U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-289
License No.	DPR-50
Report No.	97-03
Licensee:	GPU Nuclear Corporation
Facility:	Three Mile Island Station Unit 1
Location:	P.O. Box 480 Middletown, PA 17057
Dates:	March 3, 1997 - April 27, 1997
Inspectors:	Michele G. Evans, Senior Resident Inspector Samuel L. Hansell, Resident Inspector
Approved by:	Peter W. Eselgroth, Chief Reactor Projects Section No. 7

EXECUTIVE SUMMARY

Three Mile Island Nuclear Power Station
Report No. 50-289/97-03

This inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers an eight week period of resident inspection.

Plant Operations

Several operational challenges occurred during the period (Section O1). The responses of the operators, supervision, and management were excellent. In addition, support to the operations organization from radiological control, engineering, and maintenance personnel for these challenges was excellent and timely. Specifically, the operations response and evaluation of the March 10, 1997 reactor coolant drain tank in-leakage was thorough and resulted in determining the source location in a short period of time. The shift senior reactor operators addressed the appropriate Technical Specifications related to the reactor coolant system leakage (Section O1.2).

The operations and engineering coordination of the Reactor Building inboard damper valve operability determination was factual and timely (Section O2.1).

Maintenance

Overall, the performance of maintenance and surveillance activities was excellent. Operations personnel continued to implement the "touch the tag" self checking technique during the performance of all surveillance tests (Section M1.1).

Plant management's decision to inspect the 'B' emergency diesel generator (EDG), after returning the 'A' to service, for a potential generic lockwire problem was a positive initiative to ensure the EDG operability remained unquestioned (Section M1.1).

Engineering

During the period of about March 17, 1986 until March 23, 1997, there was not reasonable assurance that the reactor building emergency cooling fans, AH-E-1A, 1B, & 1C would have functioned under post-LOCA environment conditions because their motors did not meet environmental qualifications. There was a small length of exposed metal between the heat shrink tubing and the spark plug porcelain connector to the motor. On March 24, 1997, the licensee made a permanent repair to AH-E-1A and returned it to service. On April 25, 1997, the licensee made an interim repair to the AH-E-1B & 1C motors which the inspectors found to be acceptable. In addition, the inspectors concluded that the licensee's process for and the timeliness of addressing the condition of AH-E-1B & 1C upon identifying the condition of AH-E-1A was weak (Section E2.1).

Plant Support

The emergency preparedness remedial training was adequate and addressed the weaknesses noted during the March 5, 1997, graded exercise. The emergency action level classroom training lacked the attributes of the standard systems approach to training instruction and provided limited benefit to aid the licensed senior reactor operators' ability to classify plant events (Section P5.1).

Security management initiated proactive responses to the TMI-2 accident anniversary demonstration and the potential security concern related to the Oklahoma City federal building bombing anniversary.

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Report Details

Summary of Plant Status

Unit 1 remained at 100% power throughout the inspection period.

I. Operations

O1 Conduct of Operations (71707)[1]

O1.1 General Comments

Using Inspection Procedure 71707, "Plant Operations," the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious. In particular, the inspectors noted the continued implementation of the "touch the tag" self checking technique during all operations activities.

During the period, several operational challenges occurred including increased reactor coolant system (RCS) leakage on March 10, 1997, an Integrated Control System (ICS) module failure on March 28, 1997, and increasing reactor building sump level due to a non-RCS leak in the Reactor Building on April 24, 1997. For each of these events, the inspectors noted that all aspects of the response of the operators, supervision, and management were excellent. In addition, support to the operations organization from radiological control, engineering, and maintenance personnel for these events was excellent and timely. The March 10, 1997, increased RCS leakage is further discussed below.

O1.2 RCS Drain Line Leak

a. Inspection Scope

On March 10, 1997, the reactor coolant drain tank (RCDT) level started to increase at the rate of 2.3 gallons per minute (gpm). The inspectors reviewed the control room indications, recorders, and associated procedures for the identified leak.

b. Observations and Findings

The operations response to the 2.3 gpm reactor coolant system (RCS) leak was thorough and resulted in determining the source location in a short period of time. The RCDT level was going up at the rate of 2.3 gpm. The control room team verified that the reactor building temperature, pressure, and RCDT temperature and pressure readings were all normal, eliminating a steam or high temperature leak from the RCS as probable sources of the leak. The control room operators (CROs) also verified that the reactor coolant pump (RCP) seal leakage and temperatures were normal. In addition, the reactor building (RB) radiation monitor (RM-A2) readings were normal and constant.

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[1]Topical headings such as O1, M3, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

The shift senior reactor operators (SROs) addressed the appropriate Technical Specifications (TSs) related to RCS leakage. Plant personnel selected the most probable sources of the leak and isolated components one at a time from the control room. The leak continued and a decision was made to enter the RB to monitor RCS drain line temperatures. The 'C' RCS drain line temperature was trending up and read approximately 270°F. The two inch RCS drain line manual isolation valve, RC-V-6C, was closed 1/8 of a turn and the leak was stopped at 11:20 am. Corrective Action Process (CAP) form No. T1997-098 was initiated to review and evaluate the RCS leak activities.

c. Conclusions

The operations response and evaluation of the RCDT in-leakage was thorough and operator knowledge of the plant configuration resulted in determining the source location in a short period of time. The shift SROs addressed the appropriate TSs related to the reactor coolant system leakage.

O2 Operational Status of Facilities and Equipment

O2.1 AH-V-1B/C RB Purge Damper Motor Operator Operability Evaluation

The inspectors reviewed an operability determination for the reactor building (RB) inboard purge motor operated dampers (AH-V-1B/C). The engineering motor operator valve (MOV) calculations questioned the ability of AH-V-1B/C to close under accident conditions with a degraded grid voltage. The information was provided immediately to plant operations to address the TS operability requirements. The shift SROs promptly entered the applicable TS time clock (LCO) and applied safety tags to the valve controls to ensure the RB dampers remained closed until the operability issue was resolved. In addition, the SROs addressed the correct reportability requirements and notifications to the NRC.

In conclusion, the operations and engineering personnel coordination of the RB inboard damper valve operability determination was factual and timely and the licensee is following up on resolution of this damper issue.

II. Maintenance

M1 Conduct of Maintenance (62707)

M1.1 General Comments

a. Inspection Scope

The inspectors observed all or portions of the following maintenance and surveillance work activities:

- Job Order No. 116406, "'A' Emergency Diesel Generator (EDG) EG-Y-1A Starting Air Flexible Hose Replacement."
- Job Order No. 134887, "EG-Y-1A Air Compressor Unloader Valve

Replacement and Inspection."

- Job Order No. 132287, "EG-Y-1A Air Start Compressor Oil Change and Inspection."
- Job Order Nos. 132281 and 132282, "EG-V-010AA and EG-V-010BA starting air check valve inspections."
- Surveillance Procedure 1301-8.2C, "Diesel Generator Reduced Scope Inspection (Mechanical)."
- Surveillance Procedure 1303-4.16A, "Emergency Power System Test."
- Surveillance Procedure 1300-3B, "IST of DH-P1A/B and Valves."

b. Observations and Findings

The annual maintenance overhaul was performed on the 'A' and 'B' emergency diesel generators (EDGs) during the weeks of April 14 and 7, 1997. Maintenance procedure 1301-8.2C, "Diesel Generator Reduced Scope Inspection," was revised, with concurrence from the EDG vendor, to provide the written guidance for the reduced scope outage. The reduced scope outage is performed every other year. Fewer diesel components were dismantled for the reduced scope outage which minimized the EDG unavailability time. The major diesel overhauls are performed during the alternate years.

The performance and coordination of the EDG maintenance activities were excellent. The operation's on-line risk evaluations provided improved written configuration control guidance to maximize the availability of safety related equipment and minimize potential operational risks. If needed, the station blackout (SBO) diesel was available to restore electrical power to the 1E 4160 Volt AC bus associated with the out of service EDG. The inspectors verified that EDG outage times were tracked as required by the NRC Maintenance Rule. The inspectors also verified that safety related parts were used for the 'A' and 'B' diesel work activities.

The Quality Verification (QV) department provided continuous overview and monitoring of the outage activities. QV and engineering were proactive in the resolution of a potential operability issue related to a broken lockwire found on the 'A' EDG. The bolts associated with the lockwire in question were found secure and tightened to the proper torque value. Plant management made a decision to inspect the 'B' EDG, after returning the 'A' to service. The 'B' EDG lockwire was found intact and resolved the EDG operability question.

c. Conclusions

Overall, the performance of maintenance and surveillance activities was excellent. Operations personnel continue to implement the "touch the tag" self checking technique on all surveillance tests.

Plant management's decision to inspect the 'B' EDG, after returning the 'A' to service, for the potential lockwire problem was a positive initiative to ensure EDG operability.

III. Engineering

E2 Engineering Support of Facility and Equipment

E2.1 Reactor Building Cooler Fan Motor Equipment Qualification Issue

Background

On March 21, 1997, during the repair of a motor failure for reactor building emergency cooling fan AH-E-1A, licensee engineering personnel questioned the Environmental Qualification (EQ) acceptability of the application of heat shrink tubing which left a small (1 - 1.5 inch) length of exposed conductor at the spark plug connector to the motor. The condition was not sealed and would be exposed to the post-Loss of Coolant Accident (LOCA) reactor building atmospheric conditions (277 degF and 100% humidity.) The motor failure was not related to the heat shrink issue. Following an engineering recommendation, the motor was repaired and heat shrinking was applied over the entire area of exposed metal. AH-E-1A was declared operable and returned to service on March 24, 1997. Engineering documented the EQ concern for AH-E-1A on CAP Form T1997-0137 dated March 23, 1997. Subsequent licensee followup on the other two fan motors, 1B and 1C, resulted in similar repairs to these motors in April 1997.

Observations and Inspection Findings

On about March 24, 1997, licensee personnel identified that the remaining two reactor building emergency cooling fans (AH-E-1B & 1C) were most likely in the same EQ configuration as AH-E-1A. The motor leads for AH-E-1A, 1B & 1C were all replaced in March 1986. At that time, Job Orders (CJ251, CJ252 and CJ253) were issued for the three motors which included specific engineering written guidance in regard to the exact method of making the connections at the spark plug and to not heat shrink over the porcelain of the spark plug. Engineering approval of these job orders, dated March 17, 1986, implied EQ acceptability. Therefore, in March 1997, licensee personnel did not question further the AH-E-1B & 1C EQ condition. However, following additional engineering evaluation of the EQ condition of these motors, no documentation could be found to support EQ acceptability of the 1986 job orders. Therefore, on April 24, 1997, the Plant Review Group (PRG) concluded that their was not reasonable expectation that the exposed conductors would continue to function under post-LOCA environmental conditions for AH-E-1B and 1C. The PRG concluded that prior to the March 1997, AH-E-1A repair, which resulted in a qualified heat shrink configuration supported by EQ testing, all three fan motors were in this condition and would have been considered inoperable. At 3:15 p.m. on April 24, 1997, the PRG concluded that the AH-E-1A fan was operable, but the AH-E-1B and 1C fans were inoperable. Therefore per Technical Specification 3.3.3.c, the licensee entered a 7 day limiting condition of operation (LCO). In addition, the licensee made a 4 hour report to the NRC via the Emergency Notification System (ENS) per 10 CFR 50.72(b)(2)(iii)(D) for a condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident.

On April 25, 1997, Material Nonconformance Report (MNCR) 97-242 was issued to address taping repairs to be made to the AH-E-1B & 1C motors, and to justify continued operation until a permanent qualified repair could be made during the refuel outage in September 1997. Both motors had approximately 1 inch of exposed metal between the heat shrink and spark plug porcelain. The taping repairs were made to both motors and AH-E-1B & 1C were returned to service and declared operable on April 25.

NRC Region I, Division of Reactor Safety management and staff personnel reviewed MNCR 97-242, and concluded that the interim repair to the AH-E-1B & 1C motors was acceptable. In addition, the resident inspectors questioned why it took about 33 days from the identification of the EQ condition of AH-E-1A, until identification and repair of the condition of AH-E-1B & 1C. Based on interviews of engineering, maintenance, quality verification, and operations personnel, the inspectors concluded that the process for and the timeliness of addressing the condition of the other fans was weak. This is an apparent violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action" (EEI 50-289/97-03-01). The inspectors identified the following items which contributed to the delay in identifying and correcting the condition of AH-E-1B & 1C:

- Although the generic issue with the 1B & 1C motors was questioned at the 6:30 a.m. meeting on March 24, 1997, during review of CAP Form T1997-0137, management believed that the condition was unique to the 1A fan because it's motor was replaced in 1993.

- Following identification of the March 1986 job orders, licensee personnel did not appear to pursue in a timely fashion identification of the necessary EQ documentation to support the EQ acceptability of the job orders.

- There were communication problems between site engineering and the corporate engineering office. For example, an electronic mail message was distributed among TMI engineering personnel and management on April 11, 1997 which stated that per corporate EQ personnel, there was no EQ concern with the existing installation on the 1B & 1C motors. This conclusion was incorrect.

- Apparently, EQ personnel at the corporate office did not understand the as found condition of the 1A motor until photographs were provided on April 17, 1997.

- Upon review of the photographs on April 17, 1997, an EQ engineer indicated that he then realized that a prompt operability decision for the 1B and 1C motors was needed, and he believed that prompt meant that he had about five to ten days to complete his evaluation.

As described in FSAR section 6.3.1, the reactor building emergency cooling system is provided to limit post accident ambient pressures to design values. Reactor building air recirculation and cooling units, backed by the reactor building spray system are used during emergency cooling periods. The systems are designed so that the heat removal capability required during the post accident period can be attained by operating spray systems and cooling units in the emergency mode with the following combinations:

Spray Systems	Emergency Cooling Units
Full Capacity (two systems)	None
Half Capacity (one system)	One
None	Three

During the period of about March 17, 1986 until March 23, 1997, there was not reasonable assurance that the fans would have functioned under post-LOCA environment conditions because of this EQ issue. In addition, discussions with licensee engineering personnel indicate that one reactor building spray pump was removed from service for maintenance and/or testing an average of about 8 days per year during this period. Therefore, for approximately 8 days per year for an 11 year period, the reactor building emergency cooling system may not have been able to function to meet the design basis described in the FSAR for reactor building heat removal capability post accident. This is an apparent violation of 1) TS 3.3.3.c for the operability of the reactor building emergency cooling system; and 2) 10 CFR 50.49(f) which requires that each item of electrical equipment important to safety be qualified by testing identical or similar equipment under environmental conditions identical or similar to those postulated for an accident, with analysis to show that qualification based on similarity is acceptable. (EEI 50-289/97-03-02)

Conclusions

During the period of about March 17, 1986 until March 23, 1997, there was not reasonable assurance that all three reactor building emergency cooling fans, AH-E-1A, 1B, & 1C would have functioned under post-LOCA environment conditions because their motors were not environmentally qualified. Approximately one inch of metal was exposed between the heat shrink tubing and the spark plug porcelain connector to the motor. On March 24, 1997, the licensee made a permanent repair to AH-E-1A and returned it to service. On April 25, 1997, the licensee made an interim repair to the AH-E-1B & 1C motors which the inspectors found to be acceptable. In addition, the inspectors concluded that the licensee's process for and the timeliness of addressing corrective actions for the condition of AH-E-1B & 1C was weak. This results in two apparent violations.

IV. Plant Support

P5 Staff Training and Qualification in EP

P5.1 Emergency Plan Exercise Remedial Training related to CAL No. 97-011 Item 1

a. Inspection Scope

The inspectors reviewed the TMI emergency preparedness (EP) remedial training documentation related to confirmatory action letter (CAL) No. 97-011, Item 1. The remedial training was completed for all personnel by April 15, 1997, and covered the weaknesses noted during the March 5, 1997, EP exercise.

b. Observations and Findings

The inspectors reviewed the emergency action level (EAL) instructor lesson plan No. 11.4.06.092, "Emergency Action Levels," to determine the scope and quality of the training. The training instructor lesson plan listed the learning objectives on the cover page but, did not include all of the objectives or associated material in the actual lesson plan. An example was noted for EAL learning objective 'C', "List the two fission product barriers that cannot be recovered once breached." The information was not included in the lesson plan or slide handout to the trainees. All personnel interviewed remembered that the information was covered during training and knew the correct response to the objective. After talking to the instructor, it was noted that the EP training personnel also noted that the lesson plan did not include some of the learning objective information.

All EAL and protective action recommendation (PAR) weaknesses from the March 5, 1997 exercise were covered in the classroom training. However, the event classification training for the senior reactor operators (SROs) was limited to one event classification exercise. Based on the missed General Emergency classification from the exercise, the inspectors concluded that the training provided limited benefit for the plant SROs to improve the CAL weakness. After discussions between EP and plant management, additional event classification scenarios were provided to the SROs.

Four SROs failed the initial examination administered after the EAL classroom training. All four subsequently passed a re-take examination. A review of the EAL training critique sheets noted that overall the trainees thought the instruction was adequate but did document some aspects of the training that could have been improved. In particular, three of the sessions were performed in the shift supervisor's office in the plant. The trainees questioned the interruptions that occurred during the training and in general thought a different training environment would have been more appropriate.

In addition, the inspectors interviewed 5 shift SROs, operation staff personnel, and the EP exercise root cause evaluation (RCE) coordinator. The SROs commented that the classroom training added limited or no value to their knowledge and use of the EALs. Training was also completed for the once through steam generator (OTSG) leak rate calculation methods, EOF response and interaction with the Commonwealth of Pennsylvania personnel and other outside agencies, PAR data beyond 10 miles, technical support center training and "coaching" for all members of the emergency response organization, and validation of plant data that is sent to the emergency director (ED).

c. Conclusions

The emergency preparedness remedial training was marginal, completed on time, and addressed the weaknesses noted during the March 5, 1997, graded exercise. The emergency action level classroom training lacked the attributes of the standard systems approach to training instruction and provided limited benefit to aid the licensed senior reactor operators' ability to classify plant events.

S1 Conduct of Security and Safeguards Activities

S1.1 TMI Security Responses for the TMI-2 Accident and the Oklahoma City Anniversary Dates

a. Inspection Scope

The inspectors reviewed the TMI security response to the TMI-2 accident anniversary demonstration and a potential security concern related to the Oklahoma City federal building anniversary.

b. Observations and Findings

On March 28, 1997, six protestors arrived at the Three Mile Island front gate at approximately 3:00 a.m. in an attempt to block the normal entrance lane into the plant. The protesters walked in a circle, outside of the company property, blocking the plant entrance lane. The protest was peaceful, but did at times pose a personnel safety hazard at the gate. Plant workers were able to drive around the group and cross the river onto the island. The Security Manager, site investigator and four extra site security officers were at the front gate to monitor the situation. As an added protection, the heavy duty steel gate was closed to prevent an intruder from crossing the bridge onto the island. The Pennsylvania State Police were also on call and in close proximity in case the protesters entered GPU property.

At 6:10 a.m. the six protesters were warned that they were not authorized to continue the protest because of the safety hazard to plant workers and themselves. The State Police arrested three of the protesters that chose to ignore the warning. The other three people left the area without a disturbance. The plant access was returned to normal and all protesters left the area.

GPU notified NRC headquarters to inform them of the event. The Pennsylvania State Police issued a press release about the issue. GPU Nuclear did not issue a separate press release about the protest.

In a separate issue, TMI security responded to the potential security concern related to Oklahoma City federal building bombing anniversary.

c. Conclusions

Security management initiated proactive responses to the TMI-2 accident anniversary demonstration and the potential security concern related to Oklahoma City federal building bombing anniversary. In both cases the security department provided additional armed security personnel at the main gate. TMI security and the State Police coordinated their efforts to minimize any potential impact on the plant and alleviated the personnel safety concerns on the TMI-2 anniversary.

V. Management Meetings

X1 Exit Meeting Summary

At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting with TMI management on May 1, 1997, summarizing Unit 1 inspection activities and findings for this report period. TMI staff comments concerning the issues in this report were documented in the applicable report section. No proprietary information was identified as being included in the report.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Ethridge, Acting Radiological Controls/Occupational Safety Director
J. Grisewood, Emergency Preparedness Manager
D. Hosking, NSA Manager
*J. Langenbach, Vice President and Director
R. Maag, Plant Maintenance Director
L. Noll, Plant Operations Director
M. Ross, Director, Operations and Maintenance
J. Schork, Regulatory Affairs
G. Skillman, Technical Functions Site Director
P. Walsh, Engineering Director
J. Wetmore, Manager, Regulatory Affairs

* senior licensee manager present at exit meeting on May 1, 1997.

NRC

W. Ruland, Chief, Electrical Engineering Branch, NRC Region I
L. Cheung, Senior Reactor Engineer, NRC Region I

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 40500:	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92901:	Followup - Plant Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

EEI 50-289/97-03-01
EEI 50-289/97-03-02

Closed

None

Updated

None

LIST OF ACRONYMS USED

CAL	Confirmatory Action Letter
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CRO	Control Room Operator
EAL	Emergency Action Level
ED	Emergency Director
EDG	Emergency Diesel Generator
ENS	Emergency Notification System
EQ	Equipment Qualification
IR	Inspection Report
IST	Inservice Testing Program
JO	Job Order
LOCA	Loss of Coolant Accident
LCO	Limiting Condition of Operation
MNCR	Material Nonconformance Report
NRC	Nuclear Regulatory Commission
OTSG	Once Through Steam Generator
PAR	Protective Action Recommendation
PRG	Plant Review Group
QV	Quality Verification
RB	Reactor Building
RCDT	Reactor Coolant Drain Tank
RCE	Root Cause Evaluation
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
SBO	Station Black Out
SRO	Senior Reactor Operator
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation