U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket No. 50-423 Report No. 50-423/85-26

License No. CPPR-113 Category B

Licensee: Northeast Nuclear Energy Company
PO Box 270
Hartford, Connecticut 06101

Facility: Millstone Nuclear Power Station, Unit 3

Inspection at: Waterford, Connecticut

Inspection conducted: May 28 - July 8, 1985

Inspectors: Robert J. Summers for T. Rebelowski 8/1/85
T. A. Rebelowski, Senior Resident Inspector Date

Robert J. Summers for F. Casella 8/1/85
F. A. Casella, Reactor Engineer Date

Approved by: E C McCabe 8/5/85
E. C. McCabe, Chief, Reactor Projects Section 3B Date

Summary: Inspection 50-423/85-26 (May 28 - July 8, 1985)

Routine resident (139 hours) and region-based (12 hours) inspection. Major tasks observed during this report period were the Steam Generator "J" Tube Modifications, the initial Eddy Current testing and tube plugging, and preparations for the Containment Structural Integrity Test.

No unacceptable conditions were identified.

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DETAILS

1. Persons Contacted

Numerous members of Northeast Utilities and Stone and Webster (S&W) Corporation including engineers, technicians, craftsmen, and members of staff management were contacted.

2. Licensee Action on Previous Inspection Findings

(Closed) Inspector Follow Item (85-05-02). Service Water Intake Pump Bearing Sleeve Failure Analysis (Lab Test B-85-008). The licensee determined that the bearing sleeve coating was chrome plated Nitiomic 50 material. The plating was acceptable except for 5% of the bearing surface where spalling was caused by porosity. Damage was caused by the spalling and by the chrome particles it produced. The bearing sleeve was replaced by one made of S-Monel, which is used in this application in the other service water pumps. The inspector had no further questions on this item.

(Closed) Inspector Follow Item (423/85-16-02). Repair methods for service water elbow. The licensee's method of repair of the copper-nickel alloy lining for this carbon steel pipe was reviewed and found acceptable. Repairs included weld filler repair on the carbon steel pipe and replacement of the copper-nickel layer defects. The cause of the corrosion was attributed to a manufacturing defect in the copper-nickel lining. Licensee checks of the other similar elbows found no problems. Liquid penetrant testing of the reworked area was performed and no deficiencies were identified. This item is closed.

(Closed) Inspector Follow Item (423/85-16-03). Steam generator tube repairs. This item is discussed in Detail 10 of this report.

3. Steam Generator "J" Tube Replacement

References: a) Westinghouse procedures SSS2.7.2 Gen 33 Rev 2
b) Field Change Notice NEUM 10592
c) Purchase Requisition 901943
d) NRC Report 50-423/85-12

The licensee removed all "J" tubes from the Steam Generators and replaced them with material with improved erosion/corrosion resistance (higher chrome content) (Ref d).

a. Purchase orders were reviewed and found to document QA/QC requirements to assure proper material verification (Ref. c).

Certifications of conformance and chemical analysis reports were available to assure that material conformed to ASME SB-167 (Alloy 600 seamless pipe).

b. "J" Tube Removal Requirements

The licensee's nuclear vendor specified training for tooling and required special training of personnel in NSID FF SG-83-072, Rev I, "J" Nozzle Removal.

Oil-lubricated rotating tools were used inside the steam generator. When that was recognized by licensee QC, all work was halted until analysis of oil vapor contamination and the effect on the secondary feed system were reviewed and found acceptable. The removal continued after protective measures were taken to limit oil vapor spread.

In addition, one removed "J" nozzle was dislodged and set adrift in steam generator "D". This item was recovered upon completion of the field change. The inspector had no further questions in this area.

c. Replacement of "J" Tubes

The "J" tube installation was observed. Welder certifications, certification of weld filler material, and Inspection Procedure NDE-110, Visual Examination, were reviewed by the inspector.

Prior to installation of replacement "J" tubes, the licensee found linear indications in the area of tube removal. These indications were determined to be the remains of the previous carbon steel "J" tubes. Based on a review of the design change, it was determined by the licensee that the new Inconel attachment weld would incorporate the carbon steel remains. The deficiency was reviewed and found acceptable.

A closeout inspection and tool review was performed by the licensee and found acceptable. The inspectors observed a number of "J" tubes being removed and replaced. Inspector follow item 423/85-12-01 addressing review of "J" tube replacement is closed.

4. IE Bulletins

IE Bulletins issued to plants under construction for information only and for the identification of anomalies in hardware were reviewed. The following bulletins are closed.

75-04,75-04A,75-04B. This bulletin, "Cable Fire at Browns Ferry Nuclear Power Station" was issued during the early stages or construction at Millstone 3. The licensee has prepared a Fire Protection Evaluation Report which describes the Millstone 3 fire protection program. This matter is part of the NRR fire protection review. This bulletin is closed.

75-06. This bulletin, "Defective Westinghouse Type OT-2 Control Switches," was addressed to facilities under construction. Stone and Webster Engineering Corporation performed an investigation and determined that no Westinghouse Type OT-2 switches are used on Millstone 3. The licensee responded to this bulletin by letter dated July 31, 1975. This bulletin is closed.

75-08. This bulletin, "PWR Pressure Instrumentation", was issued to facilities under construction for information only. The bulletin requested that operating facilities provide a description of reactor coolant system temperature and pressure instrumentation showing adherence to the temperature/pressure limitations in the technical specifications. FSAR Section 7.5 describes instrumentation to perform this function. This bulletin is closed.

76-04. This bulletin, "Cracks in Cold Worked Piping at BWRs," was only addressed to BWRs, is not applicable to Millstone 3, and is closed.

76-06. This bulletin, "Diaphragm Failures in Air Operated Auxiliary Actuators for Safety/Relief Valves," described air operator diaphragm failures due to excessive heat. Millstone 3 does not use air operated Target Rock Safety/Relief Valves. This bulletin does not apply to Millstone 3 and is therefore closed.

79-05, 79-05A, 79-05B, 79-05C, 79-06A, 79-06A Rev. 1, 79-06B, 79-06C. These bulletins, revisions, and supplements addressed actions to be taken as a result of the nuclear incident at Three Mile Island Unit 2. Millstone 3 was not required to reply to the bulletins because it was not an operating facility. TMI action plan items are addressed in the Millstone 3 FSAR. The NRC Safety Evaluation Report for Millstone 3 evaluates the adequacy of the facility in light of the TMI-2 event and associated action items. Review of TMI items is conducted incident to inspections for conformance to FSAR commitments. In addition, special inspection of TMI items is to be performed incident to inspection of NRR requested inspection items in preparation for license issue. These bulletins are therefore closed.

79-10. This bulletin, "Requalification Training Program Statistics," asked licensees to provide statistics relative to the requalification training program. It did not apply to Millstone 3 at the time. FSAR Section 13 describes the retraining program and specifies record retention adequate to respond to this bulletin, should similar information be needed. This bulletin is closed.

5. Concrete Placement - Control Building

On June 3 the inspector witnessed placement of ten (10) of a total of forty-three (43) cubic yards of Concrete Mix 301. The air entrainement and slump tests were satisfactory. Ten cubic yards of a different batch were found rejectable, although about five cubic yards were poured prior to recognition of this. The A/E specification called for air content of 3.5 to 6.5% for concrete mix 301. The nonconforming batch was found to have a 3.0% air content. A nonconformance report was written (13243). Two test cylinders of the low air concrete were made.

The seven day comprehensive strength for the low air content pour tested on 6/10/85 was 2895 psi. The minimum acceptable vaue is 2650 psi. In addition, application of clear epoxy along the surface in the pour between elevation 47' to 51' was performed to prevent weather effects in areas of low air entrainment. The inspector had no further questions in this area.

6. Main Steam Isolation Trip Valve Testing

The inspector observed a number of full strokes of Main Steam Isolation Trip valve 3MSS-CTV 27 A on June 4,1985 and made additional observations of stroking of 3MSS-CTV 27 C. These valves are solenoid-actuated Sultzer Manufacturing valves. Nitrogen was used to actuate the valve pistons. Testing showed minor flow leakage past the solenoid valves.

Based on the full stroke testing of the MSIV's and additional testing of solenoids during calibration of position sensors, the vendor concluded that the valves have no mechanical defects. Additional checks of energizing and checking of the position of each solenoid during various modes is to be performed. When all vendor concerns are satisfied, the licensee will further test the valves during Hot Functional Testing (HFT).

The inspector will review results of testing during normal results review and HFT witnessing and evaluation.

7. Review of Spent Fuel Pool (SFP) Siphon Breaking Provisions

The inspector reviewed the following piping systems diagrams to determine if any installed piping run could siphon the Spent Fuel Pool and thus expose irradiated fuel in the future.

	Line #	ISO #
SFC Cooling Suction	3-SFC-012-91-3	3-SFC-524
SFC Cooling Discharge	3-SFC-010-6-3	3-SFC-520
SFC Purif. Suction	3-SFC-004-93-4	3-SFC-521
SFC Purif. Discharge	3-SFC-003-204-4	3-SFC-511
RWST/QSS Fill Line	3-SFC-004-177-4	3-SFC-518
Service Water Fill Line	3-SFC-004-41-3	3-SFC-16

Two lines, the Spent Fuel Cooling discharge line (3-SFC-010-6-3) and Spent Fuel Cooling Purification Suction line (3-SFC-004-93-4), have the ability to siphon the pool due to the location of openings at the 12'9" and 13'9" elevation inside the SFP. Both these lines have 1/2" antisiphoning holes that were verified to exist by observations made by the inspector. The inspector had no further questions on this item.

8. Interfacing System LOCA Review

The inspector began a limited review of the surveillance and maintenance activities covering ECCS isolation valves that are potentially subject to overpressurization. The following systems were reviewed: High Pressure Safety Injection (HPSI) including charging pump and HPSI pump suction and discharge lines; Low Pressure Safety Injection (LPSI) including Safety Injection Tank Discharge Lines, Residual Heat Removal System, and Containment Recirculation System.

All of these systems contain some piping having design pressure less than 70% of the design pressure of the Reactor Coolant System (RCS). The inspector reviewed the elementary (ESK) drawings for these systems to determine the high to low pressure interfaces and to identify the valves which function to isolate the low pressure portions from the RCS. The review identified no air-operated testable check valves.

Two of these ECCS Systems were chosen for a review of surveillance and maintenance activities pertaining to the above described isolation valves. The first was High Pressure Safety Injection, whose NUREG/CR-2069 formatted configuration is RCS-CK-CK-I-MOV(NC)-CK-P-H/L (220 psi). The second was Residual Heat Removal, whose configuration is RCS-CK-CK-I-MOV(NC)-H/L-MOV(WO)- HX-P (600 psi). Valves singled out for review were 3-SIH-V110 (HPSI to RCS Loop 1, Hot Leg), 3-SIH-V112 (HPSI to RCS Loop 3, Hot Leg), 3-SIL-V986(RHR/SI to RCS Loop 2, Cold Leg) and 3-RCS-V69 (RHR/SI to RCS Loop 2, Hot Leg). At the time of this inspection, Appendix J leak test procedures were the only completed procedures in effect for these valves. Upon completion of the surveillance and maintenance procedures for the above valves, a subsequent inspection will be conducted to verify implementation.

This item is open (50-423/85-26-01).

9. Containment Structural Integrity Test

References: (1) Millstone Nuclear Power Station Unit 3 - Set up preparation for Structural Integrity Test.

Preparations for the Structural Integrity Test (SIT) were witnessed, including containment walk-throughs, review of the SIT data accumulation center, and placement of several radial and vertical taut wire installations at the 71' and 76' levels. In addition, selected strain gauges in the areas of equipment and personnel hatches were observed for cleanliness of attachment areas. No deficiencies were identified. SIT adequacy will be addressed in Report 50-423/85-33.

10. Eddy Current Testing (ECT) Inspection - Non-Ferromagnetic Heat Exchanger Tubes

Observations were made of licensee's preparation and implementation of MIZ-18, Multi-Frequency/Multi-Parameter Base Line Data for steam generators B and C. Licensee analysis of steam generators A, B, C, and D identified the need to mechanically plug 5 tubes, as follows:

a. Steam Generator A

Row 29 Column 60 - 68% through wall defect
Row 25 Column 61 - 73% through wall defect

b. Steam Generator B

Row 1 Column 1 - 68% through wall defect

c. Steam Generator C

Row 32 Column 71 - Double Expansion bulge

d. Steam Generator D

Row 1 Column 122 - 67% through wall defect

These tube repairs have been completed, resolving inspector follow item 423/85-16-03. This item is closed.

Additionally, inspector concern about the numbering system used to identify steam generator tubes was resolved by proper identification points being included on deficiency reports.

The inspector monitored the licensee's program to maintain tool control, cleanliness, and procedural adherences during ECT inspection. No anomalies were identified.

The inspector had no further questions on this item.

11. Auxiliary Feedwater Pump - Initial Runs

The inspector witnessed the initial auxiliary feedwater (AFW) pump (P1B) run. The run was on recirculation flow to the demineralized water tank. The motor outboard bearing temperatures did not appear to stabilize and reached 162 degrees F. This test will be repeated to assure bearing acceptance. The licensee is required to repeat data runs during pump performance curve acceptance tests. Further licensee testing of AFW will be observed under routine inspection requirements.

12. Licensee Reports of Potential Significant Deficiencies

Items where licensee action remains outstanding:

a. Triaxial Connector - Failure to Meet Seismic Qualification (SD-84) 85-00-17

On June 14,1985 the licensee reported a potential significant deficiency involving triaxial connectors (Amphenol) used on Nuclear Instrumentation Systems (NIS) equipment. The connector that failed was Amphenol part number 34875, Version 2. The connector failed during seismic testing at low "g" levels. Westinghouse and the licensee are reviewing the reports. This item will remain open pending completion of corrective action.

b. Qualification of Support of Flux Mapping Equipment (SD-85) 85-00-18

On June 26, 1985, the licensee identified a potential significant deficiency with a seismic interaction between the flux mapping equipment and the associated seal table. The Category 2 flux mapping equipment is in-

stalled directly above the seal table which constitutes the pressure boundary between the guide tubes for the incore instrumentation and the reactor coolant system. Failure of the equipment support during a seismic event could damage the instrument tubing/seal table, resulting in a small break LOCA. The flux mapping equipment has not been analyzed for structural integrity during an SSE event. This condition is currently under evaluation by the Nuclear Vendor and the AE. This item remains open.

13. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable or not. An unresolved item identified during this inspection is discussed in Detail 8.

14. Exit Meeting

At periodic intervals during the course of this inspection, meetings were held with senior plant management to discuss the scope and findings of this inspection. No proprietary information was identified as being in the inspection coverage. At no time during the inspection was written material provided to the licensee by the inspector.