U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No. 50-289/85-17

Docket No. 50-289

License No. DPR-50 Priority -- Category --

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

Facility At: Three Mile Island Nuclear Station, Unit 1

Inspection At: Middletown, Pennsylvania

Inspection Conducted: May 6, 1985 - May 31, 1985

Inspectors: E Z Conner for 6/14/85
F. Young, Resident Inspector (TMI-1) date

E Z Conner for 6/14/85
R. Conte, Senior Resident Inspector (TMI-1) date

E Z Conner for 6/20/85
R. Urban, Reactor Engineer date

Approved By: E Z Conner 6/20/85
E. Conner, Chief, date
Reactor Projects Section No. 1A


Inspection Summary
This routine safety inspection (218 hours) reviewed routine shutdown plant activities, safety system low pressure piping integrity, condensate storage tank design review, implementation of licensee's security plan, saturation margin instrument error, physical plant restart readiness, and licensee action on previous inspection findings.

Results
Licensee management continued their detailed involvement in plant activities. The licensee continues to work on making the plant ready for restart. Implementation of the Security Plan in response to an anti-restart demonstration was thorough and in accordance with their plan. Implementation of proposed license conditions had been properly tracked and pursued in a timely manner to support restart. The licensee either initiated appropriate action or completed commitments related to previously identified inspection findings.



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# DETAILS

## 1.0 Introduction

This inspection report documents the activities conducted by the resident and region-based inspectors. The overall purpose of the inspection was to assess the licensee's activities as they relate to reactor safety and worker radiation protection for the shutdown mode and to assess plant readiness for the restart of TMI-1.

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

## 2.0 Plant Operations During Long Term Shutdown

### 2.1 Routine Review

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

- review of selected plant parameters for abnormal trends;
- plant status from a maintenance/modification viewpoint including plant housekeeping and fire protection measures;
- control of ongoing and special evolutions, including control room personnel awareness of these evolutions;
- control of documents including log keeping practices;
- implementation of radiological controls; and,
- implementation of the security plan including access control, boundary integrity and badging practices.

The inspectors focused on the following areas:

- the control room during licensee regular and back shift hours; this included selected sections of the shift foreman's log and control room operator's log for the period May 6, 1985, through May 31, 1985, and selected sections of other control room daily logs;

-- areas outside the control room on May 6, 7, 8, 10, 14, 15, 17, 21, 22, 23 24, 29, and 31; and,

-- selected licensee planning meetings.

Based on the review of the various licensee activities noted above, the inspector identified no conditions adverse to nuclear safety or regulatory requirements.

Personnel stationed in the control room presented a posture of overall control of daily activities, including problem areas that needed resolution. The planning meetings indicated attentivenss to proceed safely with daily activities, including surveillance and maintenance, and to resolve any inter-department interface problems. Licensee upper management continued their detailed involvement in site activities.

## 3.0 Reactor Safety Study (WASH-1400) Event V: Plant Review

The Event V scenario was first identified in WASH-1400 (Reactor Safety Study) published in October 1975. The NRC has determined that certain isolation valve configurations in systems connecting the high-pressure Reactor Coolant System (RCS) to low-pressure systems extending outside containment are potentially significant contributors to an inter-system loss-of-coolant accident (LOCA). Such configurations have been found to represent a significant factor in the risk computed for core melt accidents.

The sequence of events leading to the core melt is initiated by the concurrent failure of two in-series check valves to function as a pressure isolation barrier between the RCS and a low-pressure system extending beyond containment. This failure can cause an over pressurization and rupture of the low-pressure system, resulting in a LOCA outside containment.

The NRC has determined that the probability of failure of these pressure isolation valves can be significantly reduced if the pressure at each valve is continuously monitored or if each valve is periodically inspected by leakage testing, ultrasonic examination, or radiographic inspection.

As with other Region I facilities, Region I conducted a fact finding inspection at TMI-1. Areas examined were: (1) ECCS Systems subject to a potential Event V overpressurization; (2) surveillance and maintenance activities ssociated with the valves of these suspect systems; (3) any instances of an actual or potential Event V overpressurization; (4) formalized training in regard to procedural requirements for proper surveillance and/or maintenance of the pressure isolation valves; and, (5) identification of any inadequacies or weaknesses in the current facility design or procedures that would lead to an Event V overpressurization. The collective information will be reviewed by Region I to determine if any additional special inspections would be warranted at the various operating reactor facilities in Region I.

### 3.1 System Review

The inspector reviewed all ECCS systems and other low-pressure systems that contain components or piping with design pressures equal to or less than 70% of the design pressure of the RCS. The following drawings were reviewed:

-- C-302-650 (Rev. 24), Reactor Coolant System;

-- C-302-711 (Rev. 12), Core Flooding System;

-- C-302-640 (Rev. 34), Decay Heat Removal System;

-- C-302-196 (Rev. 8), OTSG Chemical Cleaning;

-- C-302-620 (Rev. 21), Intermediate Cooling;

-- C-302-630 (Rev. 14), Spent Fuel Cooling System;

-- C-302-660, 661 (Rev. 17, 23), Make-Up and Purification;

-- C-302-610 (Rev. 31), Nuclear Services Closed Cycle Cooling Water; and,

-- C-302-673 (Rev. 1) Post-Accident Reactor Coolant Sampling.

The inspector determined that the Low-Pressure Injection System (LPIS) was the only system that was within the scope of the NRC's Event V configuration. This finding was in agreement with the Franklin Research Center Technical Evaluation Report for Three Mile Island, Unit 1. The six valves of concern in the LPIS are check valves CF-V5A and B, DH-V22A and B, and motor operated valves (MOVs) DH-V4A and B.

Four other systems were possible Event V configurations, but were ruled out because they were not within the scope of all five Event V criteria. These systems were the: (1) Decay Heat Auxiliary Spray Line; (2) Core Flooding System; (3) High-Pressure Injection System; and, (4) Shutdown Cooling Mode of the Residual Heat Removal System. Various check valves and isolation valves in each of the above four systems are open issues with NRR for the Pump and Valve Inservice Testing (IST) Program. Although these sub-systems are not within the Event V configuration, NRR is requesting the licensee to add these pressure isolation valves to Technical Specification Section (TS) 3.2.6, Leakage.

### 3.2 Surveillance Activities

All four LPIS check valves were leak tested during the last Hot Functional Testing (HFT) conditions prior to achieving hot shutdown. Surveillance Procedure 1300-3T, Revision 7, "Pressure Isolation Test

of CF-V4A/B, 5A/B, and DH-V22A/B" was used to perform these tests. This procedure contained various prerequisites and precautions to prevent overpressurizing the LPIS. After completion of the tests, independent position verification was performed. Leakage rates were in accordance with TS Table 3.1.6.1 based on a review of data obtained for the above noted tests.

3.3 Maintenance/Modification Activities

Since January 1980, no corrective maintenance or modifications were performed on check valves CF-V5A/B. Minor maintenance and modifications were performed on check valves DH-V22A/B such as internal valve inspection and repair, and lock plate and stop pin modifications. Corrective maintenance (CM) performed on MOVs DH-V4A/B consisted mostly of repacking/replacing valve packing; replacing a one-amp fuse with a two-amp fuse, a minor modification, was also performed.

A preventive maintenance (PM) program does not exist for check valves DH-V22A/B and C -V5A/B. Surveillance and maintenance, if needed, was performed on these valves as specified by TS. MOVs DH-V4A/B are under a specific preventive maintenance program. Procedure E-1, Revision 10, "Limitorque Valves" and E-59, Revision 4, "Limitorque Valve Operator Brake Maintenance" are used to perform preventive maintenance on these valves at least every eighteen months.

The PM and CM procedures were reviewed. Each procedure has a prerequisite section that is adequate in addressing precautions related to interfacing areas before removing equipment from service. The procedures have post-maintenance tests for the components to assure operability; acceptance criteria are also listed in the procedures.

3.4 Miscellaneous

As reported by the licensee, there were never any actual or potential overpressurization events of low-pressure ECCS piping or components at TMI-1. Licensee personnel are familiar with the surveillance and maintenance of these check valves because they are present in the TS; the surveillance procedure is written in such a manner that personnel are aware of their TS related importance. Since inception of the Event V scenario, the licensee has focused attention in this area. Independently, the licensee also reviewed piping flow diagrams to identify similar susceptible valve arrangements.

Industry wide experience related to isolation barrier failures was reviewed. There are no apparent weaknesses in the current facility design or procedures that could lead to similar overpressurization events.

## 4.0 Condensate Storage Tank Modification Design Review

As part of a continued review of the licensee's activities in the area of Plant Modifications, the inspector attended a Preliminary Engineering Design Review (PEDR) Meeting on Condensate Storage Tank Oxygen Control. The purpose of this modification is to maintain the oxygen level below 100 ppb in the Condensate Storage Tanks during plant operations by use of nitrogen stripping.

This meeting was held onsite and was attended by representatives from corporate engineering, plant engineering, plant chemistry, maintenance and construction, plant maintenance, and plant operations. The inspector determined that the meeting identified areas where further review was required. The project engineer presented the modifications and discussions were held at key points to receive comments. The PEDR was found to be thorough and appeared to address all aspects of the modifications from regulatory requirements to actual daily operations of the system. In general, the inspector considered the PEDR to be a useful mechanism to identify many practical problems prior to construction of the modifications.

## 5.0 Security Plan Implementation

During facility tours, the inspector verified that the licensee was properly implementing selected portions of the TMI-1 security plan. The inspector verified that protected area gates were locked or guarded, entry and egress into protected and vital areas was proper, the required security force was present, and security posts were properly manned. In response to an inspector identified problem at TMI-2 on control of vehicles in a protected area, the inspector reviewed these activities at TMI-1. The licensee's handling of vehicles at TMI-1 was found to be in accordance with approved security procedures. Several security guards were questioned as to their responsibilities in this area and were found to be knowledgeable.

While touring the Auxiliary Building and while inside a vital area, the inspector noted that an individual had left his photo identification badge, and key card, and associate information required to use the card hanging in an RWP clothes change area (not in the individual's possession). The licensee's security plan implementing 10 CFR 73.46(d)14 requires personnel to display their badges while within protected or vital area and is silent with respect to personal control of key cards. The inspector expressed concern to the Security Manager that photo badges and key cards were not positively controlled by the individual within the vital area as intended by 10 CFR 73.46(d)14. The Security Manager acknowledged the inspector concerns and stated that their security plan procedures would be revised to address this issue. In addition, the Security Manager reissued to all plant personnel who enter Unit 1, a letter from the Vice President of TMI-1 which provided guidance for individuals to positively control their badges and key card while in RWP areas. The inspector considered the item unresolved pending the review of revision to the security plan implementing procedures (85-17-01).

Approximately two weeks prior to May 29, 1985, the licensee was informed by Pennsylvania State Police that an anti-restart group planned a demonstration at the plant entrance that would involve civil disobedience on the part of some of the demonstrators. At that time, the licensee started to review and make preparation to exercise applicable portions of their Civil Disobedience Security Procedure. A week prior to the demonstration a briefing was held with representatives of Corporate Security and Site Security from both units, the utility's licensing organization, site labor and administrative representatives. The resident inspector attended this meeting to assess the licensee's ability to handle this type of situation. Specific work lists with responsibilities assigned were generated. The meeting was also used to brief key licensee managers.

On May 29, 1985, at approximately 7:00 P.M. demonstrators arrived at the North Gate and blocked the entrance. The Pennsylvania State Police who were present at the time began making arrests for blocking a utility right of way. The right of way was restored at the North Gate at approximately 8:30 P.M. In preparation for the demonstration, the licensee assured the plant staff, including contractors, that this event would be handled in a peaceful and restrained manner. They kept the resident inspectors fully informed of their plans and actions. The licensee had contingencies for a long term blockade (in excess of one shift) with respect to minimum shift manning and supplies (e.g., food, beds, etc) for critical shift personnel. No conditions adverse to safeguard requirements were noted.

6.0 Saturation Margin Calculation Instrument Error

NRC Inspection 50-289/85-15 verified that the surveillance/ calibration procedure acceptance criteria tolerances were consistent with the licensee saturation margin monitor (SMM) loop error analysis (Office of Nuclear Reactor Regulation Restart Certification Item No. 154). During this inspection period, NRR requested that Region I perform a similar verification for the instruments used in the hand calculation for saturation margin. The NRR staff in its safety evaluation attached to SECY 85-189 relied on the licensee's error analysis for this hand calculation to be less than 25°F (loop instrument error plus configuration error).

This inspector verified that the surveillance/calibration procedure criteria tolerance for instruments used in the hand calculation were consistent with the licensee instrument error analysis. During this verification, the inspector discussed this issue with cognizant site and corporate engineers and reviewed the following documents and records:

-- SECY 85-189 (Draft), Three Mile Island, (TMI-1) Restart Certification on Subcooling Instrumentation;

-- GPU Nuclear Calculation Sheets, dated May 20, 1985, TMI-1 Manual Saturation Margin - Allowable Surveillance Testing Error;

-- GPU Nuclear Calculation Sheets No. C-110-665-5350-009, Revision 0, dated December 20, 1984, and Revision 1, dated January 11, 1985, (TMI-1) TSAT Margin - Manual Method Error Analysis;

-- GPU Nuclear Letter, dated January 11, 1985 (Serial No. 5217-85-2001) from Hukill, GPUN, to Stolz, NRC, Error Analysis - Subcooling Margin Indication;

-- Surveillance Procedure (SP) 1302-22, Revision 0, dated July 20, 1984, BIRO Thermocouple Calibration and data obtained August 14, 1984, and March 14, 1985 -- Preoperational Test data per MTX (Master Test Index) 123.5.3; and,

-- SP 1302-6.6, Revision 3, Saturation Margin Channel Calibration.

To verify the consistency of the calibration procedure tolerances to those in the overall hand calculation error analysis, licensee representatives performed an additional calculation (dated May 20, 1985) separating those factors affecting the error analysis into two groups. One group of factors was those affected by the calibration procedure such as the accuracy of transmitters and test devices in the instrument loop tested. The other group of factors was those not affected by the calibration procedure such as accuracy of RTD (Resistance Temperature Detectors), post-accident pressure and temperature affects, and instrument drift/stability.

The subject instrument loops are associated with the safety grade pressure channel/pressure transmitter (PT 949) and the computer calculation of the five highest incore thermocouples.

On May 20, 1985, licensee representatives performed calculations using the calibration tolerance values to represent the factors affected by the calibration procedure. They also used individual information supplied from the vendor or previously accepted (by NRC staff) calculations to represent the individual factors not affected by the calibration procedure. Based on the licensee's assumptions, the licensee calculated a worse case error of 19.89°F. Considering the previously NRC accepted 1.5°F configuration error (location of detectors in relation to lowest pressure point in the RCS), the instrument plus configuration error met the intent of the licensing appeal board item to be less than 25°F. This calculation was also consistent with the overall instrument loop error analysis noted in a docketed submittal to NRC.

The inspector verified that the assumed tolerances in the error analysis were those used in the SP. However, the inspector questioned the use of a accuracy factor of 1/2 for a stability term when no stability tolerance was provided by the vendor for certain components. Licensee representatives did not provide an adequate justification for the use of this factor. In light of this, the inspector conservatively recalculated the error analysis using a factor of one (very conservative). The results still indicated the instrument plus configuration error was less than 25°F,thus meeting the intent of the appeal board.

7.0 Restart Readiness

During the inspection, the resident inspectors continued to review selected areas in order to assess the readiness of the plant for startup. The selected areas reviewed in previous inspections included the licensee's prerequisite list for hot functional testing/ criticality (Flag "2B"); open material non-conformance reports or quality deficiency reports; surveillance program open exceptions and deficiencies and related outstanding regulatory retest equipment; important to safety system valve lineups; and, overall plant conditions including housekeeping and storage of equipment. The objective was to identify equipment operability problems that would adversely affect safe operation of the facility.

In this period the inspectors conducted a plant walkdown to determine if there were any generic items that may be adverse to restart of the plant. During the walkdown, the inspector noted a significant amount of scaffolding erected in the plant. In some areas the scaffolding was in close proximity to important to safety or safety-related equipment. The inspector noted this to licensee management who stated that the scaffolding would either be removed, properly secured, or engineering review performed to ensure the scaffolding would not create an adverse condition. Final review of erected scaffolding will be performed in subsequent walkdowns prior to restart.

In addition, while touring the Reactor Building, the inspector noted that portions of the piping associated with RC-V-1204 was loosely supported. A subsequent system walkdown and discussion with the licensee identified this piping as part of the new reactor coolant water inventory system. The licensee acknowledged the inspector's concerns and stated they would review this condition for adequacy. Because the valve formed part of the reactor coolant system boundary, the licensee agreed to review the concern prior to restart.

In general, the inspector noted the licensee was making adequate progress and, generally, most spaces were found to be acceptable. The above noted items will continue to be routinely reviewed in preparation for restart. Additional areas to be reviewed will be outstanding job tickets and modification incomplete work list items.

8.0 Licensee Event Reports(LERs) Review

The inspector reviewed LER-84-07 to verify that it was submitted to the NRC Region I office in accordance with 10CFR 50.73.

LER-84-07-01 addressed defective tubes in the "A" and "B" Once Through Steam Generators (OTSGs) identified during Eddy Current Testing performed during latter part of 1984. The licensee concluded that the source of the new indications is enhanced visibility of pre-existing indications previously below the threshold of detectability. The licensee also concluded that the corrosion failure mechanism identified in 1983 is still the correct description of what the OTSGs have undergone and no new sulfur corrosion attack has occurred.

In a safety evaluation report dated April 17, 1985, the NRC staff agreed with the licensee's determination that the ECT indications resulted from grain dropout from previously existing intergranular attack caused by thermal strain and hydraulic forces which occurred during hot functional testing. Also, these indications were not due to initiation of new corrosion sites and the NRC staff concluded that the licensee's findings were consistent with conclusions in NUREG-1019, Supplement 1.

The inspector also reviewed the 1984 Eddy Current data to verify the accuracy of data used as a basis in Technical Document Report (TDR) 666. The details of the event were clearly reported including an accurate description of the cause. The inspector determined that the described corrective actions were adequate.

## 9.0 Follow-Up on Previous Inspection Findings

The following items were reviewed to assure that the licensee took adequate corrective action in a timely manner and/or met their commitments as stated in applicable inspection reports.

(Closed) Unresolved (289/84-07-02): Updated Chemistry Procedures to Include Requirements of NUREG-1019

NRC Inspection Report 50-289/84-07 identified procedures associated with control of contaminants into tanks. Procedures associated with the Reactor Coolant Bleed Tank (RCBT) and Boric Acid Mix Tank (BAMT) were in draft form at the time of the inspector's review. In addition, new procedures implementing the new limits on primary water chemistry were to be reviewed. The inspector reviewed the revised procedures and found them to adequately meet the intent of NUREG-1019.

(Closed) Inspection Follow Item 289/84-10-04: Licensee Action Relative to Certain Decay Heat System Valve Leakage

NRC Inspection 50-289/84-10 identified that, during the Reactor Building Integrated Leak Rate Test with the reactor coolant system (RCS) open to building pressure, there was leakage past DH-V5B from the RCS to the Borated Water Storage Tank (BWST). A later NRC inspection identified that the leakage was also through DH-V5A and associated check valves on the piping connecting the Decay Heat System suction header to the BWST. Further, NRC Inspection 50-289/85-12 noted that the licensee committed to document a safety evaluation that would analyze whether or not there would be a transfer of highly radioactive water from the Reactor Building (containment) back through the leaking valves to the BWST.

The licensee issued the necessary safety evaluation by Document Release Form (DRF) No. 032607, dated May 9, 1985, "DH-V5A/B Leakage Effect," and the inspector reviewed that evaluation for adequacy. The inspector also discussed the referenced evaluation and calculations with the engineer who originated the safety evaluation. The license's conclusion was that the

leakage was not a significant problem from a plant operational safety viewpoint.

The cover page of the DRF evaluation reflected that the licensee made the evaluation in accordance with 10CFR 50.59 and that no unreviewed safety question existed.

Calculations made in the evaluation were based on the updated Final Safety Analysis Report, Section 6, during the post LOCA (Loss of Coolant Accident) Recirculation phase. The evaluation indicated that with both DHV/5A or B and DH-V6A or B (isolation between Reactor Building Sump and DH suction piping) open respectively, the water level in BWST "drop line" would be at the 296 foot elevation. Since the inlet to the tank is at 305 feet, no highly radioactive water would back up into the tank causing a radiological hazard in the yard area adjacent to the BWST.

The licensee representative had radiological engineering perform a dose calculation on the indefinite off-gasing of the assumed RB sump water in the BWST drop line. The results of that calculation showed that the most limiting isotope (iodine) cummulative dose at the site boundary was well within 10CFR 50 Appendix I limits.

The inspector concluded that the licensee's evaluation was conducted in accordance with 10CFR 50.59 and that it included very conservative assumptions on the worst case scenario analyzed.

(Closed) Unresolved Item 289/84-31-01: Licensee to Evaluate Operational Pressure-Temperature Curves for Consistency Regarding Natural Circulation Cooldown

NRC Inspection 50-289/84-31 identified that the Natural Circulation Cooling Procedure (OP 1102-16) had operators rely on primary system pressure-temperature operating curves in the Plant Cooldown Procedure (1102-11), but these curves did not incorporate the minimum pressure curve to be used in a natural circulation cooldown. The minimum pressure curve was to prevent a void formation at the relatively hotter reactor vessel head when compared to bulk primary temperature. (A void formation occurred at the St. Lucie Nuclear Plant on June 11, 1980, during natural circulation cooldown.) The licensee committed to operate within the limits of that curve by letters dated April 4, 1984, (Serial No. 5211-84-2081) and June 26, 1984, (Serial No. 5211-84-2156). This inspection also identified that the Anticipated Transient Operator Guidelines (ATOG) Procedure ATP 1210-10 was also inconsistent with the OP 1102-11 curves.

Although licensee representatives apparently conducted additional engineering analysis for this item, the licensee took no formal action prior to restart to resolve this issue. During this inspection period, the inspector examined current copies of the above noted operating procedures and determined that the OP 1102-11 minimum pressure curves were still non-conservative with respect to the minimum pressure curve for the

natural circulation cooldown commitment. The inspector verified that, based on licensee accepted (by NRR staff) submittals, the ATOG procedures are not within the scope of this issue. (The ATOG plan, programs, and procedural implementation were separately reviewed by NRC staff.)

Subsequent to the inspector's identification of this problem, licensee representatives initiated a procedure change request (PCR 1-08-85- 0360) to OP 1102-16, "Natural Circulation Cooldown," which incorporated a separate Minimum Pressure Curve. This curve was consistent with the committed curves in the licensee submittals except at the low RCS temperature range (290°F - 320°F) and low RCS pressure range (300 - 400 psig). Licensee representatives indicated that there was updated technical analysis to support the proposed minimum pressure curve.

The inspector acknowledged the licensee's intent to implement this new curve in accordance with 10 CFR 50.59. However, licensee representatives also committed to provide the new minimum pressure curve for natural circulation in a docketed submittal to NRC staff since the previous staff's Safety Evaluation relied on a different operating curve to prevent a reactor vessel void formation during natural circulation.

Licensee representatives also indicated that an internal technical document report would be issued summarizing their additional analysis of various natural circulation cooldown rates from 10°F per hour to 100°F per hour. Apparently, this additional analysis justified that the updated minimum pressure curve is the most conservative to prevent a reactor vessel void formation. The inspector had no further comments on this matter.

(Closed) Inspector Follow Item (289/84-33-01): License Condition on TMI-1 Solid Waste Handling Exclusively for TMI-1 Generated Waste

As a result of a License Condition Meeting in November 1984 and as documented in NRC Inspection Report No. 50-289/84-33, the licensee was to re-review their implementation of this license condition designated A.2. The Licensing Board ruling indicated that TMI-1 solid waste handling capabilities were not to be relied upon for TMI-2 decontamination. However, the board was more emphatic in their discussions of unit separation. Due to licensee testimony, the board used the word "exclusive" in describing separation of the two units.

The inspector reviewed subsequent licensee action related to this item. The licensee revised Operating Procedure (OP) 1104-63, "Control of Radioactive Material," Revision 9, dated February 11, 1985; this action revised the list of TMI-1 personnel responsibilities concerning the handling of radioactive material. The OP mandates that Unit 1 facilities never be used to process concentrated waste, spent resin, mechanical filters, or dry compactible/noncompactible trash generated at Unit 2. The restriction also covered decontamination of material and/or storage of material generated in Unit 2. The licensee's Quality Assurance Modifications

Operations Monitoring Report No. FEK-320-85, dated May 15, 1985, verified compliance with the proposed licensee conditions.

Based on these restrictions, the inspector concluded that TMI-1 solid waste handling capabilities will be exclusively used for TMI-1 generated waste.

(Closed) Inspector Follow Item (289/84-33-02): Licensee and NRC Staff to Re-Review Licensee Condition on Upgrading the Emergency Plan With Respect to Changing Capabilities of Plant Instruments.

As a result of the above noted license condition meeting, the licensee submitted a letter to NRR, dated February 20, 1985, which provided information to explain why the applicable license condition designated A.19 was not an ongoing requirement but ended with the completion of Task Action Plan Item II.F.1, Post Accident Instrumentation. Based on review of the Commission Order (CLI-79-8), which initiated the TMI-1 restart proceeding, the staff concluded that the condition had ongoing applicability. The NRC staff (NRR) will be responding to the licensee with that position. The inspector concluded that the licensee and NRC staff re-reviews were complete with the proposed licensee condition to remain as a final condition in the restart license amendment.

(Closed) Licensee Event Report (289/84-LO-07): New Eddy Current Indications Found in 1984.

See paragraph 8 for details.

10. Exit Interview

The inspectors discussed the inspection scope and findings with licensee management at the exit interview conducted on May 6, 1985. The following licensee personnel attended the meeting:

-- T. Hawkins, TMI-1 Startup and Test Manager
-- H. Hukill, GPUN, Director, TMI-1
-- C. Incorvati, GPUN, TMI Audits Supervisor
-- S. Otto, TMI-1 Licensing Engineer
-- L. Robinson, TMI-1 Communications
-- C. Smyth, TMI-1 Licensing Manager
-- R. Toole, Operations & Maintenance Director, TMI-1

As discussed at the meeting, the inspection results are summarized in the cover page of the inspection report. The licensee cautioned that details of the security plan implementation section may contain safeguards or proprietary information if not carefully written. The inspectors acknowledged the licensee's comment. The inspectors discussed the licensee plans for making the plant physically ready to restart.

Unresolved Items are matters about which information is required in order to ascertain whether they are acceptable items, violations or deviations.

Unresolved item(s) discussed during the exit meeting are documented in paragraph 9.

Inspector Follow Items are matters that warrant NRC verification of licensee completion as a result of commitments made to the NRC. Inspector follow items are addressed in paragraph 9.