U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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| Report No. | 97-09 |
| Licensee: | GPU Nuclear Corporation |
| Facility: | Three Mile Island Station, Unit 1 |
| Location: | P.O. Box 480<br>Middletown, PA 17057 |
| Dates: | September 7 - November 1, 1997 |
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## EXECUTIVE SUMMARY

Three Mile Island Nuclear Power Station
Report No. 50-289/97-09
September 7, 1997 - November 1, 1997

This integrated inspection included routine resident inspector activities and announced inspections in the areas of licensee operations, engineering, maintenance, and plant support over an eight-week period.

GPU Nuclear (GPUN) conducted outage activities, including reactor refueling, maintenance, and unit restart safely over the period

**Plant Operations**

- Generally, GPUN conducted the 12R refueling outage well. Operators, despite several lapses, displayed excellent control of plant and equipment conditions, including draining to mid-loop and in refilling the reactor coolant system (RCS) and in unit restart activities (Sections O1.1 and O1.2.1)

- The inspectors questioned the safety significance of GPUN management's decision to drain the RCS to mid-loop with only one decay heat removal (DH) system available. While apparently allowed by the Technical Specifications (TS), the wording of the TS bases would lead to a reasonable safety interpretation that this condition should not occur. The inspectors considered this an Inspection Followup Item, pending further NRC review of the safety significance of this issue. (Section O1.2.2) **(Inspection Followup Item (IFI) 50-289/97-09-01)**

- During the RCS filling operations a shift supervisor (SS) displayed a poor procedure compliance standard. The SS directed the plant operators to increase the fill rate, from a flow path not included in the applicable operating procedure. This resulted in an excessive flowrate and led to the overflow of approximately 50 gallons of RCS water out of the control rod drive mechanism (CRDM) vent openings. This appeared to be a violation, based on failure to follow approved station operating procedures. (Section O1.2.3) **(Escalated Enforcement Issue (EEI) 50-289/97-02)**

- Operations department and training management continue to coordinate licensed operator requalification classroom, simulator, and on-the-job training well, providing intensive preoutage training. In addition, all operators involved with the core offload and reload completed on-the-job training and a qualification card that contained refueling operation tasks from the job task analysis. (Section O5.1)

Executive Summary

**Maintenance**

- Routine outage meetings provided good insight into upcoming work activities and equipment problems. For the Job Order activities observed, the work packages contained the needed planning information and the workers properly documented the completed work. GPUN used lessons learned from the prior refuel outages and other plants to prevent recurring problems. (Section M1.1)

- Maintenance rework was required for the pressurizer power operated relief valve (PORV), following identification of a miswired operating solenoid valve (Section M2.1) and repeatedly on the 'A' DH pump following seal replacement. The rework on the 'A' DH pump resulted in an extended time with only the 'B' DH pump operable for core heat removal. (Section M1.1)

- Plant management's decision to replace the remaining 31 old design thermal barriers during the 12R refuel outage with the new design displayed a clear commitment to resolve the slow CRDM drop time issue. All control rod drop times were less that the TS limit when tested before the plant startup. (Section M1.1)

- GPUN responded proactively to the generic makeup (MU) system high pressure injection (HPI) thermal sleeve cracking issue at Babcock and Wilcox (B&W) plants. After finding two cracks in the 'B' thermal sleeve, plant management decided to complete a visual internal inspection of the remaining three sleeves, detecting no additional flaws. Replacement of the 'B' thermal sleeve was well conducted and supervised. (Section M1.1.1)

- The pressurizer PORV was inoperable for the two-year operating cycle from October 1995 to September 1997 due to a wiring error and the failure to do a post-maintenance test (PMT) after the valve replacement in the September 1995 refuel outage. This issue involved the unavailability of the PORV during plant depressurization situations as directed by the emergency operating procedures and on the calculated increase in core damage frequency (4.18E-5/year to 4.85E-5/year). This appeared to be a violation of TS IST requirements. (Section M2.1) **(EEI 50-289/97-09-03)**

- The nuclear safety assessment group (NSA) reviewed PMTs conducted during 12R on safety-related equipment, on a sampling basis, because of the PORV root cause analysis. This assessment was very comprehensive and appropriately expanded after a few minor documentation problems were found. Plant management's decision to evaluate and resolve the minor PMT issues before plant restart was prudent. (Section M2.1)

- The inspector identified a question concerning the appropriateness of testing an emergency diesel generator (EDG), following simulated loss of offsite power (LOOP) and loss of coolant accident (LOCA) conditions, with the output breaker in the pull-to-lock position. This issue was considered an Unresolved Item pending further NRC staff review. (Section M2.2) **(URI 50-289/97-09-04)**

- The licensee's corrective actions were appropriate and timely to prevent recurrence of violations regarding scaffold construction in safety related areas of the plant. **(Closed - Violation (VIO) 50-289/96-07-01)**

- GPUN conducted inservice inspection (ISI) activities at TMI following the American Society of Mechanical Engineers (ASME) Section XI, 1986 Edition and 10 CFR 50.55a(g). The inspectors found the flow accelerated corrosion (FAC) program thorough, effective, and capable of predicting the depletion of piping wall thickness. (Section M8.2)

- The eddy current inspection program was well planned and organized, and could determine the integrity of the once through steam generator (OTSG) tubes, according to the ASME Code, Section XI and TS. (Section M8.3)

<u>Engineering</u>

- The PORV problems were recognized due to the diligent review and questioning attitude of an electrical engineer. The engineer recognized and pursued the connection between the PORV problem found in the current refuel outage and the possibility of the same problem existing with the previously installed PORV. (Section M2.1)

- Based on a detailed system review, GPUN has maintained the core flood (CF) system in good material condition. Engineering maintained the design basis following 10 CFR 50.46. The inspector also concluded the design basis document (DBD) was "easy to use" and thorough in describing the design basis. Documents were found consistent with the applicable sections of the DBD, updated final safety analysis report (UFSAR), TS, IEEE standards, procedures, system drawings and system layout. Adequate procedures were in place to operate the CF within its design basis. (Section E.1)

- GPUN took adequate actions to improve the content by resubmitting LER 97-003. This LER now correctly recounts the events in a more clear and concise manner. (Section E8.1) **(Closed - VIO 50-289/97-07-01)**

- Engineering developed and maintenance installed a modification to improve the closing capabilities of MU-V-3 the RCS letdown outboard isolation valve. The modification installed an air to close function on this air operated valve, increasing the ability of the valve to close under system flow and differential pressure. The modification package was detailed and well developed and the PMT appeared complete. (Section E8.1.2)

**Plant Support**

General:

- Material conditions continued to be good. Equipment needed to meet TS requirements for shutdown conditions was maintained and operated well. (Section R1.1)

- Generally the inspectors found that housekeeping degraded over the outage. This degradation was particularly evident in the reactor building (RB), where outage related activities resulted in large amounts of debris to be left on floors and surfaces The debris observed included nails and pieces of wood and sawdust from scaffolding activities, plastic tie-wraps from the installation of temporary hoses and cables, a large roll of sheet plastic, pop-rivet stems from sheet metal installations, and tape materials left following work. (Section R1.1)

Radiation Protection:

- Adequate contamination controls and radiation survey and monitoring programs were being carried out. However, weak attributes were noted in the establishment and maintenance of contaminated areas and in the survey program. The inspectors noted instances where local postings did not agree with the area conditions and where material was allowed to cross contaminated area boundaries. In these cases the radiation protection staff corrected the conditions. (Sections R1.1 and R1.3.1)

- GPUN failed to survey adequately during the removal of the reactor vessel seal plates. As such, adequate hot particle controls were not in place and did not prevent a personnel skin contamination. This appears to be a failure to follow TS 6.11 and is a violation. (Section R1.3.2) **(EEI 50-289/97-09-05)**

- Activities to maintain personnel exposures as low as reasonably achievable (ALARA) were generally considered strong, especially the prejob reviews. (Section R.1.4)

- A contract worker failed to follow the high radiation control procedure; the action led to an unlocked high radiation area, the 'B' OTSG shield door, with the potential for an inadvertent radiation exposure greater than personnel limits. This failure was similar to a prior problem that occurred in the 1993 and 1995 refuel outages. This issue appeared to be a violation of TS 6.8.1, in that procedures for locking high radiation areas were not followed. (Section R4.1) **(EEI 50-289/97-09-06)**

- The selection, training, and qualification of contracted radiological control technicians for the outage were in accordance with requirements. The new radiation worker coaching process set up by radiological controls (RC) Field Operations was a good initiative. (Section R5)

- NSA conducted a good quality audit of the RC area with proper scope and depth. The surveillances by RC personnel resulted in the correction of numerous minor deficiencies. (Section R7)

<u>Security:</u>

- The inspector noted no deficiencies during a night tour of the protected area. (Section R1.1)

- All openings in the protected area boundary were controlled properly by the security department for the entire 12R refueling outage. Based on this improved performance, the inspectors concluded that GPUN had taken effective corrective actions for prior problems. (Section S1)

TABLE OF CONTENTS

TABLE OF CONTENTS

## Report Details

### Summary of Plant Status

Unit 1 was shutdown at the beginning of the report period for the scheduled 12R refuel and maintenance outage. GPUN completed the outage work in 44 days. Operators took the unit critical on October 18 and synchronized the generator to the grid on October 19. The plant reached 100% reactor power on October 22 and remained there over the rest of the inspection period.

## I. Operations

**O1 Conduct of Operations (71707, 92901)[1]**

O1.1 General Comments

Using Inspection Procedure 71707, "Plant Operations," the inspectors conducted frequent reviews of ongoing plant operations. Usually, the conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the sections below. In particular, the inspectors noted that the decision to drain the fuel transfer canal and install the reactor vessel head with the 'A' DH system pump out of service placed the plant in a more vulnerable condition to ensure sufficient decay heat removal capabilities.

Plant management's decision to replace the remaining 31 old design thermal barriers during the 12R refuel outage with the new design displayed a clear commitment to resolve the slow CRDM drop time issue. All control rod drop times were less that the TS limit when tested before the plant startup.

O1.2 Refuel Outage Control

a. Scope

The inspectors routinely monitored control room activities and the establishment of specific plant conditions necessary for outage work, including review of: log books, plant status paperwork, safety systems in operation or required for standby service, and reactor vessel water level requirements.

b. Observations/Findings

Overall control of plant conditions was excellent, except for several issues discussed below. Operators conducted reactor coolant system (RCS) draining and establishment of mid- loop operations very well. Reactor vessel water level instruments were properly installed and operable. The operation department's control and oversight of the core offload and reload were done without error.

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[1] Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

During the first day on shutdown cooling the inspectors found that GPUN was recording the outlet temperature of the reactor coolant leaving the DH system heat exchanger as the RCS temperature. This did not appear correct since this did not represent the bulk coolant temperature. GPUN agreed and changed the monitored temperature to be the reactor coolant temperature at the inlet to the DH heat exchanger.

The operators retained proper control over the MU system pumps and valves to reduce the possibility of RCS overpressurization while in cold shutdown. Operators performed well during restoration of the MU system, including the racking in of circuit breakers and initial starting of the pumps, following establishment of proper RCS conditions.

Operator training handouts on specific changes/modifications completed during the outage were well prepared and provided the operators with needed information on the effects of these changes on the operation of systems and components.

The operations department conducted the plant pressurization, heatup, and startup very well. Control room command and control were very good.

O1.2.1 Draindown to the Reactor Coolant System Mid-Loop Operation

The operating crews performed the two RCS mid-loop draindown evolutions during the 12R refuel outage, without incident, in a controlled manner. In both evolutions the control room staff had two independent reactor vessel level indications available to ensure a level above the DH pump vortex limit. Prior changes to Operating Procedure 1103-11 "Draining and Nitrogen Blanketing of the Reactor Coolant System," have improved the operators' ability to control reactor vessel water level and decay heat removal pump suction during RCS draindown.

On October 5, 1997, with RCS level steady in mid-loop operation, the inspectors questioned the reason for the energized RCS "Draindown Level HI/LO Alarm." At the time of the alarm the RCS level was at approximately 14.2 inches and the low level alarm was in solid. The low level alarm provides a warning to plant operators that a leak or inadvertent water removal is in progress from the RCS. The shift supervisor contacted instrumentation and control (I&C) technicians to adjust the low level setpoint to the correct value. The I&C technicians adjusted the low level alarm to the proper band based on the plant conditions.

O1.2.2 Review of Draindown Following Refueling - Open - URI 50-289/97-09-01: Decay Heat Removal Requirements During Reactor Vessel Draining

The inspectors reviewed the sequence of events following the completion of reactor refueling on September 29. During this review the inspectors determined that GPUN took actions that placed the unit in a degraded decay heat removal condition before and during mid-loop conditions (i.e., only one DH pump operating and operable) although apparently allowed by TS. A sequence of events with a discussion of the applicable TS follows:

Rx vessel flange 321 feet
Bottom of cold leg 314', the centerline of cold leg, or 0 inches mid-loop instruments.

TIME LINE

9/24 Initial Conditions: Fuel transfer canal (FTC) flooded above 344 feet (>23 feet above the RX Vessel flange), defueled. No DH system in operation. 'A' DH operable, but not in operation. 'B' DH out of service due to pump replacement and decay heat removal closed cooling (DC)/decay heat removal river water (DR) heat exchanger work.

With no fuel in the reactor vessel TS allow no DH systems to be in service.

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| 9/26 | 1:30 a.m. | 'B' DH operable, following maintenance and testing |
| | 7:50 a.m. | 'A' DH removed from service for maintenance, cannot be restored within 24 hours. |
| | 12:00 p.m. | Fuel reload begins |

TS require two operable DH systems, with one in operation, when there is fuel in the reactor vessel. TS allow a reduction to only one DH system with FTC level maintained above 344 feet, there is no LCO for this condition.

| | | |
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| 9/29 | 06:30 p.m. | Fuel transfer completed and core verified |
| | 10:00 p.m. | Began draining FTC to < 344 |
| | 10:50 p.m. | Entered 7 day LCO on 'A' DH since level was now < 344 |

TS require two operable DH systems, with one in operation, when there is fuel in the reactor vessel. TS allow a reduction to only one DH system, if the other system is out of service for less than seven days.

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| 9/30 | 3:15 a.m. | Secured draining FTC, level at 341 feet |
| | Afternoon | CF system testing, 'B' DH pump secured for a short period when injecting the 'B' CF tank. |
| | 4:55 p.m. | Recommenced draining |
| 10/1 | 12:55 a.m. | RCS level 323 feet 5 inches |
| | 1:55 a.m. | Secured pump down. RCS level 320 feet 8 inches |
| | 2:15 a.m. | Started pumping from 'B' DH to continue lowering vessel level |
| | 2:30 a.m. | RCS level 318 feet 10 inches (approx. 58 inches above centerline of cold leg) |
| | 12:30 p.m. | 'A' DH pump ready for run, seal leaks |
| | 6:50 p.m. | RCS level 57 inches (above the centerline of cold leg) on mid-loop instrument (MLI) |
| 10/2 | 3:00 a.m. | Lowered RCS level to 50 inches MLI |
| | 5:10 a.m. | Plenum installed in reactor vessel |
| | 6:00 a.m. | RCS level lowered to 18 inches MLI |

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|---|---|---|
| | 1:30 p.m. | Reactor vessel head installed |
| 10/3 | 5:50 a.m. | Filling 'A' DH |
| | 6:50 a.m. | Lowered level to 12 inches MLI for HPI nozzle inspections |
| | 8:00 a.m. | Open drains for 'A' DH seal leaked |
| 10/4 | 12:50 a.m. | Opened MU-V94 for first nozzle inspection |
| | 1:00 a.m. | RX vessel head tensioned |
| | 6:50 p.m. | DH-V-22A opened for inspection |
| 10/5 | 8:30 a.m. | Filling 'A' DH |
| 10/6 | 5:20 a.m. | 'A' DH declared operable following testing |

TS required that the 'A' DH system be returned to service in less than seven days.

In review of the TS and the associated bases the inspectors questioned if GPUN took a conservative path during this period. Specifically, the TS Bases stated that either 23 feet above the flange would be maintained or another flow path from the borated water storage tank (BWST) established to maintain subcooled conditions for seven days, before the redundant train could be taken out-of-service. This seems consistent with the improved TS that would not have allowed the draining with only one train of DH operable.

The inspectors found that the plant review group (PRG) became involved after the 'A' DH pump seal failed following initial replacement. The PRG reviewed the TS requirement and possible ways to establish an alternate cooling path from the BWST, if the DH pump maintenance took longer than seven days, but determined that such a path had not previously been established and that developing the path would not be a worthwhile initiative at the time. The inspector did note that the BWST could have been used as a makeup source to the reactor vessel through the operating DH system using the pump or just head of the tank, but that this path would not have provided any cooling.

Overall, the inspectors believe that lowering reactor vessel water level to a mid-loop condition with only one means of decay heat removal was not the safety conservative decision. Further, the inspectors questioned if the TS wording was correct based on the wording of the bases.

O1.2.3 Reactor Coolant System Fill and Vent - Open - EEI 50-289/ 97-09-02; Failure to Follow Reactor Coolant System Filling Procedure

The inspectors observed portions of the two-day RCS fill and vent evolution at the end of the outage. OP 1103-2, "Fill and Vent of the Reactor Coolant System" provided excellent precautions, limitations, and prerequisites to ensure proper RCS and support system alignment for the important evolution. The prerequisites listed the reactor coolant bleed tank (RCBT) as the preferred water source. The procedure contained hold points and actions at specific water level heights to ensure the fill progressed without an inadvertent loss of water or over fill.

On October 15, the day shift supervisor (SS) displayed poor command and control by directing the control room operator (CRO) to increase the fill flowrate, using a flow path not allowed by the procedure. Specifically, the SS directed the opening of the manual 'B' DH pump suction throttle valve from the BWST. The auxiliary building (AB) auxiliary operator (AO) opened DH-V-5B approximately eight turns after hearing flow through the valve. Almost immediately the pressurizer level recorder, in the control room, changed from a gradual increase to a prompt rise. The CROs observed the rapid increase in pressurizer level and secured the RCS fill from the RCBT. The CROs attempted to direct the AO to close the DH-V-5B isolation valve. However, radio communication difficulties resulted in a delay before the AO received the message and closed the DH-V-5B. Before the valve was closed, approximately 50 gallons of RCS water spilled out of the CRDM vent openings onto the reactor vessel head area.

The inspector found that the procedure only allowed the flow path from the RCBT. The procedural guidance for the BWST make up directly to the RCS is contained in OP 1104-4, "Decay Heat Removal System," but was only specified for controlling RCS level and specifically warns against using this path for the filling operation. OP 1104-4, Enclosure 2, step II.1. states that "The Plant Operations Director shall establish the level to be maintained in the RCS. Controlling the level using this method is NOT considered to be nor should it be used as a major RCS fill and vent method."

Further OP1103-2, Section 3.1.2, 17.c provided good guidance on controlling the filling operation as water level neared the top of the CRDMs. In part the procedure stated, "when the level at the CRDM vent is observed at one to two feet below the top, TERMINATE THE RCS FILL and hold level."

A corrective action process (CAP) form, CAP 1997-800, was initiated to evaluate the information and determine the root cause and associated corrective actions. Based on the management review of the CAP data, a quality deficiency report (QDR) was initiated to track the root cause evaluation and completion of the corrective actions.

c. <u>Conclusions</u>

Overall, GPUN conducted the 12R refueling outage very well. Operators, despite several lapses, displayed excellent control of plant and equipment conditions, including draining to mid-loop and in refilling the RCS and in unit restart activities.

The inspectors questioned the safety significance of GPUN management's decision to drain the RCS to mid-loop with only one DH system available. While this appeared to be allowed by the wording of TS, the wording of the TS bases would lead to a reasonable safety interpretation that this condition should not occur. The inspectors considered this an Inspection Followup Item, pending further NRC review of the safety significance of this issue. **(IFI 50-289/97-09-01)**

An SS showed a poor procedure compliance standard for the RCS fill and vent evolution, by directing plant operators to increase the RCS fill rate from a flow path not included in the applicable operating procedure. The excessive flowrate led to the overflow of approximately 50 gallons of RCS water out of the CRDM vent openings. The lack of controls and failure to follow procedures by the SS during the filling process appeared to be a violation. **(EEI 50-289/97-09-02)**

## O5 Operator Training and Qualification

### O5.1 Licensed Operator Refuel Outage Training

a. Scope

The inspectors reviewed the licensed operator requalification (LOR) training before the 12R refuel outage. Also reviewed were the operator on-the-job training tasks conducted before the refueling activities.

b. Observations/Findings

The topics selected for the LOR continuing training were focused on details of the 12R outage, lessons learned from the last refuel outage, and industry information related to outage problems and events.

The classroom training included lectures on the fuel handling equipment, DH system operation, outage fuel reload topics, reactor coolant system mid-loop operation, and other refuel outage related topics. The lectures included normal operating procedures, administrative requirements, and refuel outage problems that occurred at TMI and similar plants.

In addition, all operators involved with the core offload and reload completed on-the-job training and a qualification card that contained refueling operation tasks from the job task analysis. The training included the movement of dummy fuel bundles and familiarization with the refueling equipment.

c. Conclusions

Operations and training management continue to coordinate licensed operator requalification classroom, simulator, and on the job training to provide intensive training to plant operators prior to the refuel outage. In addition, all operators involved with the core offload and reload completed on the job training and a qualification card that contained refueling operation tasks from the job task analysis.

## II. Maintenance

### M1 Conduct of Maintenance (62707, 61726, 92902)

M1.1 General Comments

a. Scope

The inspectors routinely attended plant morning and afternoon outage meetings and the morning maintenance meeting to assess the control of work and planning for upcoming activities.

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

- Job Order Nos. 112731 and 109174, "'A' and 'B' Circulating Water Pump Replacement."
- Job Order No. 132154, "'A' Decay Heat Removal Heat Exchanger Clean and Inspect."
- Job Order No. 132592, "Main Steam Safety Relief Valve MS-V-21B Inspection and Clearance Checks."
- Job Order No. 143061, "'B' High Pressure Injection Thermal Sleeve Replacement."
- Job Order Nos. 112731 and 112732, "'A' and 'B' Circulating Water Pump Replacements."
- Surveillance Procedure 1303-11.54, "'A' Low Pressure Injection Test."
- Surveillance Procedure 1300-3B, "IST of 'A' Decay Heat Removal Pump and Valves."
- Surveillance Procedure 1300-3T, "Pressure Isolation Test of FAC Valves CF-V4A/B, CF-V5A/B and DH-V22A/B."
- Refueling Procedure 1505-1, "Fuel and Control Component Shuffle."

b. Observations/Findings

Work activities associated with the circulating water pumps and other potential protected area boundary openings were coordinated with security and operation departments to ensure that all openings received the proper security response. Additional details of the work controls are discussed in Section S1 of the report.

Main steam (MS) safety relief valve work on MS-V-20B and MS-V-21B was completed with the assistance of the valve vendor. Experienced mechanical maintenance personnel conducted the work activities and coordinated with radiological controls when opening the valve internals for inspection.

The pressure isolation test of the CF valves CF-V4A/B and CF-V5A/B were coordinated effectively between multiple departments. Lessons learned from other B&W plants such as lowering the fuel transfer canal level before the test prevented repeat problems that occurred at other sites. The test was done satisfactorily with no adverse impact on the plant.

To resolve a generic B&W problem with control rods with slow drop times following a reactor trip, GPUN replaced the remaining 31 old design CRDM thermal barriers with a new design during the 12R refuel outage. This demonstrated a clear commitment to resolve the slow CRDM drop time issue. During pre-startup testing all control rods inserted within the times allowed by TS.

Except for the work on the PORV, discussed in Section M2.1 and the 'A' DH pump seal replacement, the maintenance work activities were very well controlled and done correctly the first time. The leaking seal on the 'A' DH pump was replaced 3 times prior to stopping the leak; it ultimately was necessary to get the vendor representative in for installation expertise. The rework on the 'A' DH pump resulted in an extended period with only the 'B' DH pump operable for core heat removal.

M1.1.1 <u>High Pressure Injection Thermal Sleeve Replacement</u>

GPUN performed well in conducting the internal visual inspection of the four MU system on HPI nozzles. ISI engineers performed radiography and visual inspection of the four thermal sleeve pipe sections at the RCS cold leg. The inspectors observed the internal inspection of two of the four MU injection line thermal sleeves. The ISI engineer coordinated the inspection with operations and radiological controls departments. The internal visual inspection was done with a camera and was videotaped for additional review and verification of the thermal sleeve condition.

Initially, two of the four HPI lines were scheduled for inspection. After finding the cracks in the 'B' thermal sleeve, plant management decided to perform a internal inspection of the remaining two MU line thermal sleeves. GPUN verified that the three other thermal sleeves were free of any cracks or flaws.

The visual inspection of the 'B' thermal sleeve revealed two 2 to 3 inch surface cracks. The faulty thermal sleeve was replaced and tested before the RCS heatup. The inspectors monitored the work for Job Order No. 143061, "'B' High Pressure Injection Thermal Sleeve Replacement." The work was done by Framatome personnel who were involved with a similar repair at Oconee Nuclear Power Station. The site quality verification (QV) personnel provided excellent oversight of the repair work including the use of non-destructive testing to verify the weld was performed satisfactorily.

c. Conclusions

Routine outage meetings provided good insight into upcoming work activities and equipment problems.

For the Job Order activities observed, the work packages contained the needed planning information and the workers properly documented the completed work.

Lessons learned from the prior refuel outages and other plants were used during the 12R outage to prevent recurring problems. Maintenance rework was required for the pressurizer power operated relief valve and the 'A' DH pump seal replacement. The rework on the 'A' DH pump resulted in an extended time with only the 'B' DH pump operable for core heat removal.

The GPUN responded proactively to the generic MU high pressure injection thermal sleeve cracking. After finding two cracks in the 'B' MU high pressure injection thermal sleeve, plant management decided to complete a visual internal inspection of the remaining three MU line thermal sleeves, detecting no additional flaws. Replacement of the 'B' thermal sleeve was well conducted and supervised.

Plant management's decision to replace the remaining 31 old design thermal barriers during the 12R refuel outage with the new design displayed a clear commitment to resolve the slow CRDM drop time issue. All control rod drop times were less that the TS limit when tested before the plant startup.

## M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Open - EEI 50-289/97-09-03 Power Operated Relief Valve Inoperable for an Operation Cycle

a. Background/Scope

On October 13, 1997, GPUN determined that the pressurizer PORV, installed September 23, 1995, could not be opened during the operating cycle before the 12R refueling outage, either automatically or manually from the control room. An electrical engineer detected a wiring error after the PORV failed to operate following the valve replacement in September 1997. The engineer reached this conclusion based on the failure of the PORV to operate after the valve was replaced during the 12R outage and from the observation that the removed valve had also been wired incorrectly.

The inspectors reviewed the documentation associated with the replacement of the pressurizer PORV. The PORV was inoperable for the two-year operating cycle from October 1995 to September 1997 due to a wiring error and the failure to conduct a PMT after the valve replacement in the September 1995 refuel outage. In response to the PORV PMT issue, the inspectors independently reviewed a select number of refuel outage work packages to evaluate the extent of the missed PMT problem. In addition, the inspectors monitored the NSA PMT evaluation conducted because of this issue.

b. Observations/Findings

The root cause of the event was identified as personnel error. An electrician failed to connect the PORV wires correctly during the valve installation in 11R refueling outage. Besides the wiring error, the required PMT on the valve following the installation was not performed and the independent verification of the wiring was inadequate.

The prassurizer pressure relief function is provided by two ASME Code safety valves, that are nuclear safety grade components and one PORV that is not a safety grade pressure relief device. The PORV was not considered to perform a safety related function because the valve was not required for safe shutdown of the reactor, maintaining it in a safe shutdown condition, nor to prevent or mitigate the consequences of an accident as described in the UFSAR.

The PRG evaluated the safety consequences of the inoperable PORV, TS compliance, and the ASME Section XI PMT requirements. PRG determined that the safety consequences were minimal because the accident analysis did not take credit for the PORV to open to accomplish an RCS pressure reduction. The wiring error resulted in the failure of the PORV, normally closed, to open in the automatic or manual mode. A review of the PORV refueling interval surveillance test for the 1995 refuel outage determined that the test was performed before replacement of the PORV during the 1995 refuel outage. Because the surveillance test was done before the valve replacement, it did not fulfill the ASME Code requirements for an inservice post-maintenance test.

The TMI-1 probability risk assessment (PRA) personnel analyzed the impact of the PORV failure and the associated change in the Core Damage Frequency (CDF). The PRA calculations determined that a CDF increase of 16% would occur, from 4.18E-5/year to 4.85E-5/year.

The PORV does function as an approved redundant and diverse means of providing low temperature overpressure protection (LTOP) during plant heat-up and cooldown. TS 3.1.12 addresses the TMI LTOP mitigation system function. The T.S. 3.1.12 action statement and bases allow for other means of low temperature overpressurization protection. Written procedure controls in the TMI Cooldown, Heatup, and Makeup system operating procedures require that the MU pump discharge valves MU-V-16A/B/C/D and MU-V-217 are danger tagged closed when the RCS temperature is less than 332°F. The inspectors verified that the MU isolation valves were tagged closed as required by procedure to ensure compliance with the TS action statement during the period when the PORV was inoperable.

TS 4.2.2 requires that ASME Code Class 1, Class 2, and Class 3 valves be inservice tested (IST) according to Section XI of the ASME Code and Oma-1988, Part 10, paragraph 3.4, before returning a valve to service. Administrative Procedure AP 1041, "IST Program Requirements," Section 4.2, requires, in part, "After an IST valve has been replaced and before the time it is returned to service, an IST valve test shall be performed."

Before the plant restart, the NSA independently reviewed the outage PMT activities to find out if the PORV issue represented an isolated or a programmatic problem. The review was focused on safety related equipment work packages that included multi-disciplinary coordination. The focus was on electrical maintenance and safety related motor operated valve (MOV) work. The initial review conducted by the NSA and QV organizations included 44 work packages. The evaluators found three work packages that were missing the IST documentation needed to verify proper valve stroke times. In each case the valves were stroked and the times were documented using electrical maintenance data sheets. Operations stroked all of the valves in question using the required IST procedure. All valve times were within the required IST band for satisfactory stroke times. Based on this information the NSA assessment group looked at the remaining MOV work packages (five total) and did not find any additional problems. Because of the initial paper work discrepancies, NSA expanded their review to include all other plant work disciplines. No additional problems were noted for the work packages reviewed (approximately 30 additional packages were reviewed). Based on the NSA findings, plant management authorized proceeding with the plant startup.

The inspectors independently reviewed 10 work control packages and found one case of a missing MOV IST data sheet, similar to that found by NSA. The problem was corrected satisfactorily. Four of the packages the inspectors randomly selected were also reviewed by the NSA group including two MOV work packages that lacked the proper IST documentation for the MOVs.

c. Conclusions

The pressurizer PORV was inoperable for the two-year operating cycle from October 1995 to September 1997 due to a wiring error and the failure to do PMT after the valve replacement in the September 1995 refuel outage. This issue involved the unavailability of the PORV during plant depressurization situations as directed by the emergency operating procedures and on the calculated increase in core damage frequency (4.18E-5/year to 4.85E-5/year). Failure to conduct the PMT appeared to be a violation of TS IST requirements. **(EEI 50-289/97-09-03)**

The PORV problems were recognized due to the diligent review and questioning attitude of an electrical engineer. The engineer recognized and pursued the connection between the PORV problem found in the current refuel outage and the possibility of the same problem existing with the previously installed PORV.

The NSA Review of PMTs related to the PORV root cause analysis was comprehensive and appropriately expanded after a few minor documentation problems were found. Plant management's decision to evaluate and resolve the minor PMT issues before plant restart was prudent.

M2.2 Review of Loss of Power and Loss of Coolant Accident Outage Elecuical Testing. - Open - URI 50-289/97-09-04 Emergency Diesel Generator Testing During Simulated Accidents

a. Scope

The inspector reviewed Surveillance Test 1303-11.10 Emergency Safeguards (ES) System Emergency Sequence and Power Transfer Test, to verify that it met current TS requirements for the testing of offsite power system loading and sequencing and for the operation of the EDGs in a post-LOOP and post-LOCA condition.

b. Observations/Findings

The inspector found that GPUN conducted the testing such that the EDG would start on an ES signal, with its output breaker in a pull-to-lock position. There might not be a need to have the breaker in this position since the EDG should not close onto the bus following an ES signal, since offsite power was not lost.

The next part of the test caused the simulation of an undervoltage condition, simulating a LOOP; here too the EDG breaker was left in a pull-to-lock position. Once the emergency bus deenergized following the LOOP signal the procedure instructed the operator to wait five seconds and then take the EDG breaker out of pull-to-lock. Once out of pull-to-lock the EDG breaker would close to repower the bus and then the loads would sequence on.

The inspector questioned if conducting this testing with the EDG breaker in pull-to-lock met the TS surveillance requirement of "automatically start and loading the EDG." The inspector was concerned since the test might not be conducted in a mode where the EDG would respond automatically, and as realistically as possible, to the simulated ES and LOOP signals. However, based on the June 1997 LOOP it would appear that the EDGs performed their safety function and started properly. The inspector further noted that improved TS would have allowed GPUN to take credit for the LOOP portion of the testing, based on the satisfactory performance in June 1997. The inspector considered this issue Unresolved pending further review by the NRC staff concerning suitability of this testing to meet TS requirements. **(URI 50-289/97-09-05)**

c. Conclusion

The inspector identified a question concerning the appropriateness of testing an EDG in simulated LOOP and LOCA conditions with the output breaker in the pull-to-lock position. This issue was considered an Unresolved Item pending further NRC staff review.

## M8 Miscellaneous Maintenance issues

### M8.1 Closed - VIO 50-289/96-07-01: Safety Related Scaffolding

a. Scope (92902)

The inspectors reviewed the corrective actions carried cut because of the previousl, identified scaffold violation.

b. Observations/Findings

GPUN responded to the Notice of Violation (NOV), 50-289/96-07-01, in a letter dated December 24, 1996, which provided background information regarding the scaffold construction problems and failure to follow procedure 1440-Y-3, "Scaffold Construction/Inspection and Use of Extension Ladders." The root causes, as determined by the licensee, included the failure to use standards policies, and administrative controls; lack of attention to detail by the maintenance; and in one case the operations personnel inspecting and using the scaffolds. Short term corrective actions had been previously reviewed by the inspectors and were adequate to correct and prevent recurrence of similar problems. The long term corrective actions and quality of scaffold construction in safety related areas were reviewed during the 12R refueling outage.

The inspectors observed the installation, inspection, and approval of scaffold constructed in the safety related areas of the plant including the RB. The scaffold construction and use was controlled by the procedure with no problems noted. The long term corrective actions included incorporating the lessons learned from the event in the operators' training classes, scheduling personnel for self-checking, effective observation and coaching techniques training, and management support and leadership to foster an environment that encourages attention to detail in this area. The inspectors reviewed the application of the longer term corrective actions and found that they were adequate to prevent similar events. This item is closed.

c. Conclusions

The licensee's corrective actions were appropriate and timely to prevent recurrence of violations regarding scaffold construction in safety related areas of the plant.

### M8.2 Inservice Inspection (73753, 73755)

a. Scope

The inspector reviewed portions of the ISI program and related nondestructive examination (NDE) activities that were according to the ASME Code Section XI 1986 Edition and required by 10 CFR 50.55 a(g). Specific areas inspected included:

qualification and certifications of the NDE contractors
observation of NDE activities

effectiveness of GPUN's controls over ISI NDE contractors
review of approved ISI NDE procedures and examination data
review of FAC monitoring program

b. Observations/Findings

GPUN utilizes NDE contractors to do ISI NDE examinations. The contractors must successfully complete a proficiency examination in the specific NDE method, before that NDE method can be applied to examination of plant components. Proficiency examinations are administered by a TMI NDE Level III. The proficiency examination includes demonstration of understanding the TMI NDE procedures and performance of practical examinations. The inspector verified NDE ISI contractor qualification and certifications were following the ASME Code and the TMI procedures.

The inspector observed NDE contractors performing ultrasonic testing (UT) of reactor coolant pump main flange bolts and emergency feedwater header to flange welds, using TMI procedures NDE-UT-09 and NDE-UT-02 respectively. These activities were well planned and performed according to procedure. The inspector also observed TMI's NDE Level III involvement in the examination being performed on the reactor coolant pump main flange bolts. TMI's NDE Level III audits the NDE contractors during examination of plant components to assure procedures are being followed.

The inspector reviewed the approved ultrasonic testing (UT) procedures and the completed examination data packages. These examinations were performed according to procedures, and the data packages were complete and properly documented.

The inspector reviewed the TMI technical document report (TDR) No.1065 Revision 2, which provides details and component evaluation results of the FAC program inspections completed at TMI since 1983. This TDR also presents FAC theory, component summaries, and program development. GPUN is currently using CHECWORKS, the latest released computer program from Electric Power Research Institute (EPRI), as a functional database for FAC information and to aid in component wear rate estimations. Data from CHECWORKS can be directly transferred to Microsoft Excel. This reduces the risk of human error in data transfer. TMI was scheduled to examine 110 components during the 12R outage. Of the 110 components, 40% are being reinspected from previous outages. Thirty-two feedwater risers were inspected with loss of pipewall thickness found on one. TMI's evaluation criteria for scheduling reinspection was based on the estimated safe operating life and the disposition of degraded components.

c. <u>Conclusion</u>

The ISI NDE program at TMI is well planned and organized. Observed ISI examinations were performed according to procedure. The random audits by the NDE Level III provide assurance that the NDE examinations are being done according to procedure. The examination data sheets were complete and properly documented. The FAC program was found thorough, effective, and capable of predicting the depletion of piping wall thickness.

M8.3 <u>Once Through Steam Generator Tube Eddy Current Testing and Related Work (73753, 73755)</u>

a. <u>Scope</u>

The purpose of this inspection was to review and observe the implementation of the licensee's eddy current testing (ECT) program during Outage 12R for the TMI-1 OTSG tubes, plugs and sleeves. The inspection covered the program implementation to verify the outage testing met TS requirements and ASME Section XI Division 1, Rules for ISI of Nuclear Power Plant Components.

During this inspection period (12R Outage) the tube indication acceptance criteria has been revised to provide further definition between tube "imperfections" and a "degraded tube". This new acceptance criteria has been added to the TS to enable the classification of tube indications which are outside the current criteria for "degraded". Those indications which do not meet the "degraded" criteria (indications are smaller then the minimum threshold) will be further evaluated to a new criteria defined as "imperfection." These additional criteria will enable the classification of tubes with inside diameter intergranular attack (IGA) that cannot be through wall depth sized. This new criterion provides a category for patch-like IGA indications and not those which may be indicative of a crack.

Discussions regarding the work were held with supervisory individuals responsible for these activities and with the individuals doing the inspection. Observations of work in progress were made by the inspector.

b. <u>Observations/Findings</u>

The TMI-1 RCS includes two OTSGs identified as 'A' and 'B'. Each OTSG contains 15,531 Inconel 600 tubes, fifty-six feet long with an outside diameter of 0.625" and a nominal wall thickness of 0.037".

The licensee planned to inspect 100% of the tubes in both OTSG 'A' and 'B' during this outage (12R). This level of inspection for this period, exceeds the requirements of TS 4.13, OTSG Tube ISI. The TS provides the requirements for tube inspection frequency at TMI-1. The licensee also planned to remove and replace existing tube plugs made from Inconel 600 with plugs made from Inconel 690.

The licensee had constructed on site, a full scale "mock-up" of the OTSG for training of personnel in the installation of the actual ECT test equipment. The placement of this equipment is a crucial element in enabling the installation and operating personnel to conclusively assure tube identification during the examination process. The inspector observed the completed positioning of this test equipment in both the lower and upper heads of the OTSG "mock-up."

The inspector observed that initial examination of the tubes was being performed from the inside diameter of the tube using the bobbin coil probe. No examinations were being performed from or on the outside surfaces of the tubes. Tubes identified with indications that cannot be characterized using the bobbin coil probe, those with cracklike indications below 40% throughwall, and indications 40% (or greater) of through wall thickness are then examined in the area of interest using a motorized rotating pancake probe (MRPC).

The inspector noted that the licensee had contracted with an outside vendor (Framatome Technologies Incorporated, FTI) to do this examination. The examination was performed using FTI Procedure 54-ISI-400-05, Revision July 7, 1997. Multi-Frequency Eddy Current Examination of Tubing. Data collection and Analysis was conducted following GPUN Procedure NDE-ECT-03, Revision 1, Change Number 2, Analysis of OTSG Eddy Current Data. The data acquisition, data management and resolution analysis were conducted on site at the data acquisition center (DAC). At this location, operators verified tube identification and, using the remotely operated manipulator, inserted the probe and monitored the probe position and function for each individual tube. The data was evaluated by the technicians as it was accumulated with attention to any indications of poor probe performance, signs of deterioration or excessive probe wear. When indications of such malfunctions were noted by the technician, the probe was replaced and previous affected tube examinations were repeated

The data acquired is transmitted off site for analysis at two separate analysis "stations" by a primary and a secondary analyst. Resolution of all interpretation discrepancies from the primary and secondary analysts is performed on site in the DAC by the Resolution Analyst. Provision is made for feedback to these analysts of all changes to their "calls." The original examination results are retained at the DAC to assure these data can be retrieved as recorded.

The inspector observed the data acquisition, data management and analysis resolution activities and found them to be following the above procedures, ASME Section XI and the TS. The inspector performed a verification that personnel doing the inspections and the resolution analyst had been qualified for these activities and, their qualifications had been reviewed and approved by the licensee's NDE Specialist.

At the time of this inspection during the week of September 22 through 26, 1997, the inspection of the tubes in both OTSGs was in progress. The inspection using the bobbin coil probe was essentially complete and the examination of "special interest" tubes had begun. The "special interest" tubes are those where indications have been identified which require further examination for accurate characterization. Also, extraction of the previously installed Inconel 600 tube plugs was underway. This replacement activity with Inconel 690 material, intended to prevent tube leakage, had not commenced during the inspection.

| INSPECTION STATUS AT INSPECTION CONCLUSION | | | | |
|---|---|---|---|---|
| INSPECTIONS | OTSG "A" | OTSG "B" | OTSG "B" | |
| | Scheduled | Completed | Scheduled | Completed |
| Complete Tube (0.510 dia, probe) | 14028 | 14028 | 15114 | 15114 |
| (0.540 dia, probe) | 235 | 235 | 39 | 39 |
| (0.540 HF-Expanded Scope) | 74 | 74 | 51 | 51 |
| 17 inch Kinetic Exp | 3483 | 3483 | 3470 | 3470 |
| 22 inch Kinetic Exp | 238 | 238 | 232 | 23 |
| Final Resolution of Indications (1 and 2 above) | 14028 | 14022 | 15114 | 15107 |

Sleeve, Sleeve Plus Point, Lane and Wedge, I- 690 Plugs, Westinghouse Plugs and Rerolls were 100% complete at the conclusion of the inspection. Plug extraction, retests and analysis resolution were still underway at the conclusion of the inspection.

c. <u>Conclusion</u>

The eddy current inspection program for the OTSG tubes at TMI-1 was found to be well planned and organized, and capable of determining the integrity of the steam generator tubes. The inspection met ASME Code, Section XI and TS requirements.

## III. Engineering

**E1 Conduct of Engineering (37550, 37551, 92903, 93809)**

E1.1 Core Flood System Review Introduction and Purpose

a. Scope

The CF system is part of the emergency core cooling system (ECCS) and provides a rapid injection of borated water into the reactor vessel, for core cooling and reactivity control during a large break LOCA. The inspection evaluated the CF design basis and how it complies with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling System for Light Water Nuclear Power Reactors."

E1.2 Evaluation of the Design Basis

a. Scope

The inspectors evaluated aspects of the CF DBD to verify compliance with 10 CFR 50.46. The inspectors reviewed applicable sections of TS, sections of the UFSAR, operating procedures, emergency operating procedures, test procedures, and codes and standards. In addition the inspectors evaluated if changes made to the system were appropriately reflected in the DBD. The current DBD TI-213-0, was dated June 3, 1996; 10 CFR 50.46 "Acceptance Criteria for Emergency Core Cooling System for Light Water Nuclear Power Reactors;" The Safety Evaluation Report (SER) by the Atomic Energy Commission (AEC), dated July 11, 1973; and the UFSAR.

b. Observations/Findings

E1.2.1 Licensing and Regulatory Requirements

The inspectors verified that the above CF documents agreed, and that the UFSAR and the DBD have been correctly maintained to reflect the design criteria listed in 10 CFR 50.46. The DBD was maintained in a computer-based system and updated continuously as changes were made to the CF.

The DBD was in hard copy form and contained references collected, copied and included within the document. The inspectors verified the validity of the references of the subject matter to confirm compliance with the design bases for the area of interest. For example, the inspector reviewed fourteen references, including calculations, publications, bulletins, generic letters, codes and standards, licensing documents, etc. to assess core flood tank (CFT) liquid volume (discussed in Section E1.4.2 of this report). Other similar reviews, by the inspectors, showed the DBD to be "easy to use" and thorough in describing the design basis.

The inspectors reviewed sections 3.1.6.1, 3.1.6.10, 3.3.1 and 3.3.2 of the TS and compared them with the DBD. The inspectors verified that the TS bases for the sections reviewed were discussed and represented in the DBD, showing how the design satisfies the TS basis. Both TS and the DBD related the information necessary for the inspectors to confirm that they satisfy the five design criteria set forth in 10 CFR 50.46. The five criteria stated that (1) the peak cladding temperature of 2200°F, (2) the maximum cladding oxidation of 0.17 times the cladding thickness, (3) the maximum generation of hydrogen of 0.01 times the hypothetical amount that is possible, (4) the coolable geometry, and (5) the long-term cooling will not be exceeded.

E1.2.2 Interface With Chemistry and Sampling

TS 3.3.1.2 requires that the boron concentration for the CFTs shall not be less than 2270 ppm boron. This chemistry requirement would maintain the boron concentration higher than that of the RCS. The higher concentration ensures that the injected water, in the event it is needed for a large break loss of coolant accident (LBLOCA), will not dilute the borated water present in the RCS. The inspector reviewed the chemistry sample results for the past two years and determined that the results exceeded the TS minimum requirement.

The DBD discussed the interfaces with the sampling system. The inspector noted that a design change for the sample system was recently completed to add relief valves to the sampling system to satisfy the requirements of Generic Letter 96-06. Refer to Section E1.4.3 of this report for details.

E1.2.3 Mechanical Maintenance

The inspector reviewed the corrective maintenance performed on the CF for the past five years to determine if any patterns of problems were present and to determine if longstanding problems existed. The record showed that the corrective maintenance on the system was not extensive, there was no backlog, and repetitive problems did not exist.

E1.2.4 Electrical Distribution System

The inspectors reviewed applicable portions of IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and compared those requirements to the design basis requirements in the DBD. No discrepancies were found.

The inspectors verified that operating procedure 1104-1 has appropriate procedural steps in place to energize and open valves CF-V-1A&B when reactor coolant pressure is >650 psig but less than 700 psig. The inspectors also verified that the procedure has appropriate steps to ensure that, when reactor coolant pressure reaches 2155 psig, but prior to criticality, the breakers are opened and tagged to prevent inadvertent closure of the isolation valves. Circuit breakers "1A Rad Waste Control Center (RWCC) Unit 6C" (CF-V-3A) and "1B RWCC Unit 6B" (CF-V-3B) were verified to be open and tagged by the inspector to preclude inadvertent valve movement. The inspectors noted that these actions are consistent with the DBD. The inspectors verified that during plant heatup, appropriate steps were in place to operate the breakers when necessary. The inspectors also verified that during plant cooldown appropriate steps were in place to close the circuit breakers and close CF-V-1A&B prior to depressurizing below 700 psig in the primary coolant system. The inspector noted that these actions are also consistent with the DBD.

E1.2.5 Instrumentation and Control

TS Table 4.1.1, Item 25, requires that a calibration of the CF pressure and level instrumentation be performed during each refueling outage to show proper operation of the system. The licensee extended the operation cycle from 18 months to 24 months. The inspectors reviewed calculation number C1101-213-5350-011, Revision 0, "CF Tank Pressure and level Channels 24 Month Drift Calculation." The inspectors verified that the calculational assumptions were technically reasonable, and that the calculation provided adequate justification to extend the calibration cycle to 30 months.

The inspectors verified that procedure number 1302-5.15, revision 21, "CFT Pressure and Level Channel," was properly upgraded.

The inspector also verified that the design change was accurately described in the UFSAR.

E1.2.6 Environmental Qualification

The inspector reviewed the following documents:

- GPUN Technical Data Report, TDR 598, "Methodology and List of Equipment Components Requiring Radiation Qualification for Small Break Loss of Coolant Accident Mitigation," revision 1

- GPUN Technical Data Report, TDR 648, "Methodology and List of Electrical Components Requiring Environmental Qualification for Large Break LOCA and High Energy Line Break (HELB) Mitigation," revision 0

- TMI Document Number 9901429, revision 14, "TMI-1 Environmental Qualification (EQ) Master List"

- GMS-2 Environmental Qualification, System Component Evaluation Worksheets (SCEW) sheets T1-213-001 thru 012

The inspectors noted that the above documents provided adequate justification about why specific CF components are qualified or exempt from radiation and high energy line break (HELB) environments. The inspectors verified the containment isolation valves located in the RB and the AB require radiation qualification because of the harsh environment during a small break or large break LOCA. Valves CF-V-

2A&B also require HELB classification because of their location in the RB. The remaining isolation valves (CF-V-19A&B, CF-V-20A&B) do not require HELB classification because of their location outside the RB.

The inspector verified that the CFT isolation and vent valves (CF-V-1A&B, CF-V-3A&B) do not require radiation or HELB qualification. The inspectors noted that these valves would be in a harsh environment after they complete their accident function. The inspectors verified that EQ valves motor control center (MCC) breaker cabinets for valves CF-V-1&2 do not require radiation or HELB qualification. These cabinets are located in areas which have a mild radiological environment during normal operating conditions and/or postulated small break LOCAs. The inspectors found that there was appropriate classification between the DBD and reviewed EQ documentation.

E1.2.7 Core Flood Tank Heaters

The CFT heating system was originally installed to maintain the tanks above the nil ductility transition temperature (NDTT) (85°F plus a safety factor) to reduce the potential for brittle fraction of the tank walls. B&W, however, recommended the use of a lower hydro test temperature (70°F). In 1995, GPUN reevaluated their selection of the NDTT + 30°F CFT temperature and reduced the CFT temperature limit to the hydro test temperature.

The inspectors reviewed the "CFT Temperature Limitations" Safety Evaluation, SE-00023-005. The inspectors located original correspondence regarding the selection of the CFT temperature limit. The inspectors also reviewed an independent assessment of SE-00023-005 performed by B&W, which concurred with the technical basis of the reduced CFT temperature limit.

Through the review of the above, the inspectors determined that containment temperature does not go below 75-80°F. The inspectors determined that the heaters have not been used to maintain the CFT temperature greater than 70°F. The operators provided records documenting the monitoring of the CFT tank temperature and the system engineer provided the calibration and verification schedule for the CFT temperature instrumentation. Even though the heaters are not used on a regular basis, the inspectors determined that operating procedures require the CFT heating systems to be available in standby.

E1.2.8 Inservice Testing

Information Notice 89-67, "Loss of Residual Heat Removal Caused by Accumulator Nitrogen Injection," alerted licensees to potential problems resulting associated with the injection of nitrogen from the accumulator into the RCS during shutdown conditions. In their 1989 response to this Information Notice, GPUN provided a rationale for determining that Information Notice 89-67 was not applicable, based on performing Surveillance Procedure 13003-11.21.

Calculation C1101-213-E270-014, Revision 0, "Core Flood Check Valve Test Analysis," determined a maximum final CFT pressure to avoid leakage into the RCS, based on the water height in the fuel transfer canal and the height of the CFT connection. The premise of this determination was that if the water level in the CFT remained above the height of the CFT connection, the nitrogen gas would not be able to expand to the CFT connection and be injected into the RCS.

Surveillance Procedure 1303-11.58, "Core Flood System Test for IST," was completed on September 30, 1997, and observed by the resident inspectors. Nitrogen from the CFT was not injected into the RCS.

E1.2.9 Testing

The inspectors verified that the CFT valves listed below were in the Inservice Testing Program. TMI-1 Administrative Procedure 1041 "IST Program Requirements" specified the valves that are within the scope of the IST program:

- CF-V1A/B - CFT isolation valves
- CF-V 2A/B - CF containment isolation valves
- CF-V-4A/B - Check valves outside secondary shield
- CF-V-5A/B - Check valves inside secondary shield
- CF-V-12A/B - Check valves downstream of CF-V-19A/B
- CF-V-19A/B - Makeup and nitrogen supply valves
- CF-V-20A/B - Sample and drain valves

The CFT relief valves were recently added to the IST program. The inspectors reviewed IST results from Outage 12R and verified that testing has been performed for the above valves. The inspector verified that the recently added relief valves were also tested.

c. Conclusion

The inspectors concluded that the documents, listed above, were correctly maintained and display the means to maintain the design basis of CF according to the five design criteria in 10 CFR 50.46. The inspector also concluded the DBD, for the CF, was "easy to use" and thorough in describing the design basis.

The boron concentration of the CFTs was being maintained and sampled following TS requirements.

The relatively low amount of maintenance history for the system, plus the results of the physical walkdown (see Section E1.3 of this report) led the inspector to conclude that the CF was well maintained.

Documents were consistent with the applicable sections of the DBD, UFSAR, TS, IEEE standards, procedures, system drawings and system layout. Adequate procedures were in place to operate the CF within its design basis.

The design change and calculation reviewed were adequate to support system operability extension to 30 months.

EQ documentation was consistent with the DBD, and that justification for CF component evaluation were adequate.

The CFT heaters have not been used since implementing a 1995 Safety Evaluation which decreased a CFT temperature limitation. The use of the decreased temperature limitation was consistent with industry practices and the recommendation of the Nuclear Steam System Supplier, B&W. The inspectors verified that document and procedural changes accurately reflect the new CFT temperature limitation.

GPUN adequately considered nitrogen injection in planning and performance of Surveillance Procedure 1303-11.58.

The inspectors concluded that the CF system valves that require testing by the ASME code were in the IST program and that testing was performed as scheduled.

E1.3 System Walkdown

a. Scope

The inspectors performed a walkdown of the CF to evaluate the material condition of the system and its consistency with the P&ID and electrical system drawings.

b. Observations/Findings

The material condition of the CF was good. The inspectors paid particular attention to CFTs, isolation valves, vent valves, various interface piping, valve flanges and caps, level and pressure transmitters, instrumentation and electrical system wiring interfaces, and piping insulation. The inspectors found no discrepancies between the P&ID (drawing number 302-711, revision 25, "Core Flooding Flow Diagram," the actual system layout, and the DBD.

c. Conclusions

The material condition of the CF was good, and consistent with the P&ID, the actual system layout and the DBD.

E1.4 Related Design Changes

The inspectors reviewed design changes: (1) "Replacement of the CF Valves 1A and 1B Motors," (2) "CFT Water Volume Change," (3) "Relief Valve Addition to the Sampling System," and (4) "CF transmitter Replacement" to determine if the system design control and licensing input information met the design basis.

E1.4.1 Replacement of the Valves Motors

a. Scope

Isolation valves CF-V-1A&B were intended to isolate the CF from the reactor vessel during shutdown conditions when the RCS is below 600 psig. The original motors were two pole high speed motors with a Dings break. The motors caused damage to the valves and were replaced with four pole motors without a break. The closing of the valves time increased from 10 to 20 seconds. The inspector evaluated the design change to determine compliance with changes to the facility and to review compliance with the design basis.

b. Observations/Findings

The inspectors reviewed design change T1-MM-418668-003 developed to change the motors on CF-V-1A&B. The safety evaluation and 10 CFR 50.59 assessed the motor's increased time and the lack of a break. The increased time was not a factor for the intended operation of the system, because the valves are opened and locked prior to the need for the system. The slower movement of the valve in closing was more controlled and no break was required. The safety evaluation properly concluded that the change to the CF did not affect the response time of the CF to carry out the safety function as designed. The breakers for the motors were also changed to satisfy the new motors. The inspectors reviewed the materials list and the purchase order and verified that the breakers and the motors installed were as ordered and received by the quality assurance inspection. The inspector verified that the valves were in the valve testing program and received testing following the ASME code. The inspector also verified that the UFSAR was changed to reflect the new motor's speed.

c. Conclusions

The inspectors concluded that since the CF is a self-contained, self-actuated and passive system, the changing of the valves and breakers would not affect the operation of the system during a LBLOCA.

E1.4.2 Volume Requirement Change

a. Scope

The CFT volume of borated water has changed since the original design. The inspectors evaluated the reasons for the volume change, and a newly proposed change implemented during 12R. The inspectors evaluated applicable calculations to determine if the design bases were satisfied. The design is intended to provide enough borated water to keep the core covered and provide core cooling.

b. Observations/Findings

The liquid inventory of the combined CFTs is based on the approximate volume of the reactor vessel downcomer and lower plenum. The original calculation showed the minimum level of each CFT (two installed) as 897 $ft^3$. The plant operated with CFT liquid volume of $940 \pm 45$ $ft^3$ until 1971, when ECCS rules were changed prompting new studies that resulted in increasing the volume of each CFT by 100 $ft^3$. The new volume was to compensate for water loss during blowdown, and to reduce nitrogen entering the RCS after the CFT pressure equalizes with RCS pressure, by reducing the nitrogen pressure when the tanks are emptied. In new studies performed by B&W, and in particular, engineering analysis 51-1244420-00 performed to power upgrade the reactor from 2568 MWt to 2772 MWt show a volume change was required.

These studies were performed using B&W's REFLOD3B code that simulates hydraulic behavior of the primary system during refill and reflood phases of a LBLOCA. The studies showed that the most conservative approach to the tank volume-versus pressure ratio was to reduce the volume to $940 \pm 30$ $ft^3$, and increase the nitrogen pressure to $600 \pm 25$ psig. This change will enable the water to be injected quicker. The volume was actually in the same range as the original calculation for water volume for the CFT.

The above was submitted, by the licensee, to the NRC for a TS Change. The inspectors reviewed the NRC's safety evaluation and approval to operate with the volume discussed above. The inspector evaluated the calculation for setting the level to correspond with the proper volume. The inspector verified tank volumes and alarm setpoints were changed according to the calculation, and that the appropriate changes were made to the TS.

c. Conclusions

The original CFT volume design basis was not compromised, and currently meets the latest studies conducted by B&W and accepted by the NRC. The inspector also concluded that the latest volume to pressure ratio of the CFT supports the licensees proposed power upgrade.

E1.4.3 Relief Valve Addition to the Sample System

a. Scope

GL-96-06 requested the evaluation of fluid piping systems that penetrate containment for susceptibility to thermal expansion which could cause pipes to rupture. GPUN determined that the piping between the containment isolation valves CF-V-2A&B and CF-V-20A&B on the sample and drain line was susceptible to overpressurization during abnormal conditions and thermal relief valves should be added to the system. The inspectors reviewed this design change to determine if the added relief valves could relieve the overpressurization concerns expressed in GL 96-06. The inspector also reviewed the containment leak tightness of CF-V-46 A&B as required by 10 CFR, Appendix J.

b. Observations/Findings

Relief valves, CF-V-46A&B, were installed to the piping between CF-V-2A&B and CF-V-20A&B to relieve excess pressure during abnormal conditions. The inspectors reviewed the results of the pressure integrity, lifting pressure, and re-seal pressure tests. The tests indicated that the valves lifted at the design pressure (~1500 psi) and re-sealed at the appropriate pressure (approximately 84% of the lift pressure), and that the joints maintained the pressure boundary (no visible signs of leakage).

The inspectors also reviewed the results of TMI-1 Surveillance Procedure 1303-11.18, "RB Local Leak Rate Testing." The purpose of the RB local leak rate tests was to determine the leak tightness of several valves, including CF-V-46A&B. The results showed that the tested local leak rates for CF-V-46A&B was below the calculated target leak rate criteria as defined and calculated in TMI-1 SP 1303-11.18.

c. Conclusions

Based upon the review performed, the inspectors concluded the testing performed on CF-V-46A&B provided confidence that the design modification could relieve excess pressure during abnormal conditions and maintain the RB pressure boundary.

E1.4.4 Transmitter Replacement

a. Scope

GPUN replaced the CFT level transmitters with Rosemount transmitters because of the reliability and maintainability of the originally installed Bailey transmitters. The inspectors evaluated the design change for compliance with GPUN's procedures. The inspectors also reviewed licensee documentation to insure that the new transmitters met NRC Bulletin 80-16, "Potential Misapplication of Rosemount Transmitters."

b. Observations/Findings

The inspectors reviewed design change package (DCP) 1165, 9/6/78, "Replacement of Bailey "BY" Transmitters on the CFTs." This DCP installed Rosemount transmitters in place of Bailey transmitters to monitor CFT level. The inspectors verified that the DCP contained sufficient documentation to permit evaluation of the effect of this change on the design and licensing basis. The package documented the necessary procedure and drawing changes, adequate installation instructions, and appropriate retest instructions. The inspectors verified, during the walkdown of the CF, that the Rosemount transmitters CF2-LT1\2\3\4 (Part Number

1151DP5E22P5) were installed as required by the DCP. The inspectors also reviewed maintenance records for the CF (see Section E1.2.3 of this report) and determine that no maintenance was required for the transmitters since their installation.

The inspectors reviewed Bulletin 80-16, "Potential Misapplication of Rosemount Transmitters," and the licensee's response, to verify that the Rosemount transmitters with either 'A' or 'D' output codes were not used in the CF. The Rosemount transmitters with either 'A' or 'D' output codes would provide ambiguous signals when exposed to excessive over or reverse pressure conditions. The inspectors verified that the Rosemount transmitters installed and those in the warehouse use 'E' codes and are not effected by the over or reverse pressure conditions.

c. <u>Conclusions</u>

Based on the system waikdown and the review of calibrations and maintenance records, the inspectors concluded that the replacement of the transmitters on the CFTs increased the reliability and maintainability of the level instrumentation. The inspectors also concluded that the new transmitters were not in conflict with Bulletin 80-16.

E1.5 <u>Review of NRC Bulletins, Information Notices, Generic Letters</u>

a. <u>Scope</u>

The inspectors conducted a search of NRC Bulletins, Information Notices, Generic Letters from 1979 to present. The search found that several were applicable to the CF. The inspectors reviewed the licensee's actions regarding the information presented.

b. <u>Observations/Findings</u>

The following NRC documents were applicable to the CF:

- <u>Bulletin 80-16</u> contained concerns regarding rosemount transmitters. See Section E1.4.4 for inspector findings.

- <u>Information Notice 89-67</u>, "Loss of RHR Caused by Accumulator $N^2$ Injection" see Section E1.2.8 for inspector findings.

- <u>Information Notice 91-05</u>, "Innergranular Stress Corrosion Cracking in the PWR Safety Injection Accumulator Nozzles." The inspector reviewed GPUN memo "Review of Information Notice 91-05..." that disclosed the problem was found at a Westinghouse Plant in a 304 stainless steel nozzle. The TMI design was carbon steel clad with stainless with a different weld configuration. The document also showed that there was machining of the internal diameter weld surface which removed the stresses prior to cladding. The inspector concluded that the TMI design was different than the nozzle referenced in the Information Notice.

- Generic Letter (GL) 96-06. GL 96-06 requested the evaluation of fluid piping systems that penetrate containment for susceptibility to thermal expansion that could cause pipes to rupture. See Section E1.4.3 of the report for inspector findings.

c. Conclusions

The inspectors concluded that the licensee was properly acting upon information from the NRC to enhance the operation of the CF, or to assess problems applicable to the CF.

During the inspection no UFSAR concerns were identified.

**E8 Miscellaneous Engineering Issues**

E8.1 (Closed) Violation 97-07-01: Failure to Write a Clear Licensee Event Report Narrative:

GPUN failed to provide an adequate description of multiple over pressurizations of the make up system suction piping in LER 97-003. The inspectors verified that the licensee has taken corrective actions by issuing new guidance for the formulation of LERs. GPUN also hired a root cause analysis trained person to fill the position of "Organizational Effectiveness Coordinator." The position will oversee root cause analysis and the CAP. These two functions will be key to the development of the LERs. In addition the licensee resubmitted the LER that now correctly recounts the events in a more clear and concise manner.

E8.2 Modification Review - Letdown Valve Closing Capability Upgrade

a. Scope

The inspector reviewed the modification installed on the MU outboard letdown isolation valve (MU-V-3), to improve the isolation capability.

b. Conclusion

The modification installed an air to close function on this air operated valve; this increased the ability of the valve to close under system flow and differential pressure.

The modification package was detailed and well developed and post-modification testing appeared complete.

## IV. Plant Support

### R1 Radiological Protection and Chemistry (RP&C) Controls

#### R1.1 General Plant Tours (71750)

a. Scope

The inspector made routine tours of the RB, AB, and the intermediate building (IB) during the outage looking at material and radiological conditions, and plant housekeeping. The inspectors also toured the site at night to determine that ability to monitor the protected area boundary.

b. Conclusions

Material conditions continued to be good. Equipment needed to meet TS requirements for shutdown conditions was maintained and operated well.

The inspectors found radiological conditions adequate; however, there were instances noted where local postings did not agree with the area conditions and where material was allowed to cross contaminated area boundaries. In these cases the radiation protection staff corrected the conditions. The controls in place for highly contaminated and high radiation areas, such as the OSTGs appeared proper.

Generally the inspectors found that housekeeping degraded over the outage. This degradation was particularly evident in the RB, where outage related activities cause the large amounts of debris to be left on floors and surfaces The debris observed included nails and pieces of wood and sawdust from scaffolding activities, plastic tie-wraps from the installation of temporary hoses and cables, a large roll of sheet plastic, pop-rivet stems from sheet metal installations, and tape materials left following work.

The inspector noted no deficiencies during a night tour of the protected area.

#### R1.2 Radiological Controls-External and Internal Exposure

a. Scope (83750)

The inspector reviewed the licensee's control of external and internal exposure. Information was gathered through observation of activities, tours of the radiologically controlled area (RCA), discussions with cognizant personnel, and review and evaluation of procedures and documents.

b. Observations/Findings

Radiation work permits (RWPs) and controls in place in the field for various work evolutions were reviewed, observed, and discussed with the licensee staff. The activities involving incore instrumentation replacement, CRDM work, and OTSG tube eddy current testing were especially inspected. Appropriate proper personnel protective clothing and equipment, precautions and instructions were being prescribed for the work descriptions and radiological conditions at the work sites. A pre-job briefing for stuck incore cuttings work was detailed, thorough, and emphasized an understanding of the work sequence and individual responsibilities. The pre-job ALARA review and RWP for recent diving operations in the spent fuel pool were reviewed, and these documents plus the personnel doses received during these diving operations were discussed with cognizant licensee personnel. Diving operations had been well controlled, and lessons learned from recent operational experience at another site had been incorporated. Individual and cumulative dose were examined and showed that personnel external and internal dose controls were being effectively implemented.

c. Conclusions

Personnel external and internal dose controls were being effectively implemented.

R1.3 Radiological Controls-Radioactive Materials, Contamination, Surveys, and Monitoring

R1.3.1 General Outage Controls

a. Scope (83750)

The inspector reviewed the licensee's control of radioactive materials, contamination, surveys, and monitoring. Information was gathered through observation of activities, tours of the RCA, discussions with cognizant personnel, and review and evaluation of procedures and documents.

b. Observations/Findings

Water and gatorade were placed in the uncontaminated area outside the access point of the RB and were made available to workers exiting the RB to replace lost fluids, especially for workers who were experiencing heat stress symptoms. Adequate contamination controls were in place so that the workers could be provided the liquids without leaving the contaminated area and could return to the RB without doffing and donning protective clothing. High radiation boundaries, gates, and postings were properly established and maintained. The establishment and maintenance of contaminated areas exhibited some weak attributes. In some cases, the barricade rope/tape was not continuous around the contaminated area, leaving small openings, and in other cases, structures or permanent carts were incorporated as part of the boundary. These conditions made the exact perimeter of the contaminated areas unclear for the radiation workers. There were also several examples of equipment and hardware lying on the floor across the vertical plane of the contaminated area boundary which made it unclear whether the equipment/hardware was contaminated or not. Routine and job specific radiological survey records were reviewed and, in general, were found to contain appropriate information and to be accessible to the radiation workers. However, several weak attributes were noted in the survey program which detracted from the availability of accurate and current radiological condition information to the radiation workers. Several individual surveys posted in the AB were incomplete in that they did not accurately identify all radiologically posted areas and boundaries in the areas covered by the survey maps. Deficiencies with the surveys posted at the two main RCA access points were also noted. At the main HP access control point, several location labels on the board were incorrect, and a dated survey, superseded by a more recent one, had not been removed. At the Outage Equipment Storage Building control point, two dated surveys, superseded by a more recent one, had not been removed. Also, the survey boards at both control points were subject to foot traffic congestion.

c. Conclusions

Adequate contamination controls and an adequate radiation survey and monitoring program were being implemented. However, weak attributes were noted in the establishment and maintenance of contaminated areas and in the survey program.

R1.3.2 Review of Hot Particle Contamination - Open - EEI 50-289/97-09-05: Personnel Hot Particle Contamination Due to Inadequate Surveys

a. Scope

The inspector reviewed the licensee's control of work which resulted in a hot particle exposure to a worker and the licensee's dose assessment. Information was gathered through discussions with cognizant personnel and through review and evaluation of procedures and documents.

b. Observations/Findings

On October 4, 1997, the task of raising (parking) the seal plate (a reactor vessel head activity) and removing gasket material was performed under RWP No. 141032. Based on prior surveys after decontamination, the work area was not posted or controlled as a hot particle area. On the initial entry into the work area, the workers wore a single set of coveralls, double rubber gloves, double rubber boots, wet suit bottoms, hood, hard hat, and safety glasses. After the seal plate was parked, the radiation control technician reportedly halted work to perform a survey of the newly exposed surfaces. The survey results and visual examination of smears and sticky wipes indicated that the newly exposed work area was a hot particle area meeting the criteria for Level II controls (presence of discrete particles with an activity of 50,000 net counts per minute or greater). However, the licensee's procedures stated that the preferential method for dealing with emergent hot particle areas was to eliminate hot particles and sources of hot particles from the area so that work could resume without the need for hot particle controls.

In this instance, since decontamination tools were available in the immediate area and since gasket removal was estimated to last less than one hour, the radiation control technician decided to proceed without implementing any additional protective clothing requirements or other hot particle controls and without notifying his supervision. The radiation control technician and the supervisor of the work crew, under the technician's direction, decontaminated the newly exposed work area to eliminate hot particles and to allow gasket removal work to commence in the decontaminated areas. The radiation workers in the work crew stayed back from the seal plate area during this hot particle decontamination effort and were not involved. After completion of the hot particle decontamination effort, the radiation workers in the work crew proceeded to perform the gasket removal task. At the end of this work evolution, the radiation workers were required by the technician to frisk at the exit of the RB. At the frisking location, one of the radiation workers was discovered to have two separate hot particles (16.1 and 1.2 microcuries) (each particle was approximately equal parts Zirconium-95 and Niobium-95) on the skin of his lower face. The licensee's final dose assessment for the 16.1 microcurie particle was 13.978 rem, total dose to the skin (12.825 rem, beta dose and 1.153 rem, gamma dose) and 50 millirem, total dose to the whole body. Upon discovery of the hot particle contaminations, the seal plate area was posted as a hot particle area.

A review of the licensee's hot particle control procedure on pages E8-1 through E8-3 of Procedure 6610-ADM-4110.04 indicated several inconsistencies with 10 CFR 20.1501. First, specific controls were not required during the elimination of hot particles and sources of hot particles from the area in an emergent situation so that work could resume. Second, it was not clear if Level II controls included Level I controls (Level I area: presence of discrete particles with an activity of 5,000 net counts per minute or greater; Level II area: presence of discrete particles with an activity of 50,000 net counts per minute or greater) or if Level II controls stood alone; for example, the requirement that all items removed from the area were to be labeled or marked as originating from a hot particle area appeared in both the Level I and II controls, but consideration of face shields and a requirement that all personnel were to utilize a whole body frisker after exiting the hot particle area only appeared in the Level I controls. Third, most of the listed controls were only recommendations, not requirements. For example, of the seven controls identified for Level I areas, five were required only to be considered while two were required to be implemented (i.e., frisking as soon as possible after exiting and labeling all items removed from the area as originating from a hot particle area); of the seven controls identified for Level II areas, only two were required to be implemented (i.e., a cognizant on-shift GRCS needed to be aware of all work in progress and labeling all items removed from the area as originating from a hot particle area). Fourth, there was little guidance on varying hot particle radioactivity (and, thus, potential magnitude of skin dose) versus required frisking intervals. The licensee's root cause analysis stated that the hot particle control procedure would be evaluated for lack of specificity and revised accordingly.

TS 6.11, Radiation Protection Program, requires that procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure. 10 CFR 20.1501 requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR 20 and are reasonable under the circumstances to evaluate the extent of radiation levels, and concentrations or quantities of radioactive material, and the potential radiological hazards that could be present. The licensee's hot particle control procedure on pages E8-1 through E8-3 of Procedure 6610-ADM-4110.04 was inconsistent with 10 CFR 20.1501 in that it did not cause surveys to be made reasonable under the circumstances to evaluate the extent of quantities of radioactive material and the potential radiological hazards that could be present and resulted in a radiation worker receiving a skin exposure of 13.978 rem.

c. Conclusion

GPUN failed to perform adequate surveys during the removal of the reactor vessel seal plates, as such adequate hot particle controls were not in place and did not prevent a personnel skin contamination. This appears to be a violation of TS 6.11. **(EEI 50-289/97-09-05)**

R1.4 Radiological Controls-As Low As Reasonably Achievable

a. Scope (83750)

The inspector reviewed the licensee's pre-job ALARA reviews, use of temporary shielding, and radiological goals, projections, and results.

b. Observations Findings

Numerous pre-job ALARA review packages had been generated, and the ones for incore instrumentation replacement, CRDM thermal barrier replacement, OTSG activities (tube plugging /repairs, tube eddy current testing, cold leg dams, and Roger robot work), replacement of the 'B' DH pump, and spent fuel pool diving operations for cable modification and maintenance of fuel transfer carriages were evaluated and found to be detailed and thorough. The current outage cumulative total effective dose equivalent was tracking close to the goal projection. Cumulative committed effective dose equivalent for the outage was low (less than 50 mrem). Personnel contamination goals were established, and personnel contaminations were being tracked and evaluated for cause. The amount of temporary shielding used as compared to last outage had been increased by almost a factor of two.

c. Conclusions

ALARA activities, especially the pre-job reviews, were generally considered a strong point of the radiological control program.

R1.5 Other Changes to the RP Program

a. Scope (83750)

The inspector reviewed the effect of the elimination of the Corporate Radiological Health/Safety Director position on the overall performance of both (Oyster Creek and Three Mile Island-1) radiation protection programs. Information was gathered through discussions with cognizant personnel and document review.

b. Observations/Findings

During the radiological control portion of NRC Inspection No. 50-289/97-06, it was noted that the TMI Radiological Controls/Occupational Safety (RC/OS) Director had reported to the Corporate Radiological Health/Safety Director up to approximately June 1997. This had changed because the Corporate Radiological Health/Safety Director position had been eliminated. The TMI RC/OS Director now reported to the Oyster Creek RC/OS Director. The effect of this change on the overall performance of both (Oyster Creek and TMI-1) radiation protection programs was uncertain and required further NRC review; of particular concern was the disposition of responsibilities and authorities previously maintained by the corporate director relative to review and maintenance of the GPUN Radiation Protection Plan and to the required annual review of Radiation Protection Program content and implementation for each site; during a telephone discussion after that inspection, the Oyster Creek RC/OS Director informed the inspector that those responsibilities and authorities previously maintained by the corporate director would be performed by the Oyster Creek RC/OS Director; this issue was documented as an item to be reviewed during a subsequent inspection (IFI 50-289/97-06-02). During the current inspection, the TMI RC/OS Director stated that a Safety Determination and 50.59 Review was in progress to address this issue. A review of the in-progress Safety Determination and 50.59 Review indicated that needed changes to the Oyster Creek and TMI-1 UFSAR had been identified.

c. Conclusions

The change in the corporate organization, involving the elimination of the Corporate Radiological Health/Safety Director position, vesting the responsibilities and authorities previously maintained by the corporate director in the Oyster Creek RC/OS Director, and having the TMI RC/OS Director reporting to the Oyster Creek RC/OS Director was being evaluated by the licensee. The effect on the performance of the site radiation protection programs is still uncertain and will require further NRC review.

**R4 Staff Knowledge and Performance in RP&C**

R4.1 Open - EEI 50-289/97-09-06; Inadequate Control Over Once Through Steam Generator Locked High Radiation Area

a. Background/Scope

The inspectors reviewed the licensee identified failure to maintain positive control of the 'B' OTSG locked high radiation area. The review included the radiological controls procedure 6610-ADM-4110.06, "Control of Locked High Radiation Areas," plant TS, CAP form T1997-0738, and PRG meeting minutes for prior high radiation control issues.

b. Observations/Findings

On September 30, 1997, a radiation control technician, at a remote video monitor, noticed that the 'B' OTSG upper manway shield door was unlocked and unattended. A contract worker failed to maintain positive control for the 'B' OTSG upper manway shield door. With the high radiation door unlocked, the contractor left the area unattended and exited the RB in violation of the administrative procedure for the control of locked high radiation areas. The opening was monitored continuously from a remote video monitor display. Upon recognition of the missing worker a radiological technician was sent to the 'B' OTSG to lock the manway shield door. The high radiation barrier was left unattended for approximately 60 to 90 minutes. The radiation levels in the 'B' OTSG were approximately 3 to 4 Rem/hour. This is a repetitive issue of a similar problem that occurred in the 1993 and 1995 refuel outages.

TS 6.8.1 states, in part, that written procedures shall be established, implemented, and maintained covering certain activities, including the applicable procedures recommended in Appendix 'A' of Regulatory Guide 1.33, Revision 2, February 1978, which includes radiation protection procedures for the control of access to radiation areas. The licensee's Radiation Protection Procedure Number 6610-ADM-4110.06, "Control of Locked High Radiation Areas," states in part that, "prior to a OTSG platform worker leaving the area, they must turn over the locked high radiation controls to an on-coming platform worker or have the OTSG shield door verified locked by a radiological control technician." The contract worker failed to follow the high radiation control procedure; the action led to an unlocked and unattended high radiation area, the 'B' OTSG shield door, with the potential for an inadvertent radiation exposure in excess of personnel limits. This is an apparent violation.

c. Conclusions

A contract worker failed to follow the high radiation control procedure, the action led to an unlocked high radiation area, the 'B' OTSG shield door, with the potential for an inadvertent radiation exposure in excess of personnel limits. This failure was similar to a prior problem that occurred in the 1993 and 1995 refuel outages. This issue appeared to be a violation of TS 6.8.1, in that procedures for locking high radiation areas were not followed. **(EEI 50-289/97-09-06)**

## R5 Staff Training and Qualification in RP&C

a. Scope (83750)

The inspector reviewed the licensee's selection, training, and qualification program for the contracted radiological control technicians hired for the current outage. The radiological controls (RC) Field Operations guidance for coaching radiation workers was also reviewed. Information was gathered through discussions with cognizant personnel and review and evaluation of procedures and documents.

b. Observations/Findings

Licensee's procedures and documentation for the training and qualification of contracted radiological controls technicians hired for the current outage were reviewed and discussed with training personnel. The review of training procedures and records showed that the procedures were being properly implemented. The training and qualification process was on-going at the time of this inspection, and the status of each contracted technician's progress in this process was being tracked by the licensee. Based on a review of the documented experience of selected contracted radiological control technicians and on discussions with the supervisor of RC Field Operations, selections were in accordance with the TS experience requirement. RC Field Operations had initiated additional radiation worker coaching at the start of this outage. This coaching was for radiation workers who were new to nuclear power sites or to TMI-1. The coaching was performed in accordance with written guidance, conducted in the RCA, and lasted for approximately 1.5 to 2 hours. The coaching addressed practical factors involved with RCA access and egress and involved radiological control requirements applicable to radiation workers while in the RCA.

c. Conclusions

The selection, training, and qualification of contracted radiological control technicians for the outage were in accordance with requirements. The new radiation worker coaching process implemented by RC Field Operations was a good initiative.

**R7 Quality Assurance in RP&C Activities**

a. Scope (83750)

The inspector reviewed the licensee's independent and self-assessing processes. Information was gathered through discussions with cognizant personnel and review and evaluation of procedures and documents.

b. Observations/Findings

GPUN'S NSA group performed independent reviews of the radiation protection program. NSA audits the entire program every two years in two parts. The inspector reviewed the completed, but not yet approved audit designated S-TMI-97-06 which covered organization, training and qualifications, dosimetry, TS surveillances, source accountability, use, maintenance, and calibration of radiation instrumentation, procedures, document control, and records. This audit had a broad scope and was also highly detailed and in depth. It resulted in the identification of several minor deficiencies, but no quality deficiency reports.

Self-assessment efforts by the radiation protection organization since the last inspection were examined by the inspector. Approximately thirty surveillance inspection reports for various plant locations had been performed and documented. The inspector's review indicated that these surveillances by radiological control supervision and radiological engineering staff resulted in the identification of numerous minor deficiencies. Most of these resulted in corrective action being implemented on the spot.

c. Conclusions

The scope and depth of the NSA audit of the radiological controls group was of good quality. The surveillances by the radiological control personnel resulted in the correction of numerous minor deficiencies.

**R8 Miscellaneous RP&C Issues**

While performing the inspections discussed in this report, the inspector reviewed the applicable portions of the UFSAR that related to the areas inspected. The inspector verified that the UFSAR wording was consistent with observed plant practices, procedures, and/or parameters.

**S1 Conduct of Security and Safeguards Activities**

The inspectors monitored the security department's control of protected area openings throughout the 12R refueling outage. The control of protected area boundary openings and worker sensitivity toward security requirements was a weakness during the 11R refuel outage.

All openings in the protected area boundary were controlled properly by the security department for the entire 12R refueling outage. An example of improved boundary controls was noted throughout the replacement of the 'A' and 'B' circulating water pump impellers. Security locks were placed on the pump discharge valves and associated electrical breakers to prevent an inadvertent breech in the protected area boundary. The increased involvement of security personnel at the daily work planning process, increased worker sensitivity to security requirements, and improved security references in the work packages resulted in the improved performance in the security area. Based on the improved security performance for the refuel outage, the inspectors concluded that the plant corrective actions for prior problems were effective.

## V. Management Meetings

### X1 Exit Meeting Summary

At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting with GPUN management on November 10, 1997, summarizing Unit 1 inspection activities and findings for this report period. The licensee acknowledged the findings presented. No proprietary information was identified as being included in the report.

## INSPECTION PROCEDURES USED

| | |
|---|---|
| IP 37550: | Engineering |
| 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 73753 | Inservice Inspection |
| IP 73755 | Inservice Inspection; Data Review and Evaluation |
| IP 83750 | Occupational Radiation Exposure |
| IP 92901: | Followup - Plant Operations |
| IP 92902: | Followup - Maintenance |
| IP 92903: | Followup - Engineering |
| IP 92904: | Followup - Plant Support |
| IP 93809 | Safety System Engineering Inspection |

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | |
|---|---|
| 50-289/97-09-01, | Decay Heat Removal Requirements During Reactor Vessel Draining (IFI) |
| 50-289/97-09-02, | Failure to Follow Reactor Coolant System Filling Procedure (EEI) |
| 50-289/97-09-03, | Power Operated Relief Valve Inoperable for an Operation Cycle (EEI) |
| 50-289/97-09-04, | Emergency Diesel Generator Testing During Simulated Accidents (URI) |
| 50-289/97-09-05, | Personnel Hot Particle Contamination Due to Inadequate Surveys (EEI) |
| 50-289/97-09-06, | Inadequate Control Over Once Through Steam Generator Locked High Radiation Area (EEI) |

Closed

| | |
|---|---|
| 50-289/96-07-01, | Safety Related Scaffolding (VIO) |
| 50-289/97-07-01, | Failure to Write a Clear Specific Narrative Description in LER 97-003 (VIO) |

Updated

NONE

## LIST OF ACRONYMS USED

| | |
|---|---|
| AB | Auxiliary Building |
| AEC | Atomic Energy Commission |
| ALARA | As low As Reasonably Achievable |
| ASME | American Society of Mechanical Engineers |
| B&W | Babcock and Wilcox |
| BWST | Borated Water Storage Tank |
| CAP | Corrective Action Process |
| CDF | Core Damage Frequency |
| CF | Core Flood System |
| CFR | Code of Federal Regulations |
| CFT | Core Flood Tank |
| CR | Control Room |
| CRDM | Control Rod Drive Mechanism |
| CRO | Control Room Operator |
| DBD | Design Basis Documents |
| DCP | Design Change Package |
| DH | Decay Heat Removal System |
| ECCS | Emergency Core Cooling System |
| EDG | Emergency Diesel Generator |
| EEI | Escalated Enforcement Issue |
| EPIP | Emergency Plan and Implementing Procedure |
| ESF | Engineered Safety Feature |
| FAC | Flow Accelerated Corrosion |
| FTC | Fuel Transfer Canal |
| GPUN | GPU Nuclear (Licensee) |
| HRA | High Radiation Area |
| IB | Intermediate Building |
| IFI | Inspection Followup Item |
| IPE | Individual Plant Evaluation |
| IR | Inspection Report |
| ISI | Inservice Inspection |
| IST | Inservice Testing Program |
| HELB | High Energy Line Break |
| HPI | High Pressure Injection (MU) |
| JO | Job Order |
| LCO | Limiting Condition of Operation |
| LER | Licensee Event Report |
| LOCA | Loss of Coolant Accident |
| LOOP | Loss of Offsite Power |
| LOR | Licensed Operator Requalification |
| LPI | Low Pressure Injection (DH) |
| LTOP | Low Temperature Overpressure Protection |
| MCC | Motor Control Center |
| MNCR | Material Nonconformance Report |
| MOV | Motor Operated Valve |
| MU | Makeup System |

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|---|---|
| NCV | Non-Cited Violation |
| NDE | Non-Destructive Examination |
| NDTT | Nil Ductility Transition Temperature |
| NOV | Notice of Violation |
| NRC | Nuclear Regulatory Commission |
| NSA | Nuclear Safety Assessment |
| OTSG | Once Through Steam Generator |
| PCR | Procedure Change Request |
| PMT | Post-Maintenance/ModificationTest |
| PPB | Part per Billion |
| PPM | Part per Million |
| PRA | Probabilistic Risk Assessment |
| PORV | Power Operated Relief Valve (Pressurizer) |
| PRG | Plant Review Group |
| QDR | Quality Deficiency Report |
| QV | Quality Verification |
| RB | Reactor Building (Primary Containment) |
| RC | Radiological Controls |
| RCA | Radiological Control Area |
| RCBT | Reactor Coolant Bleed Tank |
| RCS | Reactor Coolant System |
| RP | Radiation Protection |
| RWP | Radiation Work Permits |
| SALP | Systematic Assessment of Licensee Performance |
| SER | Safety Evaluation Report (NRC) |
| SF | Shift Foreman |
| SRO | Senior Reactor Operator |
| SS | Shift Supervisor |
| TI | Temporary Instruction |
| TS | Technical Specification |
| UFSAR | Updated Final Safety Analysis Report |
| URI | Unresolved Item |
| VIO | Violation |