting the safety-related equipment in the containment recirculation pump vaults of the Engineered Safety Features (ESF) Building. The SRW system configuration and equipment lineup were examined in accordance with the details and notes of the SRW piping and instrumentation drawing, EM-106D-1. The system layout was also evaluated with respect to the design criteria delineated in design change record (DCR) M3-00004, Revision 1.

The inspectors examined the structural condition of the enclosed sump, 3SRW*SUMP 6, containing the groundwater leakage within the ESF building. This sump also serves as a supplementary leak collection and release system (SLCRS) boundary; i.e., a plant secondary containment design feature. The Technical Requirements Manual (TRM) provisions (3.6.1.6) for the SRW system operability were reviewed with the system engineer to verify that trending of the groundwater in-leakage ensured that the SRW system was capable of meeting its design criteria. The inspectors also confirmed that the sump is routinely opened for inspection and discussed the results of the most recent inspection with the SRW system engineer. In addition to checking various SRW structural and pipe support configuration details and electrical design features, the inspectors verified that piping that had been abandoned in place in each ESF vault was properly capped and isolated as a SLCRS boundary.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed walkdowns of two areas (Fire Areas CB-4 & CB-6) at elevation 4'-6" in the Control Building and four areas (ESF-1, ESF-2, ESF-8 & ESF-9) at various elevations in the Engineering Safety Features (ESF) Building. The fire detection and suppression equipment located in these areas and overall area fire protection design features were checked for conformance to the general area description and design details specified in the Millstone Unit 3 Fire Protection Evaluation Report. The inspectors examined equipment (e.g., emergency lighting units, fire extinguishers) for evidence of degraded or inoperable conditions, assessed the transient combustible materials stored in each Fire Area, and discussed any identified issues with the cognizant operations and fire protection personnel. The inspectors also reviewed condition report CR-02-04291, documenting a licensee identified concern regarding fire extinguisher coverage in the ESF building. The inspectors discussed both the basis for the concern and the licensee's follow-up corrective actions with the site Supervisor of Fire Operations.

During the backshift evening hours on March 28, the inspectors observed operator actions in support of an unannounced, graded fire brigade drill, involving a simulated turbine building fire scenario. The inspectors assessed the communications and direction provided by the Unit Supervisor in the implementation of the fire fighting strategy and noted those actions taken by the plant equipment operator designated as the Fire Technical Advisor for the shift crew. Subsequent to the drill performance, the inspectors reviewed the completed drill report and verified the assessment and evaluation activities with regard to the expected fire brigade actions and the documented acceptance criteria.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the maintenance rule related system health reports and supporting materials for the following systems:

- Engineered Safety Features and Reactor Protection Systems
- Station Blackout Diesel
- Containment Structure and Containment Penetration Systems
- Emergency Lighting

The supporting material that was reviewed included an (a)(1) evaluation report for the emergency lighting system, various outstanding and completed maintenance activities, preventive maintenance documents, performance and availability trending, operability determinations, corrective actions for CRs and the designated compensatory measures. For each CR reviewed, the inspectors evaluated the applicable system's maintenance rule scoping document, corrective actions taken in response to the equipment problem, and the maintenance rule functional failure determination. The inspectors confirmed that the licensee appropriately tracked the occurrences against the systems' performance criteria, both for functional failures and unavailability time. The inspectors verified that any differences in calculational approaches that existed among the various systems and system engineers were not significant contributors to the required operability or availability determinations.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed work planning activities, priority assignments, and corrective actions for the emergent work items documented in three condition reports, as follows:

- CR-02-01133 Valve 3SWP-V257 has stem-disc separation
- CR-02-01485 Control Rod Drive Mechanism (CRDM) Fan 3HVU-FN2A tripped
- CR-02-02482 Unplanned LCO entry - cleaning of Unit 3 control room kitchen/restroom sink drains results in breach of control building pressure boundary

For all three issues, the operators on shift were interviewed and immediate response actions were reviewed. For the low lubricating water flow condition documented in CR-02-01133, the troubleshooting and action plan was reviewed and the inspectors noted that no stem-disc separation had occurred. The low-flow conditions were due to strainer sizing, rather that valve problems. The inspectors confirmed that the engineering evaluations determined the proper strainer mesh sizing for this application and the properly sized mesh strainers were installed in both trains of the circulating water pump lubricating water system.

For CR-02-01485, the inspectors verified that the immediate operator actions, in accordance with the related alarm response procedure, appropriately removed the failed CRDM fan from service, and started a spare fan. The inspectors also checked operations and health physics/chemistry actions to follow up on a containment radiation monitor (CMS22) particulate alert that alarmed around the time that the CRDM fan failed. The results of a containment air sample, along with the automatic reset of the CMS22 alarm, provided evidence that the CMS22 alarm was related to the degrading fan operation and not to a radiation event inside containment.

For CR-02-02482, the inspectors reviewed a related Millstone Unit 3 license amendment, issued on February 20, 2002. This amendment, No. 203, revised technical specification (TS) sections 3.7.7, "Plant Systems - Control Room Emergency Ventilation System," and 3.7.8, "Plant Systems - Control Room Envelope Pressurization System," and provided a 24-hour allowed outage time for an identified control room boundary breach. The inspectors interviewed the cognizant Shift Manager and Regulatory Affairs personnel and verified that this event was not reportable and did not constitute a situation requiring entry into TS 3.0.3. The licensee developed an action item, based on this issue, to post signs in the control room kitchen and restroom, warning that opening the sink drains result in breaches of the control room pressure boundary.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability determination (OD) MP3-38-01 against the criteria in Generic Letter 91-18, "Resolution of Degraded and Non-Conforming Conditions," and NRC Inspection Manual Part 9900 "Operable/Operability-Ensuring the Functional Capability of a System or Component." The criteria in these documents were used to determine that the identified condition did not adversely affect safety system operability or plant safety. Specifically, the inspectors reviewed the OD analysis for an intermittent alarm on the “A” channel of the digital rod position indication (DRPI) for one shutdown bank rod cluster control assembly to confirm the proper application of the relevant TS criteria. The system engineer was interviewed to determine the extent of the condition. The inspectors also reviewed the design requirements for the DRPI system, as described in FSAR Section 7.7.1.3.2. The inspectors noted that this alarm was due to a problem with a coil on the “A” channel for this assembly. The alarm did not come in when the “B” channel was selected.

The inspectors reviewed for OD for CR-02-02205, documenting a condition involving the nonconformance of the service water (SWP) pump shaft coupling bolts with respect to the pump manufacturer’s specification. Millstone Station Procedure, RP 5, Revision 002-04, was reviewed to determine whether the identified SWP bolting material nonconformance raised any pump design or qualification questions that required an OD to be initiated. The inspectors noted that the licensee’s evaluation of the installed bolt condition was based, in part, upon an acceptable yield stress safety factor and that the nonconforming bolts met the required design criteria. The inspectors interviewed the cognizant system engineer regarding the history of SWP pump overhaul activities, which led to the identified discrepant conditions. The inspectors determined that the licensee had considered other bolting properties (e.g., corrosion resistance) in the CR operability assessment and in the disposition of the required design documentation for the subject coupling bolts.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

A selection of existing and closed operator workarounds was inspected and discussed with control room operators. For those operator workarounds that required operability determinations, risk assessments, and/or reasonable expectation of continued operability (RECO) evaluations, the inspectors reviewed the accompanying documentation. This documentation was reviewed to ensure the system reliability, system availability, mitigating system function (if appropriate) and potential for mis-operation were considered. For those operator workarounds that were established to account for system degradation, licensee corrective actions and the difficulty of operator compensatory actions were reviewed.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance testing (PMT) activities to ensure: 1) the PMT was appropriate for the scope of the maintenance work completed; 2) the acceptance criteria were clear and demonstrated operability of the component; and 3) the PMT was performed in accordance with procedures. The PMT activities related to the following work, system restorations, and testing were reviewed:

- SP 3646A.1 “A” Emergency Diesel Generator (EDG) Operability Test
- SP 3670.5 Cold Weather Protection

The inspectors specifically reviewed the PMT activities conducted on March 8, with respect to the "A" EDG testing performed following completion of planned work. The work was described in automated work order (AWO) M3 01-21646, "Jacket Water Cooler and Engine Air Cooler Heat Exchanger Inspection," and AWO M3 01-21314, "3EGF*LS41A “A” EDG Follow Fuel Transfer Pump Level Control starts Pump at 270 gallon vice 322 gallon." Additionally, the inspectors verified the restoration of the safety-related dampers in both trains of the service water pumphouse ventilation system, from the winter mode of operation to the summer mode. The inspectors also confirmed that the Production Maintenance Management System (PMMS) restoration dates were consistent with those specified in Engineering & Design Coordination Report T-B-05710 affecting the system configuration, ventilation air flow rates, and balanced operation.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed licensee performance related to the following surveillance tests.

- SP 3610A.2 Residual Heat Removal Pump 3RHS*P1B Operational Readiness Test
- SP 3626.13 Service Water Heat Exchangers Fouling Determination
- SP 3630D.1 Charging Pump Cooling 3CCE*P1A Operational Readiness Test
- SP 3646A.9 Slave Relay Testing - Train B (Section 4.29, Safety Injection)

The residual heat removal, charging pump cooling, and slave relay tests were observed in the control room to confirm performance of the tests in accordance with approved procedures. The completed data sheets were reviewed for all tests to verify that the equipment met procedural acceptance criteria and was operable consistent with technical specification requirements. Additionally, the inspectors checked the conduct of licensee field activities in the ESF building with respect to the performance of SP 3626.13 on one train "A" ESF building ventilation air conditioning unit (3HVQ*ACUS1A). The test results for this surveillance were reviewed and checked for consistency against other common system test results.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Design Change Notice (DCN) DM3-00-0047-02, involving a setpoint revision for the actuation of the anticipated transient without scram (ATWS) mitigation system actuation circuitry (AMSAC). The setpoint change, from 5% to 15.1% narrow range (NR) steam generator (SG) water level, was initiated, in part, as corrective action to an identified error in the SG level narrow range instrumentation. This error was documented in condition report CR-02-01953, which references a generic Westinghouse SG mid-deck plate pressure loss issue. The inspectors evaluated this modification in consideration of the AMSAC description provided in Section 7.8 of the FSAR, as well as information documented in NRC Information Notice 2002-10, dated March 7, 2002.

The inspectors also reviewed a licensee engineering summary of the proposed "Resolution of SG Level Indication for Millstone Unit 3." This summary discussed the generic issues associated with the SG level indication problems documented in three Westinghouse Nuclear Safety Advisory Letters. The licensee's Safety Analysis Supervisor was interviewed regarding the implementation of the AMSAC modification, the determination of operability for all other trip setpoints potentially affected by this issue, and any plans for further design changes. With respect to 10 CFR 50.62, "Requirements for Reduction of Risk from ATWS Events for Light-Water-Cooled Nuclear Power Plants" and the Unit 3 design and safety analysis, the inspectors discussed the impacts of this issue with a member of the licensee's Regulatory Affairs staff.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

Occupational Exposure Control Effectiveness

Refer to NRC Inspection Report 50-336/02-02, Section 4OA1 for specific details.

4OA6 Meetings, including Exit

.1 Resident Exit Meeting Summary

The inspectors presented the inspection results to Messrs. D. Hicks and S. Scace and other members of licensee management on April 10, 2002. The licensee acknowledged the findings presented.

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

.2 Site Management Visit and Other Public Meetings

Millstone Annual Assessment Meeting

In the afternoon of March 21, 2002, Mr. A. Randolph Blough, Director, Division of Reactor Projects (DRP), and Mr. Curtis Cowgill, Branch Chief, DRP, Branch 6, conducted the Millstone annual assessment meeting with Mr. J. Alan Price and other members of the licensee management.

Nuclear Energy Advisory Council (NEAC) Meeting

In the evening of March 21, 2002, Mr. A. Randolph Blough and Mr. Curtis Cowgill conducted a meeting with Mr. J. C. Markowicz and Mr. E. Woollacott and other members of NEAC. This meeting was held to discuss the annual assessment of the Millstone facilities and other items of interest.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. List of Items Opened, Closed and Discussed

None

b. List of Acronyms Used

AMSAC	ATWS mitigation system actuation circuitry
ATWS	anticipated transient without scram
AWO	automated work order
CR	condition report
CRDM	control rod drive mechanism
DCR	design change record
DRPI	digital rod position indication
EDG	emergency diesel generator
ESF	engineered safety features
FSAR	Final Safety Analysis Report
OD	operability determination
PMMS	production maintenance management system
PMT	post-maintenance testing
RECO	reasonable expectation of continued operability
SG	steam generator
SLCRS	supplementary leak collection and release system
SWP	service water
TS	technical specification
TRM	technical requirements manual
SRW	underdrain system