U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report Nos.: 50-245/89-17; 50-336/89-17

Docket Nos.: 50-245; 50-336

License Nos.: DPR-21; DPR-65

Licensee: Northeast Nuclear Energy Company
P.O. Box 270
Hartford, CT 06101-0270

Facility Name: Millstone Nuclear Power Station, Units 1&2

Location: Waterford, Connecticut

Dates: Unit 1, 7/18/89-9/5/89; Unit 2, 7/27/89-9/5/89

Reporting Inspector: P. J. Habighorst, Resident Inspector

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L. M. Kolonauski, Reactor Engineer, Division of Reactor Projects

Approved by: Donald R. Haverkamp 10/18/89
Donald R. Haverkamp, Chief Date
Reactor Projects Section 4A
Division of Reactor Projects

Inspection Summary: Combined Inspection Report Nos. 50-245/89-17 and 50-336/89-17.

Areas Inspected: Routine NRC resident and specialist inspection at Millstone 1 (106 regular hours, 9 backshift hours, and 5.5 deep backshift hours), at Millstone 2 (142 regular hours, 21.5 backshift hours, and 4.5 deep backshift), of plant operations, surveillance, maintenance, previously identified items, committee activities, evaluation of licensee self-assessment, and Plant Incident Reports (PIRs).

Results:

1. General Conclusions on Adequacy, Strength, or Weakness in Licensee Programs

   Steam generator tube plug engineering reviews, corrective actions, and planning for repairs were thorough and comprehensive, reflecting a licensee program strength in the area of engineering and technical support.

2. Violations

   One licensee identified violation was reported involving failure to evaluate conditions adverse to safety with respect to two hydraulic control units. No notice of violation was issued. (See section 5.8)

3. Unresolved Items

   Twenty-one open environmental qualification items were closed: ten at Unit 1 and eleven at Unit 2. (Section 3.)

   At Unit 1, an unresolved item concerning seismic verification and operability of hydraulic control units was opened. (Section 5.8)

   At Unit 2, one unresolved item was identified regarding implementation of previous commitments to NRC Bulletin 83-03, Check Valve Failures in Raw Water Cooling Systems of Diesel Generators. (Section 5.5)

   A second unresolved item was identified regarding lack of design specifications, and the implementation and control of design for the emergency diesel generator saturable transformers. (Section 8.0)

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*The NRC Inspection Manual inspection procedure (IP) that was used as inspection guidance is listed for each applicable report section.

DETAILS

1.0 Persons Contacted

Inspection findings were discussed periodically with the supervisory and management personnel identified below.

H. Haynes, Station Services Superintendent
J. Stetz, Unit 1 Superintendent
J. Keenan, Unit 2 Superintendent
N. Bergh, Unit 1 Maintenance Supervisor
J. Riley, Unit 2 Maintenance Supervisor
W. Vogel, Unit 1 Engineering Supervisor
F. Dacimo, Unit 2 Engineering Supervisor
R. Prezkop, Unit 1 Instrumentation and Controls Supervisor
J. Becker, Unit 2 Instrumentation and Controls Supervisor
R. Palmieri, Unit 1 Operations Supervisor
J. Smith, Unit 2 Operations Supervisor
M. Brennan, Unit 1 Health Physics Supervisor

The inspector also contacted other members of the Operations, Radiation Protection, Chemistry, Instrumentation and Controls, Maintenance, Reactor Engineering, and Security Departments.

2.0 Summary of Facility Activities

Millstone 1

Millstone 1 operated at full power except for normal power reductions for routine surveillances and main condenser backwashing. On August 8, the licensee commenced a power reduction to 25% of rated power to investigate a low oil level alarm on the "A" recirculation pump motor. Oil was added to the upper and lower reservoirs. Motor bearing temperatures and vibration instrumentation in the control room were consistent with normal levels during the alarmed condition. Full power operation was restored on August 10.

Millstone 2

The unit maintained power operation during the inspection period. Two power reductions were conducted on July 31 and August 25, respectively. On July 31, the licensee downpowered to 80% of rated power to replace the "C" condensate pump seal. During power restoration, control element assembly (CEA) No. 68 reed switch position indication failed and resulted in a technical specification required downpower to less than 70%. The CEA was restored to the "full out" position in compliance with TS, and full power operation resumed on August 1. The licensee downpowered to 90% of rated power to correct water intrusion into the upper motor bearing of the "B" heater drain pump on August 25. Full power operation was restored on the same day and continued until the end of the period.

3.0 Previously Identified Items

3.1 (Closed) NC4 50-336/88-20-01: Qualification of Conax Seal (ECSA) Buttsplice Associated with Rosemount Transmitters

Qualification of the buttsplice was not established prior to the November 30, 1985 completion date specified in 10 CFR 50.49 (g).

Documentation was provided prior to the end of the August 16-19, 1988 NRC inspection to support qualification of the Conax Seal buttsplice. Qualification was based on Tennessee Valley Authority Test Report No. 86-1995, dated December 12, 1986 and Toledo Edison Test Report No. 22414-87N, dated August 27, 1986.

This item is closed.

3.2 (Closed) NC4 50-336/88-20-02: Qualification of Limitorque Motor Valve Terminations Using Bishop Tape Splices

Qualification of the Bishop tape splice was not established prior to the November 30, 1985 completion date specified in 10 CFR 50.49 (g).

Qualification was demonstrated to the NRC inspector during the August 16-19, 1988 inspection through Wyle Test Report No. 17436-17, Final Report on Thermal and Radiation Analysis for Bishop W-962 and W-963 EPR Electrical Insulation Tape for Millstone 1, dated November 26, 1980.

This item is closed.

3.3 (Closed) NC4 50-336/88-20-03: Qualification of the Auxiliary Feed Pump Motors and DC Switchgear Room Fan Motor Terminations Using Bishop Tape Splices

Qualification of the Bishop tape Splice for this application was not established prior to the November 30, 1985 completion date specified in 10 CFR 50.49 (g).

Qualification was demonstrated to the NRC inspector during the August 16-19, 1988 inspection through Wyle Test Report No. 17436-17 dated November 26, 1980.

This item is closed.

3.4 (Closed) NC4 50-336/88-20-04: Enclosure Building Fan Motor Terminations Using 3M Scotch Tape Splices

Qualification of the 3M Scotch tape splices was not established prior to the November 30, 1985 completion date specified in 10 CFR 50.49(g).

Qualification was demonstrated to the NRC inspector during the August 16-19, 1988 inspection through a Franklin Research Center Test Report No. F-C5022-2, Qualification Tests of Terminal Blocks and Splices - Insulation Assemblies in Simulated Loss-of Coolant-Accident Environment Phase B, dated November 1978.

This item is closed.

3.5 (Closed) NC4 50-336/88-20-05: Qualification of Solenoid Operated Valves (SOVs) in the Main Steam Isolation Valve (MSIV) Rooms

Qualification was not established prior to the November 30, 1985 completion date specified in 10 CFR 50.49(g).

Qualification of the SOVs was demonstrated to the NRC inspector during the August 16-19, 1988 inspection through a similarity analysis of the tested ASCO NP Series and the WPHT 8300 series. Based on the additional qualification documents and analysis, the inspector concluded that the SOVs are qualified.

This item is closed.

3.6 (Closed) NC4 50-245/88-14-01: Qualification of the Crane Teledyne Motor Operated Valves (MOVs) in a Radiation Harsh Environment

Qualification of the MOVs located outside the drywell, was not established prior to the November 30, 1985 completion date specified in 10 CFR 50.49(g).

Qualification was demonstrated to the NRC inspector during the August 16-19, 1988 inspection through an analysis performed by the licensee establishing qualification of the MOVs in a radiation only harsh environment.

This item is closed.

3.7 (Closed) NC4 50-245/88-14-02: Qualification of Components in the Standby Gas Treatment System (SGTS)

Qualification of SGTS components; PE/PVC cables, GE-EB 5 terminal blocks, Honeywell microswitches and 3M "scotch lock" wire nuts, was not established prior to the November 30, 1985 completion date specified in 10 CFR 50.49(g). The licensee issued PDCR No. M1-11-88 on April 15, 1988 to replace the unqualified SGTS components and cable with qualified materials. Implementation of the PDCR was completed during the 1989 refueling outage. Walkdown inspections to verify completion were performed in May 19, 1989 and documented on an EEQ checklist. The SGTS system was declared operable and is back in service.

This item is closed.

3.8 (Closed) NC4 50-245/88-14-03: Qualification of Electrical Terminations Using t AMP In-Line Buttsplice

Qualification of the AMP in-line buttsplice was not established prior to the November 30, 1985 completion date specified in 10 CFR 50.49(g).

Qualification of the AMP in-line buttsplice was demonstrated to the NRC inspector during the August 16-19, 1988 inspection through Wyle Test Report No. 17655-TER-5 dated August 13, 1985.

This item is closed.

3.9 (Closed) IFI 50-245/85-30-02 and 50-336/85-35-02: The Lack of Procedural Control for Revisions/Changes to Equipment Qualification Documents

The licensee has revised applicable sections of procedure GE-EQ-01, revision 2 (NUSCO Branch Instructions) to incorporate requirements for controlling unauthorized revisions/changes without signature and date of person making the changes.

This item is closed.

3.10 (Closed) IFI 50-245/85-30-03: Qualification of Unidentified Terminal Block in Limitorque Valve Operator No. MS-5

The licensee performed an inspection of the Limitorque operator in question on December 3, 1985 and determined that the correct identification of the unit was LP10A and not MS-5. The Limitorque unit is installed outside the drywell and is exposed to a radiation-only harsh environment with brief excursions of temperatures far below the qualified temperature of 312 degrees F. The terminal block in the Limitorque was identified as a Curtis type "L" block and qualified to 312 degrees F in Limitorque Test Report No. B-0119.

This item is closed.

3.11 (Closed) IFI 50-245/85-30-04 and 50-336/85-35-03: Qualification Criteria Specified for Procurement of Replacement 10 CFR 50.49 Components

NUSCO procedure NE&O 6.02, revision 3, Preparation Review of Quality Purchase Requisitions, dated November 25, 1988 has bei evised to incorporate specific requirements for EQ reviews of procurement and compliance with the requirements of Procedure No. NE&O 2.21, revision 1. Part II.F(1) of the NE&O procedure requires specification upgrade to current qualification criteria.

This item is closed.

3.12 <u>(Closed) IFI 50-245/85-30-05 and 50-336/85-35-04: Plant Review of all SCEW Sheets and Issuance of all EQ Related Maintenance Procedures</u>

The licensee has replaced all SCEW sheets for Millstone Units 1&2 with Component Replacement Schedule sheets for each item on the EQ Master List. The Component Replacement Schedule sheets were issued on February 3, 1986 via document transmittal letter No. GEE-DT-004 & 005. All EQ maintenance procedures were completed and issued in August 1986.

This item is closed.

3.13 <u>(Closed) IFI 50-245/85-30-07: The SCEW Sheet Instruction Directing the Licensee to Add a 1/4 Inch Weep Hole at the Conduit System Low Point for Valves 1-MS-1A, B, C, and D</u>

The licensee completed this task on November 19, 1985 with quality control inspection and approval on November 30, 1985. The licensee indicated that the existing junction box installation, with drain holes, was adequate to support qualification based on ASCC Test Report; No. AQR-6738 revision 1.

This item is closed.

3.14 <u>(Closed) IFI 50-245/85-30-08: The Adequacy of the Motor Maintenance Requirements for Unit 1 Low Pressure Coolant Injection (LPSI) Pumps</u>

The licensee has discontinued use of SCEW sheets and replaced them with component Replacement Schedule sheets (reference SCEW VI-6, EEQ File No. MP1-144) which incorporate the manufacturer's required maintenance instructions to preserve qualification.

This item is closed.

3.15 <u>(Closed) NC4 50-336/87-15-02: Bishop Tape (Qualified to DOR Guidelines) as a Qualified Replacement for NUREG-0588 Qualified Terminations of the Power Field Cables Connected to the High Pressure Safety Injection (HPSI) Pump Motor</u>

Qualification data to support qualification of the Bishop tape to NUREG-0588 requirements was not available at the time of the July 15-17, 1987 inspection. Subsequent to the inspection, the licensee provided "sound reason to the contrary" as required in 10 CFR 50.49(I) to justify use of the Bishop tape in this application. While only qualified to DOR guidelines, the licensee was able to demonstrate that the radiation only environment in which the tape was used is less severe than that for which the tape is qualified.

This item is closed.

3.16 (Closed) NC4 50-336/87-15-03: Qualification Status of Ideal Model 74B Wire Nuts

The licensee was not able to provide documentation to support qualification of the Ideal wire nuts during the July 15-17, 1987 inspection. Justification was provided for interim operation in a letter to the NRC dated December 19, 1986. The wire nuts have been replaced with qualified Raychem splices during the 1986 outage.

This item is closed.

3.17 (Closed) NC4 50-336/87-15-04: Qualification Status of Tape Splices for Power Cable Terminations to Motors

The licensee was unable to provide documentation to support qualification of tape splices for motor nos. P-43A and P-43B prior to the November 30, 1985 completion date specified in 10 CFR 50.49(g).

The licensee has subsequently added the motors to the EQ Master List and has replaced the tape splice with qualified Raychem splices. Full compliance was achieved on September 20, 1986.

This item is closed.

3.18 (Closed) NC4 50-245/87-17-01: Qualification of Curtis Terminal Blocks in Limitorque Motor Operated Valves

The licensee was not able to provide documentation to support qualification of the Curtis type "L" terminal block for use in containment prior to the November 30, 1985 completion date specified in 10 CFR 50.49(g).

The licensee has replaced the Curtis type "L" terminal block in Limitorque operators with qualified Raychem splices. Work was authorized on July 17, 1987 on work request No. WO M1-87-08389. Work was completed on August 8, 1987.

This item is closed.

3.19 (Closed) BU 85-00-03: NRC Bulletin 85-03, Motor-Operated Valve Common Mode Failure Due to Improper Switch Settings - Millstone 1 and 2

The NRC issued Bulletin 85-03 and Supplement 1 to Bulletin 85-03 on November 15, 1985 and April 27, 1986, respectively. The bulletin requested that licensees develop and implement a program to ensure that switch settings on certain safety-related motor-operated valves are selected, set, and maintained correctly to accommodate maximum differential pressures during normal and abnormal events within the design basis. Supplement 1 to NRC Bulletin 85-03 clarifies which motor-operated valves are to be included in the program and defines the phrase "inadvertent equipment operation".

Past NRC inspection of this bulletin and supplement for both Millstone 1 and 2 was documented in inspection reports 50-245/86-17, 50-336/86-19, 50-336/88-06, 50-336/88-07, and 50-336/88-16.

On June 28, 1989, the NRC issued Generic Letter (GL) 89-10, Safety-Related Motor Operated Valve Testing and Surveillance. GL 89 10 supersedes NRC Bulletin 85-03 and its supplement. Licensee documented results of tests or other surveillances that were used to satisfy the recommended actions of Bulletin 85-03 or the supplement may be used to the extent practical for response to GL 89-10.

Based on past review of Bulletin 85-03 and its supplement, and issuance of GL 89-10, the NRC considers this item closed. Future routine inspection followup of licensee response and programs for GL 89-10 will be implemented.

3.20 (Closed) UNR 50-245/88-17-04: Low Pressure Coolant Injection (LPCI) Minimum Flow Valve Found Closed

While performing a routine tour of the Millstone 1 control room on August 29, 1988, the inspector noted a closed indication for the "A" LPCI minimum flow valve (1-LP-26A). The normal standby position for 1-LP-26A is open. The inspector reported the finding to the operators, who corrected the position of the valves.

Further inspector and licensee review did not determine the specific reason for the mispositioning of the valve. While this incident constitutes a violation of the LPCI valve lineup procedure and was identified by the inspector rather than the licensee, no enforcement action is being taken at this time because: (1) mispositioned valves are not a recurring problem at Millstone 1; (2) the licensee took prompt corrective action; and (3) the event was not required to be reported to the NRC. The incident had low safety significance; if LPCI had automatically initiated, the operator could have opened the valve by using the control room switch before pump damage occurred. This item is closed.

3.21 (Open) UNR 245/87-18-01: Reverse Direction Test, 10 CFR 50 Appendix J, Type C

Inspection Report 50-245/87-18 documented a concern regarding the method by which certain containment isolation valves were type "C" leak tested. Although the valves (primarily atmosphere control system valves) were leak tested in the reverse direction, the valves' packing was not exposed to test pressure. Since test results did not include the valve packing as a potential release path, they were potentially less conservative than direct leak testing of the valves. 10 CFR 50, Appendix J allows reverse direction testing of containment isolation valves provided that such testing would not result in less conservative results. This concern was left as an unresolved item pending the final disposition of this issue or the submittal of an exemption by the licensee.

The licensee reviewed all valves which were tested in the reverse direction and evaluated the conservatism of the test. The results of the review were submitted to the NRC in May 1988. Additionally, an exemption request for the affected atmospheric control system valves was submitted to the NRC for approval. The licensee concluded that the reverse direction testing of some atmospheric control system valves (valves AC-3A, AC-3B, AC-5, AC-6, AC-7, AC-9, AC-11, and AC-12) was less conservative than direct testing because some containment pressure boundaries, including actuator shaft seal, body-to-bonnet joints and double O-ring flange seals, were not tested. Since reverse direction testing does not test these boundaries and the system configuration does not allow direct testing of these valves, the licensee has proposed in its exemption request to reverse the valves. Reversing direction of the valves and use of testable double O-ring seals would adequately test all the software of the valves. The licensee recognized that although the valves' software would be tested, the reversal of these valves would result in non-conservatism in the testing of the seat leakage. The basis for accepting the non-conservative leak testing of the seats is the historical data that concludes that the valve seats are equally leak-tight in either direction.

The inspector concluded that the basis for the proposed reversal of the valves is weak. The conservative testing of valve seat leakage should not be eliminated. Additionally, reversing the valves would result in containment pressure tending to unseat the valves under postulated accident conditions. This item remains unresolved pending the final disposition of the exemption request.

4.0 Facility Tours

The inspector observed plant operations during regular and backshift tours of the following areas:

Control Room	
Vital Switchgear Room	Diesel Generator Rooms
Turbine Building	Intake Structures
Enclosure Building	ESF Cubicles

Control room instruments were observed for correlation between channels, proper functioning, and conformance with Technical Specifications. Alarm conditions in effect and alarms received in the control room were discussed with operators. The inspector periodically reviewed the night order log, tagout log, plant incident report (PIR) log, key log, and bypass jumper log. Each of the respective logs was discussed with operations department staff. No inadequacies were noted.

During an inspection tour on July 17, the inspector verified proper completion of equipment tagging activities at Millstone 1 under switching and Tagging order 1597-89-1. This review was completed to verify that the diesel generator and service water systems were properly restored to service following maintenance. No inadequacies were identified.

During an inspection tour on August 28, the inspector verified proper completion of equipment tagging activities at Millstone 2 under tagging orders 2-2414-89, 2-2437-89, 2-1899-89, and 2-1700-89. The reviews included corrective maintenance activities on the 'B' reactor building component cooling water pump, and vital AC chillers. No inadequacies were noted.

During plant tours, logs and records were reviewed to ensure compliance with station procedures, to determine if entries were correctly made, and to verify correct communication and equipment status. No inadequacies were noted.

5.0 Plant Operational Status Reviews

5.1 Review of Plant Incident Reports (PIRs) - Units 1 and 2

The plant incident reports (PIRs) listed below were reviewed during the inspection period to (i) determine the significance of the events; (ii) review the licensee's evaluation of the events; (iii) verify that licensee response and corrective actions were proper; and, (iv) verify that the licensee reported the events in accordance with applicable requirements. The Unit 1 PIRs reviewed were numbers 89-57, 89-60, 89-61, 89-62, and 89-64. PIR 89-62 involved a problem with the gas turbine generator control system. Review of this matter is described in section 10.1 as followup of AWOs 89-8459 and 89-9071. The Unit 2 PIRs reviewed were numbers 89-87, 89-88, 89-89, and 89-90. PIR's 89-87 and 89-88 involved problems with the 'A' emergency diesel generator. Review of this matter is described in section 8.0.

No inadequacies were identified.

5.2 NRC Bulletin 89-01, Failure of Westinghouse Steam Generator Tube Mechanical Plugs - Unit 2

Numerous plants have experienced primary water stress corrosion cracking (PWSCC) and leaks of Westinghouse steam generator tube mechanical plugs. On February 25, 1989, North Anna Unit 1 experienced a mechanical plug failure following a reactor trip during a feedwater isolation transient. The plug failure caused a 75 gallon per minute (gpm) primary to secondary leak rate. The failure mechanism involved a full circumferential severance of the top portion of the plug from the body of the plug. The top portion of the plug was propelled up the length of the tube by primary system pressure to a point just above the u-bend transient point where it impacted and punctured the outer curvature of the tube. The top portion of the plug subsequently impacted and dented an adjacent tube.

The plug manufacturer determined that under critical conditions of stress and temperature, and for certain metallurgical characteristics, the material used for the steam generator tube plugs can be susceptible to PWSCC. In approximately 75% of the cases involving PWSCC, axial cracks in the plugs were observed. If these axial cracks propagate throughwall they normally allow a small leak path into the plugged tube. If the tube was plugged as the result of a leak the normal methods of leak detection will reveal the problem. If the tube did not previously leak, the tube will fill with water and the pressure equalize with no adverse effects. In the remaining 25% of the cases, circumferential cracking has been found in the tube plugs. This was the case at North Anna.

For the plug top to release as at North Anna, several conditions must be satisfied:

-- The PWSCC must progress to the point where the remaining ligament cannot maintain the integrity of the plug.

-- The plug lands above the expander must be leak tight to prevent the tube from filling with water.

-- The tube must not be filled with water.

-- During original installation, the tube must have been expanded for the full thickness of the tubesheet.

If any of these conditions are not met, the plug top cannot release into the tube.

The plug manufacturer has determined that three heats of material used to fabricate plugs have the metallurgical characteristics required to make these plugs susceptible to PWSCC. The plug manufacturer developed an algorithm based on the operating pressures, temperatures, and metallurgical characteristics of the plug material to determine the expected life of the plugs installed in steam generators. Based on this algorithm, the licensee determined that the expected life of plugs in the cold side of the Millstone Unit 2 steam generators was sufficiently long that no immediate corrective action was required. However, the licensee determined that many plugs on the hot leg side required corrective action. To repair these hot leg plugs, the licensee elected to install a custom designed cap screw in the threaded expander of the existing plugs and tack weld this cap screw in place. At the conclusion of the repairs, 446 hot leg plugs had received the plug-in-plug (PIP) repair. Five plugs were removed for evaluation and replaced with metallurgically acceptable material.

After completion of the refueling outage and returning to power, Westinghouse informed the licensee that an additional heat of plug material was found cracked due to PWSCC at two other utilities. The licensee was also informed that 50 plugs manufactured with this material were installed in the hot legs of the Millstone Unit 2 steam generators. However, of these 50 plugs, 27 were installed in tubes that had sleeves and the other 23 were installed in stabilized tubes. Based on this information, the licensee prepared a Justification for Continued Operation (JCO) that requested operation with the 27 sleeve plugs in place until the next refueling outage in October 1990. The bases for this request are as follows:

-- The 27 plugs are in sleeves that are expanded only one or two inches above the plug. Therefore a severed plug top would be captured by the unexpanded portion of the sleeve.

   This installation is considered similar to tubes partially rolled into the tubesheet consistent with NRC Bulletin 89-01, paragraph 3, Actions Requested, which states: "Remedial actions at plants where the steam generator tubes are partially depth expanded within the tubesheet may be deferred on a one-time basis to the next scheduled refueling outage if the outage that immediately follows receipt of this bulletin ends before October 1, 1989."

-- Millstone Unit 2 is analyzed for a primary to secondary leak in excess of that postulated to occur due to a North Anna type tube plug failure.

The licensee has also prepared a JCO until December 1, 1989 for the standard 3/4 inch diameter steam generator tubes containing mechanical plugs and which have integrally attached stabilizers (stainless steel cables and collars). Using the plug manufacturer's algorithm, the expected life of these plugs was calculated to be 293 effective full power days from the end of cycle nine.

The licensee was requested to verify that information contained in references 1 and 2 of Bulletin 89-01 relating specifically to their plants is correct for plugs supplied from heat numbers 3279, 3513, 3962, and 4523. The specific information to be verified is the number of Westinghouse mechanical plugs installed in the hot and cold legs broken down by steam generator number, heat number, and date of installation. If information from these references is incorrect, appropriate corrections should be identified. The licensees also were requested to state if their plants have not installed Westinghouse mechanical plugs from the subject heats.

No errors were found regarding this information; however, other data contained in References 1 and 2 of Bulletin 89-01, such as temperature scaling factors, plug type, (i.e., stabilizer plug, sleeve plug, standard plug) and hot leg temperatures were found to be in error. These discrepancies were identified to Westinghouse and will be included in a revision to WCAP 12244 "Steam Generator Tube Plug Integrity Summary Report." The correct information was provided in Attachment 1 to the licensee's response letter to NRC Bulletin 89-01.

The bulletin requested the licensee to take the following six actions, to be implemented initially during any refueling outage or extended outage (greater than four weeks) which ends 30 days or more following receipt of this bulletin and during all future refueling outages:

(1) Steam generator tube plug lifetime for plugs from heats 3279, 3513, 3962, and 4523 should be estimated, and appropriate remedial actions (i.e., repair and/or replacement) should be implemented for all plugs whose estimated lifetimes do not extend to the next refueling outage.

    Attachment 1 (to the licensee's letter) identifies a total of 298 tube plugs from Millstone Unit 2 which will exhaust their predicted lifetimes before the end of the present fuel cycle (Cycle 10). Of this total, 243 were repaired by the PIP method during the 1989 refueling outage and 5 plugs were replaced. The remaining 50 plugs are from heat 4523 which was identified as being susceptible after Millstone Unit 2 had returned to power. Of those, 27 are insleeved tubes which are partially depth expanded as described in NRC Bulletin 89-01. The remaining 23 plugs are in tubes which have stabilizers installed. The JCO for these plugs is discussed above.

(2) For plug repairs or replacement, the licensee was reminded that their responsibilities under As-Low-As-Reasonably-Achieveable (ALARA) criteria require analysis of the various plug repair or replacement methods available to determine which method will result in the lowest overall personnel radiation exposure while still remaining cost effective.

    The licensee stated that it intends to address ALARA concerns.

(3) The licensee was requested to discontinue installation of Westinghouse mechanical plugs from heats 3279, 3513, 3962, and 4523. The licensee stated that the use of Westinghouse plugs made from these four heats had been discontinued.

(4) Regardless of heat number, Westinghouse mechanical plugs removed from steam generators should be examined for PWSCC on a sample basis for each heat number. The licensee stated that they plan this examination of removed plugs.

(5) Remedial actions at plants where the steam generator tubes are partially depth expanded within the tube sheet may be deferred on a one time basis to the next scheduled refueling outage if the outage that immediately follows receipt of this bulletin ends before October 1, 1989.

The licensee did not directly respond to this request since the tubes in the Millstone steam generators have been expanded through the full thickness of the tube sheet. However, the licensee has submitted a JCO to justify operation with plugs from heat 4523 installed in partially expanded sleeves.

(6) Remedial actions for "sentinel related" mechanical plugs may be deferred. The licensee stated that no "sentinel related" plugs have been installed in Millstone Unit 2.

On August 29, the licensee prepared a revised JCO superseding the NRC Bulletin 89-01 response. In addition to the previous information, the revised JCO includes an evaluation of SG plug heat lot 5222. The heat lot is considered suspect to PWSCC; however, the industry and the plug vendor have not experienced cracking of plugs from this heat lot or reached the same conclusion. The susceptibility of this material was based on licensee evaluation of material heat microstructure and corrosion test data provided in WCAP-12244 "SG Tube Plug Integrity Summary Report." The inspector reviewed the material microstructure and corrosion test results and concerns of the licensee that the material conditions behaved similar to a suspect heat lot NX 4523.

The licensee identified two NX 5222 mechanical SG plugs which will exhaust predicted lifetime before the end of cycle 10 operation. The two plugs are located in the SG hot legs with Combustion Engineering (CE) stabilizers. The CE stabilizer differs from the previously mentioned Westinghouse stabilizers in that they are not integrally attached to the plug and are constructed of different material and design. The licensee contracted mock-up testing of the SG plug CE stabilizer configuration. The results of the testing did not produce penetration of the SG tube wall.

The duration of the revised JCO is December 1, 1989. The overall licensee justification for continued operation for all suspect SG tube plugs not previously repaired is:

-- Industry operating experience indicated that 75% of the potentially cracked plugs are expected to leak, thereby equalizing the pressure differential across the plug preventing a high energy plug release.

-- Probabilistic evaluation indicates that continued operation results in a small increase in core melt probability (3 X 10 -6 or .1% of the licensees corporate goal for the remaining licensed lifetime).

-- Based on an administratively controlled limit below 0.2 micro-curies per gram reactor coolant activity, a tube rupture would result in radioactive release below 10 CFR 100 requirements.

-- Mock-up testing configuration does not show that stabilizers will rupture the SG tube or additional tubes.

On August 30, the licensee documented planned acceptance criteria and scope for SG eddy current testing for the mid-cycle outage scheduled to begin on October 21, 1989. The inspection scope documents repairs to SG plug heat lots NX-4523 and NX-5222 with a plug-in-plug design.

In conclusion, the inspector will continue to follow licensee commitments and controls for repair of susceptible SG plugs during the scheduled mid-cycle outage.

5.3 Bypass Jumper (BJ) Review - Unit 2

A detailed review of two bypass jumpers was conducted. The two bypass jumpers reviewed were:

-- 2-89-61 Facility 1 Vital Switchgear Room Cooling
-- 2-89-64 Control Element Assembly (CEA) #68

The inspection elements consisted of licensee adherence to administrative control procedure ACP-QA-2.06B, Station Bypass Jumper Control; unreviewed safety question (10 CFR 50.59) determination, effects on design and accident analysis commitments in the Final Safety Analysis Report (FSAR), and licensee piping and instrument drawings (P&IDs).

Bypass jumper 2-89-61 disabled the low pressure signal for service water flow to vital A/C switchgear cooler X-182. In addition, the inlet service water bypass valve 2-SW-180B was opened to increase flow by 5-10 gallons per minute (GPM). The purpose of BJ 2-89-61 is to remove control room annunciator C06-5C, and to provide additional cooling to cooler X-182. Annunciator C06-5C provides indication to CR operators of low discharge pressure indicative of a service-water-to-switchgear coolers line break. FSAR section 9.9.15.2.1 takes credit for the low pressure signal to isolate the inlet trip valve (2-SW-178B) to prevent flooding of the switchgear room. Installation of BJ 2-89-61 removes the designed automatic feature. The licensee's safety evaluation addresses the consequence of a service water pipe break, and actions and impact to replace the automatic trip function. The inspector reviewed the licensee's actions and basis for disabling the automatic trip feature and found them to be acceptable. Acceptability was based on review of the licensee's flooding potential evaluation and design of the fire suppression system. The inspector also reviewed the licensee's 10 CFR 50.59 evaluation, adherence to procedure ACP-QA-2.06B, safety and technical evaluations for BJ 2-89-61 and had no further questions.

Bypass jumper 2-89-64 was approved by the licensee Plant Operation Review Committee on August 11. The bypass jumper installed a variable resistance potentiometer in the reed switch indication circuit for CEA #68. On August 1, the licensee documented loss of position indication of CEA #68, and complied initially with technical specification requirement 3.1.3.3.b and subsequently with requirement 3.1.3.3.c. The installation of the bypass jumper re-established position indication response as described in FSAR 7.5.3.2.c., as well as annunciator interlock alarms (CEA motion inhibit, CEA withdrawal prohibit, and CEA deviation) for the remaining CEAs. The installation of the jumper did not provide for partial rod insertion indication. The licensee's alternate indication of CEA #68 misalignment is ex-core or in-core nuclear instrumentation response, and verification of the requirements in TS for the upper limit switch indication on CEA #68.

The licensee documented a review of analyzed accidents, and determination of an unreviewed safety question. No unreviewed safety question existed. Inspector review and verification of the licensee evaluation found it to be acceptable.

On August 21, a review based on ACP-QA-2.06B, step 6.10.5, of BJs installed six months or longer was performed to assess the status of the items. Two BJs were installed in 1985, one in 1986, and three in 1988. BJ 2-85-113 and 2-85-114 are internally documented to be addressed by plant design change or removed during the next shutdown plant condition. The remaining outstanding bypass jumpers will be internally evaluated or corrected by December, 1989. The inspector considers continued licensee efforts to remove "temporary" jumpers an appropriate action.

5.4 Engineered Safety Feature Walkdown - Unit 1

The inspectors conducted a walkdown of the engineered safety feature systems listed below. The major valves in each system were verified in the proper positions by visual and/or local/remote position indication. The valve lineup procedures were consistent with system drawings and accurately reflected the present system configuration. Process parameters were reviewed and evaluated to be consistent with system status. No inadequacies in the physical condition of the valves or instrumentation were noted. The systems reviewed were:

-- Low Pressure Coolant Injection System
-- Core Spray System
-- Isolation Condenser System

A few minor discrepancies were noted during the walkdown. These discrepancies included deteriorated lagging and missing or erroneous valve labels, i.e., labels indicating "locked closed" versus "locked". These minor deficiencies were brought to the attention of the licensee.

5.5 IE Bulletin 83-03, Check Valve Failures in Raw Water Cooling Systems of Diesel Generators - Unit 2

Licensee response to IE Bulletin 83-03 was reviewed in routine resident inspection report 50-336/87-01. The licensee committed by letter dated May 31, 1983 to incorporate in the in-service test (IST) program disassembly and inspection of one service water discharge check valve on the emergency diesel generator (EDG) cooling system. In inspection report number 50-336/87-01, the inspector identified that the licensee had failed to document a forty month inspection of the valves in their ISI program for the second ten-year period (December 26, 1986 to December 26, 1996).

The two valves described above (2-SW-13A and 2-SW-13B) are eight inch swing check valves. The purpose of the check valves is to prevent backpressure surge to the EDG coolers which could cause extensive damage and potentially render the EDGs inoperable. A backpressure surge could result from adverse conditions of the ultimate heat sink (i.e. tidal waves, hurricanes).

In April 1983 both check valves were repaired after wear was noted upon disassembly. Licensee inspection in 1985 indicated that the 1983 repairs were only marginally effective. These results are documented in licensee internal memorandum EN2-85-072 dated August 27, 1985. Based on that inspection, the licensee engineering staff recommended replacement of 2-SW-13A and 2-SW-13B. In early 1987 the valves were replaced under plant design change request 2-94-86.

The inspector reviewed the plant modification which replaced the check valves. The original valves had cast iron bodies and carbon steel pin retainers. The body, step pin retainer, and hinge pin retainer of the replacement valves are fabricated with 410 stainless steel. The inspector also reviewed the seismic evaluation, material compatibility, unreviewed safety question determination, and re-test acceptance criteria. The plant modification technical support documentation was clear and concise.

The check valves are subjected to quarterly in-service inspection (ISI) surveillance to verify stroke to full open per procedure EN-21132, Quarterly ISI Testing. The inspector's review of completed ISI surveillances on 2-SW-13A and 2-SW-13B between April 1986 and April 1988 revealed no unacceptable performance of the valves.

The inspector found, however, that disassembly and inspection of the service water check valves is not currently established in the ten-year IST program. The licensee acknowledged the inspector's concern, and at the end of the inspection period was preparing an addition to the current IST program to include internal inspection. The licensee attributed the failure to include the committed inspection of the service water valves to personnel error. An additional concern was the inadequate follow-up of routine inspection report 50-336/87-01, and lack of follow-up of bulletin commitments to the NRC. The inspector considers this an unresolved item pending licensee actions to implement committed action to IE bulletin 83-03 (UNR 50-336/89-17-01). This bulletin remains open.

5.6 Radiological Controls - Unit 1

On August 19, 1989, a reactor building plant equipment operator (PEO) realized that he had failed to use the personnel contamination monitor after exiting from the radiologically controlled area (RCA). When the PEO frisked, he discovered that he was contaminated with readings between 1,000 to 10,000 dpm. Health physics personnel retraced the PEO's activities since his exit from the reactor building. Affected areas were wiped down and surveyed without identifying any spread of contamination.

The licensee issued a plant incident report and reviewed the circumstances surrounding this event. Results of that review indicate that areas in the reactor water cleanup pump room were inadvertently contaminated during an evolution to fill and vent the "B" cleanup demineralizer. During the PEO's tour of the reactor building, the PEO entered the reactor water cleanup pump room where he became contaminated. While in the room, the PEO had put on a partial set of anticontamination clothing to enter a roped off contaminated area. A personnel frisk is required if anticontamination clothing is used.

The PEO failed to frisk as required upon exiting the pump room. The PEO also failed to use the personnel contamination monitor after exiting the reactor building. Licensee procedure SHP 4909, Personnel Monitoring and Decontamination, states that "[I]t is each individual's responsibility to frisk his/her body for contamination after exiting a contaminated area or at designated frisking areas and to notify Health Physics if he/she is contaminated." In this event, the PEO failed twice to adhere to procedures.

The licensee issued a memorandum to all site personnel emphasizing the importance of contamination control and reiterating the requirements and policy for frisking. Additionally, efforts to determine the cause and methods to prevent the inadvertent contamination of the cleanup pump room during demineralizer fill and vent were initiated. The demineralizer fill line and vent line share a common pipe. This configuration prevents simultaneous filling and venting. As a result the licensee normally fills the demineralizer with the influent valve. The licensee postulates that during the evolution excessive flow through the vent line causes a backup in one of the drains in the cleanup room. The contaminated liquid then overflows into the clean areas of the cleanup room. The licensee is reviewing additional procedural guidance to require a radiological survey after this evolution. The licensee is also exploring the possibility of a modification to prevent recurrence of this problem.

The inspector concluded that the immediate licensee actions in response to this event appear appropriate. Upon exiting the reactor building, however, the inspector observed that a couple of people had not frisked hand held items as specified in the recently issued memorandum. The inspector concluded that continued efforts to improve the area of contamination control are warranted. The inspector will continue to review contamination controls in future inspections.

5.7 Closed Cooling Water System Heat Exchangers - Unit 1

During the summer, the licensee has had a significant problem with mussels entering the closed cooling water heat exchangers. The inspector reviewed the maintenance performed on the reactor building closed cooling water system and the turbine building closed cooling water heat exchangers for the period July 26, 1989 through August 26, 1989. During this period, each heat exchanger was cleaned on a daily basis, and, on several occasions, each was cleaned every shift.

The inspector was concerned that the emergency service water (ESW) heat exchangers may have a similar problem with mussel accumulation which could affect ESW system operability. During the inspector's review of ESW system surveillances, the inspector concluded that there is no surveillance which would indicate the condition of the heat exchangers. Additionally, the ESW system has no instrumentation to determine the differential pressure across the heat exchanger or to provide some other indication of heat exchanger performance.

The licensee stated that it believes that the ESW heat exchangers are functional and free of mussels based on the following:

-- The source of the mussels is believed to be from the service water system piping. During May, while the plant was in an outage, the service water system, except for a portion of underground piping, was drained. Conditions for a period of eleven days were evaluated by the licensee environmentalist to be optimal for mussel seeding. As a result, the mussels have been growing in the piping and subsequently collecting in the heat exchangers. Since the keep fill line to the ESW system taps off of the service water system at a point upstream of the postulated mussel colonies, no mussel accumulation is expected in the ESW heat exchangers.

-- The ESW system is only operated during monthly surveillance when the pumps are run for a few minutes each. The licensee environmentalist states that the stagnant water is oxygen deficient and not conducive to mussel growth or collection.

-- During previous outages, inspection of the ESW heat exchanger water boxes showed no accumulation of mussels.

-- The intake bays were cleaned after the last outage and are surveyed periodically to determine the condition of the bays.

The licensee further stated that the rate of mussel collection in the closed cooling water system has abated and that levels of chlorination have been increased. A review of potential changes to the ESW system to enable the assessment of the heat exchanger condition was initiated. Additionally, actions taken in response to NRC Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment, are in planning. The inspector had no further question.

5.8 Hydraulic Control Unit (HCU) Seismicity - Unit 1

During a tour of the reactor building on August 21, the inspector noted that the accumulator straps on two HCUs (HCU's 18-47 and 30-31) were not installed. This deficiency was brought to the attention of the licensee and the straps were promptly installed. During licensee walkdown of the HCUs, two additional straps were discovered broken or missing on HCUs 14-47 and 18-47.

The inspector questioned whether the straps were part of the seismic verification test for the HCUs and whether they are required for seismic qualification. At the end of the inspection period, the straps were confirmed to be a part of the original seismic verification test. Since the question of HCU operability during a seismic event has not been analyzed, the licensee decided to leave this issue open as an unanalyzed condition and on September 1, the licensee notified the NRC operations center of the condition pursuant to 10

CFR 50.72 (b)(1)(ii)(B) and 10 CFR 50.72 (b)(2)(iii)(A). A licensee event report (LER) will be prepared by the licensee on this event. This event will be left as an unresolved item pending review of the licensee's LER and corrective actions (UNR 50-245/89-17-03).

The inspector noted that a plant incident report was not written for the missing HCU accumulator straps. Licensee Quality Assurance Program Topical Report, section 16.2.1, requires that conditions adverse to quality be evaluated to determine the need for corrective action in accordance with approved procedures. Although the licensee took corrective actions and was evaluating the impact of the missing accumulator straps, licensee corrective action was not implemented fully in that a plant incident report was not issued until questions were raised by the resident inspector on August 30. Failure to evaluate conditions adverse to quality in accordance with established procedures is a violation. This violation is not being cited because the criteria specified in Section V.A of the Enforcement Policy were satisfied. (NCV 50-245/89-17-04)

5.9 Mid-Cycle Outage Planning Meeting - Unit 2

On August 24, the inspector attended the fourth mid-cycle outage planning meeting. Meeting attendees included department outage coordinators from operations, maintenance, engineering, instrumentation and controls, production test, construction, chemistry, health physics, and security.

The meeting agenda included review and discussion of the mid-cycle schedule bases, open items list, and overall review of critical path activities. The cycle 10 mid-cycle outage is currently scheduled for 31 days 22 hours (100% to 100% power operations). Major work activities will include steam generator (SG) eddy current examinations, repair of SG tube plugs, 'A' reactor coolant pump motor/seal replacement, inspections of steam supply check valves for the turbine-driven auxiliary feedwater system, repair of the 'A' EDG voltage regulator, RPS power supply replacements, and SDC suction valve pressure interlock modifications.

In conclusion, the inspector evaluated the meeting as well-organized, with knowledgeable planners, and good inter-departmental interface.

5.10 Auxiliary Feedwater Pump (AFP) Room Instrumentation - Unit 2

On July 7, due to the relatively warn ambient temperature in the AFP room, the inspector questioned the licensee regarding the environmental qualification of instrumentation located therein. The licensee documented the following response on July 8.

The equipment in the AFP room included pressure transmitters (PT) 5284, 4190, sump pumps P72A and P73A, and motor-operated valve SV-4188.

The pressure transmitters are GE model 551, and are rated for temperatures up to 180 degrees F. The temperature measured at these transmitters was 107 degrees F. Therefore, the transmitters will function as required in their normal ambient environment for pump discharge pressure and turbine steam pressure indications, respectively.

Ambient temperature at sump pumps P72A and P72B approximated rated temperature. Therefore, it is expected that these motors will function as required in their normal ambient environment. These pumps are not safety related items.

The thermal lifetime of Terry Turbine steam admission valve motor operator (SV-4186) was evaluated previously in NUSCO calculation number PA 78-771-241-GE. The evaluation showed that the motor would have a lifetime well in excess of 40 years if exposed to 140 degrees F. Therefore, this motor will function as required in its normal ambient environment.

No inadequacies were noted.

6.0 Observation of Physical Security

Selected aspects of site security were verified to be proper during inspection tours, including site access controls, personnel searches, personnel monitoring, placement of physical barriers, compensatory measures, guard force staffing, and response to alarms and degraded conditions. On August 23, the inspector walked down the Millstone Protected Area parameter. Compensatory security personnel were found to be knowledgeable of their duties and responsibilities during interviews. No inadequacies were found.

7.0 Service Water Leak Repairs - Unit 2

On August 31, a licensee ISI technician reported service water leakage from a previously identified localized through-wall pinhole leak. This leakage was initially identified on May 15. Non-Conformance Report (NCR) 289-141 was initiated and dispositioned "use-as-is" based on non-destructive examination readings greater than nominal wall thickness (.280 inch) circumventing the leak at 0.5 inch, 1 inch, and 2 inch radius, and evaluation concluding that structural integrity did not adversely affect the design basis.

The service water leak was on a six-inch header at an inner radius elbow. The header supplies service water to the sodium hypochloride system. The leak is unisolable from the main service water header.

On September 1, the licensee attempted to quantify the growth of the original leak. The attempt was unsuccessful due to the excessive amount of service water leakage from the temporary repair. At approximately 11:19 a.m. the licensee removed the 'B' service water header from service and entered the required technical specification action statements regarding two independent service water headers (3.7.4.1) and inoperability of the 'B' emergency diesel generator (3.8.1.1.a).

The inspector observed control room operators implement procedure OP 2326A, section 5.20, Removing Service Water Header from Service in Operating Modes 1-4. The evolution was well organized, pre-planned and professionally conducted.

The 'B' service water header was removed from service and tagged per clearance 2-2415-89. The inspector independently verified the adequacy of the equipment configuration utilizing piping and instrument drawings (P&ID's) 25203-26008 sheet 3, 25203-26008 sheet 4, and 25203-26009 sheet 2. No inadequacies in the tag-out were noted.

The replacement service water elbow was configured by the licensee under authorized work order (AWO) M2-89-06340. The replacement elbow was constructed to the original code ANSI B31.1-1971, and followed the weld inspections and hydrostatic test requirements of ASME Section XI repair and replacement plan. The replacement spool piece was completed on June 30 after the original leak was identified on May 15.

The AWO M2-89-07453 was used to replace the elbow spool piece on September 1. The retest was a visual in-service leak test at system operating pressure. No leaks were identified and the licensee restored the service water system to normal operation at 4:20 p.m.

In conclusion, work activities, required operational manipulation of the service water system, and management attention to ensure service water system integrity during power operations were assessed as good.

8.0 Emergency Diesel Generator (EDG) Operation

Routine resident inspection report 50-336/89-16 documented under-excitation events with the 'A' EDG during the end of July, 1989. On August 11, the licensee completed a EDG operability evaluation. The plant on-site review committee approved the evaluation during meeting 2-89-137. Licensee and voltage regulator vendor investigations have identified a saturable transformer in the exciter as a probable cause for the delayed start times. The series rated saturable transformers convert the EDG output voltage (4160 volts) to an AC voltage that is rectified by the field rectifier network into a DC voltage. Malfunction of one of the three saturable transformers delays the field current and subsequently retards the buildup of field voltage.

Technical specification requirement 4.8.1.1.2 a.2 verifies that the diesel starts at ambient conditions and accelerates to greater than or equal to 90% of rated speed and greater than or equal to 97% of rated voltage in 20 seconds or less. During the inspection period the licensee has demonstrated that the EDG will fulfill the requirements. However, the average start time is approximately 13 seconds vice the historical average since 1975 of 9-10 seconds.

Licensee actions to increase the reliability of the 'A' EDG are: increase the surveillance frequency from monthly to weekly and monitor voltage control parameters (i.e. field flash duration, automatic voltage regulator control signal, and field voltage) to trend further degradation of the saturable transformer output. Maintenance on the alternate diesel (22) will be kept to a minimum to reduce reliance on the 'A' EDG.

The licensee has initiated action to replace the saturable transformers. The vendor has notified Northeast Utilities that the original design documents have been discarded, and that no spare saturable transformers exist due to unique diesel generator characteristics. Quality Assurance Topical Report section 3.2.4 assigns NUSCO/NUPOC responsibility for assuring that design documents generated by contractors are adequately generated, approved, and maintained.

The inspector discussed the above design documentation control with licensee engineering personnel. The inspector considers this an unresolved item pending licensee acquisition of design specifications, replacement saturable transformers, and control of the specifications (UNR 50-336/89-17-05).

During the week of August 15 numerous surveillance runs were conducted on the 'A' EDG. On three occasions during the performance of the surveillance tests, the EDG output breaker tripped on reverse power. The total number of surveillance runs during the week of August 13 was fifteen.

The licensee retained the services of the EDG governor vendor (Woodward) to assist in the causal determination of the reverse power events. The governor installed in the EDG is a Woodward Model EG-B10C governor/actuator with an electronic governor assembly (EGA) electronic control box. The EGA electronic control box receives inputs from EDG output potential transformers, current transformers and a reference speed/load demand signal from a motor-operated potentiometer. The input signals are converted to reference demand setpoints in the EGA control box. The output signal from the EGA control box is sent to the governor/actuator to change fuel to the engine.

Licensee troubleshooting concluded that for all reverse power trips, the EDG initially loaded to a value higher than setpoint and rapidly unloaded, resulting in reverse current and trip of the unit.

On September 6, the licensee PORC approved an evaluation and statement of operability of the 'A' EDG governor. The evaluation included system capability and functionality, actions to increase reliability, corrective actions, operator awareness of the issue, and duration of the evaluation. The committee concluded that EDG governor reliability is not affected in the isochronous mode for the following reasons; in this mode the EDG will service all electrical loads on the bus, and the EGA control box is not in service. In the event of EGA control box output failure in the parallel mode, acceleration/deceleration of unit frequency will be dampened by emergency loads on the bus and will not result in an automatic trip. The statement of operability will be effective through the mid-cycle steam generator inspection outage scheduled to commence on October 21, 1989.

The inspector reviewed the operability evaluation, the Fairbank Morse maintenance manual, 50205872, volume I, NRC Regulatory Guide 1.108, Actions to Address EDG Reliability, and licensee troubleshooting activities. Discussions were also held with licensee engineers. The inspector noted no inadequacies and will monitor future licensee activities in this area.

9.0 <u>Committee Activities - Unit 2</u>

The inspector attended Plant Operations Review Committee (PORC) meetings 2-89-133, 2-89-142 and 2-89-144 on July 26, August 28, and September 5, respectively. The inspector observed that committee administrative requirements were met for the meetings, and that the committees discharged their functions in accordance with regulatory requirements. The inspector observed a thorough discussion of matters before the PORC and a good regard for safety in the issues under consideration by the committee. No inadequacies were identified.

10.0 <u>Observation of Maintenance</u>

The inspector observed and reviewed selected portions of preventive and corrective maintenance to verify compliance with regulations, use of licensee administrative and maintenance procedures, compliance with codes and standards, proper QA/QC involvement, use of bypass jumpers and safety tags, personnel protection, and equipment alignment and retest. The following activities were included:

<u>Unit 1:</u>

-- AWO M1-89-08459, Gas Turbine HMA Relays
-- AWO M1-89-09071, Gas Turbine HMA Relays
-- AWO M1-89-07947, Diesel Generator (DG) Emergency Stop Alarm
-- AWO M1-89-08153, G Exhaust Manifold Gasket

-- AWO M1-89-5671, D Service Water (SW) Flow Meters & Isolation Valves
-- AWO M1-89-5706, D SW Pump Strainer
-- AWO M1-89-5707, D SW Pump Strainer

Unit 2:

-- AWO M2-89-08452, Emergency Diesel Generator Exhaust Manifold Repair
-- 2701J-19 Monthly/Quarterly Preventive Maintenance on 'A' EDG

10.1 Millstone 1 AWO 89-08459, Gas Turbine Generator HMA Relays (PIR 89-62)

On July 24, while testing the gas turbine generator (GTG) following the replacement of the hydraulic power unit pump in the control circuit, the licensee noted that a logic "lock-up" condition existed when the "ready to start" light was illuminated concurrent with the "sequence in progress" light. The logic problem would have affected the normal engine shutdown sequence of the GTG and was cleared by manually toggling contacts in the 3K6X relay in the GTG normal stop control circuit.

Licensee investigation under automated work orders (AWOs) 89-8459 and 9071 identified a crack in the casing of the GE Type HMA relay that caused the relay contacts to bind. The failed relay was replaced with a new unit from stores. Further inspections of all similar relays in the control circuit (about 90 units installed as part of the GTG upgrade during the recently completed refueling outage) identified one additional relay in an annunciator circuit (3K7X) that was cracked but still functional. The licensee also verified proper movement of each HMA control relay in the GTG control circuit.

Licensee evaluation concluded that the two faulty relays were damaged when the board on which they were mounted was flexed during installation of the design change package and that its investigation under the AWOs was sufficient to assure all damaged relays had been identified. The licensee further determined that the failed relays would not have rendered the GTG inoperable in the as-found condition and that the matter was not reportable under the criteria of 10 CFR 50.

The NRC inspector subsequently confirmed the failure mechanism by examination of the faulty relay. The inspector reviewed the licensee's corrective actions and concluded that all suspect relays had been identified. The inspector requested the licensee to consider an additional inspection of the control relays for cracks to assure that the relays remain operable and to confirm the proper identification of root cause. The licensee agreed to conduct a visual inspection of the HMA control relays in conjunction with a future outage of the GTG allowing the deenergization of the control circuit to verify that the crack mechanism was positively identified. The inspector identified no inadequacies in the licensee's followup actions.

11.0 Observation of Surveillance Testing

The inspector reviewed completed surveillance tests and observed portions of surveil'ance performance to assess compliance with approved procedures and limiting conditions for operation, removal and restoration of equipment, and deficiency review and resolution. The following tests were reviewed:

-- SP IC 406BB, Stack Gas High Range Monitor Functional Test, 7/18/89 (Unit 1)
-- SP 668.1, Diesel Generator Operational Readiness Demonstration, 7/18/89 (Unit 1)
-- SP 668.2, Gas Turbine Emergency Fast Start Test, 8/9/89 (Unit 1)
-- SP 412P Isolation Condenser High Pressure Actuation Functional Test (Unit 1)
-- SP 2402P Spec. 200 Safety Parameter Functional Test (Unit 2)
-- SP 2613A Diesel Generator Operability Test, Facility 1 (Unit 2)
-- OP 2346A Emergency Diesel Generators (Unit 2)

No inadequacies noted.

12.0 Storage of Transient Equipment in Safety-Related Areas (Temporary Instruction 87-03) - Unit 2

NRC Information Notice (IN) No. 80-21, dated May 5, 1980, raised a number of issues associated with the seismic design of nuclear power facilities including the temporary use of non-seismic equipment such as tool storage cabinets and scaffolding. The licensee responded to IN 80-21 by making changes to their housekeeping program as described in procedure ACP-QA-4.01, Plant Housekeeping. Section 6.4.7 of the subject procedure requires the following seismic considerations when working in safety related areas:

-- Temporary ladders are to be secured to prevent seismic interaction with safety-related equipment.

-- Unattended equipment mounted on wheels is to be secured by chocks, locked wheels, or curbs.

-- Equipment with a high center of gravity is to be secured against tipping.

-- Staging or scaffolding should be erected and secured in accordance with ACP-2.19, "Scaffolding Program". Section 6.5.2 of ACP-2.19 contains requirements to prevent scaffolding from tipping or sliding.

-- Equipment is not to be stored such that a falling hazard is created (e.g. a tool box stored above safety-related equipment).

-- Cranes are to be secured against swaying or swinging interaction with safety-related equipment.

The staff consider the above provisions to be responsive to the concerns in IN-80-21 regarding storage of transient equipment in safety related areas.

Safety-related areas of Millstone Unit 2 were inspected and determined generally to be in compliance with licensee procedure ACP-QA-4.01. However, the following conditions were noted:

-- Chain falls consisting of a chain/pulley assembly on an overhead trolley are routinely used in safety related areas. The licensee does not routinely secure this equipment. The inspector noted that chain falls in the intake structure, emergency diesel generator A room, and on the -45 foot and -25 foot levels of the Auxiliary Building should be secured.

-- A scaffold on the -45 foot level in the auxiliary building is not properly securing.

-- The service water pump shroud in the intake structure is not secured.

-- Equipment mounted on wheels in the following locations should be properly secured against rolling: intake structure (welding machine), turbine building near heat exchangers X169A and B (welding machine), auxiliary building 14'6" west electrical penetration room (test equipment).

-- A temporary ladder in the turbine driven auxiliary feed water pump room is not secured.

The inspector presented the above items to the licensee for disposition, and had no further questions.

13.0 Periodic Reports

Upon receipt, a periodic report submitted pursuant to technical specifications were reviewed. This review verified that the reported information was valid and included the required NRC data. The inspector also ascertained whether any reported information should be classified as an abnormal occurrence. The following reports were reviewed:

-- Monthly Operating Report, Millstone 2, May 1989
-- Special Report for Seismic Monitoring System per Technical Specification 3.3.3.3 requirement (Millstone 2)

No inadequacies were noted.

14.0 Management Meetings

Periodic meetings were held with station management to discuss inspection findings during the inspection period. A summary of findings was also discussed at the conclusion of the inspection. No proprietary information was covered within the scope of the inspection. No written material was given to the licensee during the inspection period.