U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-336/88-19

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, Unit 2

Inspection At: Waterford, Connecticut

Dates: July 30 - September 12, 1938

Reporting Inspector: Peter J. Habighorst, Resident Inspector

Inspectors: William J. Raymond, Senior Resident Inspector
Scott Barber, Millstone Unit 3 Resident Inspector
Lynn Kolonauski, Millstone Unit 1 Resident Inspector
Peter Habighorst, Millstone Unit 2 Resident Inspector
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Approved by: Ebe C. McCabe, Jr 9/23/88
E. C. McCabe, Chief, Reactor Projects Section 1B Date

Inspection Summary: 7/30 - 9/12/88 (Report 50-336/88-19)

Areas Inspected: Routine NRC resident and specialist inspection of plant operations, surveillance, maintenance, physical security, previous identified items, Justification of Continued Operation for high Service Water Inlet temperature, dual role SS/STA qualification, periodic reports, and committee activities.

Results: No unsafe conditions were identified. Good operator awareness, detection, and correction of unidentified leakage on August 4 was identified (Report Detail 2.0). One unresolved item was reviewed concerning electrical separation between Class 1E and non-class 1E power supplies to the vital 120 volt AC buses. This item and the preventive maintenance program for the power supplies will be followed during future inspections (Report Detail 5.3).

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DETAILS

1.0 Persons Contacted

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

S. Scace, Millstone Station Superintendent
J. Keenan, Unit 2 Superintendent
J. Riley, Unit 2 Maintenance Supervisor
F. Dacimo, Unit 2 Engineering Supervisor
D. Kross, Unit 2 Instrument and Controls
J. Smith, Unit 2 Operations Supervisor

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

2.0 Summary of Facility Activities

The unit began the inspection period at full power and remained at this level for the duration of the period. On August 3, the licensee conducted a containment entry to obtain routine monthly safety injection tank boron samples, and to investigate the cause of the increase in unidentified leakrate from less than 0.1 GPM to approximately 0.3 GPM. The technical specification (TS) limit for unidentified leakrate is 1.0 GPM. The licensee identified a body-to-bonnet steam leak from valve 2-RC-252 (Loop I spray valve manual isolation valve). On August 4, the licensee re-entered containment and isolated the leak. The unidentified leakage decreased and remained less than 0.1 GPM throughout the remainder of the inspection period. Good licensee awareness, detection, and correction of the leakage was observed. Other items of interest during the period were: the JCO for service water inlet temperature (Report Detail 9.0); loss of vital instrument bus VA-20 (Report Detail 5.3); and seismic qualification of Engineering Safety Actuation System (ESAS) power supplies (Report Detail 5.2).

3.0 Licensee Actions on Previously Identified Items (92701)

3.1 (Open) Unresolved Item 88-07-03: Inadequate Surveillance on Low Temperature Overpressure (LTOP) Setpoint for Power Operated Relief Valves (PORVs)

This item concerned the licensee's 18-month surveillance testing of solenoid-actuated PORVs per TS 4.4.9.3.1.d. The licensee sends the PORVs to Wyle Laboratories for testing and refurbishing during refuelings. The inspector reviewed Certification Test Reports 49134-10 and -11, dated January 28, 1988, for each of the two PORVs. The tests showed the PORVs operated within the allowable band of 2385 + 10 psig in less than 100 milliseconds. The inspector questioned the licensee on why no test data was available for LTOP plant conditions. The licensee reviewed the basis for TS surveillance 4.4.9.3.1.d. The American Society of Mechanical

Engineers (ASME) Section XI valve testing criteria, on which the TS is based, requires that Class I power-operated relief valves be: visually examined; seat tightness determined; verify operability of pressure sensing and valve activation equipment; verify integrity of balancing device; and checked for operation and electrical characteristics of position indicators. The ASME Code criteria establish valve operability and valve seat tightness. The surveillance test demonstrated that the valves operated at the high pressure setpoint of 2385 psia. The licensee concluded the valve testing program performed at 2385 psig demonstrated operability at both required setpoints of 2385 psig and 450 psig based on past test data correlation between the two pressure setpoints. The valve actuation is identical except for the Reactor Coolant System pressure on the pilot valve. This item remains open pending NRC review of past data to verify the licensee's conclusion.

3.2 (Closed) Unresolved Item 88-07-04: PORV Actuation Data Discrepancies

This item concerned valve operation time assumed in the licensee's LTOP evaluations. In the NRC Safety Evaluation supporting TS Amendment 50, issued in 1979, mention is made of the quick opening time of 10 milliseconds; however, typical valve opening times from the Electrical Power Research Institute (EPRI) for similar valves was 170 milliseconds with water at 600 psia. The inspector questioned the licensee on the PORV operation opening time discrepancy. The licensee reported the 10 ms opening time corresponds to the time required for the valve to open once the solenoid arm has been lifted, while the 170 ms time obtained from EPRI includes the time required for the signal to reach the solenoid and lift the solenoid arm and open the valve (full stroke time). The actual full stroke time for the licensee's PORVs is less than 100 ms. No inadequacies were noted.

The LTOP design basis pressure transient is the starting of a Reactor Coolant Pump (RCP) with a secondary to primary side temperature differential of 43 degrees F. This transient results in an RCS pressure of 500 psig (assuming failure of one PORV) based on actual valve opening characteristics. Licensee evaluation determined that if 100 ms were added to the PORV opening time, the resultant RCS peak pressure transient was approximately 504 psig, which is less than 10 CFR 50 Appendix G limits. This item is closed.

3.3 (Closed) Inspector Follow Item 87-24-01: Measurement Control Evaluation Non-Radiological Chemistry

On completion of the analyses of water samples (spiked samples) by the licensee and Brookhaven National Laboratory, a statistical evaluation was to be made. The analyses were completed and an evaluation was performed. The analytical comparisons were acceptable. The table is shown on the following page.

Millstone Units 1 & 2 Split Samples with Spikes

Analyte	Matrix	Spike	Millstone 1&2	Brookhaven
Flouride (ppb)	Steam Generator	D1	14.7+/-0.1	14.3+/-0.1
		D2	7.2+/-0.2	less than 10
Chloride (ppb)	Steam Generator	D1	16.1+/-0.1	14.7+/-0.9
		D2	8.4+/-0.2	9.5+/-2.7
Sulfate (ppb)	Steam Generator	D1	16.1+/-0.1	11.3+/-0.5
		D2	8.1+/-0.1	7.9+/-2.1
Hydrazine (ppb)	Steam Generator	none	20.5+/-0.7	19.2+/-0.4
Iron (ppb)	Feedwater	E1	750+/-10	731+/-15
		E2	533+/-6	465+/-16
Copper (ppb)	Feedwater	E1	760+/-10	700+/-0
		E2	503+/-6	450+/-0
Nickel (ppb)	Feedwater	E1	730+/-0	720+/-16
		E2	477+/-6	750+/-20
Chromium (ppb)	Feedwater	E1	733+/-6	750+/-20
		E2	493+/-23	500+/-0
Boron (ppm)	Standby Liquid Control Tank	none	24,921+/-140	26,240+/-90

3.4 (Closed) Unresolved Item 88-16-02: Determination of Equalizing Charge on Vital Batteries per Vendor Recommendation

This item concerned equalizing charges on the vital station service batteries for situations other than restoration from a Battery Performance Discharge Test and a Battery Service Test. For situations where the battery charger is operable but is providing a float charge at a voltage below recommended value, the vendor (C&D) recommends an equalizing charge when the lowest cell drops more than 0.04 volts below minimum float voltage. The inspector asked the licensee why the vendor recommendation was not incorporated in the station service battery surveillance program. The station batteries have an acceptable temperature-corrected specific gravity of 1.200 per surveillance SP-2736A "Battery Pilot Cell Surveillance," Rev. 1. In accordance with the vendor manual, the minimum acceptable voltage per cell, minus 0.04 volts, is 2.08 volts. For battery operability, TS 4.8.2.3.2.a.3 requires each battery cell to have a voltage equal to or greater than 2.08 volts under a float charge with specific gravity greater to or equal to 1.200. Thus, if the individual cell voltage decreased to the point (2.08 volts) where the manufacturer would recommend an equalizing charge, the battery also would not be in compliance with the associated TS surveillance. In plant operational modes 1 through 4, the battery then must be restored to operable status within 2 hours or the plant must be in cold shutdown within 36 hours. The inspector checked the licensee's calculations and reached the same conclusion. Since the criterion for performing the equalizing charge based on individual cell voltage is enveloped by the TS, this item is closed.

4.0 Facility Tours (71707)

The inspector observed plant operations during regular 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)

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 control of 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 reviewed included various operating logs, turnover sheets, and tagout logs. No inadequacies were identified.

5.0 Plant Operational Status Reviews (71707)

5.1 Review of Plant Incident Reports (PIRs)

The plant incident reports (PIRs) listed below were reviewed to (i) determine the significance of the events; (ii) review the licensee's evaluation of the events; (iii) verify the licensee's response and corrective actions were proper; and, (iv) verify that the licensee reported the events in accordance with applicable requirements, if required. The PIRs reviewed were: number's 88-57, 88-58, 88-59, 88-60, 88-61, 88-62, 88-63, 88-64, and 88-65. The following item warranted inspector followup: PIR 88-65 (Report Detail 5.3).

5.2 Seismic Qualification of Power Supplies in Engineering Safety Actuation System (ESAS)

On August 26, the licensee discussed with the inspector the seismic qualification of Lambda power supplies (+15 VDC, and +60 VDC) and Deutsch relays in the ESAS circuitry. The 15 VDC power supplies are used for ESAS logic; the Deutsch relays are used for ESAS actuations, and the 60 VDC power supplies are used for refueling water storage tank level. According to Final Safety Analysis Report (FSAR) Table 1.4-1, the ESAS system and status panel (C01X) are classified as seismic category I components. The power supplies were supplied by Consolidated Controls Corporation (CCC).

The licensee's Quality Services Department (QSD) audited CCC between May 20 and 26, 1988. The audit consisted of verification of implementation of applicable criteria of CCC's Quality Assurance Manual, American National Standards Institute (ANSI) standard N45.2, 10 CFR 50 Appendix

B, and 10 CFR 21 as it applies to the licensee's Purchase Orders. A major finding in the licensee's QSD audit was lack of traceable seismic qualification information from CCC.

On August 26, site and corporate Engineering conducted an operability evaluation of the seismic capability of the ESAS power supplies per Institute of Electrical and Electronic Engineers (IEEE) standard 344-1971 and Final Safety Analysis Report (FSAR) Section 7.3.1.2.5, "Seismic Evaluations of ESAS Components." The inspector reviewed the licensee's "Operability Evaluation of Lambda Power Supplies and Deutsch Relays" report. The evaluation report addressed: dimensional analysis with "known" seismically qualified Foxboro control cabinet power supplies; MIL-STD-801B Method 514 and 516.1 for vibration and shock qualification of the affected power supplies and relays; power supply and relay mounting in accordance with CCC recommendations; a seismic simulation test performed by Wyle Laboratories of a similar form, fit, and function, with respect to the affected Lambda power supplies; and review of Bechtel Engineering Report No. 863 on seismic analysis of ESAS cabinet test results. The technical evaluation performed by the licensee concluded the replacement Lambda power supplies and Deutsch relays are seismically qualified. No inadequacies were noted in the licensee's evaluation.

The inspector questioned the licensee on future replacement Lambda power supplies and the potential for case-by-case seismic evaluations. The inspector will follow this item in future inspections.

The inspector questioned the licensee on the timeliness between the seismic engineering evaluation and the initial licensee QSD audit of CCC. The licensee provided the inspector a sequence of events between May 20 thru September 2, 1988. The inspector reviewed the sequence of events, and expressed concern about the close to three-month delay in taking action on potentially significant findings (the site was notified about the problem on June 21). The inspector will continue to review associated interfaces in future inspections.

5.3 Loss of Vital Instrument Bus (PIR 88-68)

The 120 volt a.c. power supply and distribution system consists of four essential buses for vital instrumentation and control. The four vital a.c. buses are supplied from d.c./a.c. static Inverters. To provide increased reliability, each of the four vital instrumentation panels have an alternate power supply through a "zero break" static transfer switch. The static transfer switch monitors both the primary and back-up Inverter frequencies to maintain phase displacement within ten degrees. Under this condition, transfer between sources occurs with minimal perturbation in frequency or voltage. The static switch will transfer a.c. loads to the alternate source under low a.c. voltage, load overcurrent, or inverter failure.

While the plant operated at 100% power under steady state conditions on September 7, power to vital instrument bus VA-20 automatically transferred, on undervoltage, at 1:12 p.m., from Static Inverter #2 to the back-up inverter, #6, by way of the static transfer switch. At 1:50 p.m., Inverter #6 output voltage degradation resulted in complete loss of VA-20. The licensee detected a ground fault alarm on Inverter #6 at the time of transfer. The loss of VA-20 resulted in a loss of: the "B" reactor protection system (RPS) channel; "B" Engineering Safety Actuation System (ESAS) sensor cabinet; ESAS Actuation Cabinet 6; and other vital instrumentation and controls supplied through the CE Spec 200 controls cabinets. The control room operators took manual control of pressurizer level and pressure. The pressurizer pressure and level variation was a result of power loss on VA-20 to the pressurizer control channels and temperature, pressure and level inputs. No operator response inadequacies were noted.

Upon loss of VA-20, no RPS or ESAS actuations occurred. The licensee entered the following TS limiting conditions for operation: 3.8.2.1, "Onsite Power Distribution Systems;" 3.3.2.1, "ESAS Instrumentation;" 3.3.1.1, "Reactor Protective Instrumentation;" 3.3.3.1.b, "Radiation Monitor Instrumentation," and 3.3.3.8, "Accident Monitoring Instrumentation." The inspector reviewed Operation Form 2388-5, "120 volt a.c. Vital Instrument Panel (VA-20) Loads" and the TS for affected limiting conditions for operation. No inadequacies were noted in licensee identification and entrance into the required limiting conditions for operation.

At 2:20 p.m., the licensee returned Inverter #2 to service. Licensee investigation found no failure of components in the power supply. The licensee returned power to the main steam line radiation monitors (RE4299A, RE4299B, and RE4299C), the containment high range monitor (RE8241), and other vital instrument and controllers. No electrical grounds resulted from re-energizing the above loads. At approximately 3:00 p.m., licensee workers and supervisors held a meeting with the plant manager to discuss actions required for complete restoration of VA-20. The inspector attended this meeting.

At 7:25 p.m., the licensee re-energized ESAS Sensor Cabinet B and ESAS Actuation Cabinet 6 per procedure OP-2384 Rev. 3, "Engineering Safeguards Actuation System Operation." The inspector observed the restoration, and no inadequacies were noted. At 7:40 p.m., the licensee reenergized the "B" RPS channel and shut the affected reactor trip circuit breakers (TCB's 1,2,5,6). No electrical ground fault indications occurred during the restoration. At 8:39 p.m., the Spec-200 cabinet (Hot Shutdown electrical supply) was reenergized. No indications of an electrical ground fault existed. At 9:00 p.m., all TS limiting conditions were satisfied. At the end of the inspection period, the licensee continued troubleshooting and repair efforts for Static Inverter #6.

The inspector reviewed the design criteria for the affected vital inverters. According to the FSAR Section 8.6.1.2, the static inverters have the same design criteria as the vital station batteries. Thus, the applicable standards for inverters are: Safety Guide 6, Section 5.3 of IEEE 308, 1971; 10 CFR 50 Appendix A Criteria 1, 2, 3, 17, 18; and Section 4 of IEEE 279-1971. The inspector reviewed the Millstone 2 Safety Evaluation (May 10, 1974) dealing with the vital 120 volt a.c. system. No inadequacies were noted. The inspector reviewed the design criteria guidelines concerning the separation between the Class 1E Inverter #2 and the Non-Class 1E Inverter #6. Class 1E is a safety classification for electric equipment and systems needed for emergency reactor shutdown, containment and reactor heat removal, or for preventing a significant release of radioactive material. The inspector questioned the electrical separation of the two power supplies for VA-20 because the loss of Inverter #2 and, shortly thereafter, Inverter #6 resulted in a loss of VA-20. Also the design uses a common neutral between Inverters #2 and #6. To this end, the inspector discussed IEEE 384-1977, "Criteria for Independence of Class IE Equipment and Circuit" with the licensee's engineering staff. The discussion reviewed the as-built electrical connection between Static Inverters #2 and #6; the effects of a common neutral connection; and the licensee's investigation of the "root" cause of the event on September 7. The inspector concluded the electrical separation between the inverters was a parallel network of silicon-controlled rectifiers (SCRs). The inspector asked the licensee if an SCR is considered an "isolation" device between class 1E and non-class 1E power supplies, and the relationship to IEEE 384-1977 criteria. The inspector considers the separability of Class IE and non-class IE power sources, as it relates to the 120 volt a.c. vital instrument system, open pending further review. This is an unresolved item (UNR 88-19-01).

The inspector reviewed the preventive maintenance program for Static Inverters 1 through 6. The program consists of MP-2701J, "Battery Chargers and Inverters" and PT-21415, "Inverter and Static Switch Test." The time interval for preventive maintenance on the static inverters is the refueling outage interval. Routine inspection report 50-336/88-02 describes a previous failure of Static Inverter #1 in January 1988. The inspector discussed with the licensee the adequacy of the preventive maintenance program for static Inverters. This review will be continued in future inspections.

6.0 Observations of Physical Security (81064)

Selected aspects of site security were verified to be proper during inspection tours, including site access controls, personnel searches, personnel monitoring, placement of physical barriers, compensatory measures, guard force staffing, and response to alarms and degraded conditions. No inadequacies were noted.

7.0 Observation of Maintenance Activities (62703)

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

-- M2-88-09716, "Troubleshoot and Repair of Inverter 6."

No inadequacies were identified.

8.0 Observation of Surveillance (61726)

The inspector observed portions of surveillance tests to assess performance in accordance with approved procedures and Limiting Conditions of Operation, removal and restoration of equipment, and deficiency review and resolution. The following tests were reviewed:

-- SP2604M3, "Facility II Low Pressure Safety Injection (LPSI) Valve Operability Test" - August 26, 1988

-- SP21136 Rev. 5, "Safety Injection and Containment Spray System Valve Operational Readiness Test" - August 25, 1988

-- OP2384 Rev. 3, "Engineering Safeguards Actuation System Operation" - Section 7.0, Power Restoration - September 7, 1988

No inadequacies were noted.

9.0 Justification for Continued Operation on Service Water Temperature (93702)

The function of the service water system is to supply a dependable continuous flow of cooling water to the reactor building closed cooling water (RBCCW) heat exchangers, turbine building closed cooling water (TBCCW) heat exchangers, diesel engine cooling water heat exchangers, vital AC switchgear room cooling coils, chilled water heat exchangers, service water pump bearings and circulating water pump bearings. FSAR Section 9.7.2.1.3 describes the service water cooling system serving the TBCCW and RBCCW heat exchangers as the ultimate heat sink. The water source for the service water cooling system is Long Island Sound.

During the first week of August, service water intake temperature was trending towards the design basis value of 75 degrees F. Actual service water temperatures ranged between 70 - 72 degrees F.

On August 13, the licensee prepared a JCO (Justification for Continued Operation) evaluation for service water temperature up to 78 degrees F, not to exceed 30 minutes of operation. The licensee's evaluation reviewed the potential impact of increasing service water injection temperature to 78 degrees F for: Post-Loss of Coolant Accident (LOCA) Containment Response; Post-LOCA Core Cooling Consideration; High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) Pump post LOCA Operability; Normal Reactor Coolant system cooldown; 10 CFR 50 Appendix R cooldown; Effect on Safety-Related Components; and Environmental Qualification (EQ) concerns.

The licensee's corporate generation engineering department, along with site engineering, developed the JCO evaluation. The most limiting condition evaluated at Millstone 2 was the impact on the safety-related RBCCW heat exchangers, which occurred under analyzed accident conditions with service water inlet temperature of 78 degrees F . The design basis RBCCW outlet temperature during a design basis accident is 130 degrees F. This temperature is calculated to be reached during accident conditions when service water inlet is 78 degrees F, assuming design service water and RBCCW cooling water flows. No discrepancies were noted.

The inspector evaluated the impact of increasing service water inlet temperature from 75 degrees F to 78 degrees F, as it related to the design basis accident containment parameters, Emergency Core Cooling System (ECCS) pump operability, core cooling and effects on the RBCCW heat exchanger, vital D.C. switchgear coolers, service water pump bearings, and the emergency diesel generator heat exchanger performance. No inadequacies were noted in the licensee's evaluation. No evaluated design parameters were exceeded, and no safety-related equipment operability concerns were noted.

At the close of the inspection period, the service water inlet temperature had decreased to 68 degrees F. The licensee had implemented a tracking system for inlet service water temperatures at the time of temperature rise in late July and early August. The licensee's JCO in this case, as explained by the licensee's Vice President, Nuclear and Environmental Engineering, was a one-time justification based on existing equipment conditions, and any future high temperature conditions will have to be reviewed/analyzed separately. The inspector had no further questions.

10.0 Reactor Coolant System (RCS) Flow Degradation (92701)

On June 1, the licensee summarized the results of a study of reactor coolant flow at Millstone 2. The study was predicated on an unpredicted decrease in RCS flowrate following the refueling cycle 9 start-up in February 1988. The licensee informed the inspector of this issue on July 29.

The following conclusions resulted from the licensee's study: RCS flow loss is predominantly in loop 2; the amount of flow loss is estimated at 3,000 gallons per minute (GPM) per cycle since the beginning of cycle 5; and the total RCS flow reduction is 12,000 GPM. An estimated maximum of 400 additional steam generator (SG) tubes may be plugged at the end of cycle 9, further reducing RCS flow.

RCS flow rate is a significant input for the thermal-hydraulic analysis of Millstone Unit 2. A minimum value is assumed in the accident analyses to assure acceptable results for design basis accidents. The inspector reviewed FSAR Table 14.6-1, "Key Parameters used in the Loss of Flow Analysis," and verified the minimum value for RCS flow in design basis accidents is 340,000 GPM. A 13,000 GPM allowance is utilized for measurement error (4% of design flow) to give a minimum allowable measured RCS flow of 353,000 GPM.

The inspector reviewed the licensee's surveillance procedure SP21006, "Core Flow Determination." The surveillance measured RCS flow on a monthly frequency as determined by TS 4.2.6. The inspector discussed the observed surveillance on September 7. No inadequacies were noted in the performance or review of SP 21006. The measured RCS flow on September 7 was 363,185 GPM. This value of RCS flow is 10,185 GPM greater than minimum allowable measured flow (353,000 GPM) in TS 4.2.6 and the FSAR Table 14.6-1.

The inspector reviewed the licensee's TS amendment change for RCS flow from 350,000 GPM to 340,000 GPM October 27, 1986, and the subsequent NRC TS Amendment No. 113 issued on December 8, 1986. The change addressed plant operational restrictions based on a small margin between the TS limit for minimum RCS flow and future plugging and/or sleeving of SG U-tubes.

The NRC concluded, based on licensee submittals, that 340,000 GPM RCS flowrate was acceptable with the provision that the licensee perform additional Loss-of-Coolant Accident (LOCA) analysis to support extended coastdown operation beyond Cycle 8 at the reduced RCS flowrate. The additional LOCA RCS flowrate calculations support TMI Action Item II.k.3.31, "Plant Specific Calculations to show compliance with 10 CFR 50.46."

The inspector reviewed the licensee's engineering evaluation in light of cycle 9 start-up RCS flow calculations. Calculated RCS flowrate at the beginning-of-cycle (BOC) was 361,000 GPM. The licensee's evaluation reviewed flow results for loop 1 and loop 2, steam generator plugging and sleeving effects, reactor coolant pump motor performance, instrument errors, and Loop 2 reactor coolant pump performance. A cause of RCS unpredicted flow degradation that was not initially ruled out by the licensee was the loop 2 reactor coolant pumps (RCPs). The licensee conducted a follow-up review of RCP performance on June 2. The inspector reviewed the results. The review considered pump vibrations, RCP speed, RCP differential pressure trend plots, Baltimore Gas and Electric Calvert Cliffs Units 1 and 2 (sister plant) RCP performance, and the nuclear industry experience with Byron-Jackson RCPs. The licensee's review was inconclusive. Additional actions were identified to collect on RCP differential pressure vs. motor current data ,and to use existing RCS flow models to evaluate pump wear scenarios.

The inspector will continue to follow future licensee actions concerning the "root cause" for degradation of RCS flow in future inspections.

11.0 Concentration Reduction in Boric Acid Storage Tank (BAST) (37701)

On April 29, the licensee submitted to the NRC a proposed revision to the TS to reduce the BAST boric acid concentration to the level where heat tracing of the boric acid make-up system would no longer be required. The current system utilizes heat tracing and Boric Acid Storage Tank heaters to maintain the solubility of borated water. The BAST water has a boric acid concentration between 6-12% by weight. The requirement for minimum temperature vs. percent weight is covered in TS 3.1.2.7 and 3.1.2.8.

The proposed change as described in PDCR 2-15-88, "Boric Acid Correction Reduction" eliminates heat tracing and tank heating between the BAST and the suction of the charging pumps. The change also reduces the boric acid concentration in the system to 2.5% to 3.5% weight percent. The inspector reviewed PDCR 2-15-88 and three general areas were discussed with the licensee: the interface between the TS change evaluations and the new cycle 10 fuel analysis study; procedures affected by the proposed change; and the details of the safety evaluation. The inspector had no questions on the PDCR at this stage. The licensee's engineering department presented the PDCR to the PORC members for comment. The PORC is scheduled to review the PDCR near the end of September 1988.

The purpose of the TS change request is to prevent equipment unavailability due to piping blockage from precipitated boric acid, and to reduce radiation exposures by reducing the maintenance of heat tracing.

The licensee plans to do the boric acid reduction at power utilizing in-service test T-88-42, if the NRC approves the TS amendment.

The inspector will follow future actions on the implementation of boric acid reduction.

12.0 Update on the Dual-Role SRO/STA Issue (71707)

As initially reported in NRC inspection Report 50-336/88-02, the dual-role Shift Supervisor/Shift Technical Advisor (SS/STA) does not meet the current Commission policy statement for degreed engineering expertise on shift.

In a memo dated August 9, 1988, the Secretary of the Commission informed the NRC Executive Director for Operations that the Commission had voted on this matter. Existing dual role SS/STA personnel and the thirty people who had already graduated from the Memphis State University and Thames Valley State Technical College programs, as well as the eleven candidates enrolled prior to October 1, 1987, may serve as dual role SSs/STAs upon successful completion of their studies. Subsequent dual role SS/STA candidates are to meet the current NRC policy that such personnel are to have an engineering or technical BS degree.

13.0 Review of Periodic Reports (90713)

Upon receipt, a periodic report 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 performance specifications. The inspector also ascertained whether the reported information should be classified as an abnormal occurrence. The following reports were reviewed:

-- Monthly Operating Report for Millstone 2 for July, 1988
-- Monthly Operating Report for Millstone 2 for August, 1988
-- Refueling and Maintenance Outage Report for 1988

No unacceptable conditions were identified.

14.0 Committee Activities (40700)

The inspector attended meetings 2-88-137, 2-88-141, and 2-88-144 of the Plant Operations Review Committee (PORC) meetings on August 8, August 30, and September 7, respectively. The inspector noted by observation that committee administrative requirements were met for the meetings, and that the committees discharged their functions in accordance with regulatory requirements. The inspector observed a thorough discussion of matters before the PORC during meetings and a good regard for safety in the issues under consideration by the committee. No inadequacies were identified.

15.0 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. No written material was given to the licensee during the inspection period.