U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-289/85-10

Docket No. 50-289

License No. DPR-50 Priority -- Category C

Licensee: GPU Nuclear Corporation
Post Office Box 480
Middletown, Pennsylvania 17057

Facility: Three Mile Island Nuclear Station, Unit 1

Inspection At: Middletown, Pennsylvania

Inspection Conducted: March 8, 1985 - April 8, 1985

Inspectors: E Conner for 4/23/85
F. Young, Resident Inspector (TMI-1) Date

E Conner for 4/23/85
R. Conte, Senior Resident Inspector (TMI-1) Date

Approved By: E L Conner 4/23/85
E. Conner, Chief, Reactor Projects Section Date
No. 1A, Division of Reactor Projects

Inspection Summary: This routine safety inspection (169 hours) reviewed: routine shutdown plant activities including those related to steam generator repair; selected equipment operability; administrative control implementation; and licensee action on previous inspection findings.

Results: Licensee management involvement in daily operations of the plant was aggressive and proper attention was given in resolving plant problems. Review of Safety Evaluations associated with Reactor Coolant Pump Cracked Shaft and Once Through Steam Generators missing plugs found the licensee's review to be complete and accurate. The licensee properly implemented administrative control procedures associated with emergency shift duty rosters. The licensee either initiated appropriate action or completed commitments related to previously identified inspection findings.

DETAILS

1.0 Introduction

This inspection report documents the activities conducted by the resident inspectors. The overall purpose of the inspection was to assess the licensee's activities as they relate to the reactor safety and worker radiation protection for a plant in a shutdown mode and to assess plant readiness for restart.

On a sampling basis, the inspectors made this assessment by reviewing licensee's advancements, through licensee interviews, actual observation of activities (where possible), measurement of radiation levels, and review of listed documents or records. Within each area, the inspector listed the specific purpose of review (or verification), scope of the review (or specific inspector activity) and findings.

2.0 Plant Operations During Long Term Shutdown

2.1 Routine Review

The resident inspectors periodically inspected the facility to assess the licensee's compliance with general operating requirements of Section 6 of the Technical Specifications in the following areas:

-- review of selected plant parameters for abnormal trends;

-- plant status from a maintenance/modification viewpoint including plant housekeeping and fire protection measures;

-- control of ongoing and special evolutions, including control room personnel awareness of these evolutions;

-- control of documents including log keeping practices;

-- implementation of radiological controls; and,

-- implementation of the security plan including access control, boundary integrity and badging practices.

The inspectors focused on the following areas:

-- the control room during regular and backshift hours which included the selected sections of the shift foreman's log and control room operator's log for the period March 8, 1985, through April 8, 1985, and selected sections of other control room daily logs;

-- areas outside the control room during regular and/or back shift hours on March 11, 12, 15, 18, 19, 20, 21, 25, 27, 28, and April 2 and 5, 1985; and,

-- selected licensee planning meetings.

Based on the review of the various licensee activities noted above, the inspector identified no conditions adverse to nuclear safety or regulatory requirements. Personnel stationed in the control room presented a posture of overall control of daily activities, including problem areas that needed resolution. The planning meetings indicated an attempt to proceed safely with daily activities, including surveillance and maintenance, and to resolve any inter-departmental interface problems. Licensee upper management continued their detailed involvement in site activities.

2.2 Once Through Steam Generator (OTSG) Repairs

On March 29, 1985, GPUN decided to plug all tubes to the 40% through-wall criterion in the existing Technical Specifications (TS). The licensee had proposed modified criteria which would have allowed some tubes with indications greater than 40% through-wall wastage to remain in service. Previously, the licensee had plugged all tubes that did not meet this new plugging criteria. Although the licensee proceeded to plug these tubes, the utility intended to pursue the approval of the new criteria with the NRC staff. The plugging of these tubes began on April 2, 1985, and was to be completed by April 9, 1985. The OTSG Hot Functional Testing (HFT) was scheduled for the week of April 8, 1985, to establish a new base line primary to secondary leakrate.

During the inspection period, NRR representatives and the Resident Inspector reviewed site records to resolve some questions associated with Eddy Current Testing and the licensee's analysis. This analysis was to be used to justify continued operations of tubes with indications greater than 40% through wall. The staff was still reviewing the licensee analysis and new information from the site when the licensee decided to plug the tubes in question.

In July 1984, following a HFT, it was discovered that seven rolled plugs, developed by Westinghouse (W) to plug the OTSG tubes, had dislodged from their installed positions. Four of the plugs were from the bottom tube sheet of OTSG "A", two were from the bottom tube sheet of OTSG "B", and one was from the upper tube sheet of OTSG "A". The plug from the upper tube sheet of OTSG "A" has been recovered, but the remaining six plugs are still missing. The licensee reported this matter by LER 84-04. Further, by letter dated October 23, 1984, in accordance with the provisions of 10 CFR Part 50.59, the licensee submitted safety analysis reports documenting its review of the Westinghouse rolled plug qualification program, the cause of the dislodged plugs and the test and repair program implemented to ensure the installed plugs have adequate integrity under postulated transient and accident conditions. The reports also addressed the effects of the loose plugs on the core and other reactor coolant system components and on the safety of plant operation.

The NRC staff issued an SER, dated March 5, 1983, to evaluate the GPUN program to identify and correct defective W rolled OTSG plugs, and evaluate safety aspects of operations with loose/missing W rolled OTSG plugs.

The inspector reviewed the licensee submittal and the NRC staff SER to ensure that the information was consistent. On a sampling basis, the inspector reviewed the field records to independently determine accuracy of the information used as the basis for the evaluation. The field records were found to reflect what was stated in the SER. This item (289/84-LO-04) is resolved.

2.3 Reactor Coolant Pump 1B Crack Shaft

On January 27, 1984, with TMI-1 in cold shutdown and reactor coolant pump 1B (RCP-1B) in operation, pump shaft vibration increased from the normal range of 9 to 12 mils to 12 to 15 mils. On January 30, 1984, vibration increased to 19 mils and then continued to increase to 28 mils on January 31. At this point the pump was shut down. Ultrasonic inspection of the pump shaft indicated an area of discontinuity in the shaft near the impeller, which coupled with analysis of vibration data and available failure history of similar RCP's suggested a crack in the shaft. After dismantling and examining the pump, it was determined that the shaft was cracked more than half way through in the vicinity of a 3/8" drilled hole. It was further determined that the impeller vanes were eroded significantly. A spare shaft and impeller was installed.

To determine if there was any safety significance, the NRC staff evaluated the licensee submittals and issued a safety evaluation dated February 9, 1985. The NRC staff concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation of TMI-1 with the existing reactor coolant pumps. This was based on the fact that the pump shaft failure is bounded by the FSAR locked rotor analysis and degradation of the other impellers is not significant from a safety standpoint. Reasonable assurance was demonstrated that the integrity of the reactor coolant pressure boundary is not threatened by RCP shaft cracking. The failure of the RCP-1B shaft was caused by fatigue and was not related to the sulfur corrosion problem previously observed in the steam generators.

The inspector reviewed the licensee's records and verified the information submitted to the NRC. In addition, the inspector reviewed and discussed the periodic monitoring to be conducted on all RCP's. All pumps will continue to be monitored for vibration, including periodic analysis of vibration data for components felt to be prone to shaft cracks. The data and proposed RCP monitor were found acceptable. This item (289/84-11-01) is resolved.

2.4 Power Operated Relief Valve (PORV) Operability

The inspector reviewed licensee maintenance (preventative and corrective) and surveillance activities to assure PORV operability. Specifically, the inspector was to verify:

-- Procedures required by TS 6.8.1 properly implement TS 3.12, Table 4.1-1 and TS 4.2.2 (per NRC approved Licensee Program for Inservice Valve Testing) related to PORV operability;

-- Applicable procedures have the proper format and technical content in accordance with applicable sections of ANSI 18.7-1976;

-- Surveillances/Calibration/Preventive Maintenance were conducted at the proper frequency; and,

-- Machinery history records and related surveillance/calibration/ preventive maintenance records were retrievable.

In addition to discussions with cognizant licensee representatives (maintenance, operation, and engineering personnel), the inspector reviewed selected sections of the following licensee documents/ records:

-- Surveillance Procedure (SP) 1302-6.16, Revision 2, September 5, 1984, PORV Setpoint and Remote Position Check, including data obtained April 24, 1984;

-- SP 1303-11.45, Revision 3, February 25, 1985, PORV Setpoint Check, including monthly data obtained in 1984 and January and February 1985;

-- Preventive Maintenance Procedure M-132, Revision 0, April 13, 1982, PORV Inspection, including annual data obtained during 1983 through 1985;

-- Selected job tickets from machinery history files: 18443, 18821, 19671, 20438, C3349, C9775, CC617, CD333, and CD354;

-- Wyle Laboratories Test Report No. 47439-0, dated September 6, 1984, for PORV Serial No. BS03989;

-- Wyle Laboratories Test Report No. 47122-1, dated February 1, 1984, for PORV Serial No. BL08905;

-- Operating Procedure (OP) 1102-1, Revision 73, January 7, 1985, Plant Heatup to 525°F;

-- OP 1101-11, Revision 49, December 17, 1984, Plant Cooldown; and

-- OP 1102-4, Revision 37, January 7, 1985, Power Operation.

Machinery history provided a useful summary of work activities on the PORV. Referenced job ticket records were retrievable using the licensee's microfische and microfilm systems. Both the installed PORV (BL08905) and the spare PORV (BS03989) received extensive refurbishment and testing in which the licensee used the services of Dresser (the valve manufacturer) and Wyle Laboratories. (This occurred subsequent to the corrosion problem noted on both valves in LER's 82-11 and 83-03, which were reviewed by NRC staff in Inspection Report No. 83-34). Preventive Maintenance (PM) procedure M-132 appropriately reflected vendor recommendations to periodically check for signs of inhibited movement caused by corrosion. The PM also specified checking for proper applied solenoid voltage. The refueling Surveillance Procedure (SP), 1302-6.16, required a check for actual valve movement during exercise testing.

The applicable Operating Procedures (OPs) properly reflect TS limiting conditions for PORV operability (TS 3.2.23.2, .3, and .4). The applicable SPs properly reflect frequency requirements and setpoints listed by TS 3.1.12.2 and Table 4.1-1 (Item 48). Adequate checks for relief setpoint and temperature interlock (RC5A-TS1) for change in setpoint were reflected in the applicable SP. The SPs provide for independent checks of lifted lead positions during the course of implementation along with measures to restore the components to normal.

The inspector found an inconsistency in how technicians performed the reset check of the temperature switch (bistable) for the calibration/functional check of RC5A-TS1. The TS temperature setpoint of 275°F (plus/minus 12°F) corresponded to -2.5 volts. Input scale ranges from -10 V to +10 V DC corresponding to temperature ranges of 50°F to 600°F. The technician, by procedure, deals in two test signals, pressure signal greater than 485 psig, and temperature signal less than 275°F (corresponding voltage less than -2.5 volts). The temperature signal is increased from below -2.5 volts to 0 volts until a PORV shut signal occurs. The reset check is then performed by reducing temperature signal towards -10 volts. The procedure directs the technician to set the reset setpoint at 0.1 volts below the temperature setpoint. Depending on the technician performing the monthly test, the reset voltage was inconsistently left at either -2.4 or -2.6 volts.

The inspector concluded that the calibration was adequate to meet TS tolerances of 275°F (plus/minus 12°F), but the licensee needed to provide additional guidance in the SP to assure a consistent calibration/functional check. The licensee representative agreed to review the discrepancy for an appropriate procedure revision.

Based on a review of applicable and current electrical drawings for the PORV control circuits, the inspector verified that a Loss of Non-Nuclear Instrumentation (NNI) power will not cause the PORV to open at normal operating temperature and pressure assuming no failure of any instrument modules in the control circuit. An event of this type occurred in the control circuit for TMI-2 in 1978, but modifications have been made at TMI-1 to preclude this type of recurrence.

Overall, the inspector concluded that the licensee's records were well kept and that these records reflected applicable procedures were properly implemented.

3.0 Administration Control Implementation

The inspectors reviewed TMI-1 Administrative Control Procedure 1014 "Administration of the TMI-1 Initial Response and Emergency Support Duty Roster," Revision 13 to verify the following:

-- the administrative procedure adequately reflected requirements and/or commitments in the Emergency Plan and ANSI 18.7-1976 and other applicable regulatory documents;

-- the administrative procedures were properly implemented by licensee representative;

-- licensee personnel listed on the different duty rosters were aware of their responsibilities as defined in AP 1014; and,

-- duty personnel were qualified to hold the position as stated on the duty roster.

The inspector found that the licensee was properly implementing this administrative procedure in conformance within requirements of ANSI 18.7-1976 and the Emergency Plan.

4.0 Licensee Action on Previous Inspection Findings

4.1 (Closed) Unresolved Item (289/84-11-01) - Review of licensee's evaluations and conclusions.

See paragraph 2.3 for details.

4.2 (Closed) Licensee Event Report (289/84-LO-04) - Steam generator tube not plugged as required.

See paragraph 2.2 for details.

5.0 Exit Interview

The inspectors discussed the inspection scope and findings with licensee management at the exit interview conducted on March 8, 1985. The following licensee personnel attended the meeting:

-- R. Barley, TMI-1 Manager, Plant Engineer
-- J. Colitz, Plant Engineering Director
-- C. Incorvati, TMI-1 Audit Supervisor
-- R. Neidig, Jr., TMI-1 Communications
-- S. Otto, TMI-1 Licensing Engineer
-- C. Smyth, TMI-1 Licensing Manager

As discussed at the meeting, the inspection results are summarized in the cover page of the inspection report. The licensee representatives indicated that none of the subject matter discussed contained proprietary information. Also, discussed were licensee plans for making the plant physically ready to support an April 11, 1985, Hot Functional Testing date.

Unresolved items are matters about which information is required in order to ascertain whether they are acceptable items, violations, or deviations. One unresolved item, discussed during the exit meeting, is documented in paragraphs 2.3.