U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report 50-245/85-10; 50-336/85-13

Docket Nos. 50-245; 50-336 License Nos. DPR-21; DPR-65

Licensee: Northeast Nuclear Energy Company

Facility: Millstone Nuclear Power Station, Waterford, Connecticut

Inspection at: Millstone Units 1 & 2

Dates: March 25, 1985 through May 13, 1985

Inspectors: Ebe McCabe, for 6/7/85
John T. Shedlosky, Senior Resident Inspector Date

J T Wiggins for 6/7/85
Robert A. McBrearty, Reactor Engineer Date
(April 3-4, 1985)

Approved: Ebe McCabe 6/7/85
E. C. McCabe, Chief, Reactor Projects Section 3B Date

Summary: Routine NRC resident (106 hours) and region-based (10 hours) inspection of plant operations, equipment alignment and readiness, radiation protection, physical security, fire protection, plant operating records, maintenance, modifications, surveillance, calibration, and reporting to the NRC. This inspection also included review of Unit 2 Steam Generator inspection and maintenance, Steam Generator secondary chemical cleaning, and Steam Generator primary decontamination. No violations or unacceptable conditions were identified. Unplanned actuations of engineered safety features were noted for further review during evaluation of licensee performance.

DETAILS

1. Plant Status

Unit 1

The reactor operated at full power except for planned power reductions for surveillance testing and preventive maintenance.

A fire occurred within a degreasing unit located in a tool decontamination facility at 0950, April 4. That unit, which uses heated freon as a solvent, apparently boiled dry and ignited the residual grease. The fire remained contained within the degreasing unit. Assistance was requested from the Waterford Fire Department. There were no serious personnel injuries, but fire fighting personnel did receive minor skin burns from airborne acid vapor (see Report Detail 5).

The emergency gas turbine generator is tested monthly. In a test at 1007 on April 30, the unit failed to start. A defective air start valve was the cause. The valve was replaced and the unit was successfully tested at 1830 that day.

Unit 2

The reactor has been shutdown for a refueling/maintenance outage since February 16. There is no fuel in the reactor. While the fuel has been in the fuel storage pool, inspections were conducted to identify assemblies with leaking fuel pin cladding. Nineteen (19) pins in sixteen (16) assemblies were so identified. These will not be used during the next operating cycle.

The secondary side of both steam generators was cleaned chemically to a height of about two (2) feet above the tube sheet. That and high pressure water hydrolazing (sludge-lancing) removed about 1330 pounds of sludge.

The steam generator primary channel heads were chemically decontaminated to reduce the general area radiation fields from 28 R per hour to 1.5 R per hour. The activity which was removed was deposited in ion exchange beds which are to be de-watered and shipped off site. The reduction in radiation levels was accomplished to allow personnel entry into the steam generators with reasonable stay times for installation of sleeves and plugs for tube repair.

2. Steam Generator Tube Inspections - (Unit 2)

In-service inspections are conducted each refueling outage to verify the integrity of steam generator tubes. The primary examination technique is eddy current testing. The program is described in Technical Specification 4.4.5.0. Because pitting corrosion had been discovered during a previous outage, all tubes are inspected to the height of the first tube support plate. Metallic copper in the secondary sludge pile is considered responsible for the pitting corrosion. During the 1983 refueling outage, 2022 sleeves were installed in

pitted tubes. During the current outage, both the hot and cold leg ends of each tube were inspected using techniques designed to detect pits in areas having significant copper interference. A total of 28,372 tube ends were to be inspected. In addition, a combination of sludge-lancing and chemical cleaning was used to remove the sludge pile. Because of scheduling the work activities between the two steam generators, the No. 2 generator cold leg side was examined prior to chemically removing the secondary sludge from that generator. Based on that inspection, 304 sleeves were installed. Eddy current examinations were made following chemical cleaning. A significant number of new flaws appeared and previously recorded flaws showed an increase in measured depth by an average of 40 percent. The entire generator was re-examined and an additional 780 sleeves were installed (cold leg side).

Copper has been known to interfere with the eddy current tests, but the licensee selected test methods, probe coils, and operating frequencies to allow detection and measurement of pits with copper present. Nonetheless, the data now indicates that copper in a pit may completely mask a rejectable defect. Ater chemical cleaning during this outage, rejectable defects were recorded on tubes which previously had no defects measured and no strong copper signal. All tubes were therefore eddy current tested after chemical cleaning. The data from each tube test was subject to two independent analyses. A third analysis was performed if there was greater than a ten percent difference in defect size from the first two. A second eddy current test was conducted of all tubes with rejectable defects.

Based on these inspections, a total of 2,958 sleeves are to be installed and 18 tubes are to be plugged (in both steam generators). A total of 3,592 tube ends were recorded with pitting defects. Shallow defects will not be sleeved this outage and are to be reinspected during future outages.

The 2022 sleeves installed during the 1983 refueling outage were also examined. Six of these tubes are to be plugged because of indications on the inside surface of the tube. These are believed to be tooling marks. One additional tube is to be plugged because of blockage in the sleeve.

No unacceptable conditions were identified. The matter of copper completely masking rejectable defects may be generic and has been referred to the NRC Office of Nuclear Reactor Regulation (NRR) for review.

3. Steam Generator Secondary Sludge Removal - (Unit 2)

The licensee has implemented a program to remove the sludge pile from the steam generator secondary. The sludge, which is sixty (60) percent copper, appears to accelerate the corrosion of Alloy 600 tubes. The secondary side of both steam generators was cleaned chemically to about two (2) feet above the tube sheet (just below the first tube support plate). High pressure water hydrolazing (sludge-lancing) was used before and after chemical cleaning. The hydrolazing removed about 380 pounds of sludge from the No. 1 Steam

Generator and about 385 pounds from No. 2. Chemical cleaning removed about 287 pounds of metal from No. 1 steam generator and about 246 pounds from No. 2. Of this, about 42 pounds of iron were dissolved from base metal. In all, about 1330 pounds of sludge, including oxides, were removed. The process used was one qualified in EPRI testing. Corrosion of plant components was monitored with coupons placed inside the steam generator and inside a slip stream corrosion monitor. That monitor also provided corrosion information in real time using data analyzed by computer. The maximum corrosion was measured to be 5.9 mills in corrosion coupons.

There were no unacceptable conditions identified.

5. Tool Decontamination Facility Fire (Units 1 and 2)

A fire occurred within a degreasing unit located in a tool decontamination facility at 0950, April 4. The unit involved uses heated freon as a solvent. Its electrical heating coils apparently ignited residual greasy sludge when the unit boiled dry. The fire remained contained within the degreasing unit and was reported out at 1013. The fire brigade responded. Assistance was requested from the Waterford Fire Department, and their Fire Department trucks arrived at 1010. Station security personnel acted in accordance with emergency procedures by restricting access to the site to all but emergency vehicles. There were no serious personnel inquiries. Several minor injuries were reported from skin exposure to acid vapors. These originated from an electro-polishing unit located next to the degreasing unit.

The licensee has not used equipment in this area since the fire. A special investigation was undertaken by licensee personnel. That investigation addressed personnel and procedural deficiencies and the possible need for fire detection and fire suppression equipment in this area. These matters are under licensee review.

Sampling was performed at all the doors to the tool decontamination facility for airborne radioactive contamination during and after the fire. None was detected. The facility is located in a shielded building which had originally been a solid radioactive waste drum storage area. The building is ventilated directly to the unit stack, an elevated release point, and there were no problems identified by the licensee or resident inspector in the area of airborne exposure control. Safety-related equipment was not involved in or in proximity to the fire. The inspector had no further questions on this matter.

6. Engineered Safety Features Actuations (Unit 2)

Inadvertent actuations of Engineered Safety Features Equipment occurred four times during the inspection period. In each case there was no fuel in the reactor and equipment performed as designed.

At 1105, March 25, a containment isolation occurred while troubleshooting an intermittent radiation monitor alarm. The process radiation monitors are provided with two reset functions, an alarm bistable reset and a count rate integrator reset. The alarm bistable provides an alarm output only; the Engineered Safety Features Actuation System (ESFAS) contains a trip function bistable. The count rate integrator reset operates to drive the output to zero. The process radiation monitors are provided with a down-scale bistable to detect an instrument failure. In the case of the signal from the four containment radiation monitors to the ESFAS bistables, that signal is interrupted by the instrument failure trip and results in the equivalent of a high radiation ESFAS trip. On March 25, when the monitor reset button on RM8123A was pushed during troubleshooting, the ESFAS processed a Containment Purge Valve Closure in a one-out-of four logic. To limit inadvertent actuations in the future, the licensee has placed a protective cap over the instrument reset push-button.

At 1425, March 29, the ESFAS processed various actuations when a relay was removed for replacement. Technicians were replacing various actuation relays for surveillance testing of relay response time. The relay replaced provides a blocking function to prevent system actuations with the reactor shut down. Removal of the relay allowed Channel II Safety Injection, Enclosure Building Filtration, and Containment Isolation Actuation Signals to be processed. All equipment functioned as designed with the exception that valve 2-AC-11, the main purge exhaust damper, could not be verified because of a dual indication. Subsequent testing verified proper operation of 2-AC-11. Procedural changes have been instituted to prevent further actuations under such circumstances.

At 0850, April 1, during a test of breaker control circuitry, a feeder breaker which was supplying safeguards bus 24D was tripped. This resulted from the failure to block all of the interconnecting logic from the breaker being tested (2253-24D-2,A411), which is the feed from the Reserve Station Service Transformer (RSST), to safeguards bus 24D. The trip signal indicated a fault condition on the bus and tripped the feed (24D-1T-2, A410) from bus 24B. The ESFAS sensed the bus undervoltage and processed a loss of power actuation, starting the 13U Emergency Diesel Generator. However, the generator output breaker was blocked from closing because the testing which had caused the loss of power also energized two lockout relays. Those relays signaled a bus overcurrent condition and isolated the bus from power sources. The tests could have been accomplished if all the points needed were blocked. The licensee has cautioned the personnel involved to use more care in researching circuits prior to performing retests. Additionally, the licensee has urged personnel to use pre-test briefings, and cautions have been added to functional test procedures.

At 1445, April 13, the ESFAS processed a Containment Purge Valve Closure. Following routine leak rate testing of the containment hydrogen analyzer and containment radiation monitors RM8262A and RM8262B, a spike occurred in the radiation monitor when the sample pumps were started. It is believed that

the radiation spike occurred when particulate material came loose in the sample piping. Station procedures have been updated to inform the control room operators of the potential for this happening in similar circumstances.

There were no unacceptable conditions identified following the review of these individual events. Routine resident inspection will continue to cover ESFAS actuations, and the matter of unnecessary actuations of safety features will be considered during NRC evaluation of licensee performance.

7. Maintenance Observations

The inspector observed portions of corrective and preventive maintenance to determine if the work was conducted in accordance with requirements regarding plant conditions, maintenance, and retest. These included:

Unit 1

-- The Emergency Gas Turbine Generator failed to start during surveillance testing at 1007, April 30, 1985. The cause was found to be an inoperable starting air pressure regulating isolation valve. That valve is normally shut but opens in a pressure regulating mode to supply air to an air starting motor for the gas turbine. A replacement valve was installed and the unit was successfully tested at 1830, April 30. The defective valve had been installed during the April-June, 1984 refueling outage. That valve is an aircraft engine part and is serviced by a specialty vendor. A failure analysis is to be provided. It will be reviewed by the NRC (IFI 245/85-10-01).

Unit 2

-- The steam generator feedwater and auxiliary feedwater check valves (2-FW-5A,5B, 12A, 12B) were overhauled during the current refueling outage. This was in response to a November 28, 1984 plant trip in which feedwater check valve 2-FW-5B remained partially open after a main feedwater pump trip. The licensee discovered that binding was occurring between the valve disk and the valve body. This was due to compression on the disk hinge caused by the shaft follower. The follower bolts to the outside of the valve body and extends into the inside of the valve. The follower to valve body gasket, which had been a flat composition gasket, had been replaced with a metallic flexible gasket. The flexible gasket compressed more than the original gasket, allowing the shaft follower to interfere with the valve disc. The gasket modification had been handled as a design change, documented as PDCR 2-116-83 for the main feedwater check valves and PDCR 2-121-83 for the auxiliary feedwater check valves. Because the work was accomplished with the valve assembled, it had not been possible to check for interference. The gasket change was made in an effort to correct a chronic leakage problem between the shaft follower and the valve body. The licensee's corrective actions were to restore the original clearances with machined shaft spacers. This resolves a previously identified item (336/84-24-01).

8. Observation of Surveillance

The inspector observed parts of surveillance tests to determine for conduct in accordance with requirements. These tests included:

Unit 1

-- Emergency Gas Turbine Generator Surveillance on April 3 per SP 668.2.

-- Emergency Diesel Generator Functional Testing on April 30 per SP 668.1. At the conclusion of the test, the generator breaker tripped on reverse power. The machine was restarted and successfully loaded and unloaded. The tripping appears to have been related to unloading the machine. This test is performed weekly; there were no additional problems during this reporting period.

Unit 2

-- "B"-Emergency Diesel Generator overspeed trip testing on April 9.

-- Inservice Inspection of Steam Generator No. 1 & 2 dome weld defect.

-- Integrated loss of power test facility components on April 26 per Test Procedure T85-15.

9. Potential Generic Problems Identified at Other Facilities

Another licensee recently determined that Rosemont 1153 Series B level transmitters on the scram instrument volume were not qualified for the reactor building environment following a break in the HPCI steam line. These transmitters are not in use at Millstone 1, where the reactor scram is provided by level switches by a different manufacturer. The inspector had no further questions on this matter.

10. Licensee Action on Previous Inspection Findings

Unit 1

(Closed) Violation 245/85-02-02. The licensee's letter to the NRC dated April 26, 1985 detailed their corrective actions concerning incomplete surveillance testing. No further occurrences have been noted. There are no additional questions at this time. This area is covered by resident and region-based inspections, and routine inspection will assess the ongoing effectiveness of corrective actions.

(Closed) Violation 245/85-02-05. The licensee's letter to the NRC dated April 26, 1985 detailed their corrective actions to assure that isolation zones remain clear. No recurrence has been noted. There are no additional questions at this time. This aspect receives continual NRC review.

(Closed) Open Item 245/84-27-04, Procedure PT 1430, RPS MG Set and Backup Protection Surveillance, Revision 1, dated February 13, 1985 was reviewed. The procedure now provides testing by using external voltage and frequency sources. There were no unacceptable conditions identified.

Unit 2

(Open) Deviation (336/85-08-01), The licensee has conducted an investigation to determine the acceptability of Reactor Coolant System (RCS) hot leg and cold leg piping ultrasonic examinations which were done using calibration block number UT-6. Block UT-6 is flat and the licensee has committed to perform ultrasonic examinations in accordance with the ASME Code, Section XI, Appendix III, which requires the use of a curved calibration block.

The investigation, which was witnessed by the inspector, consisted of comparing the acoustic characteristics of the piping and of block UT-6. In addition, the licensee used calibration block UT-15, a 36 inch diameter block fabricated of cast stainless steel welded to carbon steel with roll band cladding on the I.D. surface. Side drilled holes at 1/4T and 3/4T (T = thickness) are included for calibration purposes.

The ASME Code, Section XI, Appendix III requires that examination sensitivity be established from notches in the I.D. and O.D. surfaces of the calibration block. The licensee stated that piping examinations at Millstone Unit 2 were based on side drilled hole (SDH) calibration, which routinely results in greater sensitivity than the sensitivity resulting from notch calibration. Comparison of the two types of calibration was included in the licensee's investigation.

The inspector reviewed calibration records of RCS piping examinations performed in 1977, 1980 and 1982 to ascertain the type of calibration reflector upon which the examinations were based for those years. The records indicated that the examinations were based on SDH reflectors at 1/4T, 1/2T and 3/4T in block UT-6.

The investigation was done using the following equipment:

1. Krantkramer - Branson USL-38 ultrasonic instrument.

2. Kranthramer - Branson USD-1 ultrasonic instrument.

3. Aerotech, 2.25 MHz, 0 degree, 1 inch diameter transducer S/N A 31369.

4. Sonic, 2.25 MHz, 45 degrees, 1/2" X 1" transducer S/N 00841T.

5. Calibration block UT-6, flat, 3.7 inch thick, SDHs and I.D. and O.D. notches 2% into base metal.

6. Calibration block UT-15, 36 inch diameter, 3.2 inches thick, SDHs.

7. Hot leg piping - 42 inch diameter, 3.5 inches thick.

8. Cold leg piping - 30 inch diameter, 3 inches thick.

The investigation was done by adjusting the instrument to produce an 80% of full scram height (FSH) signal amplitude from the 1/4 T SDH in block UT-6. This was considered the reference sensitivity.

The results of the investigation were as follows:

-- Block UT-6 is 13 dB more transparent to ultrasonic energy than is the RCS piping.

-- Block UT-6 is 13 dB more transparent to ultrasonic energy than is block UT-15.

-- Block UT-15 and the piping are acoustically similar.

-- SDH sensitivity is 18 dB more sensitive than notch sensitivity in block UT-6.

The investigation was done using the zero degree, 1 inch diameter transducer. Notch and SDH sensitivity also were compared using a 45 degree shear wave transducer with blocks UT-6 and UT-15. Similar results were noted.

Based on the above, the licensee determined that the examinations performed at the facility were done at 5 dB greater sensitivity than would have resulted from basing the examination sensitivity on the notch calibration reflectors of the ASME Code, Section XI. Therefore, the RCS piping examinations are acceptable because they exceed applicable Section XI requirements.

At the exit meeting on April 4, 1985 the inspector stated that, based on his observations, he agreed with the licensee's determination as documented above. This item remains open pending licensee resolution of the discrepancy between the use of a flat calibration block and the the licensee commitment to use a code acceptable curved calibration block.

(Closed) Unresolved Item 336/85-09-03; The licensee has removed the rotary switch handle associated with the power switch for valve 2-SI-652.

(Closed) Unresolved Item 336/85-03-01, The licensee's letter to the NRC dated April 11, 1985 addressed the previously reported error in the small break LOCA analysis. The information presented in that analysis agrees with the re-analysis data, which is known to the inspector.

(Closed) Violation 336/85-03-03; The licensee's letter to the NRC dated April 26 detailed their corrective actions to ensure better adherence to radiation protection procedures and radiation work procedures. There have been no additional lapses in procedural compliance noted.

(Closed) Unresolved Item 336/84-24-01; Operability of steam generator feedwater check valves. This item is addressed in report paragraph 7.

11. Exit Interview

At periodic intervals during the inspection, meetings were held with senior licensee site management to discuss the inspection scope and findings. The inspector addressed the details of NRC Information Notice 85-27 and the importance of making reports promptly to the NRC.

At no time during this inspection was written material concerning inspection findings provided to the licensee by the inspector. Information which may have been proprietary and which was addressed during this inspection period was discussed with licensee representatives. No information identified as proprietary was included in this report.