May 4, 2001

Mr. R. P. Necci, Vice President - Nuclear Technical Services
C/O Mr. D. A. Smith, Process Owner - Regulatory Affairs
Dominion Nuclear Connecticut, Inc.
Rope Ferry Road
Waterford, Connecticut 06385

SUBJECT: MILLSTONE UNITS 2 AND 3 - NRC INSPECTION REPORTS 05000336/2001-002 AND 05000423/2001-002

Dear Mr. Necci:

On March 31, 2001, the NRC completed inspections at your Millstone Units 2 & 3 reactor facilities. The enclosed reports document the inspection findings which were discussed on April 18, 2001 with Mr. E. Grecheck 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.

Based on the results of these inspections, the inspectors identified one Unit 2 issue of very low safety significance (green). This issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of these inspection reports, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Millstone facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document

Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Curtis J. Cowgill, Chief
Projects Branch 6
Division of Reactor Projects

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

Enclosures:
(1) NRC Inspection Report 05000336/2001-002
Attachment 1: Supplemental Information

(2) NRC Inspection Report 05000423/2001-002
Attachment 1: Supplemental Information

cc w/encls:
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ENCLOSURE 1

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	05000336
License No.:	DPR-65
Report No.:	05000336/2001-002
Licensee:	Dominion Nuclear Connecticut, Inc.
Facility:	Millstone Nuclear Power Station, Unit 2
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	February 11, 2001 - March 31, 2001
Inspectors:	S. R. Jones, Senior Resident Inspector, Unit 2 P. C. Cataldo, Resident Inspector, Unit 2 P. Frechette, Physical Security Inspector, Division of Reactor Safety (DRS) T. A. Moslak, Heath Physicist, DRS D. M. Silk, Sr. Emergency Preparedness Inspector, DRS G. C. Smith, Sr. Physical Security Inspector, DRS
Approved by:	Curtis J. Cowgill, Chief Projects Branch 6 Division of Reactor Projects Region I

SUMMARY OF FINDINGS

IR 05000336/2001-002; on 02/11-03/31/01; Dominion Nuclear Connecticut, Inc., Millstone Nuclear Power Station; Unit 2. Operator Work-Arounds, Licensee Identified Violations.

The inspection was conducted by resident and regional inspectors. The inspection identified one green issue, which was a Non-Cited Violation. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 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/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- **Green.** The licensee failed to implement timely and effective corrective actions to address recurrent lifting of a letdown line relief valve during periods when two charging pumps are placed in operation, such as during implementation of the abnormal operating procedure for a rapid downpower. This failure is considered a violation of 10 CFR 50, Appendix B, Criterion XVI. This condition is of very low safety significance because, although the multiple relief valve lifts slightly increased the frequency of initiating events involving a loss of reactor coolant system inventory, mitigating equipment was unaffected. The violation is being treated as a Non-Cited Violation. (Section 1R16)

B. Licensee Identified Violations

A violation of very low significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. This violation is listed in Section 4OA7 of this report.

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Report Details

SUMMARY OF UNIT 2 STATUS

The plant operated at essentially 100 percent power throughout the inspection period, with the exception of the following unplanned, short-term power reductions:

March 4, 2001	Power reduction to 92 percent power due to temporary failure of the in-core neutron flux monitoring system;
March 9-10, 2001	Power reduction to 94 percent power due to a feedwater heater drain system transient caused by operator actions to isolate a minor leak from a feedwater heater level sensing line;
March 19-20, 2001	Precautionary power reduction to 55 percent power when operators were unable to reset an “A” main feedwater pump trip condition during testing (the trip condition was blocked from causing an actual feedwater pump trip during the testing and subsequent maintenance activities).

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

1R04 Equipment Alignment

.1 Partial System Alignment Checks

a. Inspection Scope

Inspectors performed the following partial system alignment checks:

- Following surveillance testing that realigned low pressure safety injection (LPSI) valves, the inspector verified that the injection piping for LPSI was correctly aligned in accordance with Surveillance Procedure (SP) 2604L-2, “LPSI System Valve Alignment Check, Facility 1,” SP 2604M-2, “LPSI System Valve Alignment Check, Facility 2,” and system piping and instrumentation diagram 25203-26015.

- While the “A” channel of the reactor protection system (RPS) was declared inoperable and partially removed from service due to problems with the nuclear instrument power signal for that channel, the inspector verified that the remaining three RPS channels were correctly aligned in accordance with Operating Procedure 2380, “RPS and NI Safety Channel Operation,” and Unit 2 Technical Specification 3.3.1.1.

b. Findings

No findings of significance were identified.

.2 Full Auxiliary Feedwater System Alignment Check

a. Inspection Scope

The inspector verified that the accessible portions of the auxiliary feedwater (AFW) system were correctly aligned in accordance with SP 2610C-2, "Auxiliary Feedwater System Lineup Verification," and system piping and instrumentation diagrams 25203-26002 and 25203-26005. The inspector also verified that outstanding condition reports (CRs) generated to address deficiencies and adverse conditions associated with the AFW system did not impact the system's ability to perform its required safety functions. The CRs that were reviewed are listed below:

M2-00-3028	M2-00-3105	M2-00-3161	M2-00-3192	M2-00-3205
M2-00-3310	M2-00-3318	M2-00-3392	M2-00-3426	M2-00-3535
M2-00-3541	M2-01-0022	M2-01-0028	CR-01-00594	CR-01-01159

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspector reviewed the licensee's fire hazard analysis for the following plant areas: (1) Auxiliary Building Cable Vault, Fire Area A-24; (2) Auxiliary Building Ventilation Equipment Room, Fire Area A-33; (3) East 480 Volt Load Center Room, Fire Area A-28; (4) West 480 Volt Load Center, Fire Area T-6; and (5) Vertical Cable Chases Connecting the Turbine Building and Auxiliary Building Cable Vaults. The inspector toured these areas to verify the functionality of fire detection devices (based on the absence of trouble alarms on local fire monitor panels), the integrity of penetration seals and other fire barriers, and the adequate control of transient combustible materials located in these areas. The inspector used procedure OP 2341A, "Fire Protection System," to verify the correct operational alignment of the wet-pipe fire suppression sprinkler systems and deluge systems protecting the Auxiliary Building Cable Vault and the vertical cable chase connecting to the Turbine Building East and West Cable Vaults.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On March 29, 2001, the inspector observed the conduct of an licensed operator requalification simulator training exercise. The inspector observed licensed operator performance with a focus on the following activities: communications, implementation of normal and emergency procedures, command and control, and technical specification compliance. The inspector verified that the evaluators addressed operator performance issues that were identified during the exercise, and that applicable training objectives had been achieved.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed the licensee's implementation of the maintenance rule for the following systems. The inspector verified that scoping tables associated with each system had appropriate performance criteria consistent with the plant configuration, and in accordance with Integrated Maintenance Program, Program Instruction PI-3, "Performance Criteria." The inspector reviewed associated CRs to verify that the identified issues for these systems were correctly evaluated and classified based on Engineering Department Instruction 30710, "Maintenance Rule Functional Failures":

- Enclosure building filtration system (EBFS) and condenser air removal system with associated CRs M2-00-2134, M2-00-2135, M2-00-2494, M2-00-3261, and CR-01-00978.

- Safety Injection Tanks and associated CRs M2-00-2696, M2-00-2725, and M2-00-3137.

- 125 VDC distribution system and associated CRs M2-00-0382, M2-00-0768, M2-00-1306, M2-00-2751, M2-00-2776, M2-00-2782, M2-00-2809, M2-00-2873, and M2-00-3191

- Emergency diesel generator (EDG) system and associated CRs M2-00-2195 and CR-01-00783. The inspector also verified that the license was correctly tracking train unavailability in accordance with their maintenance rule implementation guidelines.

- Pressurizer heaters and associated CR-01-03224, which documented a failure of the Group 1 pressurizer proportional heaters to generate adequate heat.

- Inadequate core cooling monitor system and associated CRs M2-00-0245, M2-00-0380, M2-00-1778, M2-00-1786, M2-00-1936, M2-00-2931, M2-00-3062, and

M2-00-3487, which involved failures of individual in-core temperature sensors and computer software problems.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

.1 Feedwater Pump Protection System Malfunction

a. Inspection Scope

The inspector reviewed work activities during emergent work on March 19 and 20, 2001, to correct a malfunction in the feedwater pump protection system that occurred during surveillance testing. With the protection system manually blocked from initiating a pump trip, operators were unable to reset a component in the protection system that was stuck in a tripped state. The inspector verified that the licensee reduced reactor power to a level within the capability of one main feedwater pump before initiating maintenance activities on the affected pump, thereby significantly reducing the potential for the maintenance activities to initiate a reactor trip at reduced steam generator water inventory due to a partial loss of main feedwater flow.

b. Findings

No findings of significance were identified.

.2 Charging Pump Maintenance Activities

a. Inspection Scope

The inspector reviewed the licensee's work coordination during planned maintenance activities affecting the "A" charging pump and "A" AFW pump on February 27, 2001. Due to the cumulative risk associated with the simultaneous removal from service of both pumps, the inspector verified that the licensee appropriately assessed and managed the plant's increased risk through scheduling and control of maintenance activities. Specifically, the inspector verified that the licensee ensured maintenance work on the "A" AFW pump was not allowed to commence until all work had been completed on the "A" charging pump and the charging pump was returned to service.

b. Findings

No findings of significance were identified.

.3 Pressurizer Heater Emergent Work

a. Inspection Scope

The inspector reviewed the licensee's work coordination when emergent work to correct an inoperable bank of pressurizer proportional heaters was implemented at a time when work was scheduled on the "B" atmospheric dump valve. The inspector verified that the licensee appropriately managed the two work activities by deferring the scheduled work on the "B" atmospheric dump valve.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

Operator Performance during a Rapid Reduction in Reactor Power

a. Inspection Scope

In response to a significant reduction in reactor power following problems with main feedwater pump trip mechanism testing on March 19, 2001, the inspector performed a detailed review of plant process computer data related to reactor core parameters, reactivity control systems operation, and feedwater system parameters. Based on these parameters, the inspector verified that significant operator actions reflected by changes in these parameters were in accordance with abnormal operating procedure (AOP) 2575, "Rapid Downpower."

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 Reactor Building Closed Cooling Water System Water Hammer

a. Inspection Scope

The inspector evaluated the licensee's actions following the identification of the potential for reactor building closed cooling water (RBCCW) system relief valves to stick open following a water hammer event. The licensee concluded that a waterhammer could occur when RBCCW flow was restored to the containment air recirculation coolers during a large break loss-of-coolant accident with a concurrent loss of normal power, but their original evaluation did not address the effect the water hammer could have on relief valve operation and the maintenance of adequate coolant inventory within the RBCCW system. The inspector reviewed operability determination OD MP2-051-01, which addressed operability of the RBCCW system. The inspector verified that the licensee provided an adequate basis for continued operability in that:

(1) All thermal relief valves in the RBCCW system are currently gagged.

(2) The ungagged relief valve protecting the letdown heat exchanger is distant from the containment air recirculation coolers, which reduces the magnitude of the

potential pressure spike, and located where it has substantial margin between normal operating pressure and its setpoint, which reduces the likelihood the relief valve would lift.

(3) Four relief valves designed to protect against inter-system loss of coolant accidents have higher setpoint pressures, which reduces the likelihood these relief valves would lift.

(4) In the event any of the five ungagged relief valves were to lift, they are likely to immediately reseat, which would not release sufficient coolant inventory to affect the operability of the RBCCW system.

b. Findings

No findings of significance were identified.

.2 Auxiliary Feedwater System and Tool Crib Seismic Interaction

a. Inspection Scope

The inspector evaluated the licensee's actions following the identification that a potential seismic interaction existed between an unrestrained tool crib and certain AFW system components located in the west penetration area of the auxiliary building. The inspector reviewed the licensee's initial operability determination for the AFW flow transmitter FT-5278A as documented in operability determination OD MP2-056-01. The licensee based this determination on the following:

(1) A seismic event would not cause the tool crib to topple or slide.

(2) Expected motion of the tool crib would not cause significant damage to the terminal box containing cable for the AFW flow transmitter.

(3) The AFW flow transmitter cables within the potentially affected terminal box are installed as a continuous run with enough flexibility to limit any significant damage to the subject cables during a seismic event.

The licensee subsequently closed the OD following the removal of the tool crib, which eliminated any potential seismic interactions.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed the licensee's lists of operator workarounds, control room panel deficiencies, and alternate plant configurations. The inspector discussed the identified issues with operations department personnel and reviewed the associated CRs to evaluate the impact on the functional capability of the systems, and the operator's ability to effectively respond to transients. The inspector also evaluated the cumulative effects of the identified conditions on the ability of operators to respond to and implement abnormal or emergency operating procedures during a transient.

The inspector reviewed the following CRs relative to a potential unidentified operator burden that involved a requirement for operators to locally throttle open letdown ion exchanger (IX) outlet isolation valves from 1-1/2 turns to 2-1/2 turns when starting a second charging pump:

CR-95-04011	July 21, 1995
M2-00-0592	March 10, 2000
M2-00-0880	April 15, 2000
CR-01-03115	March 21, 2001

These CRs documented four occasions where the letdown relief valve, 2-CH-354, had lifted during two charging pump operation because the IX outlet valve was not sufficiently open. The inspector discussed this apparent operator burden with operation department and corrective action department personnel.

The licensee had established a corrective action to perform troubleshooting regarding the relief valve lifting issue as documented in CR-95-04011. This troubleshooting plan was completed in April 1999, and concluded that the IX outlet isolation valves should be reset to approximately four turns open. However, the inspector identified that this recommendation was never implemented. Consequently, the normal position of the IX outlet isolation valves remained at 1-1/2 turns open.

When the relief valve again opened in May 2000, as described in CR M2-00-0592, the licensee implemented a procedure change to OP 2304B, "Purification Portion of the CVCS [Chemical and Volume Control System]," that required the IX outlet isolation valves to be throttled open 2-1/2 turns during two charging pump operation.

The lifting of relief valve 2-CH-354 on March 20, 2001, was caused by the failure to throttle open the IX outlet isolation valve 2-CH-378 following the start of a second charging pump during a rapid downpower. Step 4.6 of AOP 2575, "Rapid Downpower," Rev. 02-02, specified that operators ensure two charging pumps are operating, but the procedure did not contain instructions regarding the throttling open of the purification IX outlet isolation valves to accommodate the increase in letdown flow during two charging pump operation. Two charging pumps are required for adequate core reactivity control through boration during a rapid downpower. The inspector found that the need to open the IX outlet valve constituted an operator burden in that operator action would be required during implementation of an AOP (i.e., AOP 2575) to preclude relief valve operation during two charging pump operation.

b. <u>Findings</u>

The NRC identified that the licensee failed to implement timely and effective corrective actions to address recurrent lifting of a letdown line relief valve during periods with two charging pumps in operation. Although the condition increased the potential for reactor coolant system (RCS) inventory loss, the condition was of very low safety significance (Green) because mitigating equipment was unaffected.

The inspector evaluated the letdown line relief valve lifts using the NRC's Significance Determination Process (SDP). The inspector concluded that the relief valve operations had a credible impact on plant safety in that failure of the relief valve to reseat would require actuation of mitigating equipment (e.g., letdown isolation) to preclude RCS inventory loss. The SDP classified this condition as one of very low safety significance (Green) because, although the condition increased the frequency of initiating events involving RCS inventory loss, it was unlikely to affect mitigating equipment.

The licensee has initiated a root cause investigation regarding the long standing problem with lifting of relief valve 2-CH-354 during two charging pump operation and the related corrective action issues. The licensee implemented an alternate plant configuration to position the IX outlet isolation valves to four turns open pending completion of the root cause investigation and associated corrective actions. These issues have been entered into the licensee's corrective action program as CR-01-03179.

The licensee's failure to implement timely and effective corrective actions to prevent relief valve lifts when two charging pumps were placed in operation was considered a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." This violation is more than minor because the multiple relief valve operations had a credible impact on plant safety. The violation is being treated as a Non-Cited Violation **(NCV 05000336/2001-002-01)**, consistent with Section VI.A of the NRC Enforcement Policy, NUREG-1600. This violation is in the licensee's corrective action program as CR-01-03179.

1R19 Post-Maintenance Testing

.1 Enclosure Building Filtration System Flow Rate Testing

a. Inspection Scope

On February 27, 2001, the inspector observed testing associated with work order MP-00-08673, which involved EBFS flow rate verification performed in accordance with the following procedures:

- EN 21063C, “Enclosure Building Filtration System and Auxiliary Exhaust Actuation System Ventilation Testing.”
- EN 21063A, “HVAC Test and Balancing.”
- OP 2314G, “Enclosure Building Filtration System.”

The flow rate testing was performed to ensure compliance with applicable licensing and design basis acceptance criteria following the permanent de-energization of the Unit 1 main exhaust fans in support of the decommissioning of Unit 1. The inspector reviewed the test data and verified that the post-maintenance tests were adequate given the scope of the activities, and provided adequate assurance that the EBFS would continue to perform its required safety function.

b. Findings

No findings of significance were identified.

.2 Charging Pump Testing

a. Inspection Scope

The inspector observed testing associated with work orders M2-016783, M2-00-19945, and M2-01-00846. These work orders involved corrective maintenance activities on the “A” charging pump, and were followed by post-maintenance testing performed in accordance with SP 2601G, “”A” Charging Pump Operability Test. The inspector reviewed the test data and verified that the post-maintenance test was adequate given the scope of the activities, and provided adequate assurance that the “A” Charging Pump would continue to perform its required safety function.

b. Findings

No findings of significance were identified during this inspection.

.3 Turbine-Driven Auxiliary Feedwater Pump (TDAFP) Steam Supply Check Valve Repair

a. Inspection Scope

The inspector reviewed post-maintenance testing associated with the weld repair of 2-MS-4B, the TDAFP steam supply check valve, as detailed in work order M2-01-01865. The weld repair involved seal welding of mechanical joints on the check valve hinge pin cover assembly to resolve minor steam leaks from these joints. The inspector reviewed the post-maintenance testing that was performed in accordance with the following procedures:

- EN 21218, "Post Repair/Replacement Leakage Test."
- SP 2610E, "2-MS-4A and 2-MS-4B Part Stroke IST."

The inspector reviewed the test data and verified that the post-maintenance tests were adequate given the scope of the activities, and provided adequate assurance that the check valve would continue to perform its required safety function.

b. Findings

No findings of significance were identified.

.4 Atmospheric Dump Valve Positioner Replacement and Controller Refurbishment

a. Inspection Scope

On March 7, 2001, the inspector observed a portion of the post-maintenance testing associated with work orders M2-00-03290, which involved refurbishment of the No. 1 Atmospheric Dump Valve (ADV) Controller, and M2-01-00988, which involved replacement of the No. 1 ADV positioner. The inspector reviewed the post-maintenance testing conducted using portions of procedure SP 2402F, "Atmospheric Dump Valve (ADV) Pressure Controller Calibration," and SP 2610E, "MSIV Closure and Main Steam Valve Operational Readiness Testing." The inspector verified that the test data satisfied applicable acceptance criteria and that the post-maintenance tests were adequate to assure that the No. 1 ADV would continue to perform its design functions.

b. Findings

No findings of significance were identified.

.5 Diesel Generator Preventive Maintenance

a. Inspection Scope

The inspector reviewed post-maintenance testing associated with semi-annual preventive maintenance activities for the "B" EDG, which were performed under work orders M2-00-09980 and M2-00-0694719945, and M2-01-00846 on March 14, 2001. These work orders involved inspection and cleaning of EDG service water-cooled heat exchangers and inspection of the generator. The post-maintenance testing was performed in accordance with SP 2613L, “Diesel Generator Slow Start Operability Test, Facility 2.” The inspector reviewed the test data and verified that the post-maintenance test was adequate to demonstrate that the “B” EDG would continue to perform its required safety function.

b. Findings

No findings of significance were identified.

.6 Reactor Protection System Maintenance

a. Inspection Scope

The inspector reviewed the maintenance activities and post-maintenance testing associated with troubleshooting and repair of power supply problems in the core protection calculator drawer of Channel “C” of the reactor protection system. This work was performed under work order M2-01-01282 on February 22, 2001. The post-maintenance testing was performed in accordance with SP 2401 MC, “RPS Channel ‘C’ TM/LP Calibration.” The inspector reviewed the test data and verified that the post-modification test was adequate to demonstrate that the affected Channel “C” power supply would perform its required safety function.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Control Element Assembly (CEA) Partial Movement Surveillance

a. Inspection Scope

The inspector observed CEA partial movement testing on March 16, 2001, which was performed in accordance with the SP 2620A, “CEA Partial Movement.” The inspector verified that test results satisfied the applicable acceptance criteria, and that performance of the test adequately demonstrated equipment operability and capability to perform the intended safety function.

b. Findings

No findings of significance were identified.

.2 Emergency Diesel Generator Fast-Start Surveillance Test

a. Inspection Scope

The inspector observed the semi-annual fast-start surveillance test of the “B” EDG on February 14, 2001, which was performed in accordance with the SP 2613B, “Diesel

Generator Operability Tests, Facility 2." The inspector verified that test results satisfied the acceptance criteria of the surveillance procedure and the requirements of Technical Specification 4.8.1.1.2.d, and that performance of the test adequately demonstrated equipment operability and capability to perform the intended safety function.

b. Findings

No findings of significance were identified during this inspection.

2. RADIATION SAFETY

Occupational Radiation Safety [OS]

2OS2 ALARA Planning and Controls

a. Inspection Scope

During the period February 26 through March 2, 2001, the inspector conducted the following activities to evaluate the effectiveness of administrative, engineering, and operational controls to minimize personnel exposure for recent tasks performed during the Unit 3 refueling outage, and while Unit 2 was conducting power operations.

The inspector reviewed pertinent records regarding cumulative personnel exposure, current exposure trends, and ongoing activities in order to assess the licensee's effectiveness in establishing exposure goals and in keeping actual personnel exposure as low as is reasonably achievable (ALARA). Actions taken by the licensee to address radiological challenges caused by a primary system crud burst, while powering down the plant, were reviewed. Included in this review was ALARA Council minutes, Unit 3 outage ALARA Reviews (AR) for reactor disassembly/reassembly (AR 3-01-01), steam generator eddy current inspections (AR 3-01-02), and snubber inspections (AR 3-01-06). The inspector attended daily Radiation Protection Department staff turnover meetings and a Unit 3 outage daily planning meeting.

The inspector evaluated the effectiveness of exposure controls specified in ALARA Reviews for selected jobs completed during the Unit 3 refueling outage. The actual cumulative exposure was compared with the estimated exposure and evaluated using the criteria contained in the relevant NRC's Significance Determination Process. Jobs that were reviewed included steam generator eddy current testing, snubber inspections, and primary system insulation removal/re-installation.

Performance was observed of selected work groups preparing for and conducting Unit 3 reactor cavity drain-down, cavity decontamination, reactor head flange inspection and cleaning, and reactor head set. In evaluating the ALARA controls applied to these tasks, the inspector reviewed the relevant Radiation Work Permits (301, 302, 303) and associated ALARA Review (3-01-01), attended the pre-job ALARA briefing, observed the tasks in progress, and attended the post-job debriefing. The inspector interviewed selected workers to evaluate their knowledge of radiological controls applied to their tasks.

The inspector evaluated the adequacy of Work-In-Progress ALARA Reviews for various dose intensive tasks such as Unit 3 primary system mechanical preventative/corrective maintenance activities and Unit 2 reactor coolant pump oil addition in which the licensee determined that the actual cumulative dose received could exceed original estimates.

Independent radiation surveys were performed in areas of the Unit 3 reactor building, Unit 3 auxiliary building, and Unit 2 auxiliary building to confirm posted survey results and assess the adequacy of radiation work permits and associated controls. Technical Specification locked high radiation areas were verified to be properly secured.

The inspector reviewed the following CRs relating to the control of personnel exposure and work activities involving radioactive materials to determine if the issue was identified in a timely manner and that appropriate actions were taken to evaluate and correct the issue: CR-01-00951, CR-01-01260, CR-01-01262, CR-01-01280, CR-01-01302, CR-01-01339, CR-01-01432, CR-01-01480, CR-01-01651, CR-01-01683, and CR-01-02096

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Physical Protection [PP]

3PP3 Response to Contingency Events

a. Inspection Scope

The following activities were conducted to determine the effectiveness of the licensee's Response to Contingency Events:

On February 13, 2001, performance testing of the intrusion detection system was conducted. This testing was accomplished by touring the entire perimeter and selecting areas of potential vulnerability in the intrusion detection system. As a result of this tour, twelve specific locations were selected for testing. An inspector observed the NRC contractor perform crawl, jump and run testing at these locations. A second inspector was positioned in the alarm station during the tests, to observe audible and visual alarm annunciation, and to evaluate the licensee's camera coverage of the perimeter.

Firearms proficiency was observed on February 14, 2001. The course of fire for stress firing was observed. Four security officers demonstrated their proficiency on this course of fire. In addition, a selected review of fifteen firearms qualification training records was performed.

A review was conducted of the licensee's defensive strategy, response time lines, target sets and relevant implementing procedures. Upon completion of this review, four table top drills were conducted with a security shift supervisor and a response team leader.

The scenario selections, including the adversary entry points and targets, were made by the inspectors for each table top drill.

On February 15, 2001, a review of documentation associated with the licensee's drill and exercise program was conducted. This review included the documentation and critiques for response drills conducted during the four quarters prior to the inspection.

The inspectors also reviewed seven CRs generated and entered into the licensees corrective action program, to address concerns identified during the inspection. The CR's reviewed are identified in the list of documents contained in Attachment 1 of this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

.1 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector selectively examined records used by the licensee to identify occurrences involving locked high radiation areas, very high radiation areas, and unplanned personnel exposures. The information contained in these records was compared against the applicable criteria contained in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 0, to verify that all conditions that met the NEI criteria were recognized, identified, and reported as a Performance Indicator. The records reviewed included CRs and ARs addressing individual and collective exposures.

b. Findings

No findings of significance were identified.

.2 Emergency Preparedness Performance Indicators

a. Inspection Scope

The inspector reviewed the licensee's process for identifying the data that is utilized to determine the values for the three emergency preparedness performance indicators, which are: (1) Drill and Exercise Performance, (2) Emergency Response Organization Participation, and (3) Alert and Notification System Reliability. The review assessed data from the fourth quarter of 1999 through the end of 2000. Classification, notification and protective action opportunities were verified by reviewing selected scenarios. Attendance records for drill and exercise participation were reviewed. Details of the siren testing and data collection were discussed with individuals responsible for that program and test data was reviewed for completeness and accuracy.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

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

b. Findings

The NRC found that the licensee did not implement timely or effective corrective actions regarding recurrent lifts of letdown relief valve 2-CH-354 during periods where two charging pumps were in operation. (Section 1R16).

4OA3 Event Follow-up

The inspector reviewed an event report (Event Number 37871) submitted on March 29, 2001, in accordance with 10 CFR 50.73(a) (1) regarding an invalid specific system actuation of the "A" emergency diesel generator on January 31, 2001. The issues and related findings are described in Section 4OA7.

4OA6 Meetings, including Exit

.1 Physical Security Exit Meeting Summary

The inspectors met with licensee representatives at the conclusion of the inspection on February 16, 2001. At that time, the purpose and scope of the inspection were reviewed, and the preliminary findings were presented. The licensee acknowledged the preliminary inspection findings.

.2 Resident Inspector Exit Meeting

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

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

4OA7 Licensee Identified Violations

The following finding of very low safety significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG 1600, for treatment as a Non-Cited Violation (NCV):

NCV 50-336/2001-002-02 Technical Specification 6.8.1.c. requires that procedures covering surveillance activities be adequately implemented. On January 31, 2001, an operator failed to adequately implement the surveillance procedure addressing a periodic air-roll of the “A” emergency diesel generator (EDG) (OP 2346A, “Emergency Diesel Generators”) in that the operator failed to effectively trip the diesel engine fuel rack prior to the air roll. As a result, the diesel started, control room operators emergency tripped the diesel, and an additional hour of unavailability accrued for the “A” EDG. This condition is in the licensee’s corrective action program as CR-01-00783, and the licensee has identified corrective actions to enhance OP 2346A by adding steps to ensure the effective tripping of the EDG fuel racks.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. List of items Opened, Closed and Discussed

Opened and Closed During this Inspection

05000336/2001-002-01	NCV	Licensee's failure to implement timely and effective corrective actions to prevent relief valve lifts when two charging pumps were placed in operation (1R16)
05000336/2001-002-02	NCV	Inadvertent start of the "A" emergency diesel generator (4OA7)

Discussed

None

b. Partial List of Documents Reviewed

Safeguards Event Reports - 2nd, 3rd, and 4th Quarter, 2000
Millstone Training and Qualifications Plan
Millstone Physical Security Plan
Selected personnel training records
CR-01-01656, Level 2, Develop a training schedule to insure security force stress fire training, for all members of the security force, is completed, Due 03/09/2001
CR-01-01661, Level 2, Evaluate Target Sets for additional detail, Due 03/09/2001
CR-01-01667, Level 2, Evaluate dispatch and response time lines, Due 03/09/2001
CR-01-01716, Level 2, Evaluate weapons requalification failure rate, Due 03/12/2001
CR-01-01664, Level N, Complete design change and installation of upgrades to the Security radio system, Due 04/01/2001
CR-01-01658, Level N, Develop Intrusion detection system testing training curriculum and train individuals, Due 05/17/2001
CR-01-01719, Level N, Evaluate weapons manipulation training, Due 05/18/2001

c. List of Acronyms Used

ADV	atmospheric dump valve
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
AOP	abnormal operating procedure
AR	ALARA reviews
CEA	control element assembly
CRs	condition reports
CVCS	chemical and volume control system
EBFS	enclosure building filtration system
EDG	emergency diesel generator
IX	ion exchanger
LPSI	low pressure safety injection
NEI	Nuclear Energy Institute
RBCCW	reactor building closed cooling water
RCS	reactor coolant system
RPS	reactor protection system
SDP	significant determination process
SITs	safety injection tanks
SP	surveillance procedure
TDAFP	turbine-driven auxiliary feedwater pump

ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No.:	05000423
License No.:	NPF-49
Report No.:	05000423/2001-002
Licensee:	Dominion Nuclear Connecticut, Inc.
Facility:	Millstone Nuclear Power Station, Unit 3
Location:	P. O. Box 128 Waterford, CT 06385
Dates:	February 11, 2001 - March 31, 2001
Inspectors:	A. C. Cerne, Senior Resident Inspector, Unit 3 B. E. Sienel, Resident Inspector, Unit 3 T. F. Burns, Reactor Inspector, Division of Reactor Safety (DRS) P. R. Frechette, Physical Security Inspector, DRS T. A. Moslak, Heath Physicist, DRS D. M. Silk, Sr. Emergency Preparedness Inspector, DRS G. C. Smith, Sr. Physical Security Inspector, DRS
Approved by:	Curtis J. Cowgill, Chief Projects Branch 6 Division of Reactor Projects Region I

SUMMARY OF FINDINGS

IR 05000423/2001-002; on 02/11-03/31/01; Dominion Nuclear Connecticut, Inc., Millstone Nuclear Power Station; Unit 3.

The inspection was conducted by resident and regional inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 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/NRR/OVERSIGHT/index.html.

No findings of significance were identified.

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Report Details

SUMMARY OF UNIT 3 STATUS

The plant began the period on February 11, 2001, in Mode 6 (Refuel) with the seventh refueling outage (3R07) in progress. After reactor refueling, substantial completion of all 3R07 work activities, including the overhaul of the three low pressure turbine stages, and heat-up of the plant to normal operating temperature and pressure, the operators placed the plant in Mode 2 (Startup) on March 21 to perform low power physics testing (LPPT). Following the completion of LPPT on March 22, the operators returned the plant to Mode 3 (Hot Standby) to await restoration of final outage work activities. The reactor was subsequently placed in Mode 2 on March 29. Criticality was achieved at 10:19 am., later that day. With the breaker closure connecting the turbine generator to the grid at 4:29 am., on March 31, 3R07 came to a close. At the end of the report period on March 31, the operators were controlling the plant at approximately 32 percent power with a normal power ascension to 100% in progress.

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

1R04 Equipment Alignment

a. Inspection Scope

The inspector performed a partial equipment alignment walkdown of both trains of the spent fuel pool cooling (SFPC) system while the reactor core was completely offloaded into the spent fuel pool. This period was risk significant in that SFPC was the only credited method of decay heat removal for the spent fuel. The walkdown also included confirmation of reactor plant component cooling water supply to the SFPC heat exchangers. The inspector performed the partial walkdowns by comparing actual equipment alignment to approved licensee piping and instrumentation diagrams to confirm correct system lineup.

Following the completion of maintenance on the service water cooling return line from the “A” train of safety injection pump cooling (CCI), the inspector performed a complete equipment alignment walkdown of the accessible portions of both trains of CCI. The walkdown included service water supply and return piping and was performed by comparing actual equipment alignment to approved piping and instrumentation diagrams, operating procedure lineups, and the licensee’s Final Safety Analysis Report description of the system. The inspector confirmed that identified equipment tags and trouble reports would not affect operability of the system.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspector performed walkdowns of two separate floor elevations in the engineered safety features building (Fire Area ESF-5), housing the turbine driven auxiliary feedwater (TDAFW) pump and related equipment. An adjoining room, within the supplementary leak collection and release system (SLCRS) boundary, containing the control and isolation valves for the steam lines supplying the TDAFW pump, was also inspected. These areas were examined for the design features and fire detection and suppression capabilities described in the Millstone 3 Fire Protection Evaluation Report (FPER).

Given the SLCRS area boundary controls and different design features of the adjacent areas and elevations, the inspector interviewed the cognizant licensee fire protection engineer regarding the controls and equipment in place to respond, in the event of a fire, to the various safe shutdown equipment. The construction, penetrations, and combustible material fire loading within these areas were evaluated with respect to both the FPER descriptions and analyzed consequences of a postulated fire.

The inspector also performed a walkdown of the cable spreading room (Fire Area CB-8). The inspector verified the fire detection and suppression equipment described in the FPER was available with the exception of the automatic carbon dioxide suppression system. Since an inadvertent actuation of the system in January 1999, this system has been locked out physically and has been administratively controlled such that its manual actuation in order to fight a fire is not allowed. As a result of this system unavailability, the licensee has posted a continuous firewatch, as required by the technical requirements manual (TRM). The inspector verified the firewatch was in place and aware of his responsibilities. In addition, the inspector verified the fire suppression equipment available was accurately described in the Unit 3 Fire Fighting Strategy for this area. The available equipment, and the strategy, were revised following an aborted test of the carbon dioxide suppression system in February 2001. Further discussion of this aborted test is documented in Section 1R19 of this inspection report.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

The inspector selected samples of nondestructive examination (NDE) and ASME Section XI code repair/replacement activities for evaluation based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. Also, the inspector evaluated the effectiveness in the resolution and corrective action of problems identified during inservice inspection (ISI) activities. The inspector reviewed a sample of examination data sheets, inspection reports and condition reports initiated as a result of problems identified during ISI examinations (see Partial List of Documents Reviewed in Attachment 1 of this report).

The inspector reviewed three types of NDE activities including volumetric, surface and visual examinations to verify the effectiveness in monitoring degradation of risk significant systems, structures and components and to evaluate these activities for compliance with the requirements of ASME Section XI of the Boiler and Pressure Vessel Code. This review included evaluating the disposition of non-conforming conditions identified and verifying analyses were performed for acceptance and continued operation without repair. The inspector reviewed the ultrasonic (UT) and magnetic particle (MT) test results for the reactor pressure vessel (RPV) weld 101-101 (closure head to flange) and UT and liquid penetrant (LP) test results for weld CHS-31-FW-1 (pipe to valve) in the high pressure safety injection (HPSI) system. In addition, the inspector reviewed the radiographic test (RT) results of the thermal sleeve between welds RCS-20-43-1 and 46-1 (HPSI to RCS Loop “D”).

The inspector reviewed a sample of selected eddy current test data collected from examination of tubing in steam generators “B” and “D”. The data review was performed to evaluate the data analysis practices used and confirm data was analyzed in a consistent manner. Also, the inspector reviewed this activity to assure that an appropriate requirement was established for degradation and sizing of indications and that data evaluation was accomplished using equipment, techniques and personnel qualified for the site in accordance with an established guideline. The inspector also reviewed the activities performed by the independent qualified data analyst representing NNEC to assess the licensee’s level of involvement and oversight of the tube examination effort. The inspector evaluated analysis and calibration techniques specified in the data analysis reference manual to assure tube degradation would not go undetected due to poor data quality (e.g. poor signal to noise ratio, excessive signal variation and other undesirable variations). The inspector reviewed the calibration standards to assure the test method would identify the tube imperfection depth prior to reaching the technical specification plugging limit. The inspector reviewed the indications identified in the tubes of both steam generators and the licensee’s disposition of these indications. The disposition included the plugging of eleven tubes which exceeded the maximum imperfection depth in steam generator “D”. Forty one additional tubes in steam generator “D” which did not exceed the acceptance criteria were evaluated by the licensee and either plugged as a conservative measure or were accepted for continued operation without repair. Regarding steam generator “B”, no tubes were required to be plugged.

The inspector reviewed a sample of video recordings of the remote in-vessel visual inspection (IVVI) of the vessel flange, baffle plate bolting and baffle plate vertical seam welds. Condition report (CR) 01-02330, was initiated to address the speed of the camera and lighting used during this examination. The inspector also reviewed a sample of visual examination reports and condition reports initiated as a result of the visual inspection performed during this outage (3R07) of the containment liner for coating failure, corrosion of liner and damage to moisture barriers for compliance with the requirements of ASME Section XI, IWE (requirements for class MC and metallic liners of class CC components).

The inspector reviewed welding activities associated with the repair and replacement of selected components to verify the activities were performed in accordance with the requirements of ASME Section XI and IX. The inspector reviewed work orders, M3-00-18619, 18620 and 18621, and the completed work documentation for the replacement of charging pump minimum flow line restriction orifices, 3CHS-RO46A, B and C, in the chemical and volume control system. The review included welding at the three affected pumps and two weld repair activities performed at pump “C” using NNEC weld procedure specifications (WPS) 100 Rev. 3 and 101 Rev. 3. The inspector reviewed WPS 100 Rev. 3 and 101 Rev. 3 for compliance with the qualification requirements of ASME Section IX. CR-01-02323 was initiated to address heat input control during the welding of three hundred series stainless steel.

The inspector interviewed the licensee’s radiographic and ultrasonic Level III inspection personnel responsible for the review and approval of test results. Radiographs of welding activities were reviewed to ensure proper identification, characterization and size of rejectable indications for welds W31, 32, 45, 51 and 52. The review included the radiographs of the two repairs (R1 and R2) made during the installation of the orifice plate on charging pump “C”.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed licensee actions taken in response to the problems identified and documented in the following CRs:

- 01-00265 Chemical and volume control system exceeded maintenance rule functional failure criteria
- 01-00403 “B” service water pump discharge check valve failed to close during operational readiness test
- 01-00785 Service water header leak in the supply line to the “B” train safety injection pump cooler and “B” train ESF air conditioning units
- 01-00851 Quench spray system manual valve misalignment

For the first two CRs identified, the inspector reviewed the applicable system’s fourth quarter system health report, corrective actions taken in response to the related equipment problem, maintenance rule functional failure (MRFF) determination, related CRs, and a(1) evaluation, where applicable. The inspector confirmed that the licensee appropriately tracked any MRFF against the system performance criteria.

The inspector reviewed the corrective actions implemented for the service water header leak, including analysis of the system operability until repairs could be effected. This equipment problem was assessed with respect to a similar leak in the opposite service water train, identified earlier in the operating cycle, for which the licensee had requested ASME Code relief and installed a temporary, non-code repair in accordance with the provisions of NRC Generic Letter 90-05. The inspector discussed these degraded conditions with the cognizant system engineer and confirmed appropriate consideration of the maintenance rule criteria and the determination that the leaking piping/structural integrity concerns did not represent MRFFs. The inspector verified that code repairs were implemented during 3R07 to restore both service water piping trains to a fully qualified status.

With respect to CR-01-00851, a discharge valve in each of the redundant trains of the quench spray (QSS) system was found by the licensee to be closed, contrary to the expected system lineup and analyzed operational configuration. The licensee performed a root cause investigation of this problem and reported it to the NRC in accordance with 10 CFR 50.73. System surveillance testing did not disclose the valve misalignment because of adequate system flow measurements through the bypass piping. Thus, while no MRFFs were observed as a result of this condition, further NRC assessment of the safety consequences of this event will be performed with a review of the subject licensee event report, LER 2000-001-00, during a future inspection period.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

During 3R07, the licensee identified minor service water leakage in several brazed joints in the supply and return lines to the cooling units for the motor control center/rod control portions of the reactor plant ventilation (HVR) system in the auxiliary building. A condition report, CR-01-02175, was initiated to document this degraded condition and the licensee performed a technical evaluation that concluded that the HVR system remained operable, despite the minor but detectable leakage. At the time of discovery of this problem, the plant was in an operational mode (Mode 5 - cold shutdown) where HVR system operability was not required by the unit technical specifications.

Subsequently, the licensee decided to replace all the leaking HVR joints with new brazed fittings or welded pipe connections. These repairs were effected during 3R07, prior to taking the unit critical (Mode 2). The inspector examined the repair work in progress and assessed the contingency plans the licensee had established for HVR system operability, if required for room cooling in Modes 3 and 4.

The licensee’s technical evaluation, supporting a position of HVR system operability with the leaking brazed joints, was determined by the NRC to be inconsistent with NRC guidance on ASME Code compliance for such cases. However, because the degraded HVR system conditions and related code interpretation problems were discovered with the unit in Mode 5, no violation of the Unit 3 technical specifications was identified. Likewise, since repairs were effected during the refueling outage, all ASME Code requirements for a Code Class 3 pressure boundary were met and the HVR system was restored to an operable and fully qualified status.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

.1 Response to Low Instrument Air Header Pressure

a. Inspection Scope

During the 3R07 refueling outage, temporary air compressors were used to provide compressed air to the instrument air (IAS) and service air (SAS) systems, with both system headers cross-tied. On several occasions during 3R07, control room operators were required to respond to an IAS low pressure alarm, typically caused by problems with the SAS temporary air compressor.

The inspector observed the operator response to one such incident on the evening shift of February 14, 2001. At that time, the control room operators appropriately suspended core offload activities that were in progress, isolated the IAS and SAS cross-connection to maintain the IAS header pressure, and subsequently fixed the SAS compressor problem.

On February 21, 2001, a more significant loss of IAS occurred, requiring the operators to use Abnormal Operating Procedure (AOP) 3562 to respond to the loss of instrument air. During this incident, one non-safety related valve repositioned as a result of the IAS header pressure degrading to approximately half the nominal value at which it is maintained. The operating crew was again able to restore the system before any operational transient occurred. During this latter event, the inspector responded to the control room and assessed both the operator response and potential for adverse consequences on the safety-related systems in service.

The inspector subsequently discussed the cause of this event, along with the related work control and communications problems, with the Unit 3 Operations Manager and Master Process Owner for Station Operations. Additional corrective measures were implemented by the licensee to prevent similar IAS event recurrence.

b. Findings

No findings of significance were identified.

.2 Reactor Startup and Low Power Physics Testing

a. Inspection Scope

The inspector observed portions of the reactor startups performed during the refueling outage; the first was to perform low power physics testing and the second was at the completion of the outage to begin power operations. These activities included taking the reactor critical using dilution and rod withdrawal, respectively. During these activities the inspector observed operators take the reactor from Mode 3 (Hot Standby) to Mode 2

(Startup). Following the first reactor startup, the inspector attended the brief for and observed portions of low power physics testing. In the course of these plant evolutions, the following procedures were used.

- OP 3202 Reactor Startup
- SP 31008 Low Power Physics Testing (IPTE)

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds

a. Inspection Scope

The inspector reviewed the Unit 3 Operator Work-Around Management Summary, assessing the impact of the required operational actions on the affected system availability and overall operator response capabilities. For three of the open work-arounds on safety-related systems, the inspector discussed the documented corrective actions with a cognizant licensee manager and reviewed the status of engineering actions planned to address the adverse conditions that necessitated the compensating operator activities.

Certain of the work-around items were scheduled for work during the 3R07 refueling outage. The inspector verified that field work was conducted to effect the system repairs, as scheduled. However, in each case, the anticipated results were not achieved resulting in the need for further engineering review and a continuation of the operator work-around activities.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

As documented in Inspection Report 50-423/2000-14, the inspector periodically examined the progress of the installation of the plant design change for the removal of groundwater intrusion into the engineered safety features building sumps. This modification, as detailed in design change record, M3-00-004, involved the replacement of air-driven sump pumps with an electric submersible pump and new groundwater collection and storage tanks. In accordance with licensee commitments to the NRC, based upon past groundwater removal problems, the modified system was required to be available at the time of the unit restart from refueling outage 3R07.

During this inspection period, the completed modification was examined, verifying that a working system was in operation at the end of the outage. A TRM 6.1 specification for containment structural integrity was issued to delineate the groundwater in-leakage limit for which this modification has been qualified and to provide allowed outage times and actions if the new submersible pump becomes inoperable or if in-leakage rate exceeds the assumed limit. The inspector discussed the implementation of these TRM controls with operations personnel and confirmed the appropriate revision of other related Unit 3 procedures.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspector reviewed documentation associated with the following post-maintenance testing (PMT) activities, as controlled by the associated automated work orders (AWOs):

- AWO M3-01-00030, directed performance of special procedure, SPROC ENG-00-3-11, as the PMT for installation of a new “A” charging pump minimum flow line recirculation orifice. The installation was performed under design change, DM3-00-0364-00 and AWO M3-00-18619.

- AWO M3-01-01275, directed overhaul and belzona repair of the “B” service water pump discharge check valve, 3SWP*V003, due to leakage past the seat.

The inspector reviewed the scope of the work activities for AWOs, M3-01-00030 and M3-01-01275, and verified that the PMTs planned and performed were appropriate to restore the operability of the affected systems.

Additionally, the inspector observed conduct of the CO_2 Discharge Test for the Cable Spreading Room CO_2 System on February 19, 2001. This test, classified as an Infrequently Performed Test and Evolution (IPTE), was conducted during back-shift hours with the reactor de-fueled and was controlled by special procedure SPROC

ENG01-3-001, supported by Safety Evaluation (SE), S3-EV-01-0008. The purpose of the test was to verify that CO_2 leakage into areas of the plant requiring operator habitability, particularly the switchgear rooms where the alternate shutdown capability outside the control room is achieved, had been adequately addressed by licensee repairs, maintenance, and testing activities. These corrective actions were required as a result of an event that occurred in January 1999, involving the inadvertent actuation of the cable spreading room CO_2 system. In response to this event, the licensee locked out the CO_2 fire suppression capability to the cable spreading room, utilizing both physical and administrative controls, and instituted compensatory actions as required by the Unit 3 Technical Requirements Manual.

The inspector reviewed the SPROC and SE and witnessed the pre-test briefing in the control room. The test termination criteria were specifically addressed and discussed. Shortly after the CO_2 discharge was initiated, the test was terminated when a cable spreading room door failed open and the licensee lead test personnel determined that some of the test termination criteria had been met.

Because the test was unsuccessful, this system remained locked out when the unit was returned to power following 3R07. The licensee developed a new Fire Fighting Strategy for the cable spreading area, which was further reviewed by the inspector, as discussed in Section 1R05 of this inspection report. Also, as a result of CO_2 migration outside the cable spreading room, which was identified by the licensee after the unsuccessful test, questions arose regarding operation of the control building purge system (CBPS) and whether it may have contributed to the CO_2 concentration levels measured in various areas of the control building. The licensee subsequently decided to also restrict the CBPS usage in the cable spreading room. While the CBPS is still available for use in other areas of the control building, its use is precluded in the cable spreading room. The smoke removal function for this area is now provided by other equipment, as part of the revised Fire Fighting Strategy.

On March 21, 2001, the licensee submitted a letter (B18359) to the NRC responding to a request for the evaluation of the CO_2 discharge test on February 19, 2001. The licensee provided the requested information to address questions regarding the impact of the test results on the Unit 3 licensing and design basis for maintaining the plant safe shutdown capability in the event of a fire and for maintaining control room habitability in accordance with governing regulations.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspector reviewed the following areas related to the 3R07 refueling outage for conformance to technical specification requirements and approved procedures. Selected activities were verified for each evolution.

- Shutdown risk evaluations
- Top nozzle inspections
- Eddy current testing - “B” & “D” steam generators and in-core flux thimbles
- Refueling operations
- Core fuel loading verification
- Control of reactor vessel level while reduced for reactor vessel head replacement
- Reactor trip breaker adjustments
- Actions in response to Westinghouse Nuclear Safety Advisory Letter, NSAL-00-016, Rod Withdrawal from Subcritical Protection in Lower Modes
- Plant Heatup
- Multiple rod drop time testing
- Initial criticality for low power physics testing
- Low power physics testing
- Criticality leading to power ascension
- Power ascension

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Occupational Radiation Safety [OS]

2OS2 ALARA Planning and Controls

Refer to NRC Inspection Report 05000336/2001-002, Section 2OS2 for specific details.

3. SAFEGUARDS

Physical Protection [PP]

3PP3 Response to Contingency Events

Refer to NRC Inspection Report 05000336/2001-002, Section 3PP3 for specific details.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

.1 Heat Removal System (Auxiliary Feedwater) Unavailability

a. Inspection Scope

The purpose of this inspection was to confirm that the information presented in the licensee's December 2000 Safety System Unavailability Performance Indicator (PI) for the auxiliary feedwater system was complete and accurate. The inspector reviewed selected operator logs; system engineer equipment out-of-service records; and licensee Technical Evaluation, M3-EV-00-0029, which addresses the unavailability monitoring requirements and PI calculational controls for the auxiliary feedwater system. Reported plant information was compared against industry guidance provided by NEI 99-02, Regulatory Assessment Performance Indicator Guideline, and was discussed with system engineering personnel.

b. Findings

No findings of significance were identified.

.2 Residual Heat Removal System Unavailability

a. Inspection Scope

The purpose of this inspection was to confirm that the information presented in the licensee's December 2000 Safety System Unavailability PI for the residual heat removal system was complete and accurate. The licensee's plant configuration requires monitoring of both the residual heat removal (RHR) system and the containment recirculation (RSS) system for this PI. The inspector reviewed selected operator logs; system engineer equipment out-of-service records; and licensee Technical Evaluation, M3-EV-00-0029, which addresses the unavailability monitoring requirements and PI calculational controls for the RHR and RSS systems. Reported plant information was compared against industry guidance provided by NEI 99-02, Regulatory Assessment Performance Indicator Guideline, and was discussed with system engineering personnel.

b. Findings

No findings of significance were identified.

.3 Emergency AC Power System Unavailability

a. Inspection Scope

The purpose of this inspection was to confirm that the information presented in the licensee's December 2000 Safety System Unavailability PI for the emergency ac power source (i.e., emergency diesel generator [EDG] system) was complete and accurate. The inspector reviewed selected log data, interviewed the EDG system engineer, and compared the out-of-service time for the EDG system, including supporting components, with the unavailability information reported. The inspector also reviewed the licensee technical evaluation, MP3-EV-00-0071, which addresses the monitoring requirements, methodology for determining unavailability, and PI controls for the EDG system. The licensee's emergency power PI data, compiled as both train and system unavailability percentages, were evaluated against industry guidance provided by NEI 99-02, Regulatory Assessment Performance Indicator Guideline.

b. Findings

No findings of significance were identified.

.4 Occupational Exposure Control Effectiveness

Refer to NRC Inspection Report 05000336/2001-002, Section 4OA1.1 for specific details.

.5 Emergency Preparedness Performance Indicators

Refer to NRC Inspection Report 05000336/2001-002, Section 4OA1.2 for specific details.

4OA6 Meetings, including Exit

.1 Resident Inspector Exit Meeting

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

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

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

a. List of Items Opened, Closed and Discussed

Opened

None

Closed

None

Discussed

50-423/2001-001-00	LER	Quench Spray System Manual Valve Misalignment (1R12)

b. Partial List of Documents Reviewed

Millstone Nuclear Power Station ISI Activities Inspection, March 26-30, 2001
Inspection Procedure 71111.08, Inservice Inspection Activities
Documentation Review

Radiograph Review

M3-1029	Radiographic Inspection Report of “A” Charging Pump Weld 51
M3-1030	Radiographic Inspection Report of “A” Charging Pump Weld 52
M3-1031	Radiographic Inspection Report of “B” Charging Pump Weld 31
M3-1032	Radiographic Inspection Report of “B” Charging Pump Weld 32
M3-1034	Radiographic Inspection Report of “C” Charging Pump Weld 45

NDT Examination Reports

MP3-MT-003	Magnetic Particle Test (Dry) of RPV Closure Head to Flange Weld 101-101
MP3-UT-012	Ultrasonic Test of RPV Closure Head to Flange Weld 101-101
MP3-PT-005	Liquid Penetrant Test of Valve to Pipe Weld Safety Injection CHS-31-FW-1
MP3-UT-088	Ultrasonic Test of Valve to Pipe Weld Safety Injection CHS-31-FW-1
M3-1035	HPSI to RCS Loop D Area between FW-43 and FW-46 Thermal Sleeve
MP3-01-002	IWE Data Sheets for Visual and Thickness Inspection of Containment Liner thru 007

Welding Procedure Specifications

WPS 100	Welding Procedure for Gas Tungsten Arc Welding of P8 to P8 Materials
WPS 101	Welding Procedure for Gas Tungsten Arc/Shielded Metal Arc Welding of P8 to P8

In Vessel Remote Visual Examination

VT-3	Visual Examination (VT-3) of Baffle Plate Bolting and Flange
VT-3	Visual Examination (VT-3) of Baffle Plate Vertical Seam Welds

Examination Procedures

NU-VE-4,R2	Visual Examination of Unit 3 Reactor Vessel Interior
IWE/IWL	Visual Examination ASME IWE/IWL of Containment Liner

Unresolved Indication Report

MP3-042	Linear Indication at RPV Closure Head to Flange Weld 101-101

Repair/Replacement Work Orders

M3-00-18619	Replace Existing Charging Pump Min Flow Restricting Orifice 3CHS*RO46A
M3-00-18620	Replace Existing Charging Pump Min Flow Restricting Orifice 3CHS*RO46B
M3-00-18621	Replace Existing Charging Pump Min Flow Restricting Orifice 3CHS*RO46C (Includes two repairs)

Condition Reports

CR-01-02330	Reactor Vessel Internal Visual Examination
CR-01-02323	Heat Input Control in Weld Procedures 100 and 101
CR-01-02222	AWO M3-00-10419 Pipe Replacement Deferred without Documentation

Steam Generator Documentation

U3-24-SIP-REF01 Rev.000	Unit 3 Steam Generator Eddy Current Data Analysis Reference Manual - Refueling Outage Seven
MP-24-SIP-GDL01 Rev.000	Steam Generator Tube Examinations Independent Qualified Data Analyst Guidelines
Tube Plugging Report	Steam Generator Tube Plugging (Draft) Report (15 Day Report- 3/1/01)
SGD-Plug-1	Millstone Unit 3 Plugging Approval Letter S/G “D”
IQDA Report	Independent Qualified Data Analyst Report
ECT Status Report	Eddy Current Examination Status Report (2/17/01)

c. List of Acronyms Used

3R07	Unit 3 refueling outage number seven
AOP	abnormal operating procedure
AWOs	automated work orders
CBPS	control building purge system
CCI	safety injection pump cooling
CR	condition report
EDG	emergency diesel generator
ESF	engineered safety features
FPER	Fire Protection Evaluation Report
HPSI	high pressure safety injection
HVR	plant ventilation system
IAS	instrument air
IPTE	infrequently performed test and evolution
ISI	inservice inspection
IVVI	in-vessel visual inspection
LER	licensee event report
LP	liquid penetrant
LPPT	low power physics testing
MRFF	maintenance rule functional failure
MT	magnetic particle test
NDE	nondestructive examination
NRC	Nuclear Regulatory Commission
PI	performance indicator
PMT	post-maintenance testing
QSS	quench spray
RHR	residual heat removal
RPV	reactor pressure vessel
RSS	containment recirculation
RT	radiographic test
SAS	service air
SE	safety evaluation
SFPC	spent fuel pool cooling
SLCRS	supplementary leak collection and release system
SPROC	special procedure
TDAFW	turbine driven auxiliary feedwater pump
TRM	technical requirements manual
UT	ultrasonic test
WPS	weld procedure specifications