U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report No. 50-289/88-03 License: DPR-50

Licensee: GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Facility: Three Mile Island Nuclear Station, Unit 1

Location: Middletown, Pennsylvania

Dates: February 7, 1988 - March 5, 1988

Inspectors: R. Conte, Senior Resident Inspector
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Accompanied by: S. Peleschak, Reactor Engineer, Region I (RI)
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Approved by: C Cowgill
C. Cowgill, Chief, Reactor Projects Section 1A 4/20/88
Date

Inspection Summary:

Areas Inspected: The NRC staff conducted routine safety inspections during routine operations. Plant operational items reviewed were: plant shutdown and startup activities, decay heat removal river pump strainer foundation bolt problem, shift and daily surveillances, and emergency feedwater pump inservice testing. Items reviewed in other functional areas included: engineering planning for Regulatory Guide (RG) 1.97 modifications; Safety Issues Management System (SIMS) items related to reactor trip system modifications, natural circulation cooldown, and emergency feedwater (EFW) upgrades; and, licensee fitness for duty policy.

Inspection Results: The licensee continued to operate the plant safely. The short outage for a main generator stator coolant problem and the related plant shutdown and startup activities were conducted in an acceptable manner. Outage planning was good which resulted in the completion of repairs to NI-2 and several significant unisolable Once-Through Steam Generator (OTSG) skin valve leaks.

The seismic concern associated with the defective bolts on the 1A Decay Heat River Pump strainer foundation was tentatively resolved by engineering personnel. Region I will review the licensee calculations in a future report. One unresolved item resulted from this review.

Licensee corporate engineering personnel appeared adequately prepared and knowledgeable on upcoming modifications to comply with the provisions of RG 1.97.

SIMS items reviewed resulted in the closeout of two issues -- SIMS Item No. 75 (B-80), Reactor Trip System - Vendor-Related Modifications, was adequately addressed, and SIMS Item No. MPA-8-66 Natural Circulation Cooldown Issue, was closed. SIMS Item No. II.E.1.2, EFW Upgrades, was reviewed and the status of remaining open items was documented.

The licensee responded to NRC staff inquiries concerning fitness for duty policy and provided answers to the staff questions. Licensee action on previous inspection findings was acceptable.

DETAILS

1.0 Introduction and Overview

1.1 NRC Staff Activities

The purpose of this inspection was to assess licensee activities during the power operations mode and during transition periods as they related to reactor safety, safeguards, and radiation protection. Within each area, the inspectors documented the specific purpose of the area under review, acceptance criteria and scope of inspection, along with appropriate findings/conclusions. The inspectors made this assessment by reviewing information on a sampling basis through actual observation of licensee activities, interviews with licensee personnel, measurement of radiation levels, or independent calculation and selective review of listed applicable documents.

1.2 Licensee Activities

During the report period, the plant operated at full power with the exception of a three-day outage. The plant was shut down on February 16, 1988, and restarted on February 19, 1988. Full power was reached on February 20, 1988.

During the outage, repairs were made to the main generator stator cooling system, which had previously showed signs of fouling. The licensee also completed several repairs to Once-Through Steam Generator (OTSG) unisolable valve bonnet leaks and completed repairs to a channel of source range nuclear instrumentation, NI-2.

As of 8..0 a.m. on March 5, 1988, TMI-1 was operating at full power with the reactor coolant system (RCS) at normal operating temperature (579 F average) and pressure 2155 psig).

2.0 Plant Operations (71707, 71715)

2.1 Criteria/Scope of Review

The resident inspectors periodically inspected the facility to determine the licensee's compliance with the general operating requirements of Section 6 of the Technical Specifications (TS) 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 logkeeping practices;

-- implementation of radiological controls; and,

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

The inspectors focused on the areas listed in Attachment 1.

2.2 Findings/Conclusions

2.2.1 Plant Shutdown/Outage/Startup

In the early part of February 1988, the licensee noted indication of poor heat removal transfer across the main generator in the stator cooling system. There had been an apparent buildup of copper oxides at the heat transfer surfaces in the main generator stator. Subsequent to licensee evaluation of the problem, the licensee started to decrease power to hot shutdown condition at 7:00 p.m., February 16, 1988. In conjunction with its vendor, General Electric, the licensee conducted a chemical cleaning and backflush process of the system.

Safety-related work was also completed because of the outage opportunity, such as "Furmaniting" steam leaks in the reactor building, cable replacement for one-of-two channels of source range nuclear instrumentation, and leak repair of a reactor water level indication sensing line.

On Friday, February 19, 1988, the licensee returned TMI-1 to power operations after the unplanned outage of three days. The generator stator cooling system was cleaned and returned to normal. The cause of the fouling remains to be determined. The plant reached 100 percent power on Saturday, February 20, 1988.

During the power transition periods, the resident inspectors provided 24-hour inspection coverage. The purpose was to more adequately stay abreast of plant status and to assess licensee performance during the transition periods.

There were overall good direction and control of the plant shutdown and startup. The plant startup procedures were used and strictly followed. Operators were alert and attentive and they were responsive to alarms that were both expected or unexpected. Overall, the licensee's performance was professional with respect to entry into and out of the "unplanned" outage.

The inspectors identified no unacceptable conditions.

2.2.2 Decay Heat River Pump Strainer Foundation Bolts

During the evening of February 25, 1988, the Manager of Safety Review informed the senior resident inspector of a problem noted on the foundation support bolts for the "A" loop decay heat removal river water (DR) system strainer (DR-S-1A). During foundation repair work, the licensee found that an approximate 4-inch section of a 36-inch foundation bolt (embedded in the strainer pedestal) was sheared off, apparently due to corrosion. Subsequent examination of the studs revealed that during a previous modification, the anchor studs were lengthened by approximately four and one-half inches by welding to the existing stud. The resulting weld exhibited partial penetration and this caused the stud failure. By ultrasonic testing, a similar problem was noted for the remaining three bolts on DR-S-1A and four bolts on DR-S-1B. Testing on other river water system strainers did not reveal a similar problem.

The licensee considered themselves to be on a 72-hour time clock with respect to TS 3.16, "Shock Suppressors (Snubbers)" as the bolts provided support for the strainer. The TS also required an engineering evaluation as to the affect on system operability and/or to make repairs as necessary.

Corporate-based personnel performed that evaluation and on February 26, 1988, concluded that the strainer foundation bolts were not needed with respect to the seismic installation of that section of the DR system (strainer and adjacent piping). The licensee provided the resident office a copy of their evaluation, which also included a 10 CFR 50.59 evaluation and this was tentatively found to be acceptable. This is unresolved pending Region I specialist review of this information (289/88-03-01).

2.3 Plant Operations Summary

The plant continued to be operated safely. The plant shutdown, outage, and startup activities related to the repair of the main generator stator coolant problem exhibited good planning and overall control. Response to minor plant control system transients was good and the operating staff was able to control the events without major problems.

3.0 Maintenance/Surveillance - Operability Review (61726, 62703)

3.1 Criteria/Scope of Review

The inspectors reviewed selected activities to verify proper implementation of the applicable portions of the maintenance and surveillance programs. The inspector used the general criteria listed under the plant operations section of the report. Specific areas of review are listed in Attachment 1. A more detailed review of equipment operability is also addressed.

3.2 Findings/Conclusions

3.2.1 Surveillance Shift and Daily Checks

The inspector verified by direct observation of licensee activities that surveillances are being conducted in accordance with Technical Specification (TS) requirements. The aspects which were reviewed included:

-- conformance to TS's;

-- calibration of instrumentation;

-- conduct of the surveillance test;

-- adequacy of documentation review;

-- accuracy and completeness of test data;

-- qualification of personnel performing the test; and,

-- performance of the test within the required frequency.

The inspector observed the performance of Surveillance Procedure (SP) 1301-1, Revision 69, dated September 4, 1987, "Shift and Daily Checks." The surveillance procedure was found to be in conformance with TS requirements. The conduct of the test was satisfactory. Adequate reviews were evident. The results of the surveillance were accurate, complete, and contained the required surveillance acceptance criteria. The personnel who performed the surveillance were found to be qualified. By review of surveillance records, it was determined that the test was completed within the required frequency.

The inspector observed that calibration records of instrumentation were difficult to track. The inspector did not find any instrumentation which had not been calibrated at the required frequency, but it is noted that control of calibration records is not uniform for important-to-safety (ITS) and not important-to-safety (NITS) systems. Currently, numerous surveillances exist for calibration of instrumentation used in ITS systems. These records are kept either in the control room or are kept on microfiche in the records department. Calibration records for instrumentation in NITS systems (and some ITS) are kept in a library outside the control room. The system, master test index (MTX), by which these records are kept is not controlled. Calibration records may, therefore, be retained in several locations.

In summary, the performance of the surveillance was within requirements. The personnel performing the test were knowledgeable and qualified. Although difficulty was experienced in verifying calibration of instrumentation, all instrumentation chosen to be verified was found to be properly calibrated.

3.2.2 Inservice Testing Activities

The inspector witnessed surveillance being performed on motor-driven emergency feedwater pumps EF-P-2A/B. The surveillance was performed on February 12, 1988, in accordance with Surveillance Procedure (SP) 1300-3F to ensure compliance with Technical Specifications, Section 4.2.2. This requires testing of pumps and valves in accordance with Section XI of ASME (American Society of Mechanical Engineers) boiler and pressure vessel code, as well as 10 CFR 50.55 a(g).

Recording of the test data, procedure compliance, and communication between the personnel at the pump site and the control room was thorough and effective.

3.3 Operability Summary

Maintenance and surveillance activities continue to be accomplished in a satisfactory manner and housekeeping is adequate.

4.0 Corporate Inspection/Briefing (Regulatory Guide 1.97) (37700)

4.1 Background/Scope of Review

During the upcoming outage (7R), presently scheduled for June 1988, the licensee plans to complete various modifications, calibrations, and licensing actions to meet commitments concerning the provisions of RG 1.97, "Post-Accident Monitoring Instrumentation." The inspectors were briefed by corporate licensing and engineering personnel on February 23, 1988, on the status of completed modifications, planned modifications, and anticipated licensing actions concerning RG 1.97. The inspectors reviewed various licensee/NRC correspondence concerning this issue in order to assure that the licensee was adequately prepared to complete the required actions to ensure compliance with commitments for RG 1.97 at this time.

4.2 Findings

The license had previously completed several modifications during the 6R outage that related to RG 1.97. Those included: the Reactor Coolant Inventory Tracking (RCITS) and void function monitoring system, effluent monitoring from the main condenser offgas system, safety-grade condensate water storage tank (CST) level indicating system, and redundant OTSG pressure indication in the control room.

These modifications have been evaluated by the NRC staff in previous inspections. The only major concern remaining is the acceptability of the accuracy of the CST level indicating system. The NRC staff had a concern that the level indicators give a lower than actual reading when the EFW pumps are running due to flow induced pressure changes at the point where the level is monitored. The licensee has addressed this issue and has evaluated the approximately 3/4-foot lower than actual reading as acceptable and has documented this review in a letter to NRR, dated June 29, 1987. Remaining action of this item consists of NRR staff approval via a supplement to the Safety Evaluation Report (SER) for RG 1.97.

Several open questions remain concerning five other design issues raised during the initial NRR SER. Those are:

-- accumulator tank level or pressure;

-- pressurizer level;

-- pressurizer heater status;

-- containment sump water temperature; and,

-- containment heat removal system instrumentation classification.

The issues surrounding the acceptability of the type of indication for the safety injection accumulators and the containment sump water temperature are being evaluated by NRR on a generic basis. The acceptability of the licensee response will be based on this review. The type of indication available for pressurizer level and whether or not it is temperature compensated with environmentally-qualified instrumentation is still in question. The type of instrumentation to be used for pressurizer heater status is still unresolved. NRR is presently requiring instruments to measure current draw vice just using heater switch position.

The last issue involves the type of instrumentation used for monitoring the performance of the reactor building cooling river water system, which is presently Category 3. NRR is presently requiring Category 2 instruments. The licensee is relying on containment temperature/pressure to monitor containment heat removal performance.

During 7R, the licensee plans to complete the following modifications:

-- Reactor Coolant Drain Tank (RCDT) temperature range extension;

-- Reactor Coolant System (RCS) pressure range extension;

-- redundant wide range neutron flux monitoring (NI-11/12); and,

-- control room indication (qualified) for low pressure indication (LPI) flow, high pressure indicator (HPI) flow, and borated water storage tank (BWST) level.

During 8R the licensee will complete modifications for RG 1.97. This will consist of providing "on demand" recordings of several important plant variables via the new plant computer.

The inspectors have reviewed the system design descriptions for the new instrumentation and instrumentation modifications. Those areas will be examined as they are completed in the upcoming outage.

The inspectors are presently tracking three unresolved items related to RG 1.97. These are: (1) 289/85-21-03, which is the NRR review of design adequacy of the new non-nuclear instrumentation (NNI); (2) 289/87-09-03, condensate storage tank (CST) level indication oscillations; and, (3) 289/87-09-06, adequacy of differential/pressure (DP) instrumentation used in high pressure service. The resolution of these issues is awaiting NRC staff review as the licensee has completed their action on these items.

4.3 Conclusion

The licensee is aware of the remaining issues to be resolved by NRR prior to achieving full compliance with RG 1.97. Modification status for those items to be accomplished during 7R appears to be on schedule.

5.0 Safety Issue Management System Item Verification

5.1 Introduction

The inspector verified proper implementation of licensee actions related to the below-listed NRC Safety Issue Management System (SIMS) item. The generic inspection approach for the SIMS item was:

-- research various licensee and NRC correspondence, including safety evaluation reports (SER's) to identify key assumptions, commitments, or other licensee actions to be taken to resolve the safety issue;

-- identify an additional items which need to be verified as delineated in the related NRC Temporary Instruction or other inspection procedures; and,

-- verify proper implementation of the items planned above.

5.2 Generic Letter 83-28, Reactor Trip System - Vendor-Related Modifications (SIMS No. 75 (B-80)) (25591)

5.2.1 Background/Review

The Generic Letter 83-28, Item 4.1 required that all vendor-recommended modifications on the reactor trip breakers be reviewed to verify that each modification has, in fact, been implemented or a written evaluation, including justification for not performing the modifications, exists.

5.2.2 Findings/Conclusions

The reactor trip breakers utilized at TMI-1 are of type AK-2-25 and AK-2-15 manufactured by General Electric (GE). A GPU Nuclear Corporation (GPUN) letter of November 8, 1983, references a GE letter to Babcock and Wilcox (B&W), dated September 7, 1983, concerning vendor-recommended modifications on reactor trip breakers. GE did not recommend any modification to their subject breakers. The letter recommended that the shunt trip should be actuated simultaneously with the undervoltage trip mechanism for the reactor trip function. This function was recommended by Generic Letter 83-28, Item 4.3.

Subsequent to this, the B&W Owners Group (BWOG) undertook a program to determine long-term actions to improve reliability of the reactor trip breakers; in particular, the undervoltage trip device. The ensuing recommended action, documented in their April 8, 1985, letter to the NRC, included two modifications to the reactor trip breakers. The BWOG recommended that the utilities replace the trip shaft and latch roller bearings with bearings lubricated with Mobil 28 and that Mobil 28 lubricated bearings be used exclusively in GE AK-type reactor trip breakers. The second recommendation was the addition of a reactor protection system (RPS) type signal to the d.c. shunt trip device. A GPUN letter, dated August 23, 1985, endorses the recommended equipment upgrade and stated that they were implemented in TMI-1.

The inspector reviewed implementation of the first recommendation (about Mobil 28 lubricated bearings) at TMI-1. The licensee stated that the existing (installed and in stock) reactor trip breakers were shipped to the vendor for bearing replacement. GE performed the replacement work and B&W coordinated the replacement work, including necessary qualification and certification of the modified breakers. The bearings were not easily accessible for inspection. However, based on discussions with licensee personnel and review of the purchase order associated with this modification, the inspector concluded that the work was performed as required.

The second BWOG recommended modification was reviewed and found to be acceptably completed by the NRC under Generic Letter 83-28, Item 4.3.

The inspector also reviewed two vendor advisories issued by GE for the AK-Series reactor trip breakers during 1985. Advisory No. 300, dated September 26, 1985, deals with: (1) improper application of paint on the mating surface of the armatures and pole pieces for certain undervoltage trip devices made between mid-1978 and May 1985; and, (2) insufficient clearance between the armature and the undervoltage device mounting studs for devices with assembly number 568B309G. Licensee's follow-up actions for advisory were reviewed under 10 CFR Part 21 report (289/85-PT-07) and was found satisfactory in Inspection Report No. 50-289/87-23, dated January 27, 1988.

The second GE advisory (number 9.21, dated February 7, 1985) deals with the use of improperly heat treated shunt trip paddles in the shunt trip assembly on AK-Series reactor trip breakers. The suspected trip paddles were used in production lots from February 1983 through April 1984. The affected devices include AK-Series reactor trip breakers which have code dates from F307 to F419, replacement front frame assembly series which included the shunt trip assembly and replacement trip paddles.

The inspector discussed the advisory with licensee personnel. The licensee stated that the reactor trip breakers were all purchased during the early 1970's. The TMI-1 breakers have a bill of material date of March 27, 1973. The TMI-2 reactor trip breakers, now utilized in TMI-1 as spares, were also purchased before 1979. The reactor trip breakers were purchased before the date of concern. The trip paddle problem did not exist in the originally purchased reactor trip breakers. The licensee did not believe that shunt trip paddles were purchased as spare parts during the time of concern. The licensee was asked to confirm that the shunt trip paddles of concern were not introduced in TMI-1 as spare parts for the reactor trip breakers.

The licensee reviewed machinery history files to verify that no shunt trip paddles were replaced before, during, or after the dates mentioned in the service advice letter. All shunt trip paddles presently installed were verified to be original equipment. The shunt trip paddle is not kept in stock at TMI-1. Based on the above, this item is closed.

5.3 (Closed) Safety Issue Management System Item (289/MPA-B-66): Natural Circulation (NC) Cooldown (71707)

The NRC Inspection Report No. 50-289/88-01, paragraph 5.3, documented the verification of licensee actions related to the subject multi-plant action item (MPA-B-66). There was a residual concern with respect to the adequacy of instructions to the operators on void formation at the reactor vessel upper head (RVUH) during emergency conditions (ATOG - Anticipated Transient Operator Guidelines).

During this inspection period, the inspector discussed the concern with licensee engineering personnel at the corporate headquarters on February 23, 1988. Licensee representatives stated that the ATOG procedures are adequate with respect to NC cooldown because they are symptom-oriented procedures. The head bubble itself is not a safety concern; its interference with core decay heat removal is a concern. The ATOG procedures provide for indication if a loss of decay heat removal (namely, the loss of thermal coupling between steam generator and the reactor) and these procedures treat those symptoms -- loss of decay heat removal -- rather than the event void function at RVUH.

The licensee representatives also indicated that they had recent research results that indicate a cooldown rate of 100 F/hour for a steam generator tube rupture does not result in a head bubble or a bubble in the unaffected loop assuming proper Reactor Coolant System (RCS) pressure control.

The inspector views the licensee's position on the ATOG procedures as reasonable. The NRC staff shares the licensee's concern about putting too much information in the ATOG procedures such that they are no longer symptom oriented (in distinction to event oriented). Also, based on the last inspection, it appears that the training program addresses the RVUH bubble formation concern and it appears that operators are aware of the problem. The inspector confirmed that Abnormal Transient Procedure (ATP) 1210-1, Revision 15, dated January 4, 1988, "Reactor/Turbine Trip," coupled with ATP 1210-2, Revision 8, dated June 13, 1986, "Loss of 25 F Subcooled Margin," and 1210-4, Revision 10, dated December 17, 1987, "Lack of Primary to Secondary Heat Transfer," addresses indications of and the necessary actions to correct a loss of decay heat removal.

In resolving this item, the inspector places no reliance on the above-noted research results as stated by the licensee. The inspector had no safety concerns on this issue.

5.4 TAP II.E.1.2 (EFW Upgrade) (71710)

5.4.1 Background/Scope of Review

The inspector reviewed the status of open items concerning the recently completed Heat Sink Protection System (HSPS) installation and EFW system upgrade to safety status. During past inspections, many unresolved issues have been generated that affect the determination of the acceptability of licensee compliance with the requirements of Task Action Plan (TAP) Item II.E.1.2. At the beginning of the review, there were twenty-five open issues which are tabulated in Attachment 2. Some of these items were closed in this report. The status of each item and which NRC organization is responsible for review is also noted.

5.4.2 Findings/Conclusions

During this inspection period seven of these items were closed and are noted in Section IV on Attachment 2. Discussions with the licensee have noted that several of the remaining open items listed in Sections II and III are ready for review and that licensee action has been completed. The inspector will review licensee action on these items in future inspections.

For the items to be reviewed by NRR listed in Section I, NRR has committed by memorandum from B. Boger to W. Kane, dated January 4, 1988, to complete a review of these items by May 31, 1988.

In addition to these previously identified open items, the inspectors expressed a concern about some other problems with the recently installed HSPS system. One problem is the inability of the control room operators to read all four channels of both operating (OP) and startup (SU) range OTSG levels. At present, for each OTSG, only two of four channels of each range (SU and OP) are available in the control room. Since the channel being read in the control room is automatically selected by a logic system, confusion can exist as to what the OTSG level actually is and which channel is controlling automatic functions. This has caused at least one automatic initiation of emergency feedwater (EFW) in the past. The licensee has proposed some modifications to resolve this problem. The inspectors were briefed on these proposed modifications during the corporate inspection on February 22 and 23, 1988. The licensee intends to install a modification to improve the HSPS signal selection to the control room indicators. This modification is part of an overall upgrade for the Integrated Control System (ICS), which involves installation of Smart Automatic Signal Selection (SASS) modules in the ICS.

The licensee also intends to install well labelled plug-type instrument jacks in the HSPS cabinets to promote more reliable readout of the various OTSG level signals from all four channels. The cross-check is presently accomplished by Instrument and Control (I&C) personnel, using test probes on terminal block, and has the potential to cause inadvertent shorting across adjacent circuits.

Prior to final resolution of this SIMS item, the remaining unresolved items listed in Attachment 2 must be reviewed. The licensee has completed their action on these remaining open items. The NRC staff is tentatively planning on completing all reviews by July 1988.

At present, no conditions adverse to safety that would affect the present safety-grade status of the EFW/HSPS systems have been identified.

5.5 SIMS Summary

For the SIMS issues reviewed above, the licensee appropriately translated generic licensing actions into plant specific actions. Specifically, they properly incorporated requirements and commitments into procedural requirements and/or training plan/exercise.

6.0 Fitness for Duty Program/Survey (50-289 and 320/88-TI-01)

The NRC Region I issued Temporary Instruction (TI) No. 88-1, dated January 21, 1988, "Fitness for Duty - Drug Testing Information and Reporting." It required resident inspectors to obtain licensee response to certain questions about the licensee's Fitness for Duty Program. The Director of TMI-1 responded to these questions and the answers were provided to Region I staff by separate memorandum.

This action closed the subject Regional Instruction (289 and 320/88-TI-01).

7.0 Licensee Actions on Previous Inspection Findings

7.1 Introduction

For these items listed below that were previously identified violations, the inspector reviewed the licensee's response and corrective action to:

-- verify the licensee responded in a timely manner;

-- verify measures taken to correct the item and avoid recurrence were completed and within the specified time frame; and,

-- verify licensee commitments were completed.

7.2 (Closed) Unresolved Item (289/87-06-01): Labeling of HSPS Cabinets

This item concerned the possible operator confusion that might arise due to the similarity of the multiple test switches in the various Heat Sink Protection System (HSPS) logic cabinets. The licensee subsequently added additional labels to the outside cabinet doors and also added train/channel designators to the inside test switch cover panels. The individual switches have all been changed to red (Train A) and green (Train B) colored labels to further eliminate confusion. As the HSPS surveillance procedures have been accomplished routinely and without incidents for the past year, it appears that the present labeling system is acceptable. The inspector had no other concerns on this issue.

7.3 (Closed) Unresolved Item (289/87-06-03): Verify Preventive Maintenace is Scheduled for Emergency Feedwater System Flow Control Block Valves EF-V-52A/B/C/D

This item involved the new manually operated block valves EF-V-52A/B/C/D that replaced the previously motor-operated block valves EF-V-52, 53, 54, and 55 in the emergency feedwater (EFW) systems. These valves are used for maintenance isolation and to manually isolate the EF-V-30 valves (flow control valves). It was noted that no preventive maintenance was in place for these valves. The licensee subsequently scheduled these valves for a three-year open and inspect cycle commencing in 1990. Procedure E-13 is being developed for this purpose, but it is not complete at this time.

The inspector verified that the valves are presently scheduled for maintenance on the computerized system used by the licensee to schedule all plant maintenance. Additionally, the valves are being cycled during the monthly inservice testing (IST) for the motor-driven EFW pumps per SP 1300-3F. No problems have been noted with these valves. The inspector had no other concerns with this issue.

7.4 (Closed) Violation (289/87-06-06): Failure to Properly Review and Verify a Design Calculation Associated with Once-Through Steam Generator Low Level Actuation Setpoints

This violation concerned a licensee change made to a contractor design calculation No. 0370-129-001, Revision 0, which was then not properly reviewed by the contractor.

The licensee contractor subsequently reviewed the calculation which was design verified prior to use of the calculation. The licensee corrective action consisted of this correction in addition to counseling the personnel involved. The inspector reviewed design calculations for other licensee modifications to ensure that they were properly reviewed and that changes were also properly reviewed. Two calculations, C8708-21 and C8706-021, both for Regulatory Guide (RG) 1.97 work, were reviewed and design verified by the contractor. No problems were noted. The inspector discussed with various licensee engineering personnel the method by which calculations were changed. All personnel were aware of the requirements to review and design verify the calculations when changes are made. The inspector had no other concerns on this issue.

7.5 (Closed) Unresolved Item (289/87-06-07): Heat Sink Protection System Testing

This item concerned the initial testing of the HSPS logic system prior to use after the startup from the 6R outage. The licensee opted to perform preliminary surveillance procedures via the Special Temporary Procedure (STP) process in order to verify the operability of the HSPS system logic. The inspectors witnessed portions of that testing in March

1987. Subsequently, the inspectors reviewed the completed STP Nos. 1-87-0015, 0017, 0018, and 0019 and also reviewed subsequent completed version of the formal SP's 1303-11.36, 11.37A/B/C/D, 11.37, 11.38, and 11.39. These procedures have been performed on a quarterly basis since startup in March 1987. Prior to startup, the licensee reviewed the data from the completed STP's and declared the HSPS operable. Based on the inspector review of the above procedures and witnessing of selected portions of the formal surveillance procedures without problems at various times during the past ten months, this item is closed.

7.6 <u>(Closed) Unresolved Item (289/87-06-10): Licensee Completed Audit on Design and Modification Control</u>

At the time of that NRC inspection, the licensee was in the process of completing a major audit (No. O-TMI-86-11). The audit was extensive in that it spanned several months (September 8, 1986 - March 12, 1987) and it's purpose was to determine whether the NRC Performance Appraisal Team I (PAT I) inspection findings were generic to other Cycle 6R modifications. One of the modifications included in the review was the Heat Sink Protection System (HSPS), which is the safety-grade initiation and control system for emergency feedwater. The licensee selections of HSPS and other modifications were representative of major work completed in 6R. The audit report was issued April 9, 1987.

The summation finding of the audit was that there was no additional generic programmatic or technical issues. However, the audit report had some regulatory and technical issues of interest to the inspector so licensee corrective or follow-up actions were reviewed for adequacy.

The audit identified two audit findings, three quality deficiency reports, two preliminary safety concerns (PSC) and twelve recommendations. The findings, deficiencies, and recommendations were appropriately acted upon and verified to be completed by Quality Control (QC) inspectors. The PSC's were also appropriately addressed, but certain actions remained.

The PSC No. 86-006, dated September 11, 1986, identified, in part, a concern about the nuclear services closed-cycle cooling water (NSCCW) system in terms of withstanding a seismic event. Specifically, the surge tank fill valve has a manual isolation valve and there was an upstream Seismic I boundary. The licensee had tentatively decided to install a Seismic I check valve to retard leakage on upstream pipe breaks due to a seismic event until operator actions could shut the manual isolation valve. The check valve is scheduled for installation in this refueling outage, July 1988. Also, an internal engineering memorandum indicated that a plant-wide review would occur by February 29, 1988, to identify similar problems. This PSC remains open and the inspector established confidence that the licensee was heading toward proper resolution of this item. The inspector had no additional comments.

7.7 (Closed) Unresolved Item (289/87-13-02): NI-2 Replacement-in-Kind Issue

This item involved a modification where NI-1 cable was replaced with one which was not the same as the original. The replacement-in-kind review was performed as required. However, additional technical data was required to clarify the justification for the NI-1 cable replacement. The licensee committed to do so during the similar replacement of cable on NI-2 instrumentation.

During the last unplanned shutdown from February 16-19, 1988, the NI-2 cable was replaced under Job Ticket (JT) CP-025, dated October 22, 1987. The work involved replacement of the cable and the associated connectors from the pre-amp box to the NI-2 detector inside the reactor building. Plant Engineering Evaluation 88-007-E, dated February 1, 1988, satisfactorily provided the requested technical justification. The inspector reviewed the engineering evaluation as well as the JT. The inspector reviewed the adequacy and completeness of the technical data, job planning, testing, and acceptance by the operational staff and concluded that the licensee has satisfactorily resolved this concern.

8.0 Exit Interview

The inspectors discussed the inspection scope and findings with licensee management an an interim exit meeting at the corporate office on February 23, 1988 on certain SIMS items and at a final exit meeting conducted March 4, 1988. Senior licensee personnel attending the final exit meeting included the following:

J. Colitz, Plant Engineering Director, TMI-1
H. Hukill, Director, TMI-1
C. Incorvati, TMI-1 Audit Manager
R. Knight, Engineer, TMI-1
M. Ross, Plant Operations Director, TMI-1

The inspection results as discussed at the meeting are summarized in the cover page of the inspection report. Licensee representatives did not indicate that any of the subjects discussed contained proprietary or safeguards information.

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable, violations, or deviations. Unresolved items discussed during the exit meeting are addressed in Section 7.

ATTACHMENT 1

NRC INSPECTION REPORT NO. 50-289/88-03

ACTIVITIES REVIEWED

Plant Oper. .ons

-- Control room operations during regular and back shift hours, including frequent observation of activities in progress and periodic reviews of selected sections of the shift foreman's log and control room operator's log and selected sections of other control room daily logs.

-- Areas outside the control room.

-- Selected licensee planning meetings.

-- Reactor shutdown -- reactor startup activities.

During this inspection period, the inspectors conducted direct inspections during the following back shift hours.

Day/Date	Time
2/7/88	8:45 a.m. - 11:15 a.m.
2/15/88	8:00 a.m. - 12:00 Noon
2/16/88	4:00 p.m. - 12:00 Midnight
2/19/88	11:30 p.m. - 7:00 a.m.
2/20/88	9:30 a.m. - 10:30 p.m.

Maintenance/Surveillance

-- .P 1300-3F, Revision 30, effective March 20, 1987, "Motor-Driven EFW Pump IST."

-- JT CP-869, Repair of Condensate System Valves CO-V-16A/B Work.

-- SP 1300-3H A/B, Revision 24, effective January 15, 1988, "Makeup Pump and Valve Functional Tests."

-- SP 1302-3.1, Revision 49, effective January 5, 1988, "R.M.S. Calibration."

-- JO 80842, Decay Heat River Pump Expansion Joint Work.

Reactor Coolant System (RCS) Leak Rate

The inspector selectively reviewed RCS leak rate data for the past inspection period. The inspector independently calculated certain RCS leak rate data reviewed using licensee input data and a generic NRC "BASIC" computer program "RCSLK9" as specified in NUREG 1107. Licensee (L) and NRC (N) data are tabulated below.

TABLE

RCS LEAK RATE DATA
All Values GPM

DATE/TIME DURATION	L_G	N_G	(NUREG 1107) N_U	CORRECTED N_U	L_U
02/10/88 12:40 a.m. 2 Hours	0.3356	0.33	0.05	0.15	0.1601
02/15/88 1:14 a.m. 2 Hours	0.2881	0.29	0.05	0.15	0.1470
02/20/88 9:58 a.m. 2 Hours	0.2198	0.22	0.12	0.22	0.2242
02/27/88 3:07 a.m. 2 Hours	0.4366	0.44	-0.06	0.04	0.0405
03/06/88 1:10 a.m. 2 Hours	0.4383	0.44	-0.05	0.054	0.0531

G = Identified gross leakage
L - Licensee calculated
U = Unidentified leakage
N = NRC calculated

Columns 2 and 3, 5 and 6 correlate ± 0.2 gpm in accordance with NUREG 1107. N_u is corrected by adding 0.1044 gpm to the NUREG 1107 N_u due to total purge flow through the No. 3 seal from RCP's.

ATTACHMENT 2

EFW/HSPS UNRESOLVED ITEMS (II.E.1.2)

I. Unresolved Items to be Reviewed by NRR

Reference: October 21, 1987, Memorandum W. Kane to F. Miraglia

86-03-08 - Breaker Coordination Study
86-12-12 - Pump Overcurrent Protection
86-12-13 - 115 Volt a.c. Circuits
86-12-14 - Minimum Motor Start Voltages
86-12-15 - Grounding Practices
87-06-08 - Mechanical/Structural Systems Review
87-06-09 - Electrical/Instrument Control (IC) Systems Review

II. Unresolved Items Requiring Region I Review

86-05-05 - EFW Nozzle/Cracking
86-12-17 - RSP - OTSG Level Indication (Electrical Isolation)
86-19-02 - Seismic Interaction Study Review
87-02-03 - Diesel Generator Loading (DAPPER)
87-10-01 - Cable Grounding Practices

III. Unresolved Items for Resident (On Site) Inspector Review

85-20-01 - Residual EFW Concerns from Restart Review
86-03-22 - Back-Up instrument Air Quality
86-12-10 - Modification Review of EFW Pump Recirculation Valves Block
87-06-02 - ATOG Procedure Enhancements - OTSG High Level Main Feedwater Isolation
87-06-05 - Feedwater Pressure Switch Calibration Drift

IV. Unresolved Items Closed in 289/88-03 or 289/88-06

86-12-09 - IST for Three-Way Back-Up Instrument Air Valves
86-21-04 - MS-V-9A/B Inspection Frequency
87-06-01 - Labelling of HSPS Cabinets
87-06-03 - Preventive Maintenance for EF-V-52 Valves
87-06-07 - HSPS Testing Prior to Cycle 7 Startup
87-06-10 - HSPS Modification Audit
87-06-06 - Violation Concerning HSPS Calculation Review/Approval