U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report:	50-245/85-19; 50-336/85-25		
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:	June 30, 1985 through July 22, 1985		
Inspectors:	Ebe McCabe, for John T. Shedlosky, Senior Resident Inspector		7/31/85 Date
	Ebe McCabe, for David R. Lipinski, Resident Inspector		7/31/85 Date
Approved:	Ebe McCabe E. C. McCabe, Chief, Reactor Projects Section 3B		7/31/85 Date

Summary: Routine NRC resident (85 hours) inspection of plant operations, equipment alignment and readiness, radiation protection, physical security, fire protection, design changes and modifications, surveillance, and close-out of Inspection & Enforcement Bulletin 80-25. No violations or unacceptable conditions were identified.

DETAILS

1. Plant Status

Unit 1

The reactor operated at full power except for planned power reductions for surveillance testing and preventative maintenance. On July 16, Unit 1 set a record for consecutive days of service for a domestic Boiling Water Reactor (BWR) of 345 days. As of the end of the inspection period, Unit 1 has been in service for 351 days. The previous record, held by Big Rock Point in Charlevoix, Michigan, was 344 days.

Unit 2

The reactor was taken critical at 2115 on June 30 ending a 135 day refueling and maintenance outage. A period of startup and power ascension testing followed and the unit reached full (100%) power at 2210 on July 11. The unit remained at full power until 1533 on July 15, 1985 when pressurizer spray valve problems led to a decrease in plant pressure and a Thermal Margin/Low Pressure scram. Investigation of spray valve problems revealed an error in a recently implemented design change to the pressurizer spray valve control circuits. Following maintenance on the pressurizer spray valves, repairs to secondary (steam) plant equipment, and a review of all design changes implemented during the recently completed refueling outage, Unit 2 was taken critical at 0310 on July 19, synchronized to the grid at 0945 on July 20 and brought to full power on July 22. The license briefed NRC Region 1 management on the results of the design change review prior to startup.

2. Loss of Pressure and Scram of July 15 - Unit 2

   a. Overview

   On July 15, 1985 at 1533, Millstone Unit 2 scrammed from full power as pressure dropped through the Thermal Margin/Low Pressure (TM/LP) setpoint (approximately 2130 psia at the time of the scram). No Emergency Core Cooling System (ECCS) challenge occurred. All safety systems functioned properly. The minimum pressure reached was 1725 psia. The pressure drop was attributed to failures in both pressurizer spray valves. Following a containment entry, both spray valves were found stuck partially open. The valves were isolated, and normal pressure was restored. The licensee could not definitively determine the cause of the spray valves binding; however, stem/packing interference is suspected. Prior to startup, the valves were satisfactorily tested. A controls deficiency was identified during the investigation. However, this problem did not cause the transient. This matter will be more thoroughly reviewed following the licensee's report of the event.

b. Event Chronology

The plant was initially at 100% power with 4 Reactor Coolant Pumps (RCPs) operating, Presurizer (PZR) pressure 2275 psia, Cold Leg Temperature (Tc) 548 degrees F, and Hot Leg Temperature (Th) 596 degrees F. Data collection at 100% power was in progress to complete post refueling startup and power ascension testing. To ensure that boron concentration remained equalized between the PZR and the Reactor Coolant System (RCS) loops, essentially continuous pressurizer sprays were being "forced." This latter evolution assures that plant temperature changes only (not temperature induced boron concentration changes during PZR outsurges) cause plant power changes during testing conducted at several power plateaus.

A brief event chronology follows. Times are approximate and are reconstructed from installed recorders, the logs, and inspector notes.

TIME	ACTION/EVENT/OBSERVATION
1430	Minor difficulty controlling pressure using proportional heaters is experienced. Intermittent operation of backup heaters is necessary. Shift Supervisor (SS) requests maintenance to check pressurizer heaters.
1527	Reactor Protective System (RPS) TM/LP Pre-Trip Alarms received intermittently.
1528	RPS TM/LP Pre-Trips continue intermittently. Control Operators call SS to control boards because of PZR pressure decline.
1529	All PZR backup heaters are on. SS directs a gradual load reduction on the Main Turbine. RPS TM/LP pre-trips received steadily. Both spray valve controllers indicate valve shut orders.
1530	PZR pressure drops past 2200 psia. SS diagnoses problem as a stuck open spray valve.
1531	Both spray valve controllers taken to "manual" and moved from full shut to full open and back to full shut in an effort to free the stuck spray valve. PZR pressure continues to drop through 2180 psia.
1533	RPS trips on TM/LP. All scram breakers open. Main Turbine trips on reactor scram. Plant loads shift to off-site power. Normal scram transient behavior is observed (and reconstructed).
1537	PZR pressure reaches 1790 psia then begins to recover (Th is 525 degrees F).
1540	PZR pressure reaches 1853 psia then begins to decline.

1600 Preparations to enter containment to investigate problems with the spray valve.

1615 PZR pressure is at 1740 psia with Th at 530 degrees F. The pressure decline is continuing.

1623 SS directs RCPs "B" and "D" secured (one per steam Generator) to reduce spray flow.

1634 Containment is opened.

1636 Both spray valves are isolated by shutting their manual outlet isolation valves (2-RC-252 and 2-RC-253). As spray valve 2-RC-100E was isolated, the operator reported a decrease in flow noise through the valve. The pressure drop is stopped. The minimum pressure reached was 1725 psia. At that time Th was 535 degrees F. Pressure begins to rise.

1720 The outlet isolation valve for spray valve 2-RC-100F is opened.

1721 RCP "D" is started; pressure begins to drop.

1725 Spray valve 2-RC-100F is isolated.

1740 Observation of spray valve movements during troubleshooting leads to the discovery that valve 2-RC-100E responds to the controller for 2-RC-100F and vice versa.

1810 RCP "B" is started. Plant pressure is being controlled at 2250 plus/minus 25 psia using the PZR heaters. RCS temperature is 530 degrees F. All four RCPs are running.

c. Assessment of Core Conditions

The thermal hydraulic conditions in the core were closely monitored during the pressure transient by both the inspector and the licensee. Forced convection through the core and both steam generators was maintained was maintained throughout the event. Therefore, RCS Th closely reflected core exit temperature. The inspector plotted actual Tc, Th, and PZR pressure as well as the saturation temperature corresponding to PZR pressure for the period from 1525 to 1545. At the time of the scram, the subcooling margin was approximately 98 degrees F. At its minimum, just after the scram, subcooling margin was approximately 74 degrees F. At the time the lowest pressure (1725 psia) was reached, subcooling margin was at least 75 degrees F. Based on this behavior, it is concluded that the core heat transfer regime did not depart from nucleate boiling during the transient, thus core damage is unlikely.

Following the scram, dose equivalent Iodine activity in the reactor coolant was monitored as an indicator of core damage. Peak Iodine activity measured was 0.02 micro-Curies per milli-litre. This peak is low when compared to post-scram peaks observed during the preceding cycle. From this indicator, no core damage is inferred.

The Safety Limits on temperature, pressure, and power of Technical Specification 2.1 were not exceeded. At the existing conditions of temperature and power at the time of the scram, the safety limit on pressure was approximately 1750 psia.

d. Pressurizer Spray Valve Summary

Two spray valves are provided. The valves are Fisher Controls 3-inch angled valves. The valves are air operated and move in proportion to the demand. Ordered position is displayed in the control room. No valve stem position indication is provided. The control system was changed during the preceding refueling outage from a General Electric "GE/MAC" current loop to a Foxboro "Spec.200" current loop (PDCR 2-8-85). Manual inlet and outlet isolation valves are provided. No remotely operated isolation or blocking valves are provided. Two 3/4-inch bypass throttle valves are installed. These provide a small flow of water to warm the piping between the spray valves and the PZR and to cool the PZR spray head. This action limits the thermal stresses experienced when spray is initiated.

Routine preventative maintenance on the spray valves consists of valve re-packing. This was accomplished during the preceding refueling outage.

Troubleshooting spray valve 2-RC-100E showed normal valve motion through its range of travel. However, the valve was approximately 1/4 inch (stem movement) from full closure at the end of the air operator's stroke. The stroke was adjusted to fully shut the valve. Valve 2-RC-100E had no prior history of mechanical problems.

Troubleshooting spray valve 2-RC-100F showed erratic valve motion; the valve bound in intermediate positions. The valve was disassembled to attempt to identify and correct the cause of the mechanical binding. No obvious valve damage was observed; however, some scratches were found on the globe and stem and a packing ring was displaced ("cocked"). The valve was rebuilt and tested satisfactorily. Valve 2-RC-100F had no recent history of mechanical problems since 1978, when the globe separated from the stem.

The throttled position of the spray valve bypass valves was also adjusted.

Both spray valves are presently in service. A small amount of spray flow through the valves continues and is compensated by maintaining one group of PZR backup heaters (approximately 300 kilo-Watts) energized. The

licensee has deemed this to be acceptable until the next extended outage. The inspector confirmed that plant pressure is being maintained as prescribed by Technical Specifications.

e. Related Systems Effects

At Millstone Unit 2, pressure is normally maintained by two groups of proportional heaters (150 kil-Watts per group). At set pressure (nominally 2250 psia) half power is supplied to the proportional heaters. At 2275 psia, proportional heaters are off and at 2225 psia proportional heaters are at full power. At 2200 psia, decreasing, four groups of backup heaters (300 kilo-Watts each) are energized; these are de-energized at 2225 psia, increasing.

The spray valves normally do not cycle. Both spray valves begin to open at 2300 psia and both valves are fully open at 2350 psia. To "force" sprays in order to equalize PZR and RCS loop boron concentrations, the spray valve controller setpoints are reduced to initiate spray flow at normal presure and backup heaters are manually energized to maintain pressure. Operation with known spray valve leakage and partial backup heaters is equivalent to the routine evolution of "forcing" sprays.

Emergency Core Cooling System (ECCS) injection sources at Millstone Unit 2 include the normally running positive displacement charging pumps, High Pressure Safety Injection (HPSI) pumps, Low Pressure Safety Injection (LPSI) pumps, and Safety Injection (SI) accumulators. The low pressure Safety Injection Actuation System (SIAS) initiation occurs at PZR pressure of 1600 psia. Charging pumps inject via their normal flow path. Charging pump flow does not change as these positive displacement pumps are driven at constant speed. The HPSI pumps nominal shutoff head is 1600 psia. Both LPSI and SI accumulators inject only at further reduced pressures. Since the minimum pressure reached, 1725 psia, was well above the SIAS setpoint of 1600 psia ECCS was not initiated. Had the initiation occurred due to an early initiation of SIAS, an RCS thermal transient would not have resulted as the only injection source (charging pumps) would have injected via their normal paths.

Some time following the scram, main condenser vacuum was lost. Troubleshooting revealed that a steam deflector plate had carried away tearing the low pressure turbine exhaust flexible coupling ("exhaust boot") to the main condenser. Resulting repairs dominated the duration of the forced outage resulting from this event.

f. Deficiencies in Design Change Control

In the course of troubleshooting the problems with the spray valves, it was discovered that the controller for valve 2-RC-100E actually controlled valve 2-RC-100F and vice versa. The previously installed General Electric "GE/MAC" controllers had been replaced by Foxboro "Spec.200" controllers as part of Plant Design Change Request (PDCR) 2-8-85 which

was implemented during the preceding refueling outage. Although the spray valves and their controls are not themselves safety-related, other changes implemented via that PDCR involved safety-related pressure instrumentation, power supplies, and safety/control isolation. Thus PDCR 2-8-85 was handled by the licensee as a safety-related design change.

The error which resulted in the cross-operation of the controllers was not due to an installation error. Rather, the wiring drawings for the spray valve controllers were drawn in error. Three sheets, vice one long sheet, were used and the error occurred in the sheet transition labeling.

The post-design change testing included individual calibration of the safety-related PZR pressure instrument channels and was accomplished with actual test pressures applied to the pressure transmitters. Testing of the non-safety-related PZR spray valve controllers did not include observation of spray valve stem movement in response to position orders from the new controllers. Instead, controller output currents were measured. During the loop current checks, a discrepancy was identified between the control room and hot shutdown panel indications. Resolution included reversing the valve identifier tags at the hot shutdown panel. This appears to indicate a lack of thorough investigation to resolve the discrepancy, since the actual cause was the error in design as previously discussed. Following repairs and correction of the design change error, adequate operational testing of the entire PZR spray valve control system was conducted.

Prior to restart of Unit 2, the licensee reviewed all 65 plant design changes implemented during the preceding refueling outage in order to identify changes of potentially operational significance. Seventeen such changes were identified for further followup to ensure that adequate operational verification of affected system performance had been conducted. The inspector independently screened the PDCR list and found the licensee's selection to be acceptable. On July 18, 1985, representatives of the Northeast Nuclear Energy Company met with NRC Region 1 management to discuss the results of the design change review and other matters relating to change and testing deficiencies. Meeting report 50-336/85-26 discusses design change issues in detail.

3. Monthly Surveillance Observation

In order to ascertain that surveillance of safety-related systems and components is being conducted in accordance with license requirements, various testing evolutions were observed at both units 1 and 2 as described below:

a. Unit 1

-- "Reactor Low Water Level Scram and Low Low Level Isolation Functional and Calibration Test" per SP408C Revision 3 and "Reactor Low Low Water Level Functional and Calibration Test" per SP412C Revision 9 (with change 1).

These tests verify the calibration and functional operability of Yarway level switches used in the Reactor Protective System (SP408C), the Primary Containment Isolation System (SP408C), and the Emergency Core Cooling System (SP412C). The tests use the "Cold Water Cal.- Dry Method" as described in Yarway manual 21-101N "Instruction Manual for Yarway Liquid Level Indicator and Accesory Equipment for Millstone Point" section 5. A 3-point calibration check as well as a demonstration of control logic relay actuation as a result of the physically imposed differential pressure were featured in the tests. Test results were satisfactory. The test gauges had been calibrated and were included in the licensee's periodic calibration program. Restoration involved two-party checks and documentation of final instrument valve positions. The physical condition of the environmental enclosures (including the condition of door gaskets) which mitigate the effects of postulated high energy pipe breaks upon these instruments was found to be satisfactory. The inspector concluded that both tests satisfied the requirements of Technical Specifications 3/4.1.A and 3/4.2.A&B both in terms of the extent of testing and of the periodicity of testing. No unacceptable conditions or practices were observed.

-- "Manual Scram Functional Test" per SP609.1 Revision 2.

This test demonstrates, in turn, the ability of each reactor manual scram push button to initiate a scram (de-energize scram relays) in its respective trip system. The inspector found the test to meet the requirements of Technical Specification 3/4.1.A. No unacceptable conditions were observed.

-- "Drywell High Pressure Scram and Constant Isolation Functional and Calibration Check" per SP408H Revision 4.

This test verifies the calibration and functional operability of Barton Model 288 pressure switches in the Reactor Protective System and Primary Containment Isolation System. The Barton Model 288 instrument manual outlines a calibration method which exercises the instrument then measures instrument response at 3 points - at the low end of the scale, at mid-range, and at the high end of the scale. The licensee's test procedure describes testing the instruments in the "as-found" condition by venting the instrument, observing the 0 pound per square inch gauge value, then applying air pressure to the instrument to observe the contacts close at 1.8 psig (nominal) as measured by a precision gauge. The inspector discussed the discrepancy with Instrumentation & Controls Department senior technicians and foremen. The inspector determined that the instruments in question are used as an input to the Reactor Protective System rather than as an indicator, as is assumed by the manufacturer's procedure. The inspector concurred in the adaptation of test points to check the instruments' zero response and to check the response at the most significant point (the reactor trip set point). The requirements of Technical Specifications 3/4 were met. No unacceptable conditions or practices were observed.

b. Unit 2

-- "Calibration of Excore Nuclear Instruments to Incores" per SP2401E Revision 7 and "Power Range Safety Channel and Delta T Power Channel Calibration" per SP2601D Revision 4.

These tests are conducted to satisfy the channel calibration requirements for reactor power sensors as described in Technical Specifications 3/4.3.1.1. The amplifying notes 2, 3, and 4 of Table 4.3-1 are particularly germaine. SP2401E uses a desk top computer (Commodore PET Model 4032) to assist the operator in data collection and analysis. The program disc is controlled personally by the Instrumentation and Controls supervisor and provides its revision number to the operator to permit verification of its applicability. The test also uses data from the plant process computer and a digital volt meter (DVM) of suitable accuracy. SP2401E causes the individual excore detector outputs to be adjusted to reflect the neutron flux shape measured by the fixed incore detectors. SP2601D uses installed test instruments only. It adjusts neutron channel outputs (2 detectors per channel) to reflect primary calorimetric power. No unacceptable conditions or practices were noted.

-- "Facility I Containment Spray Alignment and Operability Test" per SP2606C Revision 3, "Facility I Containment Spray Pump Operability Test" per SP2606A Revision 3, and "Containment Spray Pump 'A' Operational Readiness Test" per SP21116.

These tests collectively demonstrate the operability of the "A" Containment Spray System as described in Technical Specification 3/4.6.2.1 (surveillance "a" items 1,2,3,4 and 6) SP2606C tests all remotely operated valves through a full stroke cycle and verifies the position of manual valves in the flow path. SP2606A tests the pump by remotely starting it, by verifying that the expected pressure head is developed with flow through the recirculation path, and by observing the pump complete a 15 minute sustained run. SP21116 measures pump vibration and hydraulic characteristics as required by ASME Boiler & Pressure Vessel Code Section XI Chapter IWP. The inspector verified the system flow path line-up following restoration. No unacceptable conditions or practices were observed.

-- "LPSI Pump 'A' Operational Readiness Test" per SP21114.

This test is part of the licensee's pump and valve In Service Test (IST) program which assesses pump readiness according to ASME Boiler & Pressure Vessel Code Section XI Chapter IWP standards. The testing acceptance criteria were attained. No unacceptable practices were observed.

4. Engineered Safety Features (ESF) System Walk-Down - Unit 2

In addition to cursory inspection of safety-related systems and components during routine daily facility tours, the inspector conducted a detailed walk-down of the "A" containment spray system. This walk-down included the condition of piping and valves in the suction lines from the containment sump and the Refueling Water Storage Tank; the condition of pumps, valves, piping and heat exchangers in the Facility I Safeguards Equipment Room (Auxiliary Building -45 foot elevation); the condition of flood control doors, sump pump suctions, and ventilation in the Facility I Safeguards Equipment Room; the condition of valves and piping in the Enclosure Building west penetration area; and the condition of accessible mechanical and hydraulic pipe snubbers. The inspection verified the position of all manual valves in the injection flow path. Additionally, the inspector witnessed tests of pump operability and motor operated injection valve operability. No unacceptable conditions were observed.

5. Extended Observation of the Control Room Environment - Unit 1

Observation of Unit 1 control room practices was conducted over an extended 4 hour period on July 9. The shift included 2 Senior Reactor Operators and 2 Reactor Operators as well as several non-licensed equipment operators. Logs were up-to-date and sufficiently detailed to permit reconstruction of plant conditions and events. The control room was clean and free of debris. The inspector viewed the interiors of the control boards and found only a light layer of dust with no excessive buildup on circuit components, cables, and housings.

Routine control room transactions were conducted smoothly and with due prudence. Procedures were observed to be in use. Emergency procedures were found to be conveniently available to operators. No personnel were observed to loiter in the control room. No extraneous reading material was in evidence. The recent re-arrangement of furniture has improved the Control Operators' view of the panels and their responsiveness to changing conditions.

No unacceptable practices were observed.

6. Plant Startup and Power Ascension Testing - Unit 2

Millstone Unit 2 was taken critical at 2115 on June 30. Power ascension tests were conducted per special test T19-85. The inspector observed segments of testing involving determination of incore neutron detector operability, moderator temperature coefficient, isothermal temperature coefficient, reactivity anomalies, peaking factors, and shutdown margin. The inspector noted that the results of testing conducted indicated that shutdown margin, moderator temperature coefficient, total planar radial peaking factor, and total integrated radial peaking factor were within the limits of Technical Specifications. No independent calculation of results and no assessment of computer analytical methods or software was attempted. The results of the startup and power ascension test program may be reviewed in greater detail in a future inspection (50-336/85-19-01).

7. Followup of Inspection & Enforcement Bulletin 80-25, Target Rock Safety Relief Valves"

This Bulletin discusses failures in Target Rock safety-relief valves at Boiling Water Reactors (BWRs). The Bulletin requires 3 actions of BWR licensees. Two of these are surveys with results to be reported to the Office of Inspection and Enforcement. The information has subsequently been incorporated into NUREG/CR 3794 "Close-Out of I&E Bulletin 80-25/Operating Problems with Target Rock Safety-Relief Valves at BWRs". The third action requires the inclusion of valve test requirements in plant procedures for safety-relief troubleshooting and maintenance. The licensee's submittal included a commitment to make the required procedural changes. The inspector reviewed the licensee's procedure index and found that one procedure, MP717.1, addresses troubleshooting and repairs for these valves. The inspector confirmed that MP717.1 "Maintenance of Target Rock Relief Valves" Revision 7 includes the testing guidance of the Bulletin in section 4, "Precautions." The procedure specifically references the Bulletin to help assure that future revisions will not inadvertently delete the guidance. This Bulletin is closed.

8. Review of Licensee Policies Regarding Control Rod Manipulations

As a result of several events at other operating reators which involved mispositioned control valves, the inspector reviewed licensee policies regarding the positioning of control rods and operation with mispositioned control rods.

a. Unit 1

-- Plant procedure OP204 "Power Operation" requires a Reactor Engineer to be present for all but the most minor power changes. Examples of conditions in which a Reactor Engineer is required include control rod pattern adjustment, power ascension following turbine synchronization, and fuel preconditioning. A "Reactor Engineering Information Sheet" is prepared to address times when a Reactor Engineer is not stationed. The sheet includes the name and number of the "on-call" Reactor Engineer, reactivity trends expected, trends in thermal limits, a flow reduction/rod insertion sequence for unplanned power reductions, and a log for all rod motion. The sheet is approved both by the Station Reactor Engineer and the Operations Supervisor. The resident inspector has often observed this system in use.

-- A scroll-type list of the rod sequence for startup/rod withdrawal and, conversely, for shutdown/rod insertion ("blue box") is mounted at the rod control station. Procedures direct all rod motion be conducted in accordance with the "blue box" (OP204).

-- Millstone Unit 1 uses a computer-based Rod Worth Minimizer to limit the potential for large reactivity excursions during postulated rod ejection or rod drop accidents. Plant procedures to Hot Standby or Hot Shutdown" detail conditions and precautions for the bypassing of the Rod Worth Minimizer (RWM).

-- Individual rod scram capability exists as part of the installed scram timing equipment. Use of this equipment is limited to scram testing and emergencies involving an Anticipated Transient Without a Scram.

-- Millstone Unit 1 does not incorporate a Rod Sequence Control System (RSCS). The RSCS at other plants has complicated unscheduled shutdowns at several plants. Use of the "emergency in" and "notch override" features of the Reactor Manual Control System at Millstone Unit 1 together with the "blue box" does not lead to difficulties in shutting down.

-- Training in proper control rod movements, consequences of improper rod patterns, RWM, and reactivity addition accidents is conducted as part of license training and requalification training.

-- No unacceptable policies or procedures were noted.

b. Unit 2

-- Recovery from a dropped or mispositioned Control Element Assembly (CEA) is addressed in detail in Technical Specification 3.1.3.1. Plant procedure AOP 2556 "Dropped CEA Recovery" implements Technical Specification requirements.

-- CEA position indicator operability is addressed in detail in Technical Specifiation 3/4.1.3.3. Sections 4.2 and 4.3 of plant Procedure OP2349B "Process Computer" addresses operation allowable under the Action Statement in the event of pulse counter position indicator inoperability.

-- Training in proper CEA manipulation, rod position limits, and reactivity addition accidents is included in the requalification training and license training programs.

-- No unacceptable policies or procedures were identified.

9. Facility Tours

The inspectors toured Units 1 and 2 as well as associated outbuildings and related yard areas throughout the inspection period. Improvement in general housekeeping was particularly noted in Unit 2 areas due to post-outage cleanup efforts. Continued decontamination efforts in the Unit 2 auxiliary building -45.5 foot elevation and in the west penetration room of the enclosure building appear warranted. Unit 1 areas remain relatively free of surface contamination areas. The inspectors observed the material condition and monthly checks of fire extinguishers required by Technical Specifications as well as the integrity of a sample of fire barriers. No deficiencies were observed.

The inspector noted that improvements in temporary solid waste storage to the east of Unit 1 were in progress. Personnel access to the Condensate Storage Tank and associated valves and to the nitrogen evaporator were not impaired. A walk-down of the protected area boundary revealed no deficiencies in the protective barriers. Personnel and package access procedures were observed at both the Condensate Polishing Facility (CPF) and Personnel Access Point (PAP). No unacceptable conditions were observed.

10. Steam Generator Repair Summary - Unit 2

The inspector reviewed the licensee's summary of steam generator repairs submitted in accordance with Technical Specification 4.4.5.1.5a in a letter dated June 18, 1985. Some adjustment to the extent of repairs forecast in Inspection Report 50-336/85-13 is necessary. A total of 63 tubes were plugged and 2918 tubes were repaired by sleaving. The licensee's report contains greater detail.

11. Exit Interview

At periodic intervals during the inspection, meetings were held between licensee site management concerning the inspection scope and findings. One item, a detailed technical review of Unit 2 startup test results (50-336/85-19-01), remains open.

At no time during the inspection was written material concerning inspection findings provided to the licensee.