U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Docket/Report No. 50-289/87-23 License No. DRP-50

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

Facility: Three Mile Island Nuclear Station, Unit 1

Location: Middletown, Pennsylvania

Dates: November 1 - December 5, 1987

Inspectors: R. Conte, Senior Resident Inspector (TMI-1)
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Approved by: C Cowgill III 1/20/88
C. Cowgill, Chief, Reactor Projects Section No. 1A, Date

Inspection Summary:

Areas Inspected: The NRC staff conducted safety and safeguards inspections (157 hours) during power operations. Items reviewed in plant operations were: voltage drop due to loss of a circulating water pump and turbine runback on November 17, 1987. Other items reviewed included: Abnormal Transient Procedures, physical security, 10 CFR 50.59 evaluations for special temporary procedures, and licensing actions on past inspection findings.

Results: No violations were identified in this report. One unresolved item was identified as a result of the voltage transient caused by the loss of the circulating water pump.

Operator performance during the loss of main feedwater turbine runback was professional and plant response was acceptable. Operations performance in other areas for the month was generally good.

Maintenance and surveillance activities were accomplished with no major problems. Successful completion of the yearly emergency diesel generators' inspection was the major effort evaluated. A review of safety evaluations for special temporary procedures revealed that generally procedures for this activity were followed correctly. Site physical security implementation was proper.

No problems were identified in the final closeout review of the abnormal transient procedures. License action on previous inspection findings was also adequate.

DETAILS

1.0 Introduction and Overview

1.1 NRC Staff Activities

The overall purpose of this inspection was to assess licensee activities during the power operations mode 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 inspections, along with appropriate findings/conclusions. The inspector 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 this period, the licensee operated the plant at essentially full power. The plant experienced a turbine runback to 70 percent power on November 17, 1987, due to loss of one main feed pump. The plant was restored to full power four hours after the event (see Section 2.2.1).

2.0 Plant Operations

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 Turbine Runback

On November 17, 1987, at 6:52 p.m., an automatic turbine runback to 70 percent reactor power occurred due to loss of the "B" main feed pump (MFP). Plant maintenance personnel were in the process of installing the "C" condensate booster pump (CO-P-2C) breaker in the "C" non-vital 4160 volt a.c. switchgear. Difficulty was experienced during this evolution in that proper operation of the charging of the breaker closing spring could not be verified. On the third attempt to rack the breaker in, the breaker spring charged correctly and then the breaker inadvertently closed. The pump operated for approximately one minute prior to the breaker being opened. This occurred when trip coil fuses were installed. When this occurred (starting of the "C" pump), the "A" condensate booster pump tripped, leaving only one condensate booster pump in service. The "A" condensate booster pump tripped due to a counting circuit relay that sensed two main condensate pumps were running. The "B" MFP tripped due to only one booster pump operating. The integrated control system (ICS) reduced turbine power automatically to 70 percent load, as designed.

Operators then manually reduced load to approximately 55 percent as only one condensate booster pump was running. After the plant stabilized, the "A" condensate booster pump and the "B" MFP were restarted and power was increased to 100 percent by 11:00 p.m. on November 17, 1987.

The resident inspector was notified by plant operations personnel of the occurrence about one hour after the event.

The licensee subsequently determined that the CO-P-2C breaker had malfunctioned. It should not have closed upon being "racked-in." It appears that the breaker closing was due to a malfunction of the closing coil. As a result of this malfunction, the licensee has given verbal guidance to operators to ensure that the 30 amp trip coil control fuses are installed prior to "racking-in" the breaker for all breakers of this type. This includes breakers in safety-related applications. Previously, the 30 amp fuses were installed after breaker installation. This will allow immediate control of the pump motor in the control room. With the fuses installed and the pump control switch in "pull-to- lock," the breaker would not have closed. The licensee will make formal maintenance procedure changes at a later date to implement this action. Although this type of malfunction has not been observed before, the resequencing of breaker control fuse installation should prevent this situation from re-occurring.

The inspector reviewed the plant response to the runback by examining individual parameter responses generated by the licensee transient analysis computer. No anomalies were noted. Although no "post-trip" type review was conducted by the licensee, inspector review of these parameters revealed that plant response was as expected.

A Plant Incident Report (PIR) was generated and operators were immediately appraised of the potential for failure of other 4160 volt a.c. breakers of this type. Plant procedures are being revised to reflect the new a breaker installation sequence. The inspector concluded that licensee corrective action for this event was appropriate.

2.2.2 Voltage Dip in Auxiliary Transformer System

At 3:38 p.m. on November 9, 1987, the output voltage of the "1B" auxiliary transformer (AXT) momentarily dipped. The "1B" AXT normally supplies one-of-two vital 4160 kv buses (the "1D") in addition to other non-safety buses/loads. The voltage drop was down to 2400 volts but not long enough for time delay relay to cause the associated emergency diesel generator to start. Various plant equipment responded to the voltage transient, such as alternate d.c. powered equipment starting. The main turbine experienced a runback of about 5 MW (megawatts); and, as a result, reactor power dropped from about 99 percent to about 98 percent. The plant was restored to full power shortly thereafter.

Licensee review of the event revealed that one-of-six circulating water pumps (CW) (CW-P-1F) (for secondary plant condenser) apparently experienced an overcurrent situation. In addition to the CW-P-1F breaker tripping, the phase windings had damage and one of the phase windings was open circuited. It was believed that the CW pump failure caused the voltage drop noted above. Licensee review indicates that the electrical distribution system responded appropriately. However, the licensee initiated a review of this event, along with repairing the pump.

At the close of this inspection period, the cause for the CW pump failure was under investigation. The licensee had preliminarily concluded that insulation breakdown and the close proximity of lead conductors in the motor probably caused the failure. The motor was repaired off site.

The inspector discussed the event with plant engineering personnel. It was acknowledged that the CW pump (a non-safety grade load off a non-vital 4160 volt bus) was a large load and that an overcurrent condition could cause such a voltage drop.

The magnitude of the voltage drop was unexpected and it was unclear if the disturbance should be sensed on the safety-grade or vital 4160 volt bus.

As a result of plant engineering request, Technical Functions Division is to perform a computer analysis of the event in an attempt to resolve the above-noted questions. This review is expected to be completed by January 30, 1988. Accordingly, this area is unresolved pending completion of the licensee's computer analysis and subsequent NRC staff review (289/87-23-01).

2.2.3 Decay Heat Removal System Interlock Annunciator Problem

During testing per Surveillance Procedure (SP) 1303-4.11, Revision 8, dated January 9, 1986, "HPI and LPI Analog Channels," a problem was observed by control room personnel with the function of the "DH-V-1/2 Auto-Close Actuated" annunciator. This annunciator is a status light to inform operators that both decay heat (DH) system isolation valves (DH-V-1/2) cannot be opened remotely due to plant pressure being greater than 400 psig to prevent overpressurization of the DH system. Two bistables must open parallel contacts to clear the annunciator when reactor coolant pressure (RCP) is reduced below the setpoint. In this instance, a test of the bistable setpoint for DH-V-1 was in progress where a bypass switch was closed which, in effect, takes one bistable contact out of the circuit. In this case, the annunciator should normally stay energized. However, during this particular test, the operators observed that the annunciator cleared, which was not the expected condition.

Investigation revealed that a sliding link was disconnected in relay cabinet No. 1. This link was for the DH-V-2 circuit bistable. The link was reconnected and the test completed satisfactorily.

Further review of this event and discussion with the licensee revealed the following information.

-- The disconnected sliding link may have been in this condition for an indeterminate period of time as either one-of-two contacts from the DH-V-1 of DH-V-2 bistable will keep the annunciator on during normal operations.

-- Sliding link disconnect and reconnect independent verification was implemented in late 1985 after a similar problem with the sliding link for another annunciator was found in an incorrect position.

-- The improper position of the link had no effect on the operation of the interlock which ensures that DH-V-1/2 cannot be opened at greater than 400 psi RCS pressure.

The licensee questioned various Instrument and Control (I&C) and maintenance personnel and none could recall any recent work where this link could have been disconnected. Personnel were informed of the event and its implications via entries in shift logs. No other documentation was generated.

The inspector concluded that licensee personnel alertness in observing the initial event was a positive indication of alert watch-standing. Followup and determination of cause was adequate. No safety concern existed as the DH-V-1/2 interlock remained operable. A formal documentation of the event via a Plant Incident Report (PIR) was not accomplished but could have been an enhancement of follow-up action to alert other operations personnel to the potential problem.

2.3 Plant Operations Summary

Plant operations were conducted in a safe manner and oriented toward nuclear safety. Operator performance during normal operations and transient conditions was acceptable. Follow-up on plant anomalies was adequate, but formal event review documentation of the event on the DH-V-1/2 sliding link was lacking.

3.0 Maintenance/Surveillance - Operability Review

3.1 General Criteria/Scope of Review

The inspector reviewed 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 Diesel Generator Annual Inspection

The inspector reviewed and witnessed licensee completion of the annual inspection of the emergency diesel generator (EDG). This inspection is completed on a yearly basis per technical specifications. During the most recent inspection in February of 1987, significant problems were experienced during the extensive overhaul of one of the diesel engines. This was documented in NRC Inspection Report No. 50-289/87-05.

During the present inspection, no significant deficiencies were identified. The blower on the "B" EDG was replaced due to lower clearances existing between blower lobes and the blower housing. This problem had been previously identified and the licensee had been tracking the problem. Replacement of the blower, at a time prior to possible failure, is indication of adequate licensee follow-up of maintenance problems.

Both EDG inspections were completed within the allowable seven-day technical specification time restriction. The inspector witnessed the final surveillance test for the "B" EDG post-maintenance operability verification. During the evolution, the operators were unable to fully load the machine to the 3 megawatt load as the governor was improperly adjusted. Only 2.9 megawatts could be achieved. The inspector verified proper adjustment of the governor device and subsequent retest that assured proper operation.

This evolution was accomplished in a safe manner. A vendor representative was present throughout the evolution and provided guidance to plant maintenance personnel. No significant adjustment or modifications were accomplished other than routine inspections. The inspector had no questions relative to this maintenance evolution.

3.3 Operability Summary

For the equipment maintenance and surveillance testing required above and as noted in Attachment 1, the inspector verified operability and satisfactory completion of procedures. Maintenance and surveillance activities appear to be accomplished with the appropriate emphasis on reactor safety.

4.0 Inspection of Emergency Operating Procedures

4.1 Background

The inspector reviewed certain aspects of the licensee's program governing the maintenance and implementation of the symptom oriented Abnormal Transient Procedures (ATP's). These procedures have been implemented in response to Supplement 1 to NUREG 0737, "Requirements for Emergency Response Capability," dated December 17, 1982. The inspector's review was based on Temporary Instruction (TI) 2515/79, which provides guidance for the determination of whether the Emergency Operating Procedures (ATP's) have been prepared in accordance with the NRC-approved Procedures Generation Package (PGP) and whether the procedures are adequate to control safety-related functions in the event of system or component malfunction. Since portions of the TI 2515/79 review were covered during inspection 50-289/84-11, only those elements not addressed previously were covered during this review.

4.2 Review

The inspector compared the Babcock and Wilcox Abnormal Transient Operating Guidelines (ATOG) developed for TMI-1 to the plant specific guidelines derived from the ATOG. The inspector concluded that the full complement of emergency procedures has been developed for TMI-1.

The inspector then selected five of the ten ATP's for an in-depth review. The ATP's listed below were selected.

-- ATP 1210-2, Revision 8, "Loss of 25 F Subcooled Margin"

-- ATP 1210-3, Revision 10, "Excessive Cooling"

-- ATP 1210-6, Revision 7, "Small Break LOCA Cooldown"

-- ATP 1210-9, Revision 10, "HPI Cooling Recovery From Solid Operations"

-- ATP 1210-10, Revision 13, "Abnormal Transient Rules, Guides and Graphs"

These procedures were compared to the base ATOG procedures for TMI-1 to ensure that the licensee had documented justification of major differences between the generic guidelines developed for TMI-1 and the plant-specific ATOG procedures, known as the ATP's. In addition, the inspector reviewed seven recent Procedure Change Requests (PCR's) for the subject procedures to verify that proper documentation and analysis was performed for changes to the ATP's. In particular, the inspector verified that safety evaluations had been performed as required by 10 CFR 50.59 and that independent review of the changes had been performed.

4.3 Human Factors

The inspector compared the five selected ATP's to the "Writer's Guide for Abnormal Transient Procedures," Administrative Procedure (AP) 1001E, Revision 1. This comparison focused on the format and human factors aspects of the ATP's. The inspector concluded that, with a few exceptions, the ATP's have been developed and updated in accordance with the standards set forth in the writer's guide (WG). Disparity existed between the ATP's and WG guide, such as three caution statements which followed instead of preceded applicable steps, and one example of an apparent ambiguous term in an action step. The licensee indicated to the inspector that a further review would be conducted and a change initiated if necessary.

4.4 Validation/Verification Program

Attachment 6 of the Procedures Generation Package (PGP) submitted by the licensee on January 26, 1984, describes the validation/verification program instituted for the control of the ATP's. The inspector reviewed the licensee's program and determined whether the ATP's have been developed and maintained in accordance with the validation/verification program. The inspector verified the adequacy of the program by interviewing staff engineers, training management, and reviewing both the procedures and a sampling of seven changes to the ATP's.

The PGP documented the various methods used to determine and ensure the ATP's adequacy and effectiveness during the development of the procedure. These methods included operating team reviews, simulator walkthroughs, and other procedure reviews.

The inspector's review of current practices, which are to maintain the adequacy of the ATP's, yielded results as noted below.

-- All changes to the ATP's are controlled by AP 1001A, Revision 13, effective June 26, 1987, "Procedure Review and Approval," in a fashion similar to other safety significant procedures throughout the plant.

-- Review of all potential changes is conducted on a panel approach with various expertise represented, including operations, engineering, and human factors.

-- All ATP's are regularly exercised during operator training on the TMI-1 simulator located near the site. This exercise provides a means of continuous feedback by operations and training personnel.

-- As discussed previously, human factors considerations are generally incorporated in accordance with the WG.

4.5 Training

The inspector interviewed training staff members and toured the training facility in order to determine the adequacy of the program in place to train the operators on the ATP's. The training program consists of periodic classroom training on the basis and theory of their ATP's and simulator training, which requires use of individual ATP's in a hands-on fashion. The operators are given a written exam and a simulator exam during the training period to demonstrate knowledge and understanding of the ATP's. The licensee also posts new procedure changes in the control room as a turnover item to be reviewed by on-coming operations personnel. Also, the licensee is beginning to implement an improved classroom training program aimed at increasing the knowledge of the ATP's by licensed operators. The inspector concluded that operator training on the ATP's is being conducted in an adequate manner.

Based on this review and the ATP review noted in NRC Inspection Report No. 50-289/84-11, this temporary instruction is closed.

4.6 Summary

The inspector concluded that the ATP's had been extensively scrutinized by management and had undergone numerous changes in order to improve the overall quality of the procedures.

Training is currently incorporating an improved classroom training curriculum on the ATP's to be used during operator training. This improvement demonstrates an attitude that favors quality.

The inspector concluded that licensee management was adequately involved in the development and maintenance of of the ATP's.

5.0 Physical Security

5.1 Acceptance Criteria/Scope of Review

The inspector conducted an implementation review of certain aspects of the physical security program. This inspection by the resident staff is intended to supplement the normal yearly programmatic inspections accomplished by region-based staff. Based on the latest SALP (Systematic Assessment of Licensee Performance), the physical security and safeguards program at TMI-1 is rated as category 1 and, therefore, this inspection is accomplished on a quarterly basis.

The following items were reviewed:

-- verification of minimum armed guards assigned on a shift basis;

-- verification of an individual assigned to supervise each shift;

-- review of surveillance test records for search equipments;

-- vital area (VA and protected area (PA) barrier maintenance conduct of access control procedures during shift changes; and,

-- conduct of visitor control badging and escort procedures.

5.2 Findings/Conclusions

The inspector verified that the appropriate armed guard manning was present on two consecutive shifts, as specified in the Physical Security Plan (PSP), and that an appropriate individual was assigned on each shift (shift sergeant) to supervise the activities of the shift site protection officers. Personnel appeared knowledgeable of their individual job responsibilities.

The inspector reviewed several completed weekly checks for operability verification for various search systems. All surveillance checks were completed in a timely manner and deficiencies were properly noted and resolved or work orders generated to accomplish repairs.

Tours of the PA boundaries during daylight and night time conditions revealed no deficiencies. Lighting appeared adequate. A visual verification that PA perimeter monitoring devices were operational was also accomplished. Routine tours by the inspectors included passage through various VA boundaries. No problems were noted with VA access control. No compensatory measures were needed or observed for inoperable VA boundaries during this inspection.

The inspector observed the proper implementation of procedures for visitor control (escorted access). Escorts were made aware of their responsibility to maintain contact with visitor personnel. The inspectors did not note any instances of unescorted, unbadged personnel. Access procedures for controlling package delivery were also verified. Proper searches for delivery or site vehicles entering the PA were accomplished and observed by the inspector.

Site protection officers (SPO's) are assigned to conduct frequent tours of selected VA's that contain cabling that is not protected for fire considerations. These roving fire watch patrols can become boring or monotonous routines with ample opportunity to circumvent the requirements and also cause low personnel morale. Frequent rotational shift assignments are made to mitigate the effects of boredom. This has not been observed among the personnel conducting these patrols and is considered by the inspector to be an indication of a positive attitude on the part of site protection officers.

5.3 Summary

The physical security and safeguards program was properly implemented as noted above. Personnel conducting various aspects of the program appear to be knowledgeable and appear to have proper respect for their duties. The SPO's assigned to implement fire protection measures properly performed their duties. Management involvement was noticeable, in particular, by providing measures for program implementation to mitigate excessive SPO watch boredom.

6.0 Special Temporary Procedure Safety Evaluations

6.1 Acceptance Criteria/Scope of Review

A past inspection finding in NRC Inspection Report No. 50-289/87-08 expressed concern regarding the preparation, review, and approval of Special Temporary Procedures (STP's). During this period, resident inspectors reviewed all of the STP's issued so far in 1987 (42) to assess the following:

-- proper implementation of related administrative controls: Administrative Procedure 1001A, Revision 13, effective June 26, 1987, "Procedure Review and Approval," and Corporate Procedure 1000-ADM-1291.01, Revision 3, effective October 1, 1987, "Procedure for Nuclear Safety and Environmental Impact Review and Approval of Documents;"

-- Safety Evaluations conducted in accordance with 10 CFR 50.59; and,

-- implementation of the licensee's relatively new Technical and Safety Review program.

6.2 Findings/Conclusions

Based on the review of all the 1987 STP's, the following observations were made. The inspector verified proper implementation of AP 1001A and 1000-ADM-1291.01 with respect to the production of STP's.

However, the inspector noted that the STP cancellation requirement could be better defined and could be earlier than the standard ninety-day expiration date. The inspector also reconfirmed the apparent lack of independence of the review and approval process in that one department within TMI-1 division could generate, review, approve, and implement the applicable STP. Further, there was a lack of documentation of review comments generated by the Plant Review Group (PRG). Their comments were not included in the STP package. They were documented as informal minutes of the meetings. Although this process is in compliance with Technical Specifications (TS), this issue is being followed by a previous unresolved item (289/87-08-05).

As yet, no adversity to safety has resulted from this type of review process. Generally, the STP process has been implemented in accordance with present requirements.

7.0 Licensee Action on Previous Inspection Findings

7.1 (Closed) Unresolved Item (289/85-07-01): Salem ATWS Action Plans

During the review of certain short-term actions related to the Salem Anticipated Transient Without Scram (ATWS) event, the inspector identified that the licensee had not submitted certain actions plans. These plans dealt with equipment classification, vendor manual control, and post-maintenance testing. The licensee subsequently made those submittals for the NRC staff's review.

Licensee actions are being reviewed under the following Safety Issues Management System (SIMS) Nos. 75 (B-77, B-86) dealing with equipment classification and vendor interface. Post-maintenance test issues were reviewed under SIMS Nos. 75 (B-78, B-79, B-87, B-88) and found to be acceptable by the NRC staff's letter or Safety Evaluation Reports (SER's), dated October 28, 1985, May 5, 1985, and NRC Inspection Report No. 50-289/85-07.

This item is administratively closed.

7.2 (Closed) Unresolved Item (289/87-05-01): Wire Error Problems in Appendix R Work

This item concerned wire installation errors for work during the last refueling outage; e.g., emergency diesel generator (EDG) "1B" and fire protection work on the reactor protection trip breakers (Field Change Request (FCR) No. 50817). The licensee initiated a review to identify programmatic problems. Specific errors were corrected prior to outage startup. The inspector reviewed findings as documented in GPU Nuclear's memoranda, dated April 21, 1987, from T. Hawkins to R. Toole, and April 27, 1987, from G. Troffer to R. Toole.

The licensee has concluded that the wiring errors were the result of incomplete installation due to errors of omission and/or cognitive errors of personnel performing the work. The licensee has decided that programmatic changes are unwarranted due to the small percentage of human error. An internal licensee memorandum called for a need for more thorough supervisory verification work prior to functional testing. Licensee management indicated that there will be closer contact between construction and test personnel to ensure sufficient knowledge of modifications to properly perform tests. The licensee's review indicated that 100 percent (independent) verification is neither warranted nor practical.

The inspector concurs with these findings. The inspector also noted a heavy reliance on test programs to detect and correct installation errors. The inspector had no further questions. This item is closed but will subject to review during additional extensive outage work to assess the effectiveness of the corrective actions.

7.3 (Closed) Unresolved Item (289/87-09-10): Defective Remote Shutdown Panel Relays

This item was previously discussed in Inspection Report No. 50-289/87-10. New relays were to be installed, tested, and verified to be qualified for service. The licensee was also to determine how to mechanically override the relays which would be required to operate if long-term cooling from the remote shutdown panel is necessary. Operating crews were to be briefed on this procedure.

The inspector verified the relays were replaced with qualified relays, required testing was performed, and crews were briefed on blocking procedures. The inspector reviewed Emergency Procedure (EP) 1202-37, Revision 32, dated November 10, 1987, "Cooldown from Outside the Control Room." This procedure provides the details for blocking the necessary relays for components required to operate after twenty-four hours. JT No. CH569 was reviewed to verify the relays were replaced, qualified, and tested.

The inspector concluded the licensee has met all required commitments. This item is closed.

7.4 (Open) Unresolved Item (289/87-11-05): Remote Shutdown Panel Source Range Indication

During testing performed prior to startup from the letdown cooler outage, NI-9, the source range neutron flux instrument on the remote shutdown panel (RSP) was found to be inoperable. Compensatory measures taken consisted of a roving fire watch through the relay room and normal continuous manning of the control room.

During a routine inspection from May 29 - July 9, 1987, it was reported that during the first hot shutdown condition in excess of forty-eight hours, NI-9 was to be repaired. On November 18, 1987, the licensee submitted a letter to the NRC informing of revisions to the NI-9 repair schedule. Detailed job planning has determined that an outage in excess of nine days is needed to repair NI-9. During the 7R refueling outage, NI-9 will be removed and repaired/replaced unless a shutdown of sufficient duration occurs prior to the scheduled refueling outage. NI-9 is located in the reactor building next to the reactor vessel.

The inspector erroneously reported in NRC Inspection Report No. 50-289/87-11 that a fire watch existing for fire areas which contain NI-1/2 cables is instructed to specifically observe these cables. The licensee provided a clarification. The fire watch will continue to patrol the applicable fire areas for NI 1/2 and other equipment to assure early detection and suppression of a fire, but specific attention to NI-1/2 cables is not required nor practical.

This item remains open pending replacement of NI-9 during the next outage of sufficient duration to complete this repair.

7.5 (Open) Unresolved Item (289/87-13-02): NI-2 Cable Replacement-in-Kind Issue

This item involved a change modification package for NI-1 in which the cable replacement was indicated as a replacement-in-kind, but the cable used was not exactly the same as the original. The licensee committed to clarify the justification for the NI-1 cable replacement or to provide clarification when NI-2 is replaced.

This item remains open pending NI-2 cable replacement or revision of the NI-1 modification package.

7.6 (Closed) 10 CFR Part 21 Report (289/85-PT-07): Defect in Undervoltage Devices for AK and AKR-Type Low Voltage Power Circuit Breakers

This item concerned the mating surfaces of the armature and pole pieces and clearances between the armature and mounting studs in the AK and AKR-type circuit breakers. The defects were brought to the attention of the licensee and NRC via General Electric Advice letter, dated September 13, 1985. These breakers are used in the power supply for the control rod drive circuitry (reactor protection trip system).

The inspector reviewed Preventive Maintenance Procedure E-36, Revision 15, (Electrical), effective October 8, 1987, "CRD Trip Breaker Check." This procedure has been revised to add the mechanism wear inspection and proper engagement requirements. The inspector also reviewed Job Ticket (JT) No. CH798, which involved the inspection of the circuit breakers. General Electric Tab: Switchgear Operability 07313, #300.0, was used to verify acceptability of the circuit breakers. All circuit breakers of the AK and AKR type existing in the plant were found to be satisfactory. The mating surfaces were free of paint, proper clearances were verified and tripping requirements were met. Replacement circuit breakers in the warehouse were found to be unsatisfactory and were returned to the suppliers and replaced with qualified breakers.

The inspector concluded the licensee had received the 10 CFR Part 21 report and took appropriate corrective actions. The inspector had no further questions. This item is closed.

7.7 (Open) Temporary Instruction (TI) 2500/26: Inspection for Compliance with NRC Bulletin 87-02

This bulletin was issued in order to obtain samples of various fasteners used in safety and non-safety-related applications. The inspectors were to verify that the licensee selected a representative sample of nuts, bolts, cap screws, and studs used at the site. Also, a review of licensee receipt inspection and warehouse procedures was to be accomplished. Review of test data was to be accomplished by NRR.

The inspector participated in the selection process and verified that a representative sample was obtained. Licensee documentation of material stock on site had recently been completed and allowed a relatively easy review of what was available for analysis and selection of a proper sample. The samples were appropriately marked and tagged and shipped to the licensee testing vendor. Results will be transmitted to NRC in accordance with bulletin requirements.

This item remains open pending procedure and program reviews described in the temporary instruction and final NRC review of test results.

7.8 (CLosed) NRC Staff TI 2515/79: Inspection of Emergency Operation Procedures

See Section 4 for details.

8.0 Exit Interview

The inspectors discussed the inspection scope and findings with licensee management at a final exit interview conducted December 7, 1987. Senior licensee personnel attending the final exit meeting included the following:

D. Atherholt, Operations Engineer
J. Colitz, Plant Engineering Director
H. Hukill, Director, TMI-1
M. Nelson, Manager, Nuclear Safety
J. Press, QA Auditor
L. Robinson, Media Representative
J. Randazzo, Licensing Engineer

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 Sections 2 and 7.

ATTACHMENT 1

NRC INSPECTION REPORT NO. 50-289/87-23

ACTIVITIES REVIEWED

Plant Operations

-- 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

-- 11/18/87 - Review of licensee response to turbine runback

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

Date	Time
11/1/87	9:30 p.m. - 10:30 p.m.
11/23/87	1:20 a.m. - 2:20 a.m.
11/25/87	6:00 a.m. - 7:00 a.m.

Maintenance

-- Nuclear river water pump strainer foundation work

-- Nuclear services closed cooling pump

-- NS-P-1A minor maintenance

-- Nuclear river pumps (RR, NR, DR) systems expansion joint modifications

Surveillance

-- SP 1301-8.2, Revision 32, dated August 6, 1987, "Diesel Generator Annual Inspection"

-- SP 1303-4.1, Revision 54, dated November 25, 1987, "Reactor Protection System" Monthly Checks, Channel C

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
11/03/87 00:23:19 2 Hours	1.1867	1.19	-0.01	.09	0.0944
11/08/87 08:13:23 2 Hours	0.4746	0.48	-0.08	.02	0.0191
11/09/87 16:16:28 2 Hours	0.8236	0.82	-0.08	.02	0.0239
11/16/87 07:20:02 2 Hours	0.9255	0.93	-0.08	.02	0.0178
11/21/87 17:46:23 2 Hours	0.0706	0.07	-0.10	.0	0.0063

G = Identified gross leakage   U = Unidentified leakage
L = Licensee calculated   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.