U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-336/88-13

Docket No. 50-336 License No. DPR-65

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

Facility Name: Millstone Nuclear Power Station, Waterford, Connecticut

Inspection At: Millstone Unit 2

Dates: May 3 - June 13, 1988

Inspectors: Peter J. Habighorst, Resident Inspector
David Jaffe, Licensing Project Manager, NRR
William J. Raymond, Senior Resident Inspector
James Trapp, Reactor Engineer, Division of Reactor Safety

Reporting Inspector: Peter J. Habighorst, Resident Inspector

Approved by: Jea H Battelen for 7/6/88
E. C. McCabe, Chief, Reactor Projects Section 1B Date

Inspection Summary: 5/3 - 6/13/88 (Report No. 50-336/88-13)

Areas Inspected: Routine NRC resident, region-based, and specialist inspection of plant operations, surveillance, maintenance, radiation protection, physical security, outage activities, allegations, Licensee Event Reports (LERs), Safety Issue Management System (SIMS) items, and committee activities.

Results: No unsafe conditions were identified. Additional follow-up is warranted on a 10 CFR 21 Report concerning wide range nuclear instrumentation susceptibility to moisture intrusion (Detail 4.6) and allegations (Detail 8.4, and Appendix B).

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DETAILS

1.0 Persons Contacted

NNECO

S. Scace, Millstone Station Superintendent
Mr. J. Keenan, Unit 2 Superintendent
Mr. J. Riley, Unit 2 Maintenance Supervisor
Mr. F. Dacimo, Unit 2 Engineering Supervisor
Mr. D. Kross, Unit 2 Instrument and Controls
Mr. J. Smith, Unit 2 Operations Supervisor
Mr. H. Haynes, Station Services Superintendent
Mr. R. Asafaylo, Quality Services Supervisor

Contractor Company

Contractor Superintendent
Contractor General Foreman - Electrical

Other members of the Operations, Radiation Protection, Chemistry, Instrument and Control, Maintenance, Reactor Engineering, Station Services Engineering, and Security Departments also were contacted.

2.0 Summary of Facility Activities

The unit began the inspection period at full power (100%) and operated there until May 6 at 9:00 pm, when an unscheduled normal shutdown occurred to locate the source of boric acid buildup inside containment and the associated increase in unidentified leakage. The unit remained shutdown until May 22 to replace reactor vessel head "O" rings (Section 4.2), to disposition and clean boric acid on reactor coolant system components (Section 7.1), and to evaluate dropped Control Element Assemblies (CEAs) (Section 4.7).

The unit returned to full power on May 24, and remained at this level until June 7 at 4:22 a.m., when CEA #23 dropped its full length into the core. The CEA was declared inoperable, and the plant was again shutdown. The shutdown, to replace upper gripper coils on all Control Element Drive Mechanisms (CEDMs), lasted through the rest of the inspection.

3.0 Licensee Action on Previously Identified Items (92701)

3.1 (Closed) Violation 87-16-01: Fire Protection for Auxiliary Feedwater Isolation Valves 2-FW-43A and 43B

This item concerned the lack of a 20-foot separation between redundant valves 2FW43A & B as required per 10 CFR 50 Appendix R Section III.G.2.b. NRC and licensee review during an inspection in July 1987 identified no safety concern. The licensee implemented appropriate compensatory measures by maintaining an hourly fire patrol of the area. In the October

23, 1987 response to the Notice of Violation, the licensee identified the intent to seek an exemption from the regulations, with the hourly fire patrol to be maintained until the staff approved the exemption request. The licensee submitted the request for exemption from 10 CFR 50 Appendix R Section III.G.2.b on February 29, 1988. The NRC concluded, based on information in a licensee February 29, 1988 letter, that Valves 2-FW-43A and B would go to the open position or remain closed and capable of being opened in the event of a fire. The NRC concluded the exemption request was justified and granted the exemption by letter dated April 29, 1988. The licensee suspended the hourly fire patrol on the same date. This item is closed.

3.2 (Closed) Unresolved Item 88-06-02: Control of Overtime During Outages

This item concerned the licensee's control and authorization of overtime in accordance with Administrative Control Procedure (ACP) 1.19, "Overtime Controls for Personnel Working at the Operating Station." NRC Inspection Report 50-336/88-06 identified twelve individual cases in the licensee's maintenance department where overtime approval apparently was not obtained prior to exceeding ACP 1.19 guidelines. The licensee provided additional information to the inspector concerning the applicable time report codes for the "Weekly Time Report" and the "Authorization to Exceed Overtime Limits" (NEO 1.09). The codes in question provide allowances for lunch breaks, extended continuous work hours, "on-call" time, turnover time, and vacation time. The inspector reevaluated the initial twelve cases with the appropriate time report codes. That review concluded that all twelve cases in question either did not exceed the ACP 1.19 guidelines or, if exceeded, licensee management approval had been granted. No inadequacies were noted. This item is closed.

3.3 (Closed) Inspector Follow-up: Location and Repair of Boric Acid Buildup Inside Containment

This item concerned the build-up of boric acid on the Control Element Drive Mechanism (CEDM) coolers and the Control Element Drive Assembly (CEDA) ventilation shroud as reported in Inspection Report 50-336/88-07. At the end of April 1988, the licensee completed hot water lance cleaning of the aforementioned components. Boric acid buildup was observed during subsequent power operation. The plant was shut down on May 6 to locate the source of boric acid. Based on licensee reactor vessel head inspections, both the inner and outer reactor vessel head "O" rings were leaking. The licensee repaired the vessel flange surface by honing out irregularities, replaced the head "O" rings, and completed an extensive boric acid clean-up of reactor vessel and containment components in a subsequent 15 day outage. Since the return to power on May 22, no boric acid build-up on the CEDM coolers or CEDA shroud has been identified. This item is closed.

4.0 Plant Tours and Operational Status Reviews (71707)

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

Control Room	Auxiliary Building
Vital Switchgear Room	Enclosure Building
Turbine Building	Intake Structure
Diesel Generator Room	Fence Line (Protected Area)
Containment Building	

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 reviewed and discussed with operators. Posting and control of radiation, contamination, and high radiation areas were inspected. 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. Records included various operating logs, turnover sheets, and tagout logs. Backshift inspections were performed on May 7 (8:00 a.m.), May 8 (3:30 p.m.), June 5 (7:30 p.m.), and June 7 (6:30 p.m.). Routine power operations and outage activities were observed. No abnormal conditions were observed.

4.1 Safety System Operability Review (71710)

Emergency systems were reviewed for operability in the standby mode. The systems reviewed were the containment spray and auxiliary feedwater systems. Status of the emergency diesel generators, service water pumps, and instrument air compressors was also inspected. The reviews included positioning of major flow path valves, operation of indication and controls, and visual inspections of proper lubrication, cooling, and other conditions References used included the Final Safety Analysis Report (FSAR), plant instrument and piping diagrams (P&IDs) 26015, "Safety Injection and Containment Spray Systems," 26005, "Condensate and Feedwater," and Operating Procedures OP-2309, Rev. 6, "Containment Spray," OP-2322, Rev. 9, "Auxiliary Feedwater," OP-2326A, Rev. 9, "Service Water," and OP-2346A, Rev. 9, "Emergency Diesel Generators." No inadequacies were noted.

4.2 Reactor Vessel "O"-ring Failures and Corrective Actions (93701/61726/62703)

On May 12, the licensee removed the reactor vessel head to visually inspect the reactor vessel flange and reactor head "O" rings. This licensee action was based on the determination that the head "O" ring was the only credible source of the increase in unidentified leak rate and the extensive boric acid buildup on the CEDA shroud and CEDM coolers in March and April 1988. Inspections were completed by the licensee and the vendor (Combustion Engineering). The two concentric "O"-ring gaskets

prevent reactor coolant leakage from between mating surfaces on the vessel flanges. Gasket grooves are machined in the closure head flange for the "O"-ring gaskets, which are polished and silver-plated Nickel-Chromium-Iron alloy. A monitoring device is provided in the vessel flange to detect any leakage past the inner "O"-ring.

Initial licensee inspections identified three separate "O"-ring failure locations. The inner ring failure was located in the vicinity of reactor head stud No. 51. The outer "O"-ring failed near stud locations 14 and 43. These studs are near the "A" and "B" cold leg nozzles. Upon further investigation, the licensee determined the failed areas on the "O"-ring surface were "hour-glass" shaped, approximately 0.25-inch wide with a necking down to approximately 0.1 inch. Measured vertically, all failed locations were 0.01 inch less than the compressed "O" diametric height (0.006-0.009 inch). All failures occurred at the "O"-ring interface with the vessel flange, not at the interface of the "O"-ring and the reactor vessel head. The inspector had no further questions on this matter.

Based on visual inspections, the licensee concluded the most probable cause of "O"-ring failures was relatively small foreign particles embedded in the "O"-ring surfaces The "O"-ring surface has 0.006 - 0.009 inch of silverplate to provide the sealing surface. The silver-plating was evaluated as insufficient to bridge deformities in the "O"-ring, thereby allowing a small amount of reactor coolant to pass across it. Identification of the debris is not yet conclusive. Licensee review identified no inadequacies in reactor vessel reassembly or in Quality Services Department (QSD) surveillances. In addition, the licensee's engineering department reviewed the possibility of reactor coolant flowing onto the flange during head installation. That investigation concluded, based on reactor vessel level indications during head installations in February 1988, that reactor coolant system inventory did not reach the flange.

Further licensee investigation of the failure mechanism of the "O"-rings will be undertaken. The licensee's Generation Engineering has been contacted and will attempt to provide additional visual and destructive analysis of the "O"-ring defects. The inspector will routinely follow-up on licensee actions in future inspections.

On May 14, in order to prevent debris caused problems, the licensee lowered the head to approximately 18 inches above the flange and carefully and thoroughly inspected and cleaned the head and flange again before the head was set in place.

Technical Specification (TS) 3.4.6.2 prohibits reactor operation with reactor coolant system (RCS) pressure boundary leakage, which is defined as a non-isolable fault in a RCS component body, pipe wall, or vessel wall. Leakage past a mechanical joint such as the reactor vessel head to vessel seal is not such a leak. The inspector had no further questions on this consideration.

During initial visual inspections of the reactor vessel flange area, one flawed indication on the flange surface was found. The flaw, which was smooth, was approximately 0.680 inch long with a maximum depth of 0.009 inch. It was located directly north on the flange, not on the "O"-ring mating surface. The licensee evaluated the flaw per Non-Conformance Report (NCR) 288-071, and EP-75167-100, Rev. 0, "Procedure for inspection, Cleaning, and Repair of Reactor Vessel and Closure Mating Surfaces." The licensee used a "honing stone" and hand worked the groove to the allowable one inch/0.001 inch gradient acceptance criterion, even though the flaw was not on the "O"-ring mating surface. The inspector had no further questions on this matter.

4.3 Plant Shutdown Due to an Inoperable CEDM (92700)

On June 7, at 4:22 am, with the unit at full power, Control Element Assembly (CEA) #23 dropped its full length into the core. Initial licensee investigation determined the individual breaker for the affected CEA had tripped on overcurrent.

Further licensee measurements determined that the upper gripper coil had developed an electrical short. Shortly after CEA #23 dropped, the licensee entered Technical Specification (TS) action statements 3.1.3.1.e, and 3.2.4, and decided to take the plant to cold shutdown to repair the inoperable CEA. By 7:00 a.m., the turbine was off line.

The inspector reviewed Abnormal Operating Procedure (AOP) 2556, Rev. 3, "Dropped CEA Recovery." The review considered actions taken by operators to mitigate the consequences of a dropped CEA, appropriate reference to TS requirements, operator actions if more than one CEA dropped in the core, and caution steps within the procedure. No inadequacies were noted.

The inspector interviewed the control room operators on actions taken for the CEA drop. The operators were knowledgeable of TS requirements and recovery actions found in AOP 2556. No inadequacies were noted.

Since February 15, 1988 the licensee has experienced five individual dropped CEAs: CEA #22, #23, #4, #16, and #31. The cause of the failures was overheating of the upper gripper coil.

The upper gripper coil is one of five coils used to manipulate the CEAs with the CEDM magnetic jack. The upper gripper coil functions to hold the CEA in position while four other coils remain deenergized. The cause of overheating of the upper gripper coils was degradation of CEDM cooling due to air flow blockage of the heat exchangers by previous boric acid deposition. (Boric acid deposition inside containment is described in resident Inspection Report 50-336/88-07.) The licensee replaced all upper gripper coils for all 61 CEDMs at Millstone 2. Vendor (Combustion Engineering) representatives provided 24-hour coverage of replacement of the coils. The inspector observed the replacement of the coil stack for CEDM-30. The observation included removal of the coils and extension

shroud, resistance checks for the replacement coil, and reassembly of the coil stack. No inadequacies were noted. The inspector concluded the action taken by the licensee adequately addressed the issue.

4.4 Review of Plant Incident Reports (PIRs) (71707)

The plant incident reports listed below were reviewed during the inspection period to (i) assess the significance of the events; (ii) review the licensee's evaluations; (iii) verify whether the licensee's response and corrective actions were proper; and, (iv) verify that the licensee reported the events in accordance with applicable requirements. The PIRs reviewed were: PIR 88-39, 88-40, 88-41, 88-42, 88-43, 88-44, 88-45, and 88-46. The following items warranted inspector followup: PIR 88-43 (See Report Detail 4.2), and PIR 88-46 (See Report Detail 4.5). No inadequacies were identified.

4.5 Pressurizer Level Control Channel Failure (PIR 88-46)

On May 27, 1988 at 12:15 am, control channel "110X" of the Pressurizer level control system failed low (less than 20% indicated level). The system responded as designed: both the proportional and back-up pressurizer heaters de-energized. The control room operators immediately selected control channel "110Y" to restore indicated pressurizer level and heaters. All pressurizer heaters were restored with the exception of group "B" proportional heaters. The licensee entered TS action statement 3.4.4.b, "Pressurizer Heaters" at 12:20 am. The licensee cycled breaker B0609 for the "B" proportional heaters and reclosed the breaker to exit the Technical Specification (TS) action statement at 2:00 a.m.

The pressurizer level control system is divided into two separate channels, "110X" and "110Y." Each channel functions to control letdown/charging flow and pressurizer heaters in response to deviations from programmed pressurizer level. The programmed pressurizer level is derived from the reactor regulating system. The level control channel has a differential pressure (dp) cell to sense level. This output signal is converted into a 4-20 milliamp (ma) electrical signal by a level transmitter.

The licensee stated that level transmitter LT-110X is a Rosemont 1153 instrument that was replaced in the April outage due to erratic level indications. The licensee will perform a failure analysis on the faulty transmitter to determine the cause of improper response. On May 28, the licensee commenced troubleshooting to identify the failure of control channel "110X" using I/C 2418K, Rev 2, "Installation/Calibration and Servicing of Rosemont 1153 Series Pressure Transmitter." Initial results localized the failure inside containment.

The inspector questioned the apparent failure frequency of Rosemont 1153 Series transmitters and the possible generic implications. NRC Notice (IEN) 85-100, "Rosemont Differential Pressure Transmitter Zero Point Drift," alerted licensees to the potential significant safety problem involving a loss of High Pressure Injection (HPI) flow indication because of a shift in the zero point of a Rosemont Model 1153, Series B differential pressure transmitter between the depressurized calibration condition and the pressurized operating condition. The licensee addressed the zero point drift condition in I/C 2418K, Step 5.3.1.3, by directing the technician to perform a static alignment adjustment to prevent zero point drift. At Millstone Unit 2, three Rosemont Model 1153, Series B pressure transmitters are installed (two for pressurizer level, and one for pressure indication for Low Temperature Overpressure Protection). The licensee and Rosemont concluded the failure was not attributable to zero point drift.

On June 1, the licensee installed a bypass jumper in the 110X control channel per in-service test 2-88-63 to simulate a 65% fixed pressurizer level. This action prevented the loss of pressurizer heaters if control channel "110Y" were to fail. This procedure was approved by the Plant Operations Review Committee (PORC) on the same day. A Night Order Instruction on June 1, was issued to the control room operators to emphasize appropriate actions on failure of the remaining control channel "110Y", and on a reactor trip.

On June 2, the licensee performed a containment entry at full power (100%) to investigate the failure of pressurizer control channel 110X. The licensee determined the apparent failure was the capacitance sensing module in the level transmitter. The licensee sent the failed transmitter to Rosemont for further, destructive failure analysis. The licensee replaced the "110X" transmitter on June 12, and removed the bypass jumper from channel 110X. No inadequacies in subsequent operation were noted.

4.6 Wide Range Nuclear Instrumentation, 10 CFR 21 Report (92700)

The licensee notified the inspector, on May 23, that solder connections on Gamma-Metrics (G-M) cable assemblies may be susceptible to moisture intrusion during a design basis accident (DBA). G-M made a 10 CFR 21 report to the NRC on February 19, 1988, identifying recent environmental qualification test failures which were attributed to a solder joint on a metal hose of a G-M cable assembly that failed to hold pressure at elevated temperatures. The licensee was notified of this problem by letter from G-M on February 22. G-M also provided, in a letter to the licensee on May 10, guidance for inspection of neutron flux monitor cabling and evaluation of a retrofit to provide additional sealing of the metal hose to prevent moisture intrusion at specific connections. The vendor reported the item as a safety issue since the G-M neutron flux monitor and cabling assemblies are used to provide the operator with neutron flux indication from the source range to 150% power in post accident monitoring environments to meet the requirements of Regulatory

Guide 1.97. Appendix A is a listing of plants potentially affected by the 10 CFR 21 report, which was provided to the inspector by the vendor. The inspector provided the list to the licensee.

Millstone 2 TS 3.3.3.5 specifies operability of wide-range neutron flux monitors on the remote shutdown panel. Accordingly, the FSAR documents that the remote shutdown panel is used by the operator to maintain the plant shutdown and to achieve cold shutdown following a reactor trip when the control room is deemed inaccessible. However, all plant operation including emergency shutdown is accomplished in the control room and not from the shutdown panel.

The wide range neutron flux monitors are also referenced in the following TSs:

-- TS 3.3.1.1, "Reactor Protective Instrumentation Channel"
-- TS 3.9.2, "Source Range Neutron Flux Monitor Operability"

The above TSs, applicable between hot shutdown and refueling plant conditions, monitor changes in core reactivity. The inspector had no further questions in this area.

The licensee, in response to the 10 CFR 21 letter, reviewed the Emergency Operating Procedures (EOPs) to determine if wide-range neutron flux instrumentation indication was a reference for operator action. The EOP utilized during a Loss of Coolant Accident (LOCA), EOP 2532, Rev. 3, made no specific reference to wide-range nuclear instrumentation and no operator actions were based on wide-range nuclear instrumentation power level indication. Other EOPs referenced were EOP-2525, "Standard Post Trip Actions," and 2540A, "Functional Recovery of Reactivity Control." The inspector had no further questions on this area.

The inspector reviewed past commitments for environmental qualification of the wide-range neutron flux monitors. By letter dated January 30, 1987, the licensee provided to the NRC the status of Regulatory Guide 1.97, Rev. 2 items. The wide-range neutron flux instrumentation was included in the response letter, while committed to Environmental Qualification (EQ) per 10 CFR 50.49. The inspector questioned the seeming failure to develop a Justification for Continued Operation (JCO) and the purpose for EQ if the equipment is not used for design basis accident monitoring. This item is unresolved pending licensee response (UNR 88-13-01).

The inspector reviewed Generic Letter (GL) 83-37 (NUREG-0737 Technical Specifications) and the July 25, 1984 licensee response to the generic letter. GL 83-87 does not specify wide-range nuclear instrumentation as a monitored parameter for post-accident monitoring instrumentation. Licensee TS 3.3.3.8, "Accident Monitoring Instrumentation," conforms to the commitments set forth in Generic Letter 83-37. The inspector had no further questions on this matter.

On May 27, the licensee completed an EQ evaluation of G-M Wide-Range Nuclear Instrumentation. The licensee contacted G-M to identify the specific problem and manufacturing dates referenced in the 10 CFR 21 report. The problem, as determined by the vendor, is voids in the solder joints of in-containment cable assemblies. That allows moisture to migrate to various cable connectors. According to a May 10 letter from the vendor to the licensee, moisture intrusion was detected in submergence testing at 60 PSIG. Failure results in arcing between the conductors, which carry 800 volts, to the detectors. The arcing would cause electrical noise which is interpreted as an increase in neutron flux by the electronics. The licensee noted that G-M had changed their shop fabrication procedure for pre-tinning the solder joints in question. Prior to 1984, a solder-pot dip process was used, and was tested in the original qualification testing. The latest qualification testing was done on cable assemblies fabricated using an iron-applied tinning. G-M further noted that they have compared finished solder joints fabricated using both methods, and have observed voids similar to the failed cable assemblies only in those samples using iron-applied tinning.

According to the licensee, of the four wide-range channels currently installed at Millstone 2, channels "A" and "C" were manufactured prior to 1983, channels "B" and "D" were replaced during the 1988 and 1985 outages, and only channels "B" and "D" have a potential EQ inadequacy. The licensee concluded that all TS requirements are satisfied at present. No inadequacies were noted by the inspector.

The licensee does not intend to replace or repair questionable channels "B" and "D" until the scheduled outage in February 1989. The licensee has committed to testing the questionable instruments using vendor recommended tests. The recommended test for wide-range nuclear instruments has yet to be formally resolved by the vendor. The inspector will follow this issue until successful completion of testing of the "B" and "D" wide range nuclear instruments.

4.7 Licensee's Disposition of CEDM #14

On May 7, at 1:45 am, during control rod insertion, CEA-14 and CEA 16 (group 6) dropped 43 steps and CEA-31 (group 1) dropped 17 steps, all from the initial fully withdrawn position. The licensee realigned each CEA with its respective group, and rod insertion was completed.

During the April, 1988 outage, the licensee removed the coil stacks from CEDM-14 to perform a visual inspection. The licensee inspected the upper gripper coil for high temperature effects. Resistance across the upper gripper coil was 6.98 ohms (corrected for ambient conditions). The average resistant values of all other CEDM's was between 7-8 ohms per I/C 2421C, "Reactor Vessel Head Cable Removal, Installation Data," from the April 8, 1988 shutdown (CEDM-14 Inspection Report 50-336/88-07). Inspection of the CEDM-14 upper gripper coil assembly revealed cracked potting insulation, but the coil varnish was intact. The licensee con-

tacted the vendor (Combustion Engineering) for a destructive failure analysis of the coil. The vendor's analysis concluded that extreme heat caused the potting material to crack and that, eventually, coil varnish breakdown would render the CEDM inoperable. The licensee then replaced all upper gripper coils for all CEDMs during the June 7 outage (Detail 4.3). The inspector had no further questions on this matter.

5.0 Physical Security (81064)

Selected aspects of site security were verified to be proper during inspection tours, including access controls, personnel and vehicle searches, personnel monitoring, placement of physical barriers, compensatory measures, guard force staffing, and response to alarms and degraded conditions. One security event occurred during the inspection period. The following is a description of the event:

On May 24, at 4:23 p.m. the licensee found a security guard to be inattentive to duty while stationed at a contingency post. The security guard was relieved and a replacement guard assumed the responsibilities within ten minutes. This event was reported under 10 CFR 73.71(c) at 5:03 pm.

The guard was a member of the contingency force assembled to meet security requirements during the security force strike which began on May 2, 1988.

The inattentive guard had assumed the post at 3:31 p.m. on May 24 and was discovered inattentive to duty at 4:23 p.m. Licensee investigation concluded this was an isolated situation with minimal security impact. No unacceptable conditions were noted during inspector review.

6.0 Surveillance (61726)

In response to Control Element Assembly (CEA) upper gripper failures on May 6, 1988 and preliminary vendor (Combustion Engineering) failure analysis of the failed CEDMs in April, the licensee completed a Plant Design Change Request (PDCR) evaluation to directly monitor CEDM coil temperatures. Previously, the containment air temperatures were monitored on top of the CEA ventilation shroud via temperature sensors T8095 and T8096.

On May 18, the licensee completed MP2-88-59, PDCR evaluation on "CEDM Upper Gripper Coil Temperature Sensor Installation." The plant change connected a temporary temperature sensor (T-8096) in place of the normal Resistance Temperature Detector (RTD). T-8096 previously monitored containment air temperature in the reactor head area. The temporary sensor was connected to the #22 CEDM upper gripper coil casing cover. Containment air temperature in the reactor head area is directly monitored by RTD T-8095. The added sensor will provide information on actual coil temperatures. The licensee will use T-8096 to assess operating lifetime of the CEDM coils. The PDCR evaluation was approved by the Plant Operations Review Committee (PORC) at meeting 2-88-112

on May 19. The inspector attended the PORC meeting, and found technical discussions concerning this matter to be complete and thorough. No inadequacies were noted.

7.0 Maintenance (62703)

7.1 Boric Acid Buildup on Reactor Coolant Nozzles

On May 11, the licensee performed a visual inspection of the annulus between the reactor vessel and the biological shield. The area contains the excore nuclear instrumentation detectors and six reactor coolant nozzles. The licensee conducted the visual examination of the reactor vessel nozzles for evidence of degradation caused by exposure to boric acid from the reactor vessel "O"-ring failure (Section 4.2). The licensee results were as follows:

-- On the top of the "B" cold leg nozzle, there was a bright metal area approximately 15 inches long by 2-3 inches wide. Corrosion/erosion had removed about 1/32 of an inch of material from the bright area. Investigation determined that the bright metal had no high temperature corrosion film. Several circular spots characterized the corroded area. Two spots near the top center of the nozzle contained dark oxide film. The licensee was unable to determine if this was the result of boric acid attack or if areas were not attacked enough to disturb the original corrosion film.

-- The "A" cold leg nozzle had a bright metal area approximately 24 inches long by 2-3 inches wide, starting near the nozzle top and extending approximately 90 degrees counter-clockwise. Licensee investigation determined that the bright metal area had no high temperature corrosion film. This area indicated some metal loss. In an area, near the top of the nozzle, a hard corrosion product was seen. When wire brushed, the area appeared to be the deepest area of metal loss. Corrosion/erosion appears to have removed about 1/32 to 1/16 of an inch of metal.

The inspector reviewed the results of the visual examination and the licensee video camera films of the inspection.

On May 14, the licensee conducted ultrasonic (UT) examinations of the "A" and "B" reactor coolant system cold leg nozzles per Non-Conformance Report (NCR) 288-069. The inspector reviewed the results of the ultrasonic examination. The examination indicated an actual wall thickness of 11.11 inches on the "A" cold leg nozzle, and 11.08 inches on the "B" cold leg nozzle, with no measurable wall loss. The minimum wall thickness for design of reactor coolant system inlet nozzles, calculated by the licensee utilizing ASME XI 1981 Winter Edition (IWB 3512 B-D), was 10.205 inches. The inspector reviewed the ASME XI code and the licensee's calculation of minimum wall thickness, and had no further questions.

The licensee's overall disposition on the affects of boric acid on the RCS pressure boundary components was aggressively pursued and resolved. No inadequacies were noted.

In reviewing the licensee's evaluation, the inspector referenced NRC Generic Letter 88-05, Information Notice 86-108 Supplement 1, "Degradation of Reactor Coolant System Boundary Resulting from Boric Acid Corrosion," and NUREG-2827, "Boric Acid Corrosion of Ferric Reactor Components." By letter dated May 27, 1988 in response to Generic Letter 88-05, the licensee reported the boric acid corrosion program on ferric materials at Millstone 2.

The inspector had no further questions on licensee examinations and corrective actions on reactor vessel nozzles.

7.2 Boric Acid Corrosion and Licensee Deposition of Reactor Vessel Studs

On May 13, in response to the reactor vessel "O"-ring degradation, the licensee inspected all fifty-four reactor vessel studs. The closure head is attached to the vessel by seven-inch diameter reactor vessel studs. The closure head is held in place by nuts screwed onto the studs and seated on spherical washers.

The licensee conducted an initial visual inspection per Non-Conformance Report (NCR) 288-072 and NU-VT-1, Rev. 7, "Pressure Retaining Bolting Visual Examination," of all fifty-four studs. The inspector reviewed the results of NCR 288-072 and NU-VT-1. The licensee concluded that 31 studs have reportable indications. Further licensee review concluded that nine studs were to be replaced. The remaining twenty-two were cleaned and re-installed. The nine replaced studs had indications of slight pitting on the lower 3-12 threads of the upper threaded area, and heavy corrosion indication one inch above the lower threaded region on the stud shank area. The material loss in the stud shank area ranged from between 0.050 to 0.2 inches. Licensee evaluation concluded that the corrosion was limited in extent. Reuse of all affected studs is expected after further evaluation. The inspector had no further questions.

8.0 Allegations and Telephone Call from Concerned Citizen (92720/92702)

The following allegations pertain specifically to Millstone Unit 2. Three other site related allegations not specific to Millstone Unit 2 are enclosed as Appendix B.

8.1 88-A-0015, "Health Physics Concerns inside Containment"

On February 3, 1988, a contractor employee approached the resident inspector to address three specific concerns: pigeons inside containment and the potential radiation hazard; potential egress from the auxiliary building without frisking for contamination; and improper containment access control concerning the spread of contamination.

The alleger saw pigeons inside containment during the refueling outage in February, 1988. He was concerned that the pigeons would spread radioactive contamination and that they might be contaminated and be eaten by people. NRC radiation specialists evaluated this as not a radiation safety hazard. The inspector discussed with licensee management the advisability of preventing pigeon entrance into containment for housekeeping reasons. The licensee committed to address methods of preventing pigeons from entering containment during outages.

The inspector reviewed egress routes from the frisking station, and whole body friskers (PCM-1A) on the lower elevation of the auxiliary building. The inspector did not detect a method of bypassing the frisking stations without violating posted requirements. The inspector considered the existing controls adequate. Adherence to the requirements is routinely checked by licensee personnel and NRC inspectors.

On the containment access concern, NRC review noted that the information provided by the alleger lack specificity on the date of occurrence, frisker reading, and scale reading on the frisker. During routine inspection throughout the period, the inspector observed containment access controls, individual frisking techniques, and anti-contamination clothing removal. No inadequacies were noted. The inspector discussed this specific concern with the licensee and asked that it be considered in contractor radiation protection training. No inadequacies were noted.

NRC follow-up did not substantiate the alleger's concerns in this case. Because of the potential for adverse housekeeping effects, however, birds present in containment will be routinely monitored and assessed. So will adherence to radiation protection requirements.

8.2 Telephone Conversation with a Concerned Citizen

The inspector received a telephone call at 3:00 p.m. on June 7 from a resident of Pleasant Beach in Waterford, Connecticut. This individual was concerned about the loud noise he heard from the Millstone Station all day. The inspector checked and informed the citizen that the sound was from the Millstone 2 atmospheric dump valves used as part of the normal procedure to cooldown the plant. The inspector assured the citizen that there was no offsite hazard from the non-radioactive steam emanating from the dump valves. The inspector stated that the noise should cease by about 8:00 p.m. on June 7, after cooldown via the atmospheric dump valves was completed. The citizen appeared satisfied with the information provided. No inadequacies were identified relative to licensee activities.

8.3 RI-88-A-0029, "Update on Procedural Compliance in Metrology Lab"

This alleger had two concerns dealing with metrology laboratory work. The concerns dealt with inadequate procedural compliance for safety-related instrument calibration and job discrimination by licensee management.

The calibration concern was addressed by the inspector in routine inspection report 50-336/88-07; calibrations involved were found acceptable. On the discrimination concern, the alleger filed a complaint with the U.S. Department of Labor (DOL) on April 29, 1988. Initial DOL efforts to conciliate the matter between the alleger and employer (Northeast Nuclear Energy Company) were unsuccessful. DOL began an investigation of whether job discrimination for protected activities (Energy Reorganization Act of 1974, 10 CFR 50.7) had occurred. The DOL investigator found in favor of the alleger. In a letter dated May 27, the licensee was notified by DOL of required remedial action. The licensee filed an appeal, invoking their right to a formal hearing. This aspect remains open (UNR-88-13-02). Also, as a result of NRC follow-up on another allegation, the licensee has been asked to include, with the response to a notice of violation, their actions to improve addressal of employee concerns. (5/28/88 NRC letter forwarding Inspection Report 50-336/88-07.)

8.4 88-A-0040, "Update on Improper Radiation Monitor Calibration and Other Concerns"

8.4.1 Control Room Radiation Monitors

On April 11, a licensee employee approached the inspector with the following concerns: control room area radiation monitors are not calibrated properly and nuclear concerns are not being addressed adequately by licensee management. The inspector reviewed the first concern during inspection 50-336/88-07, and concluded that the licensee was not in full compliance with the TS or with commitments in the Final Safety Analysis Report (FSAR). The response letter from the licensee on the proposed Notice of Violation and Notice of Deviation should address licensee actions to improve the effectiveness of addressal of employee concerns. This item remains open pending licensee response (due in mid-July 1988 based on a licensee telephone request for the time needed to respond) and pending implementation of planned actions.

During subsequent meetings with the alleger, the inspector reviewed other concerns. These were: Excore/Incore surveillance during the end of the refueling outage not properly calibrated, channel "D" of the Reactor Protection System (RPS) not responding properly due to an intermittent problem, and incorrect oil level indication for the Reactor Coolant Pumps (RCPs). These are discussed further in the following.

8.4.2 "D" Reactor Protection System Response

The inspector reviewed the alleger's concerns about operability of RPS Channel "D" during plant startup from the refueling outage. The alleger's specific concern on RPS Channel "D" was intermittent response of the cold leg temperature (Tc) input

into the channel. The Tc input is used in the RPS for generating the thermal-margin low pressure (TMLP) reactor trip signal. Each channel is provided two Tc inputs.

The inspector reviewed M2-88-03147, "Repair of D RPS channel," TS 3.3.1.1, "Reactor Protective Instrumentation," and the licensee's successfully completed SP-2418B-4, "RPS Temperature Inputs," for Channel D on February 22, 1988. Licensee investigation concluded a Foxboro 200 circuit card had failed. This card is part of a signal select module biased to select the highest of the two Tc inputs into channel "D" of RPS. The card was replaced on February 22, and the licensee completed a successful calibration check per SP-2418B-4 on the same day. The inspector concluded the "D" RPS channel was adequately resolved by the licensee, and all requirements for operability were maintained.

No unacceptable conditions were identified. Review did not substantiate the allegation.

8.4.3 Reactor Coolant Pump Oil Sump Levels

This concern was for reactor coolant pump upper and lower oil sump levels, inadequate level indication, and potential oil leakage from the sump. The inspector interviewed licensee personnel and reviewed the reactor coolant pump technical manual, Foxboro level transmitter calibration curves, I/C surveillance procedure 2435B, Maintenance History Logs, and maintenance records on RCP oil sumps since the past refueling outage. No inadequacies were noted. The licensee committed to review the alleged RCP oil level indication discrepancies. The inspector will review licensee actions in future routine inspections.

8.4.4 Excore/Incore Calibrations During Restart Testing

The inspector reviewed the alleger's concern about two potentially inoperable RPS channels (A&D) during plant start-up from the refueling outage. This issue concerned directions given to adjust the excore nuclear instruments in February 1988 when reactor conditions did not meet the prerequisites of I&C calibration procedure SP 2401E. Specifically, for the measurements done at 30% power, equilibrium xenon conditions did not exist as required by Step 4.1 of the subject surveillance procedures. The alleger stated he was coerced into performing the adjustments by shift and I&C supervisory personnel and that he was directed to perform an "unofficial" calibration, did so under protest, and filed a deficiency report [I&C form 2437A-1, Instrument Calibration Review (ICR) Form] with the data.

This matter was referred to licensee management for follow-up. The inspector reviewed the calibrations performed during restart from the outage with the Reactor Engineer and the I&C Department Supervisor. This review confirmed the alleger's statements, but also found no inadequacies in the calibration of the excore nuclear instruments.

Prerequisite 4.1 of SP 2401E, Calibration of the Excore Nuclear Instruments (NIs) to Incores, Rev. 8, requires the reactor to be at steady state power and at or near equilibrium xenon conditions when performing the calibrations. However, during power ascension testing, calibrations per 2401E are performed as required by the power ascension test program procedure T88-2, which is the governing document to establish the power level and core conditions under which the incore measurements and excore adjustments will be made. Core performance checks made per T88-2 and done at 30%, 50%, 80% and 96% power. Equilibrium xenon conditions are required at the 3 highest power plateaus, but not at 30% power (Step 7.4 of T88-2)

The Reactor Engineer stated that the first fine adjustment of the excores to the incore reading of axial shape index (ASI) is performed at 50% power/ The acceptance criteria for the adjustments made at higher powers is to calibrate the excores to within 1% of the incore readings. The calibration done at 30% power is a coarse measurement to check for general agreement between the excores and incores. The acceptance criteria is to adjust the excores to within 6% of the incore ASI values. This is acceptable because ASI is not a limiting core performance parameter at lower core powers. The inspector noted that Technical Specification Figure 3.2-2a does not establish ASI limits below 50% power, and further that the limits are wide at + or - 25% from 30% to 65% power. Further, the local power density limiting safety system setting (Figure 2.2-2) allows a wide ASI band at + or - 40% from 0% TO 65% power. No inadequacies were identified in the licensee's conclusions or in the excore calibrations completed during the startup in February 1988. The inspector had no further questions regarding the technical adequacy of the NI calibrations.

The inspector reviewed the test data recorded per SP 2401E on February 21 and 24 for testing at the 30% and 50% power plateaus. The recorded results confirmed the alleger's statements regarding the testing at 30%. The inspector noted that adjustments were made to RPS NI Channel A at 30% to obtain channel readings within 6% of the incore value. The other channels were within the 6% acceptance criteria at 30% FP. However, during the calibration of Channel A, a problem was noted with the detector high voltage cable and the channel was declared inoperable while repairs were affected. Power as-

cension testing continued while repairs were in progress. Turbine overspeed testing was completed at 15% power. Following the turbine test, the reactor was taken to 50% power for testing. The 50% test plateau was reached about 4:00 a.m. on February 22, and NI Channel A was returned to service at about 6:00 a.m. the same date.

Inspector reviewed noted that two sets of incore/excore calibration data were recorded at 50% power. Minor adjustments were required and made on Channels A, B, and D in the first calibration performed on February 22. However, it was noted that core xenon concentrations were near but not at equilibrium values, and a second calibration was performed at 50% power on February 24 with xenon in equilibrium. Final adjustments on all channels made the excore ASI within 1% of the incore values. In response to inspector inquiries, the licensee explained that the first calibration done at 50% without equilibrium xenon was done as a post-maintenance test following repair of Channel A, and equilibrium conditions were not required for a "coarse" check of the excore ASI. Channels B, C, and D were also calibrated to clear a channel deviation alarm that was in effect.

The inspector asked the licensee what impact the calibrations performed without equilibrium xenon had on operability of the local power density limiting safety system setpoints provided in TS Figure 2.2-2. The licensee's evaluation was addressed in a memorandum by the Reactor Engineer. The licensee concluded that the 50% calibrations did not make the trip setpoints inoperable since the maximum difference in channel adjustment from the "as-found" to the "as-left" values was 0.014 ASI units, which was well within the + or - 0.027 ASI units assumed in the licensee's safety analysis as the amount of decalibration allowed in the excore ASI values. The licensee also addressed a question by the inspector regarding an apparent large deviation in the ASI reading for Channel D at 50% power. The licensee determined that the Channel D ASI value was incorrectly recorded on the I&C data sheet (SP2401E) for February 22, 1988. The inspector reviewed graphs of excore ASI data obtained from the plant computer and control board indicators which showed that excore ASI was maintained within 1% of the incore values at 50% power and above. The inspector had no further question regarding the technical adequacy of the excore calibrations and operability of the RPS trip setpoints.

Inspector interviews with I&C supervision confirmed that the I&C technician was directed to perform the NI calibrations at 30% power without establishing equilibrium xenon conditions. The basis for this directive was that procedure T88-2 was the

governing document and Reactor Engineering was the proper authority to establish requisite reactor conditions for testing. The inspector concurred with the licensee's conclusions on a technical basis. However, the inspector expressed concerns on how the matter was handled by shift personnel on February 11 and on the lack of action to reconcile differences between SP 2401E and T88-2. The licensee acknowledged the inspector's comments on the procedure interface, and made the following changes to SP 2401E via Change 1 to Revision 8 effective 5/19/88: Prerequisite 4.1 now requires that, in general, the reactor to be at or near equilibrium conditions, but the technician should consult with Reactor Engineering to determine the status of xenon equilibrium, and if performing the test at non-equilibrium conditions is required.

The inspector had no further comments on the issue involving the excore/incore calibrations. The licensee's program for handling employee concerns is under review by the NRC and is discussed in Section 9.0 below.

8.5 87-A-0113, "Contractor Work Activities"

This review addresses issues raised by a former contractor employee (alleger) whose onsite employment was terminated in the Fall of 1987. The allegations were received during a January 13, 1988 meeting with the alleger, and were referred to licensee management for addressal. Licensee findings were periodically discussed with the inspector, and were summarized in an May 19, 1988 memorandum to the Station Services Superintendent. Supplemental information on the issues was obtained during various followup telephone conversations and during a meeting on June 8, 1988. Inspection consisted of inspector review of licensee findings and corrective actions, with independent assessment and review by the inspector.

Five (5) work control issues were identified and are addressed in Appendix B to this report. A new issue, identified during an April 25 telephone conversation, involved electricians who worked at Millstone 2 during the 1983-84 outages, and who left the site without whole body counts or termination exposure reports. This issue is also addressed in Appendix B. The alleger also alleged improprieties in the relationship between the contractor (Contractor A) and a NNECO supervisor. While these issues did not present regulatory concerns and were relayed to the licensee for his follow-up, the inspector also addressed them to the extent of whether or not Contractor A had regard for licensee procedures and whether contractor activities were controlled. The licensee did not substantiate the concern that Contractor A exerted an inappropriate amount of influence on site activities. While some problems were identified in the implementation of work control procedures, the licensee concluded that Contractor A has regard for and does follow procedures. Inasmuch as the alleged activities did not involve nuclear safety-related

work and this contractor does not normally perform safety-related work, the inspector concluded that the licensee's findings do not require further NRC review.

For Contractor A, work of NRC concern involves security systems and the Fire Protection Program. This contractor does not have an approved QA Program and is not used for Category I (safety-related) work. Until recently, most of this contractor's work was performed in parking lots and outlying areas and buildings. Contractor A was used for non-Category I work inside all three units during recent outages, with one job being work to modify gates for high radiation areas.

No unsafe work activities were identified and none of Contractor A's work involved or affected plant safety systems. Licensee follow-up of the work control issues did identify discrepancies in tagging operations and processing work orders. Corrective actions were taken and are in progress to assure full compliance with licensee procedures. NRC follow-up of these corrective actions will be routinely performed to assure that all onsite work groups are trained in and follow licensee administrative procedures, so as to provide added assurance that contractors who work at the interface between safety-related and non-safety-related plant activities do not inadvertently impact nuclear safety. Therefore, the corrective actions listed in Issues 5 and 6 of Appendix B will be followed-up on a sampling basis.

Two new issues were identified during the June 8th meeting with the alleger. These involve:

i. Information from an electrician who worked at Millstone 3 during the outage in November 1987 and who had to replace all lamps in underwater lights in the spent fuel pool (SFP). The electrician had to do a hot job that allegedly should have been done when the SFP was empty. Initial inspector evaluation of this issue concluded no significant radiation exposure occurred. This finding was based on the design, which allows pulling the lighting fixtures out of the pool to replace a lamp. Also, based on inspector checks made on June 13, 1988 with spent fuel in the pool, area dose rates were less than 0.2 mRem/hr and pool radiochemistry results showed low activity levels of 5E-4 uCi/ml.

ii. An electrician who worked onsite for the Millstone 3 outage stated that work orders were processed to replace 300 solenoid valves that were undersized and were being replaced as they burned out.

These issues have been referred to licensee management for follow-up. Inspector review of the licensee's actions will be documented in a future inspection report.

9.0 Licensee Reevaluation of Nuclear Complaints and Employee Concerns Procedures

The inspector discussed with the licensee the preliminary reevaluation of handling of nuclear concerns within the licensee organization. This item was discussed in routine inspection report 50-336/88-07. The concern for the content and/or management of programs to address employee concerns.

The existing program for nuclear concerns and employee grievance is outlined in ACP 1.14A, "Nuclear Complaints and Concerns," and in ACP 1.14, "Employee Complaints and Grievances." ACP 1.14A establishes a method for employees to register nuclear complaints and concerns, and describes the protection afforded to employees who pursue these complaints or who provide information to the NRC per 10 CFR 50.7. This procedure encourages employees to resolve their complaints or concerns with their immediate supervisor. If an employee is not satisfied with progress or resolution at any time, he may deal directly with senior licensee management, the Nuclear Review Board (NRB), the Nuclear Review Team, or the NRC. ACP 1.14, "Employee Complaints and Grievances," documents a licensee method for employees to register complaints or identify problems concerning working conditions, interpretation of policies or procedures, or any disciplinary action.

The licensee is currently investigating methods to restructure the program dealing with employee nuclear concerns and complaints. Proposed actions to improve this program include: stronger emphasis to contract workers on "who" to contact and "how" to address a nuclear concern, simplification of the procedures for the prospective alleger, and improved feedback to allegers. The licensee intends to dedicate a specific department within the organization to deal with concerns and provide appropriate "visibility" to the concern within the organization. Implementation of this program to licensee employees will be through either departmental meetings or bulletin boards. The inspector has no further questions at this time. This program will be reviewed further in future inspections.

10.0 Safety Issues Management System (SIMS) Items

10.1 License Amendment No. 116 - RCP Operation

On July 23, 1986, the NRC issued Generic Letter (GL) 86-13, "Potential Inconsistency Between Safety Analyses and Technical Specifications." The conclusions of GL 86-13 indicate that the TSs may not provide sufficient restrictions to assure that, should a continuous control rod bank withdrawal occur from subcritical conditions, the consequences are within those predicted by the safety analysis. This conclusion is based upon a Westinghouse safety analysis comparison with plant TSs. The comparison showed that a full complement of reactor coolant pumps RCPs are permitted to be operating at zero power while the safety analysis assumes that all RCP's are operable. Under such conditions, the departure from nucleate boiling ratio (DNBR) criteria demonstrated in the safety analysis might not be met in the event of a continuous control rod bank withdrawal.

Licensee letter dated November 4, 1986 submitted LER 86-010-00 which reported that the conclusions of GL 86-13 were valid for Milestone Unit 2. The licensee committed to provide an administrative control for the control element drive mechanisms (CEDMs) to assure that they are deenergized when less than four RCPs are operating; thus, a continuous control bank withdrawal from subcritical conditions would be prevented.

License Amendment No. 116 was issued on April 21, 1987 and contains two changes to the TSs.

-- TS 3/4.1.3.7, "Control Rod Drive Mechanisms," requires the CEDMs to be deactivated in Modes 3, 4, 5, and 6 when RCS boron concentration is less than the refueling concentration. The CEDMs may be energized for MODE 3 as long as four reactor coolant pumps are OPERATING, the reactor coolant system temperature is greater than 500 F, pressurizer pressure is greater than 2000 psia, and the high power trip is operable.

-- The operability and surveillance requirements for the Reactor Protection System - Power Level High Trip (TS 3/4.3.1, "Reactor Protection Instrumentation") is extended from Modes 1 and 2 to Modes 1, 2, and 3. The Power Level High Trip would terminate a continuous control rod withdrawal.

The inspector reviewed plant procedures which implement these TSs. The procedures were found to properly reflect the TS requirements. In addition, a sample of completed data was reviewed and confirmed acceptability of results and proper inspection intervals. Based upon the above, the inspector concluded that License Amendment No. 116 has been properly implemented at Millstone Unit 2. The inspector had no further questions on this matter.

10.2 <u>License Amendment No. 114 - "Spent Fuel Pool Temperature"</u>

License Amendment No. 114 was issued on December 19, 1986. Its purpose was to provide Technical Specifications (TSs) to assure that the spent fuel pool temperature will not exceed 140 degrees F for the purpose of (1) preventing degradation of the spent fuel pool demineralizer resins and (2) maintaining spent fuel pool area temperature and humidity within limits for personnel comfort. The spent fuel pool temperature is limited by the TSs in two ways:

-- TS 4 9.3.2 requires periodic verification that at least two trains of spent fuel pool cooling are operable.

-- TS 4.9.3.3 requires that the most recently discharged 1/3 core offload be decayed at least 504 hours prior to entering Mode 4. This requirement allows continued use of shutdown cooling for spent fuel pool heat removal.

The inspector identified the TS requirements and the plant procedures which implement the TS. The procedures were reviewed to assure accurate implementation of the TS; in this regard, no discrepancies were noted. In reviewing completed data forms, the following was noted:

-- The indications associated with SP-2614A show that TS 4.9.3.2 was properly implemented with regard to requirements and frequency.

-- SP-2614E was not implemented during the Winter 1988 refueling outage. However, the TS requirement was met and was logged in the shift supervisor's log as being completed. The licensee has since issued SP-2614E for TS 4.9.3.3, and SP-2614 is referenced in procedure ACP-QA-9.02B, "Unit 2 Surveillance Master Test control List." The inspector noted that a long time was taken to issue the procedure following the license amendment.

The TSs associated with License Amendment No. 114 appear to have been properly implemented. At the exit meeting with the licensee, the inspector discussed timeliness of procedure changes to reflect licensee TS amendments. The Millstone Unit 2 Superintendent acknowledged the inspector's comments.

11.0 Committee Activities (40700)

The inspector attended Plant Operations Review Committee (PORC) meetings 2-88-111, 2-88-112, 2-88-114, 2-88-117, and 2-88-118 of the Plant Operations Review Committee (PORC) on May 18, June 1, June 8, and June 9. The inspector noted by observation and from the written record that committee administrative requirements were met for the meetings, and that the committee 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. No inadequacies were identified.

12.0 Licensee Event Report (LER) Review (92700)

A Licensee Event Report submitted during the period was reviewed to assess LER accuracy, the adequacy of corrective actions and compliance with 10 CFR 50.73 reporting requirements, and to determine if there were any generic implications or if further information was required. The LER reviewed was: 88-008, "Unrecoverable Dropped CEA," on April 8, 1988. The inspector reviewed the contents of the LER in routine inspection report 50-336/88-07. No inadequacies were noted.

13.0 Review of Periodic (90713) and Special (92700) Reports

Upon receipt, periodic and special reports submitted pursuant to Technical Specifications were reviewed. This review verified that the reported information was valid and included the NRC required data, and that test results and supporting information were consistent with design predictions and per-

formance specifications. The inspector also ascertained whether any reported information should be classified as an abnormal occurrence. The following reports were reviewed:

-- Monthly Operating Report for Millstone Unit 2 for April, 1988
-- Monthly Operating Report for Millstone Unit 2 for May, 1988
-- Millstone Unit 2 Start-Up Test Report

The inspector also reviewed the licensee's Cycle 9 Startup Physics Test Report documented in accordance with TS 6.9, "Reporting Requirements." The results were summarized in a technical document titled "Millstone Nuclear Power Station, Unit No. 2, Startup Test Report for Cycle 9" and attached to the licensee's letter dated April 20, 1988. The test report contains an introduction and a summary of results for low power physics testing and power ascension testing.

The purpose of the Startup Physics Test Program is to verify that the measured parameters comply with TS limits, and to validate core design calculations. The content of the test program was compared and found it to be consistent with the FSAR and TS. The results of the startup test program were checked to assure compliance with the TS and test program acceptance criteria. All measured parameters met the TS and test program acceptance criteria except as noted below.

The Hot Zero Power (HZP), All Rods Out (ARO) Moderator Temperature Coefficient (MTC) were measured to be 0.502 X E-4 delta K/K/F. This measured value exceeds the TS limit on MTC of 0.500 E-4 for power levels less than 70% power. However, this is consistent with TS 3.10.2 which allows suspension of MTC TS requirements during physics testing. The licensee established administrative restrictions to assure MTC TS compliance following the conclusion of the start-up program.

In summary, the Start-up Physics Test Program and results were reviewed and found to be acceptable. Report submittal was in accordance with the Section 6.9 of the TS. No deficiencies were noted.

14.0 Management Meetings (30703)

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. One piece of written material was given to the licensee during the inspection period as found in Appendix A.

APPENDIX A

LIST OF FACILITIES POTENTIALLY AFFECTED BY GAMMA-METRICS 10 CFR 21 REPORT

PLANT	CUSTOMER NAME
ARKANSAS NUCLEAR ONE Unit 1 & 2	Arkansas Power & Light
BEAVER VALLEY 2	Duquesne Power & Light
BRAIDWOOD Units 1 & 2	Commonwealth Edison Co.
BYRON Units 1 & 2	Commonwealth Edison Co.
CALLOWAY 1	Union Electric Co.
CATAWBA Units 1 & 2	Duke Power Co.
COMMANCHE PEAK Units 1 & 2	Texas Utilities Gen. Co.
D.C. COOK Units 1 & 2	Indiana & Michigan Electric
CRYSTAL RIVER Unit 3	Florida Power Corp.
DIABLO CANYON Units 1 & 2	Pacific Gas & Electric
FARLEY Nuclear Units 1 & 2	Alabama Power Co.
FORT CALHOUN Station	Omaha Public Power Dist.
INDIANA POINT III	New York State Power Authority
MAINE YANKEE	Central Maine Power Co.
MCGUIRE Units 1 & 2	Duke Power Co.
MILLSTONE Units 2 & 3	Northeast Utilities Co.
MONTICELLO	Northern States Power Co.
NORTH ANNA	Virginia Electric Power Co.
PALISADES	Consumer Power Co.
POINT BEACH	Wisconsin Electric
PRAIRIE ISLAND Units 1 & 2	Northern States Power Co.
H.B. ROBINSON	Carolina Power & Light
ST. LUCIE Unit 1	Florida Power & Light Co.
SALEM Gen. Station Units 1 & 2	Public Service Electric & Gas
SAN ONOFRE Units 2 & 3	So. California Edison Co.
SEABROOK Station Units 1 & 2	Public Serv. of N.H.
SEQUOYAH Units 1 & 2	TVA
SOUTH TEXAS	Houston Light & Power
V.C. SUMMER Unit 1	So. Carolina Electric & Gas
SURRY Units 1 & 2	Virginia Electric Power
SUSQUEHANNA Units 1 & 2	Pennsylvania Power & Light
THREE MILE ISLAND	General Public Utilities
TROJAN Nuclear Plant	Portland General Electric
TURKEY POINT Units 3 & 4	Florida Power & Light Co.
WNP-1 and WNP-2	Washington Pub. Pwr. Sup. Sys.
WOLF CREEK 1	Kansas Gas & Electric

APPENDIX B

FOLLOW-UP OF OTHER ALLEGATIONS NOT SPECIFIC TO MILLSTONE UNIT 2

1. 88-A-0005, Expired Fire Watch Qualifications

On January 13, 1988 a contractor employee approached the inspector concerning the fire watch program. The alleger stated two specific concerns:

-- Individuals who serve as fire watches have expired qualifications.

-- In one case a fire watch did not have security access to all areas to which he was assigned.

The inspector reviewed the training fire watch program on February 1, 2, and 10 in response to the alleger's concerns. The program adequately addressed the roles and responsibilities of fire watch personnel (roving, and continuous watches). In review, random checks of fire watch qualification (dates and expiration) throughout the Millstone Station were conducted. No cases of expired fire watch qualifications were found. The alleger did not recontact the inspector as was planned, so no additional information or feedback was provided. This allegation was unsubstantiated by inspector review.

The inspector randomly selected assigned firewatches to review security access authorization and compared it to areas covered by the fire watch. No inadequacies were noted.

In conclusion, the inspector's review resulted in an unsubstantiated allegation; however, this matter was referred to the licensee for appropriate consideration. This matter is considered closed without more specific information on qualification inadequacies being needed.

2. Truck Trailer Marked as Radioactive

At 11:15 a.m., on June 10, a citizen from Niantic, Connecticut called the inspector concerning radioactive markings on a truck trailer located in downtown New London.

At 11:30 a.m., the inspector requested assistance from the licensee's health physics organization to investigate the alleger's concern. The licensee agreed to support the inspector with a qualified radiological technician and appropriate instrumentation. The inspector, with assistance from the licensee, evaluated the internal surfaces of the truck trailer for contamination and radiation levels and found no measurable indication. The inspector returned the call to the alleger on June 13, to inform him of the results. The alleger stated that he was satisfied and had no further questions. This item is closed.

3. 87-A-0113, Contractor Work Activities

This section addresses the issues raised by a former contractor employee (alleger) regarding the control of work activities by a contractor at the site (Contractor A). The alleger's employment at the site ended in the Fall of 1987. The alleger subsequently contacted the NRC by phone in the Fall of 1987 to discuss concerns about electrical tagging (see item B.3.6) and security lighting (see item B.3.3). The alleger also corresponded with the licensee in the Fall of 1987 to discuss concerns about electrical tagging. That concern involved two instances during work in an onsite warehouse that was being converted to Unit 2 maintenance offices. The alleger requested a meeting with the licensee. The alleger contacted the NRC by phone in December 1987 to set up a meeting.

The issues discussed below were received during a January 13, 1988 meeting with the alleger. These issues were referred to licensee management for addressal. The licensee had initiated action on the the alleger's concerns prior to NRC involvement and, based on audit findings, had initiated corrective actions. The status of actions and licensee findings on all issues were periodically discussed with the inspector by the licensee, and were summarized in an May 19, 1988 memorandum to the Station Services Superintendent. Supplemental information on the original issues was obtained during various follow-up telephone conversations and during a follow-up inspector meeting with the alleger on June 8, 1988. The licensee contacted the alleger to set up a meeting, which was scheduled after the end of the inspection period.

3.1. Issue: Installation of 16 Pole Lights in the Simulator Building Parking Lot in July 1987

This issue concerned the adequacy of lighting installed without anchor bolt templates, in the simulator building parking lot.

Licensee review found that installation of the mounting bolts without a template was acceptable. The Contractor A Superintendent, a certified professional engineer - civil discipline, used engineering judgement to direct installation of the mounting bolts. A template was not required based on discussions of the installation with the light manufacturer, who agreed that installation of 4 bolts on a 10 inch bolting circle was acceptable. The licensee concluded that no further actions were required and that no corrective actions were warranted. The inspector observed the parking lot lights and noted no obvious inadequacies with the installation.

The licensee stated that the alleger was terminated because of the dispute the Contractor A Superintendent over the use of the template and after removing the mounting bolts before the cement was poured. The alleger was reinstated after discussions with Contractor A upper management. The licensee considered this to be an internal Contractor matter and concluded no further action was required.

The inspector noted that this issue and its resolution had no impact on nuclear safety. The inspector identified no inadequacies in the resolution of this issue.

3.2 Issue: Coordination of Work Activities to Remove Temporary Security Lighting

This issue concerned alleged poor control of work activities as evidenced by a job where two contractor groups were assigned to do the same work.

Licensee review determined that two onsite contractor groups were both assigned to work on temporary security lighting adjacent to a warehouse onsite. The job was first assigned to Contractor B, who usually works lighting jobs. The start of work by this group was delayed. The alleger's employer, Contractor A, was then assigned to do the work by the Station Services Engineering Department at the request of Security. While preparing to do the work, Contractor A personnel noted that Contractor B was doing the job; the work order to Contractor A was then cancelled.

The problem of work coordination originated within the Security Department. Licensee actions were addressed in a memorandum from the Station Services Superintendent to the Security Supervisor dated February 1, 1988, and from the Quality Services Supervisor to the Station Service Superintendent dated May 19, 1988. Security was cautioned to avoid duplicate work assignments in future jobs.

The inspector noted this issue and its resolution had no impact on nuclear safety. The adequacy of lighting in the protected area has been previously reviewed by the NRC and noted discrepancies were resolved (NRC Region I Inspection Report 50-336/87-20). No inadequacies were identified in the resolution of this item.

3.3 Issue: Worker Whole Body Counts and Termination Exposure Reports

This was a new allegation raised by the alleger on April 25, 1988. This item involved two individuals who worked at the Millstone Station in the 1983-1984 time period. Both workers reportedly stated to the alleger that they terminated employment at the site without obtaining an exit whole body count, and without getting termination exposure reports from the licensee. The inspector asked the alleger to provide the name and address of both individuals to allow further followup. After conferral with both workers, the alleger identified Individual A, who wore a respirator while onsite and who also had a known cesium uptake. No information was provided on the second individual. Individual A reportedly did not want to talk directly with the NRC and no address was provided. This item was turned over to the licensee for review and evaluation.

The licensee provided, for NRC review, exposure history files on Individual A. These showed that he worked at the site periodically from 1974 until 1983, with the last work date onsite being 8/15/83. The licensee stated that Individual A worked at the station during outages on Units 1 and 2 and produced termination exposure reports corresponding to each work period. The licensee noted that the 7 termination reports were addressed to 5 different addresses used by the worker and noted it was possible the last report was not received at the last address on file or forwarded in the mail. The licensee stated that another copy of the October 1983 termination exposure report would be provided if requested by Individual A.

The inspector reviewed the last exposure report dated 10/10/83 and covering the period from 5/13/83 to 8/15/83. The inspector noted that the recorded quarterly exposures for all work periods were low and well within regulatory limits. The inspector noted further that a completed NRC Form 4 dated 5/13/83 was on file and properly reflected the exposure history record. The inspector also noted that the licensee's health physics records show that Individual A properly completed the prerequisite training, medical screenings, and respirator fit testing needed to wear respirators for work in radiological areas at Millstone.

Licensee records show the results of whole body counts (WBCs) performed for Individual A. The last WBC result on file was performed on 5/13/83, which was the entrance count usually done by the licensee as a matter of policy (and not because of an NRC requirement). The licensee's records confirmed that Individual A left the last work assignment at the station without a termination WBC. The licensee concluded, from his review of the exposure history and work activities, that a termination WBC was not needed to meet regulatory requirements based on records that show no airborne exposure time (MPC-hours) were accumulated by the individual. This item is discussed further below.

The inspector noted that radioactive potassium and Cesium-137 were reported in the WBC results for 7/23/80 and 3/7/81. Radioactive potassium is naturally occurring and is found in all people. Cesium-137 is not naturally occurring and is produced by nuclear fission. The inspector noted that the cesium levels recorded in 1980 and 1981 were 0.315 nanocuries (NCi) (+) or (-) 3.092 NCi (at two standard deviations) in 1980; and 1.792 (+) or (-) 2.437 NCi in 1981. At these levels, the isotope was present in trace amounts, just at the limits of detectability and far below the action level for follow-up investigation. The inspector noted that subsequent WBC results recorded on 11/6/81, 3/26/82 and 5/13/83 did not show any cesium. The inspector did not determine the source of the uptake. The inspector noted that the licensee has no record of an incident report involving Individual A. Thus, the cesium uptake does not appear to be related to work at Millstone.

The licensee provided for inspector review all radiation work permits (RWPs) used by Individual A for work in the Unit 1 and Unit 2 radiologically controlled areas during the period from May 13 - August 15,

1983. The inspector reviewed the reports along with the health physics survey results showing the radiological conditions in the work areas of interest. The survey results for airborne activity were reviewed in particular, as recorded in Air Activity Logs for dates corresponding to the RWPs. The surveys showed that airborne radiological conditions did not require the use of respirators because of pre-existing concentrations at the work site. Air activity results were at or below the 3.0E-9 uCi/cc MPC limit specified in 10 CFR 20 for unidentified isotopes in restricted areas.

The inspector noted from the RWP records that most work activity by Individual A in the Millstone 2 containment involved walkdowns, inspections, and installation of cables and conduits for neutron detectors and resistance-temperature-detectors. Some of the RWPs did require the use of respirators and Individual A indicated respirators were used. The use of respirators in these instances appears to have been a precaution to protect against possible ingestion of radioactive material made airborne during the course of work such as during core drilling activities on containment walls with general area contamination levels in the range of 50k-200k disintegrations per minute (dpm) per 100 sq. cm. Based on the radiological conditions at the job sites worked by Individual A, and considering that licensee records show that no MPC-hrs were recorded for Individual A, the inspector concluded that a whole body count was not required by NRC regulations. Specifically, no whole body count was necessary as part of the bioassay assessments required by 10 CFR 20.103(a)(3).

The inspector reviewed Station Procedure SHP 4970, Internal Exposure Control (Bioassays), Revision 4 dated 4/22/86, which establishes the licensee's bioassay program and sets the criteria under which WBCs will be performed. SHP 4970 requires a WBC if it is determined that a limit of 40 effective MPC-hours is exceeded for a worker. This is consistent with 10 CFR 20. Additionally, SHP 4907 requires "routine" WBCs for all personnel issued dosimetry at the site at the start and end of employment, and at least once per year. These whole body counts are (in part) a screening measurement used to validate the adequacy of other controls established for work in radiological areas. The performance of such WBCs is a licensee administrative practice that exceeds NRC requirements. Based on the above, even though SHP 4907 was not met in this instance, there is no safety significance and there is no violation of NRC requirements. The licensee stated he would perform a WBC on Individual A if he requested one. The alleger was requested to relay this information to Individual A.

The licensee stated that, since the reported cesium contamination was below their investigation threshold of 20 nanocuries, no further follow-up would have been taken when it was first noted. The licensee stated that all contractors are made aware at the start of employment that it is expected workers will get a whole body count upon termination. There is no mechanism in place to enforce this requirement in all cases. The

licensee estimated that about 5% or or fewer workers leave the site without and exit WBC. The licensee feels this is acceptable since, as an assessment tool, the 95% of the workers who do get the termination whole body counts confirm the success of the respiratory protection program. The licensee stated that all persons are evaluated per 10 CFR 20.103(a)(3) when required. The inspector reviewed the inspection record for all three Millstone units and noted that recent NRC Region I reviews of the internal exposure controls have found the program to be acceptable. Minor deficiencies have been noted, but no inadequacies have been identified in the respiratory protection and bioassay assessment function.

During a June 8 followup meeting with the inspector, the alleger provided additional information about Individual A. The alleger stated that, during Individual A's last work assignment at the site, he was working in containment without a respirator when plant operators turned on a fan which caused contamination to be blown on the workers. The alleger had no further information from Individual A as to whether Individual A was found to have contamination on his clothing or skin, or whether he showed positive counts on a nasal smear.

This matter was again reviewed with station HP personnel, who stated that an incident report would have been written for any instance involving skin contamination or a possible intake of radioactive material. No incident reports involving Individual A are on file. The inspector could not pursue this matter further without additional specific information directly from Individual A. The inspector sent messages to Individual A via the alleger. Individual A had not contacted the inspector as of June 13, 1988. Based on the above, the inspector concluded no further NRC actions are warranted on this matter.

3.4. Issue: Category I Welding by an Unqualified Welder

This issue involved a concern that Contractor A, who employed the alleger, was using an electrician to perform Category I welding inside the containment. No other specific information was provided as to the name of the worker, the plant involved, or the name or type of plant systems affected. Unit 2 was assumed to be the affected unit since that unit was in an outage at the time the information was provided. The inspector reviewed activities in the Unit 2 containment on January 23, 1988 for welding by Contractor A. None was identified. The issue was referred to the licensee for action. Subsequent routine inspections of Unit 3 and Unit 2 outage activities did not identify welding by Contractor A. Licensee follow-up identified information which partially corroborated the alleged concerns, but did not show safety significance.

Contractor A does not have a Quality Assurance program and is not used by the licensee to perform nuclear safety-related work. The licensee developed a list of work assigned to Contractor A at all three units since October 1986. Until recently, most work by Contractor A was performed for the Station Services Engineering (SSE) Department, whose re-

sponsibilities include non-nuclear projects such as grounds maintenance, and maintenance, installation, and modification of outlying buildings. Contractor A was also used in the May - August 1987 period to install and modify gates for high radiation areas (HRAs) in all three units, as discussed further below. During the 1987 - 1988 outages, Contractor A was used as a general laborer force under plant personnel supervision to (i) move shield blocks around reactor components - Unit 2 containment; and (ii) perform material accountability control in reactor work areas - Unit 3 containment. No Category I work has been assigned to Contractor A and no safety class welding has been performed by its employees.

Contractor A carpenters and electricians have welded in non-Category I work activities. Qualifications for electricians were based on welding training obtained during union apprenticeship. Also, Contractor A carpenters welded stiffeners on HRA gates to provide added reinforcement, when needed. That work was completed within the plant buildings and was controlled by work orders. One Contractor A electrician performed two non-Category I welds outside plant buildings: one job involved relocating a security intrusion detection system; the second involved mounting a microwave antenna on the side of the condensate polishing facility. The licensee concluded that completion of these welding activities in accordance with general construction work practices was acceptable. No further NRC actions are deemed warranted or are planned on this matter.

The inspector reviewed station administrative requirements to determine whether any controls should have been applied to the work. This review included specific consideration of ACP-QA-2.03A, Non-Category I welding. The identified welding activities did not involve ASME or ANSI code work; the work could not affect Category I systems; and the work did not involve an activity for which a completed weld documentation package would be required or desired. Based on the above, the requirements of ACP-QA-2.03A were deemed not applicable to welding Contractor A performed. No inadequacies were identified.

The inspector reviewed the licensee's actions on this issue and identified no inadequacies. Inspector review of HRA gates during routine inspections in all three units has found the gates to be sufficiently strong to provide an adequate barrier. No inadequate conditions were identified.

3.5. Issue: High Radiation Area Gate Modifications

This issue concerned the controls applied to work done to modify and install alarms on about 70 high radiation area gates in all three units. The alleger was responsible for completing the work in the March-August 1987 time period. The alleger stated that an inexperienced engineering technician, newly hired by the contractor, was assigned responsibility for the job. That individual's designs were reportedly aborted after about two weeks work and after working on two gates. The job was al-

legedly performed based on "blackboard" designs without written guidance and criteria. Problems reportedly included lights that were not needed and a $6,000 restock charge when another type of light was selected, and the use of alarm bells that ran on 24 vac and required the use of step down transformers. This item was referred to the licensee for review. The inspector noted that the High Radiation Area (HRA) gates and associated alarms are not nuclear safety-related systems or components. The inspector asked the licensee to specifically address whether installation work associated with the design not used resulted in radiation exposures which could have been avoided if better guidance had been provided.

The inspector reviewed the results of the licensee's review of this issue, independently reviewed the detailed design change package under which the work was accomplished, and interviewed the individual responsible for completion of the modifications as the designated plant project engineer (PE).

The work involved the modification of existing gates and the installation of new gates that provided barriers to the entrance to designated high radiation areas in the three Millstone units. The work was controlled with detailed guidance provided in Automated Work Orders and in the following plant design change requests (PDCRs):

-- PDCR 1-105-86, HRA Gate Alarms & Warning Lights (MP1)
-- PDCR 2-004-87, HRA Gate Alarms & Warning Lights (MP2)
-- PDCR MP3-87-002, HRA Gate Alarms & Warning Lights
-- PDCR MP3-86-372, HRA Wire Mesh Gates

The licensee is required to control access to high radiation areas as defined by 10 CFR 20 and to lock the access ways if the dose rates involve radiation levels in excess of 1 Rem/hr. The main purpose of the PDCR was to add warning lights and audible alarms to existing and new gates to notify personnel that the gates were not properly secured (locked closed) after passing through the gates. This action was in response to licensee and NRC-identified concerns that the HRA gates were being left open following access to the rooms. The PDCR was also used to add gates to new areas based on surveys by health physics personnel, and to stiffen existing gates. In addition to the guidance provided by the PDCRs, the following references also provided written guidance on gate fabrication, modification and installation:

-- Stone & Webster Specification 2199.241-932, Specification for Wire Mesh Doors

-- NUSCO Electrical Installation Specification SP-EE-076

-- Field Sketches LPKA 120286 and LPU-B 120286

-- Various AWOs for installation activities during May-August 1987

The design changes described in the PDCRs were prepared by an engineering technician in the Station Services Engineering (SSE) Department and were reviewed and approved by the licensee's engineering staff for each unit as required by station administrative procedures. The review by the unit staff and Plant Operation Review Committees determined that the change, while intended to improve compliance with the requirements of 10 CFR 20.203(c)(2) and Technical Specification 6.12.1, did not involve an unreviewed safety question per 10 CFR 50.59 or adversely affect the operation of safety systems or structures. Power for the circuits would be provided from non-class 1E supplies and the gates would not impact seismic walls. The function of the alarm circuit was to provide audible and visual indication that the HRA gate was open longer than 10 seconds (the time delay allowed for normal transit without alarms). An override switch was provided to bypass the alarm function for periods when the gate would be left open for extended periods to allow movement of materials. The alarm would also activate, however, if the function switch was not returned to the "auto" position when the gate was closed.

The licensee determined that the original PDCR design was developed in late 1986 with the intention of using explosion-proof alarm lights similar to those already installed on existing gates. The explosion proof lights were ordered from funds available in the 1986 operating budget. The initial circuit design was developed to use, to the extent possible, components already available in station stores, including transformers, 24 VAC alarm bells, and aluminum shield (ALS) cable. The ALS was chosen to minimize worker radiation exposure since it would allow installation of the circuit without conduits. Additional bells and relays were ordered as necessary. As the design change proceeded, it was concluded that the explosion proof lights were not needed and strobe lights were ordered instead. This action was taken even though there was a $6294 restock charge on the explosion proof lights, because there was still a net savings in excess of $10,000. Since the explosion-proof lights were never installed, there was no additional exposure required for the job as a result of the change.

The work was originally assigned to Contractor B. The licensee determined that the work was proceeding too slowly and the Station Services Engineering Department reassigned the job to Contractor A.

The licensee determined that the original circuit design as described in the approved PDCRs would work. Proper functioning was demonstrated by construction and bench testing of a test circuit in the shop. However, contract electricians (including the alleger) recognized that improvements in the circuit design would reduce the number of cable terminations needed and would result in less ALS cable being installed (estimated at 3 to 12 feet per gate) and thus reduce the work time required in radiation areas. Even though the gates controlled access to HRAs, the typical work area for the gates was the immediate area of the gate and the nearby electrical panel, which were not in "high radiation areas." (Personnel exposure required to do the job is discussed further below.) The modified circuit in its final form included the use of an additional

instantaneous relay to replace one contact from the gate closure limit switch proposed in the original design. Inspector interviews with the project engineer determined that he proposed several interim configurations that were found unacceptable during follow-up reviews with the alleger. Ultimately, the final circuit design chosen was the one proposed by the alleger. The licensee stated that the selection of the final design was left to the discretion of the Station Services Engineering Department, since unit engineering had determined that circuit changes were considered minor in scope and would not impact the conclusion of the 10 CFR 50.59 safety evaluation.

The licensee reviewed the radiation exposures for the job and concluded they were not excessive. No exposure was incurred on Unit 3. The work on Unit 1 expended 0.885 man-rem for 264 man hours, for an average dose rate of 3.3 mRem/hr. The work on Unit 2 expended 0.710 man-rem for 236 man hours, for an average dose rate of 3.0 mRem/hr. A tabulation of individual exposures for all contractor personnel who worked on the job (which included exposures for all work during the period and not just the HRA gate job) showed personnel exposures were not excessive. Inspector review of the tabulation noted that individual quarterly exposures were less than 250 mRem in all cases, except for one individual with a maximum quarterly exposure of 440 mRem. The licensee concluded, based on radiation work permit (RWP) records for the installation of the first few gates, that exposures were not excessive. Further, the licensee's administrative exposure limits were not increased for any worker during the job. Inspector review found no inadequacy in the licensee's conclusion.

The licensee further concluded that, in spite of the additional dose savings realized in going from the original to the final circuit design, reasonable measures were taken to minimize radiation exposures for the design change. These measures included use of ALS cable instead of conduits to reduce installation time, selection of power supplies to minimize time spent in radiation areas, testing the preliminary design in the shop and prefabricating and testing materials as much as possible in the shop to minimize installation time in radiation areas, and licensee supervisory monitoring of work progress and reassigning the job to another contractor when the first contractor was deemed unacceptably slow. The inspector identified no inadequacies in the licensee's findings.

The inspector noted there was a difference between the licensee's and the alleger's version of the job. The alleger stated that the critical circuit design did not work and modifications were required on the first two gates modified in the plant. While the inspector did not resolve the different versions, he did note, that based on the low dose expended for the entire job, rework of two gates in the field would not change the conclusion that exposures were not excessive.

Licensee review concluded that the engineering technician assigned as project engineer to the job had adequate experience to perform the work. The technician is a contractor personnel employed by Contractor A and assigned to a staff position in the SSE Department. The engineering technician worked for 3 years during Unit 3 construction and startup for the architect engineer. The technician gained experience in the electrical discipline while working with major electrical components and through involvement in switchgear testing. The technician also worked for 1.5 years in the SSE Department working on projects involving the electrical discipline. The licensee did note that the technician had some difficulty administratively coordinating the setups necessary for the first HRA gate job in Unit 1, which involved allowing sufficient lead time to process tagouts and RWPs so that the work could start on time. This difficulty stemmed from his lack of experience in processing the administrative controls. The licensee concluded this inexperience did not cause unnecessary radiation exposure. The inspector identified no inadequacy with the licensee's conclusion since delays in starting work or in obtaining the prerequisite tagout would not result in radiation exposure.

However, in addition to the above described difficulties with the licensee's administrative controls, the inspector noted that other problems occurred in following the requirements of the tagging procedure, ACP 7.06C, as discussed further in Issue 6 below. While the tagging for the HRA gate jobs was found to be done safely, it was not done in full compliance with the ACP, as follows: (i) operator-in-attendance tagging was performed by contractor electricians on some occasions, which did not meet the requirements of tagging procedure ACP-QA-2.06A; and, (ii) single tags used for multiple gates (7 primary breakers covered 5 gates each) were processed without using the SF 210 continuation form required by the ACP. Further, during interviews with the contractor technician, the inspector noted that the technician had been responsible for the initial preparation of the 10 CFR 50.59 safety evaluation for the PDCRs, but had not had training in the 10 CFR 50.59 process. The inspector found that the contractor had become familiar with the administrative requirements by reading the associated administrative procedures.

In response to inspector inquiry, the licensee stated that contractor personnel are provided on-the-job training in the station administrative requirement as needed, and that this training involved reading of the associated administrative procedures. That was not formally documented. Although the inspector identified no safety issue relative to the HRA job, and no inadequacies were identified in the completion of the PDCR per the requirements of NEO 3.03, the inspector identified this area as meriting further NRC review to determine the general adequacy of training provided to contractor personnel on licensee administrative requirements. Even though no safety-related (Category I) work is assigned to the SSE Department, the inspector expressed the concern that the licensee needs to formally document required training to assure contractor personnel

are fully familiar with all administrative procedures and requirements they are expected to follow. This area will be reviewed further on a subsequent routine inspection as a potential element of licensee performance.

In summary, while followup of this issue did confirm the alleger's statement in part, the licensee concluded that guidance was provided and controls were applied suitable to the activity. Further, reasonable efforts were taken to maintain radiation exposure hours less than the ALARA goal of 1 Rem per unit. The inspector agreed with the licensee's conclusions. The inspector noted that it is neither unacceptable nor unusual for approved designs to change to reflect improvements identified in the interaction between design and implementing groups during the design change process. No unacceptable conditions were identified during the inspector's review.

3.6. Issue: Adherence to Controls for Electrical Switching

Part A: Exercising 480 volt breakers without tags or AWO.

This issue concerned alleged actions, on one occasion in 1987, by the Contractor A Superintendent to manipulate 480 volt circuit breakers on a warehouse panel in order to trouble-shoot a problem with the air conditioning. This action was taken without a work order or tags to control the activity.

Part B: Moving a warehouse electrical circuit.

On a second occasion in 1987, the alleger processed an AWO and tagging order to remove an electrical wire so as to allow installation of a window in a warehouse wall. The wire was moved instead by station maintenance personnel without tags or an AWO. The alleger was fired during an ensuing argument on the control of Contractor A work activities and, he feels, for following administrative requirements for which he was held responsible. During this incident, the Contractor A Superintendent allegedly stated that he did not come under NU or NRC jurisdiction for following administrative requirements.

These items were referred to the licensee for follow-up and dispositioning and to address the following: (a) assuring work activities are conducted per established procedure controls; and (b) assuring the control of contractors is appropriate and that station policies are followed.

Part A: Licensee review of the first issue concluded that the Contractor A Superintendent operated 480 volt breakers in the Unit 2 warehouse without a tagout, but a tag was not required for the specific activity. The licensee determined that, on August 17, 1987, the Contractor A Superintendent worked with a representative of a local air conditioning (AC) company to investigate a report that air conditioning in the main-

tenance shop, recently converted from a warehouse, was not operating. The new air conditioning unit had been recently connected, in an unrelated action to a 480 volt distribution panel using a plug-in circuit breaker. In this application, the breaker is routinely used as a switch. Other circuits fed from the same distribution panel included welding machine supplies, which were hard-wired to fixed receptacles. With the AC representative at the cooling unit, the Contractor A Superintendent manipulated one circuit breaker to turn the AC on. The designated AC circuit had a temporary label. The unit was left on for the remainder of the day and then shut off using the same breaker. The licensee concluded that the actions by the Superintendent were proper since no "work" was done on live circuits and no tagout was required or needed. The inspector reviewed the statement of applicability for ACP-QA-2.06A, the licensee's station tagging procedure, and identified no inadequacies.

The inspector noted that there is a difference between the alleger's and the licensee's version of the activity. The alleger stated that the status of the air conditioning unit was not known by the the Contractor A Superintendent when he operated 6 or 7 breakers in an attempt to start the unit. The inspector could not resolve the different versions. The inspector contacted the alleger to obtain any additional specific information that shows activity on August 17th involved "work" on live circuits. No additional information was available to resolve the differences. Inspector review of this matter did not substantiate the alleger's input.

Part B: Licensee review of this issue was documented in memoranda dated 10/28/87, 11/5/87 and 5/19/88. The licensee initiated reviews of this issue in response to concerns raised by the alleger in a 10/16/87 letter to the Station Services Superintendent. The licensee's review confirmed essential facts in the alleger's statement, as follows. On September 8, 1987 carpenters required an electrical wire to be moved in order to install a window in an exterior wall of the maintenance shop. The alleger, acting as Contractor A electrical foreman, obtained a work order (M2-87-10030) and "blue" safety tags (clearance 1569-87) to perform the work - acting apparently independent of direction from the superintendent. The tags were hung on Thursday, September 10 to do the work on Friday, but the work was done on Wednesday by NNECO maintenance personnel. The work was done by NNECO after the Contractor A Superintendent suggested the utility could take the required actions at a cost less than what would be charged by Contractor A. The wire/conduit was moved by NNECO personnel without an authorized work order and by using a modified version of the "operator-in-attendance" controls specified in tagging procedure ACP-QA-2.06A. (This item is discussed further below.) The action to move the wire was completed before the alleger's tags were hung on September 10. When the alleger learned on Friday, September 10 that the work was already done, an argument occurred with the Contractor A Superintendent on the control of work. During this meeting, the Contractor A Superintendent terminated the alleger for insubordination and for

charging time for electricians who did not participate in the job since the only work activity completed was by the alleger to process the AWO and tags.

The licensee determined that maintenance personnel completed the job on September 9 in about 30 minutes using 3 workers to control the breaker on Warehouse 4 Lighting Panel 2B. This power source is fed from the non-safety related Flanders line. Workers were posted at the breaker and within line-of-sight of the work area and the breaker to ensure the circuit remained de-energized while the conduit was removed and the wire was moved down to allow installation of a window. The actions taken met the intent of "operator-in-attendance" tagging permitted by Section 6.1.9 of ACP-QA-2.06A, but Tag Log Sheet SF-210 was not used as required by the procedure. The inspector noted that information recorded on SF-210 would have included identification of the equipment covered in the order, its location, and applicable work order. No other specific information on the required position of the breaker would have been required. The inspector noted further that had the job been covered by Tagging Order 1569-87, as initiated by the alleger, Blue Tags would have been used (which essentially releases the equipment to the person responsible for the work) and would have allowed the breaker to be positioned "as required" by the work party leader. Thus, in the controls effected by either the Blue Tag or operator-in-attendance methods, the desired position of the breaker is left to the discretion of the work party.

In response to the alleger's concern, the licensee had the onsite Industrial Safety Department (ISD) review the actions taken in this particular situation. That review concluded that safety was not compromised and that the assumed risk in the operation was reasonable and controlled. Since ACP-QA-2.06A was written primarily for the control of inplant equipment, and since less stringent controls than are required by the ACP may be desirable for work in outlying plant areas, the ISD recommended consideration be given to modifying administrative procedures for non-plant system maintenance work to recognize the method used. To this end, the Station Superintendent issued controlled routing 6926 to address tagging in outlying buildings. Actions to draft a new procedure were in progress at the end of the inspection period.

Notwithstanding the above conclusion on the safety of activities on September 9, the licensee concluded the actions were not completed as required by station procedures. Corrective actions were taken by the Station Superintendent in a memorandum (MP-11440) issued to the station on January 29, 1988 which reemphasized the need to follow the requirements of ACP-QA-2.06A as the only currently acceptable process for safety tagging. The inspector identified no inadequacies in the licensee's conclusions or corrective actions.

The licensee evaluated the performance and statements made by the contractor superintendent to assess his attitude toward complying with station procedures and policies. The licensee concluded that the superintendent accepts the need to comply with procedures and appears to clearly differentiate between his authority as superintendent over craft personnel and his relationship to NNECO procedures and supervisory oversight. The licensee's conclusions were confirmed by the inspector in an independent interview with the contractor superintendent.

To further address contractor adherence with administrative controls, the Quality Services Department performed an audit of Contractor A activities. The results were reported in QSD Report QSD-88-4243 dated 2/4/88. The audit addressed, in detail, 119 of the 269 work orders involving Contractor A and issued from 1/1/87 to 1/27/88. The audit showed that: the contractor did not perform work on safety-related equipment, and that the contractor worked on non-safety-related equipment under direction of the Station Services Department with the approval of the Operations Department. Also, for 66 work orders, no tags were used when tagging would apparently have been required. Instead, work in all three units on the high radiation area gate modifications was performed with an improper use of "Operator-in-attendance" tagging where an electrician assured proper positioning of the MOV feeder circuit breaker at a panel in the vicinity of the work site, but no SF 210 was used. While control of the work was maintained, the method used did not meet the requirements of the ACP. The licensee also determined that, in some cases where a work order indicated no tags were used, one tagout for one gate was used for work on another gate when a common power supply for the alarms was involved. Minor administrative problems noted on the completion of the work order forms included not initiating revisions to tagging requirements listed as undetermined, and one work order with the tag "verification" and "cleared" blocks was not signed off.

Licensee review attributed the tagging problems to improper implementation of the administrative requirements based, in part, on a misinterpretation of the work order format and misunderstanding of administrative requirements. Licensee management reviewed the work activity associated with the high radiation area (HRA) gate job and concluded that proper administrative controls were generally used and all work appeared adequately controlled. However, further actions were deemed necessary to improve contractor understanding of administrative requirements and oversight of contractor activities by NNECO personnel.

Additional licensee corrective actions included supplemental training to be provided by Station Services Engineering (SSE) on the proper use of SF-210 for "operator-in-attendance" tagging. The SSE supervisor, by memo MP-S-GSS-88-6 dated 2/8/88, was directed to improve tagout controls. (Corrective actions documented in a memo dated 2/25/88 included emphasis on operator-in-attendance tagging and on work directions.) The SSE Supervisor held a meeting with contractor personnel to review and clarify tagging requirements and to address in particular the actions taken on

the HRA gate job. Actions were taken to revive the list of SSE personnel authorized to sign work orders, and to limit that function to NNECO personnel effective May 15, 1988. A measure is to be formalized whereby SSE supervision will be assured that contractor personnel perform work in accordance with procedures and policies. Additionally, the Quality Services Department will revise ACP-QA-2.026, Work Orders, to clarify the requirements on the authorizations needed to sign off "tagging verified" and "work completed" sections of the work order forms. The licensee is also drafting a new procedure for the control of tagging operations in outlying buildings (Controlled Routing 6926). The above actions have either been completed, or were in progress with an expected completion date by July 1988. No inadequacies were identified in the licensee's conclusions or corrective action plan, based on independent inspector review and interview of personnel.

In summary, the inspector noted no nuclear or personnel safety concerns for either the specific instance cited by the alleger, or in the HRA gate tagging discrepancies identified by licensee audit. However, licensee actions were appropriate to assure established work control and tagging procedures are both fully understood and followed. The corrective actions represent improvements in the licensee's programs to control work activities by a contractor who performs non-Category I work. These actions are also appropriate to assure that contractor personnel who might become involved in Category I work are knowledgeable of administrative controls. For this reason, the inspector will follow completion of the licensee corrective actions during routine inspection.

3.7 Issue: Control of Contractor Activities

This issue involved the question of whether the contractor superintendent followed station procedures or wielded undue influence on the site based on alleged improprieties between the contractor and a licensee supervisor. The alleged improprieties involved matters that were not related to nuclear safety at the site. The information provided by the alleger was referred to licensee management for follow-up. The inspector asked the licensee to address whether the control of the contractor is appropriate to assure station procedures and policies are followed.

Licensee review determined that oversight and control of the contractor activities is appropriate and that station procedures are generally followed. This conclusion was based on reviews by the lead licensee investigator and on the results of an audit of the contractor activities by the NUSCO Quality Services. The exceptions concerned the noted problems with processing work orders and tagging practices associated with the HRA gate job, and with the job to move the wire in the Unit 2 maintenance shop. Adequate control is assured through active involvement by the licensee supervisor being at the job sites on a daily basis. The licensee supervisor is accountable to NNECO management as evidenced by routine interactions which provide appropriate guidance and management direction on activities in the department. The licensee stated that

concerns about alleged improprieties in the relationship between the contractor and the licensee supervisor were not substantiated. The licensee stated that the statements made by the contractor superintendent regarding the need to follow procedures were made during an argument and were not indicative of the contractor's observed performance. The inspector interviewed the contractor superintendent and noted that the views he expressed regard adherence to station procedures and policies as important, especially in his role as supervisor.

The inspector had no further questions on these concerns.