U.S. NUCLEAR REGULATORY COMMISSION
REGION I

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Licensee: GPU Nuclear Corporation
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Facility: Three Mile Island Nuclear Station, Unit 1

Location: Middletown, Pennsylvania

Dates: June 12 - July 15, 1988

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8/23/88
Date

Inspection Summary: This was a routine safety inspection conducted by resident inspectors of licensee power operations, outage activities, and transition from one mode to the other. The inspectors reviewed the following functional areas: plant operations; equipment operability (maintenance and surveillance); modification/engineering support; radiological; and, security program implementation. See Table of Contents for more details.

Inspection Results: The transition from power operations to cold shutdown and outage start went relatively smoothly. The licensee anticipated certain problems based on past experiences, such as the buildup of radioactive iodine in the reactor building. In a majority of cases observed, the licensee rigorously implemented facility procedures. However, there were instances where procedure implementation was not strict, such as for the Once-Through Steam Generator initial fill and drain and for the start of plant cooldown.

Operator and licensee management response to the interruption of decay heat removal (DHR) flow reflected their sensitivity to the loss of DHR safety issue. The root cause of this event was procedure nonadherence. Overall, the licensee properly conducted refueling surveillance activities, such as for eddy current testing and check valve inservice tests. However, the inspectors noted several unresolved items, some of which need to be resolved by the licensee prior to startup from this outage.

Initially, at the start of the outage, licensee performance in the area of housekeeping and occupational safety and health measures trended downward. However, it appeared that licensee management, in most cases, similarly noted discrepancies

in this area and they took corrective action to reverse the noted trend. The reason for the trend appeared to be, as expected the arrival of the augmented labor force (of new personnel) for outage work.

In general, licensee personnel adequately planned and implemented modification activities in accordance with established procedures. The inspector noted one exception with respect to the control room ventilation fan work. In this case, personnel did not follow established administrative control with respect to the use of vendor manuals and vendor-related information (an apparent violation of NRC requirements; see paragraph 4.2). As a result, the fan blades were improperly torqued and the incorrect fan hub was described in the installation instruction, both problems were corrected by the licensee.

The licensee had well-established programs in the area of security and radiological controls. The inspector noted miror implementation discrepancies in the area of radiological controls. The licensee had well-staffed and well-trained personnel for both programs.

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ATTACHMENT 1 - Activities Reviewed

DETAILS

1.0 Introduction and Overview

1.1 Licensee Activities

During the report period, the plant continued to operate at full power until June 17, 1988. Over the weekend of June 17-19, 1988, the licensee shut down the reactor to cold shutdown conditions to start a 64-day refueling/maintenance outage.

As of July 15, 1988, the TMI-1 reactor was in a cold shutdown condition at approximately 110 F with reactor vessel level at the reactor vessel cold leg nozzles in preparation for refueling cavity fill to start fuel bundle shuffling.

1.2 NRC Staff Activities

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

NRC staff inspections are generally conducted in accordance with NRC Inspection Procedures (NIP's). These NIP's are noted under the appropriate section in the Table of Contents to this report.

1.3 Persons Contacted

During this inspection, the following key licensee personnel provided substantial information in the development of the inspectors' findings.

*G. Broughton, Operations/Maintenance Director
*J. Colitz, Manager, Plant Engineering
J. Frew, Director, TMI Site MCF
J. Garrison, Technical Support and Planning Manager
*H. Hukill, Vice President and Director, TMI-1
*C. Incorvati, Audit Manager
J. Kuehn, Radiological Controls Director
*M. Nelson, Manager, Safety Review
T. O'Connor, Lead Fire Protection Engineer
A. Palmer, Manager, Radiological Field Operations
M. Ross, Plant Operations Director
*T. Sessoms, Jr., MCP Planner

H. Shipman, Plant Operations Engineer
*D. Shovlin, Plant Materiel Director
C. Shorts, Project Manager, Site Technical Functions
*C. Smyth, Manager, Licensing

*Denoted those who attended the final exit meeting (see Section 7).

2.0 Plant Operations

2.1 Criteria/Scope of Review

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

-- review of selected plant parameters for abnormal trends;

-- plant status from a maintenance/modification viewpoint, including plant housekeeping and fire protection measures;

-- control of on-going and special evolutions, including control room personnel awareness of these evolutions;

-- control of documents, including logkeeping practices;

-- implementation of radiological controls; and,

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

During the transition to cold shutdown, the resident inspectors provided enhanced coverage during backshifts and that weekend. Detailed findings were addressed below.

2.2 Transition to Cold Shutdown and Start of Outage

During the shutdown/cooldown process of June 17-19, 1988, there was a buildup of Iodine concentration (primarily I-131) in the reactor building (RB) (primary containment), which somewhat restricted activities in the RB from about 8:00 p.m., June 18, 1988, to 12:00 noon, June 19, 1988. The licensee anticipated the buildup and sampled RB air every two hours. Highest airborne I-131 concentration in the RB was 11.5 MPC (maximum permissible concentration) over the "A" D-ring. There was minor reactor coolant system (RCS) leakage from the Power Operated Relief Valve (PORV) and the "B" reactor coolant pump seal cavity area, both in the "A" D-ring. The highest personnel exposure (initial estimate) due to these concentrations was about 15 MPC-hours (well within regulatory requirements). There was no indication of an off-site release.

Overall, the transition into the outage went relatively smoothly. In most cases observed, licensee representatives rigorously implemented procedures. Examples included procedures for: power reduction; emergency electrical power transfer surveillance test; and, RCS drain down.

However, in some cases, licensee personnel did not implement procedures rigorously as noted below. At the start of plant cooldown during the midshift of June 18, 1988, the inspector noted that operators did not sign off all of the prerequisites (Section 3.1) for Operations Procedure (OP) 1102-11, Revision 73, "Plant Cooldown." Examples included: (1) Heat Sink Protection System (HSPS) in defeat; (2) shutdown boron established; (3) sufficient availability of boric acid and demineralized water; and, (4) notification to chemistry to take Reactor Coolant System (RCS) gas samples. The operators performed items (1) and (4) and signed off items (1) through (4) after the inspector brought the discrepancies to the attention of the shift supervisor. The control room quality assurance monitor similarly noted the discrepancies and recorded the finding in a monitoring report.

Following the plant cooldown on June 20, 1988, the licensee started to implement Special Temporary Procedure (STP) No. 1-88-0013, dated June 13, 1988, "Water Inventory Control for OTSG [Once-Through Steam Generator] Secondary Side Hydraulic Cleaning." Steps C.5 and C.15 stated that the first two initial drain-downs would go to a water level of about ten inches. Paragraph C.16 provided for optional drain-down levels during periodic hydraulic cleaning evolutions to be determined by the OTSG coordinator. That option was not provided in steps C.5 and C.15.

The first two drain-downs went to 281-inches (not recorded in the control room operator narrative logs) and 24-inches, respectively. Licensee personnel indicated that the ten-inch target was not stringent to the evolution of initial cleanup of the OTSG. The inspector concluded that the procedure was not clear in reflecting that optional end point drain-down water level.

Further, during the initial fill and vent of the "A" OTSG per STP 1-88-0013, operators noticed reduced water flow from the OTSG vent MS-V84A. The reduced flow was because of a flow meter restriction (which was not supposed to be there for this evolution) in the vent line. With the flow meter removed, operators noted proper flow. The inspector concluded that operators did not provide rigorous attention to detail in initial line-up for venting the "A" OTSG.

On June 20 and 21, 1988, and during the OTSG "A" and "B" drain-down evolution, the line-up was through the OTSG drain system to a RB penetration to a filter near the turbine building (TB) sump to the powdex sump also in the TB. The temporary flange connections on the filter were leaking profusely, but there was sufficient collection to be routed to the TB sump. The filter housing was marked "potentially contaminated." Although the area around the filter was roped off, the situation reflected a "sloppy" way of using temporary connections. However, there

was no contamination in this area. Licensee management independently noticed the leaks and had them repaired. The inspector concluded that the initial set up off the temporary filter system was somewhat unprofessionally done, although no radiological contamination resulted.

The inspector discussed these negative aspects with licensee management. They acknowledged the inspector's finding and they indicated that performance on the subject procedures was poor. They indicated that they expected procedural prerequisites to be completed and signed off prior to the evolution at hand. They also acknowledged that the above noted procedures could have been better written. Licensee management indicated that they would maintain their attention to proper procedure adherence and adequacy.

The inspector identified to licensee management a broader issue on the specific discrepancy with respect to completing prerequisites on the cooldown procedure (OP 1102-11). The inspector's comments equally apply to the heatup procedure (OP 1102-1). Both procedures are a list of mandatory and optional actions needed for certain key plant pressures/temperatures. The mandatory versus optional actions are not clear. Based on this and previous reviews, the inspector concluded that both procedures were technically adequate, but the formats could be improved for operator use.

It appeared to the inspector that operators believed that all cooldown/heatup (CD/HU) procedure steps, including prerequisites, did not have to be completed in sequence. In many cases, step-by-step sequencing of these procedural steps was not crucial to the CD/HU evolution unless the procedure specifically required action before certain pressure/temperature milestones. Licensee management agreed that the subject procedures could be enhanced. They agreed to re-review these procedures during the next cycle of operations. Since no technical inadequacies were noted for these procedures, the inspector had no additional comments.

2.3 Decay Heat Removal Interruption

On June 27, 1988, at 10:41 a.m., a temporary loss of decay heat removal (DHR) occurred until 10:52 a.m. The event resulted when DH-V-2 was inadvertently closed while technicians performed a surveillance test (SP 1302-5.8, "HPI/LPI [high pressure injection/low pressure injection] Analog Channel Calibration") out of proper sequence. (The DH-V-2 is an isolation valve that provides flow from the "B" loop hot leg to the decay heat pump (DH-P-1A/B) suction). Upon the closure of DH-V-2, operators manually tripped the operating DH pump (DH-P-1A) because of high vibration due to low suction pressure. Through operator action, the valve was quickly re-opened and the pump restarted within about eleven minutes. During that time, the reactor coolant increased in temperature from 126 F to 155 F.

The inspector reviewed the applicable SP 1302-5.8. He concluded that the SP sufficiently warned the technician about the consequences of performing the steps out of sequence.

The inspector also reviewed the licensee's Plant Incident Report (PIR) No. 1-88-03, dated June 30, 1988, on the event. The PIR adequately addressed the event and it provided for reasonable corrective actions. Operator and licensee management response to the event reflected their sensitivity to the loss of DHR safety issue. In addition to personnel counseling, they planned to change the procedure to open the valve breaker during this test to preclude inadvertent closure of these valves. The PIR addressed the root cause as the technicians lacking attention to detail by being preoccupied with another portion of test, rather than with the test portions being performed.

Also, the inspector questioned licensee management as to the reportability of this event, particularly in accordance with 10 CFR 50.72(iii)(B) and 50.73(v)(B) or (vii)(B) with respect to a single failure causing the loss of safety function -- residual heat removal. Licensee representatives indicated that the safety function was not lost because two strings of DHR (borated water storage tank (BWST) to pump to cooler to RCS injection path) were operable. With the manway covers of both steam generators off, RB sump suction could be used if DH-V-2 could not be opened.

The inspector noted that these types of events were of particular interest to the NRC staff and asked the licensee to provide a voluntary report on the event. The licensee management agreed.

Accordingly, this event is unresolved pending licensee submittal of the above-noted report and pending further Region I review of that report and the reportability aspects of this event (289/88-13-01).

2.4 Reactor Coolant System Spill

The inspector reviewed PIR No. 1-88-04, dated July 5, 1988. This incident report documented the results of a licensee investigation into the details of a leak (approximately 300 gallons) from the RCS that occurred in the RB on June 26, 1988. The spill of RCS water occurred when operations personnel pumped pressurizer water to the "C" reactor coolant bleed tank (RCBT) with partially disassembled valves as boundary to the flow path. Before the pumping evolution, operators filled the pressurizer to approximately 47 inches from the RCS to control reactor vessel water level at 13-16 inches for OTSG eddy current testing.

The valves, RC-V-6A/B/C/D and RC-V-7A/B/C/D, were opened for work to replace the packing. At the time of the event, the valves had the stem packing removed and the packing nuts backed off. When operators con-

ducted the evolution, water from the pressurizer surge line via RC-V-20 and 21 and the drain header drained through the open packing in the RC-V-6 and 7 valves.

Additionally, on three previous days, June 22, 23, and 24, 1988, water also leaked from the packing during the maintenance activity, but this fact was not communicated adequately to operations personnel. Radiological controls personnel also knew of the problem and they stopped work on June 25, 1988, due to excessive contamination levels of the work site without further escalation of the problem.

The licensee evaluated the event and they concluded that the cause was inadequate communications between maintenance, operations, and radiological controls personnel. The outage coordinator (shift supervisor) was not aware of the above problems when he allowed the use of the drain path to control pressurizer level. The licensee's planned corrective actions were briefing personnel on adequate communications and "considering developing" a list to track RCS/OTSG openings.

At the end of the inspection period, the inspector had not reviewed all the details of this event. The inspector considered these initial corrective actions to be weak. This area remained unresolved pending further licensee and NRC review of outage work control measures (289/88-13-02).

2.5 (Closed) Unresolved Item (289/87-09-01): Reactor Coolant System Leak Rate Anomaly

The inspector opened this issue due to a concern that the licensee RCS unidentified leak rate calculation may not accurately determine actual unidentified RCS leak rate (TS limit - less than one gallon per minute). The problem occurred when the "leakage plus losses" leak rate was less than unidentified leakage. Normally, total leakage would be expected to be greater than the unidentified leakage as some of the RCS leakage is known and can be positively identified.

The licensee evaluated the problem and determined that the unidentified leakage calculation was as accurate as possible given the accuracy of the calculation and input parameters. Three factors could affect this calculation. The first was the accuracy of the computer calculation considering the various inputs. Small errors that occurred could combine to give an indication of the problem or a negative leak rate. The leakage was so small just after start up from cycle 6R that statistical variations in the input parameter could have caused this result. The licensee performed a statistical review of the data and they determined that the data was not random in nature.

The second factor was minor variations in the reactor coolant pump No. 3 seal purge flow. This purge flow was adjusted weekly to approximately 100 cubic centimeters/minute. A relatively large change in this flow compared to low leakage numbers could cause errors. The licensee evaluated possible variations in the RCS No. 3 seal purge flow as too insignificant to cause the type of error observed.

The third factor evaluated (and the most probable cause as determined by the licensee) was possible change in system configuration, such as valve positions that are operated as a normal course of plant operation and maintenance activities. These actions could tend to increase or reduce total plant leakage which would tend to confirm the cyclical nature of the leak rate discrepancies.

The inspector concluded that the relatively small values of RCS unidentified leak rate could be affected by the above consideration to the point where negative values could be obtained. Also, during the past cycle (cycle 6), the NRC staff independently calculated RCS leak rates and confirmed their results and the anomaly. As leakage increased, more positive leak rates occurred that compared favorably with licensee calculated leak rates and expected results.

The leak rates were as expected when large positive values of the plant parameters used to determine RCS leakrate exist. These values tend to cancel out the above-mentioned variation that occurs when actual total leakage is low. The inspector concluded that RCS leakage was accurately calculated given the actual plant conditions.

The inspector will continue to independently verify RCS leak rates following start up from the present outage.

2.6 Plant Operations Summary

The transition from power operations to cold shutdown and outage start went relatively smoothly. The licensee anticipated certain problems based on past experiences, such as the buildup of radioactive iodine in the RB. In a majority of cases observed, the licensee rigorously implemented facility procedures. However, there were instances where procedure implementation was not strict, such as for the OTSG initial fill and drain and for the start of plant cooldown. For these instances, no violation will be issued because a current unresolved item exits to address licensee corrective actions on past violations in this area.

Operator and licensee management response to the interruption of DHR flow reflected their sensitivity to the loss of DHR safety issue. The root cause of this event was procedure nonadherence. Also, licensee action on the RCS leak rate anomaly was appropriate and responsive to NRC staff concerns.

3.0 Equipment Operability Review - Maintenance/Surveillance

3.1 Criteria/Scope of Review

The inspectors reviewed selected activities (listed in Attachment 1) to verify proper implementation of the applicable portions of the maintenance and surveillance programs. The inspector used the general criteria listed under the plant operations section of the report.

3.2 Housekeeping and Industrial Safety

The inspectors conducted several plant walkdowns in various buildings and made the below-noted observations. Overall, conditions were acceptable.

-- During the early phase of the outage, the overall housekeeping, especially in the reactor and auxiliary buildings, was not satisfactory. The inspectors noticed improper bagging of contaminated material, incompletely filled out radiation tags, unidentified contaminated bags, loose anti-C's in the contaminated areas. All these deficiencies were promptly corrected.

-- In the intermediate building, 305-foot level, some loose rags were found inside the contaminated area around the ICV-13B valve The valve was apart for inspection. The valve cover plate and tha holding bolts were stored in a bucket which was outside the contaminated area. The Group Radiological Controls Supervisor had these housekeeping conditions improved.

-- The housekeeping inside the river water pump house was very good. The licensee had been upgrading this facility for sometime. The upgrade included repair of floors, equipment foundations, painting, etc.

-- On June 21, 1988, and again on June 25, 1988, the inspector noticed four fire extinguishers in the RB, which had not been inspected since April 1988. The fire extinguishers were on monthly inspection frequency. The licensee was aware of this condition and planned to inspect them immediately after the 7R outage began. The inspector acknowledged that since the licensee planned no work in the RB prior to the outage, the delayed fire extinguisher inspection was proper from an ALARA (as low as reasonably achievable) point of view. The outage began on June 17, 1988; however, the fire extinguishers still were not inspected monthly. The licensee's fire protection engineers initiated a revision to the fire extinguisher inspection procedure, requiring inspection of the fire extinguishers prior to beginning the work activity. The inspector verified that the subject fire extinguishers were inspected properly.

-- On June 25, 1988, the inspector saw certain recently installed scaffolding that had not been inspected in the RB. There was no work activity in progress; however, the licensee took immediate corrective actions to inspect these structures.

-- The cage ladder for the emergency egress between the 295 and 320-foot elevation in the intermediate building was smeared with oil from the leaky tendons. This condition could cause potential fire hazards as well as personnel injury. The licensee's fire protection engineer issued a work request to correct this condition.

-- On June 13, 1988, the inspector witnessed the installation of the strainer (RR-S-1A) in the river water pump house. The installation crew raised the strainer approximately five feet off the floor and then left the area, leaving the strainer hanging unattended. This condition could potentially cause personnel injury or damage to other safety-related equipment. Licensee management corrected this problem.

In summary, the licensee was quite responsive in correcting the above-mentioned deficiencies in a timely manner. The procedures for maintaining housekeeping, as well as industrial safety program, appear to be adequate; however, stricter implementation would be an enhancement.

3.3 Plant Material Conditions

On June 29, 1988, the inspector noted that all four bolts on one of the supports for the vertical section of the pressurizer relief valve tailpipe were loose. The inspector verified that there was no on-going work activity on this pipe. The licensee acknowledged the inspector's observations, but they could not provide immediate information related to the acceptability of this condition. They agreed to review the matter prior to cycle 7 startup.

This item will remain unresolved pending completion of the licensee's review and subsequent review by Region I (289/88-13-03).

Two w[illegible] spray pumps, FS-P-5A and 5B, as well as a vertical pipe support, are a part of the RB atmospheric cleanup system (kidney system) located on the 281-foot elevation of the RB. Even though the system is not safety-related, it is seismically qualified (Category II) not to generate missiles in the event of a seismic event. The two pumps, as well as the support, are mounted on the floor by eight bolts on each of their respective baseplates. The inspector saw six bolts missing on each baseplate (a circular flange).

The licensee acknowledged the inspector's observations, but could not provide immediate information related to the acceptability of this condition. They agreed to review the matter prior to cycle 7 startup (289/88-13-04).

3.4 Safety-Related Check Valve Inservice Testing

During this period and in accordance with licensee letter of June 7, 1988, licensee representatives requested an NRC inspection of the following valves: BS-V-52A on June 15, 1988; MS-V-9A/B on June 22, 1988; and, BS-V-30A on July 12, 1988. The resident inspector performed these inspections using the following acceptance criteria.

-- Licensee letter of December 24, 1986, on the inservice test (IST) program

-- Surveillance Procedure (SP) 1300-3L, Revision 0, June 2, 1987, "Disassembly/Inspection of Valves for IST"

These valves, among others, were the subject of an IST relief request by the licensee to be inspected on a ten-year frequency. The NRC rejected this initial request in a Safety Evaluation Report (SER), dated March 31, 1988. The data from the licensee's inspections conducted during this outage were to be a part of the justification for a reduced frequency of inspections.

Essentially the licensee's inspection commitment was to full stroke the applicable check valve by hand, since in-line testing was either impractical or not possible. The licensee demonstrated this for all four of the subject valves. However, the work packages for these inspections included a detailed preventive maintenance (PM) procedure for a thorough inspection of the physical conditions of the valves. For MS-V-9A/B, this PM could not be rigorously implemented because the hinge pin was seal welded to the body of the valve. Licensee representatives decided not to break the seal weld to permit a thorough inspection of the valve seat surfaces, hinge pin condition, or the condition of the underside of the valve disc. The design of the building spray (BS) valves permitted a thorough inspection.

The inspectors considered that the licensee fulfilled their commitments for the IST of the check valves. However, the lack of thorough inspection for check valves such as MS-V-9A/B will be referred to NRC technical reviewer to be factored into their review of the licensee's reduced frequency relief request. The reviewer will be visiting the site during the next inspection period to inspect DN-V-14A/B.

3.5 Borated Water Storage Tank Minimum Level

On February 28, 1988, licensee representatives documented a "preliminary safety concern" (No. 88-003) on the minimum borated water storage tank (BWST) level alarm (termed low low level - 3 feet, 0 inches) to require operator manual actions to realign valves for RB pump suction. The concern was that the low level alarm setting was too low to prevent vortexing on the suction pipe from the BWST to the emergency core cooling

system (ECCS) pumps during design basis accidents. Vortexing might entrap air, causing a loss of suction or interruption of flow through ECCS pumps.

Between February 28 and June 9, 1988, licensee engineers reviewed the matter; but they could not come to any conclusions on whether or not vortexing could form based on available data and analysis. At that point, licensee representatives decided to be conservative by proposing to raise the low low level setting considering the vortexing issue and ECCS pump net positive suction head (NPSH) requirements.

Licensee management also reported the matter to the senior resident inspector on June 9, 1988. At that discussion, licensee representatives pointed out that the seemingly conservative action of raising the low low level alarm was non-conservative with respect to the amount of water injected from the BWST to the RB (containment). This meant less water available (NPSH) in the RB sump for ECCS pumps performance during long-term RB sump recirculation. As a result, on June 13, 1988, the licensee documented a preliminary 10 CFR 50.59 safety evaluation (SE) on the proposed action of raising the low low level setting (to 7 feet, 4 inches).

The licensee's SE was thorough and it considered all the germane issues related to the proposed actions. For example, the SE addressed NPSH requirements for ECCS pumps during BWST draw down and on RB sump recirculation. It also considered the vortexing issue and water inventory requirements for the RB for the design basis loss of coolant accident (LOCA). In order to resolve a NPSH problem with the RB BS pump, the SE recommended throttling BS. The SE addressed the released RB iodine scrubbing efficiency as a result of BS flow throttling.

Since the plant is shutdown as of June 17, 1988, there is no immediate safety concern. During this outage, the licensee is reviewing the preliminary SE and is finalizing proposed actions for cycle 7 startup.

This area is unresolved pending completion of licensee action before cycle 7 startup and further NRC Region I review (289/88-13-05).

3.6 Low Pressure Injection Test

During the shutdown phase of the 7R refueling outage, the licensee performed SP 1303-11.54, Revision 5, "Low Pressure Injection." The test required each of the two decay heat pumps to deliver equal to or more than a 3,000-gallon per minute (gpm) flow rate (at RCS pressure of 0 psig). The inspector witnessed the test on June 19, 1988, on decay heat pump "B" and noted the fluctuation of the flow instrument needle around the 3,000 gpm mark of the flow meter on the console. Subsequent to the above test, the inspector reviewed the test procedure and the data for the entire test on both pumps and identified some concerns on the consistency of test data and accuracy of instruments or gauges on the console and on the Remote Shutdown Panels (RSP).

Also, on June 20, 1988, the inspector noted that the shift technical advisor (STA) recorded that operators allowed the pressurizer cooldown rate limit to be exceeded. Preliminary data indicated that hour-to-hour temperature points were 379 F to 253 F or a cooldown rate of 126F/hour. Technical Specification (TS) 3.1.2.3 requires less than 100 F/hour cooldown rate. Further, licensee determined that operators exceeded the pressurizer high level limit of 220 inches (for low pressure operations), contrary to TS 3.1.12.3, during one of the tests. It appeared to the inspector that the test procedure did not sufficiently caution operators concerning these TS limits.

The licensee acknowledged the inspector's concerns and agreed to re-evaluate the data for this test. They also acknowledged that they must submit a Licensee Event Report (LER) on the violations of TS. Since the plant was already in a cold shutdown condition, this area will remain unresolved pending: (1) completion of licensee re-evaluation, as noted above; and, (2) licensee submittal and NRC review of the above-noted LER prior to cycle 7 startup(289/88-13-06).

3.7 Steam Generator Eddy Current Testing

Technical Specifications (TS) required the licensee to conduct eddy current testing (ECT) on the OTSG tubes at various intervals throughout the life of the OTSG's. The licensee conducted ECT for both OTSG's during the present outage (7R) for a sample of OTSG tubes based on the requirements of TS Section 4.19.3. The inspector reviewed the results of the ECT inspection and observed selected portions of on-going activities during the conduct of actual ECT and subsequent tube plugging evolutions.

The licensee sample of tubes from both OTSG's initially included a 3 percent random sample, plus additional tubes that were previously identified as having a throughwall defect of 20-39 percent. This sample size varied and was larger for the "A" OTSG. The results of the 3 percent revealed two tube failures (greater than 40 percent throughwall indicated defect) in the "A" OTSG and one defect in the "B" OTSG. This necessitated an additional 6 percent sample from both OTSG's. The licensee found no failures in this additional sample.

The licensee additionally performed ECT on additional tubes in each OTSG for known problem areas, such as the tube lane and lane wedge tubes and some tubes which have a high probability of failure. The tubes plugged as a result of the random sample testing (three) and the failed tubes from the other sample resulted in total tube plugging of twenty-one tubes (13 from the "A" OTSG and 8 from the "B").

The inspector reviewed selected data from the ECT and verified that the number of tubes plugged was appropriate for the sample results and that the licensee conducted additional testing as required by the TS. Based on discussions with licensee personnel, the inspector verified that ECT results were verified by independent contractor reviewers. The licensee also reviewed the results, although they do not presently have a Level

II certified non-destructive examination (NDE) test reviewer similar to the licensee's contractor. All tube failures or acceptability calls are verified by contractor personnel. Additionally, a third Level III contractor reviews tubes that are in question based on the interpretation of the first two data evaluations.

The inspector also witnessed portions of the tube plugging evolution, specifically pre-planning and tube plug expanded calibration. The personnel performing this task were in possession of appropriate documentation, procedures, and plans to accomplish the task correctly. The inspector reviewed Job Order (JO) No. A25A-V2224 and Job Ticket (JT) Nos. CP-819 and 818, which the licensee issued for the "A" and "B" OTSG's tube plugging work.

The licensee conducted a safety evaluation for the job requiring additional tube plugging. They based that SE on previous SE's (Nos. 123125 002 and 000224-004), which limit tube plugging to 2,000 tube sections in both OTSG's.

As a result of this outage, total tubes plugged in both OTSG's were 1,627. The previous SE's were reviewed and accepted by the NRC.

Based on this limited review, the inspector concluded that ECT and subsequent tube plugging were accomplished in an acceptable manner. Procedures were in place and site and contractor personnel adequately planned and controlled activities. No safety concerns were noted by the inspector on the conduct of this evolution.

3.8 Equipment Operability Summary

Overall, the licensee properly conducted refueling surveillance activities, such as for ECT and check valve IST. However, the licensee obtained low pressure injection test data that needed further review by the licensee and NRC staff. Further, prior to startup following this outage, the licensee acknowledged a need to resolve the BWST vortexing issue, along with inspector questions on the proper seismic installation of selected components in the RB.

Initially, at the start of the outage, licensee performance in the area of housekeeping and occupational safety and health measures trended downward. However, it appeared that licensee management, in most cases, similarly noted discrepancies in this area and they took corrective actions to reverse the noted trend. The reason for the trend appeared to be, as expected, the arrival of the augmented labor force (of new personnel) for outage work.

4.0 Modification/Engineering Support

4.1 Criteria/Scope of Review

The inspector reviewed the following modifications to assess licensee's design review and approval process, including 10 CFR 50.59 reviews, planning of the work package, adequacy of the installation procedures, control of vendors technical documentation, as well as post-modification testing and acceptance.

4.2 Replacement of Fan Blades on the Booster Fans for the Control Building Ventilation System

The two booster fans, AH-E-95 A and B are part of the control building ventilation system and are to deliver approximately 8,074 standard cubic feet of air per minute (scfm) each to the second floor of the control building (322-foot elevation). This flow rate is specified in Final Safety Analysis Report (FSAR) Section 9.8, Figure 9.8-1. These fans are safety-related components and of seismic classification II.

Since the actual flow was well under 7,000 scfm on each fan, the licensee planned a modification to increase the number of fan blades from five to nine to increase the air flow to 9,500 scfm. The inspector observed portions of the installation and testing for the specific modification (JO No. A25A-V-1214) and determined the following.

(1) The job package did not include any torque requirement for the fan blades once they were adjusted to the specified pitch. The blades were torqued to some unknown value. However, following the review of the controlled vendor's manual, the inspector informed licensee representatives that the vendor did specify torque values. Following consultation with the vendor, the licensee revised the assembly instructions and retorqued the blades as specified. The approved vendor's manual VM-TM-1024 was not in the job package or referenced in the installation specifications. The job package did include a vendor's catalogue which lacked the required torque information.

(2) The vendor's technical manual was approved and issued as a controlled document; however, it was stamped incorrectly as "Information Copy." This problem was corrected immediately.

(3) The installation instruction with respect to vendor manual specifications for the fan blade assembly specified an incorrect type of hub. The licensee revised the instruction to reflect the correct type by issuing a Field Change Request (FCR).

(4) The repair plan for the two associated expansion joints included riveting with a "minimum" rivet spacing (pitch) of six inches. The inspector noted that the actual pitch was approximately three inches. The licensee issued Plant Engineering Evaluation

Request No. 88-048-E specifying a "maximum" pitch of six inches. The original work instructions was in error by indicating "minimum" pitch.

(5) The installation specification required measurement of fan vibrations following the installation for acceptance, as well as for comparison with pre-modification data. The inspector noted that test procedure did not identify the exact location for the vibration probe during pre- and post-modification testing; therefore, it was not possible to make a meaningful comparison. Subsequently, licensee representatives used specially designed stickers to specify the exact location for the vibration probe data during post-installation testing. This should provide an adequate future reference point. The actual vibration data was within the allowable limits.

(6) The inspection sheet for the scaffolding was signed off by the job foreman without performing the required inspection. The loose knuckle joint found by the inspector was tightened immediately and the personnel were counseled by licensee management, emphasizing the significance of the scaffolding inspection and the job safety.

(7) The post-modification testing failed the acceptance criteria for the air flow by a significant margin (approximately 7,000 scfm versus 9,500 scfm). The licensee planned to re-evaluate the modification. This will be an unresolved item (289/88-13-07).

Items (2), (4), (5), (6), and (7) addressed the deficiencies in the implementation of administrative controls, industrial safety, design review, as well as maintenance practice. They did not represent major safety significance and the licensee corrected them promptly.

Items (1) and (3), however, reflected an apparent violation of 10 CFR 50, Appendix B, Criterion VI, and NRC approved Quality Assurance Plan Section 3.2.1 requirements. Since the approved vendors manual was not utilized, improper torquing of the fan blades and incorrect fan hub description resulted. The fan blades, if dislodged, could potentially become a missile and impact the operability of the safety-related control building ventilation system (289/88-13-08).

4.3 Appendix R Modification - Rewiring of Intermediate Closed-Cooling System Containment Isolation Valves

The IC-V-3 and 4 are containment isolation valves on the supply and discharge sides of the intermediate closed-cooling water (ICCW) system, respectively. The ICCW provides cooling to the letdown coolers, reactor coolant pump thermal barriers, reactor coolant drain tank coolers, as well as control rod drive cooling coils. Based on the present control circuitry, these valves are to close in the event of a spurious actuation of the engineered safety features caused by a fire, either in the control room or in the relay room.

This modification (JO No. A-25H-30244) involved rewiring of these valves so that they can be opened through the operation of Remote Shutdown Panels (RSP) and to assure cooling of the above-mentioned components. The inspector witnessed the rewiring activity in progress. The work was well planned. The work package included all the necessary documentation, such as approved wiring diagrams, installation instructions, quality control (QC) hold points, etc. Licensee personnel accomplished the work in accordance with the procedures.

4.4 (Open) Unresolved Item (289/87-11-05): Remote Shutdown Panel Nuclear Instrumentation

The inspector opened this item because the nuclear instrumentation for the Remote Shutdown Panel (RSP), NI-9, failed. The licensee determined that it would take a nine-day outage to replace the detector in the RB. The detector was a source range BF-3 detector.

The licensee is in the process of upgrading the plant post-accident monitoring instrumentation (under NRC order) for compliance with Regulatory Guide (RG) 1.97 by cycle 7 startup. Since no shutdown of greater than nine days occurred during the last year, NI-9 is to be removed and two new full range nuclear instruments are to be installed. These instruments are to satisfy the requirements of RG 1.97 for full range nuclear instrumentation in the control room and, additionally, a remote readout from NI-11 is to be routed to the RSP to also satisfy 10 CFR 50 Appendix R requirements.

The inspector reviewed the following licensee documents to confirm that this new instrumentation will satisfy RG 1.97 requirements for post-accident nuclear instrumentation.

-- Installation Specification TI-IS-412491-004

-- Safety Evaluation No. 412491-002

-- GPU letter 5211-84-2752, dated October 1, 1984, licensee compliance with NUREG 0737, Supplement 1 - RG 1.97

-- TMI Document No. VM-TM-0688, Revision 0, Instruction Manual for Neutron Flux Monitor

The inspector tentatively concluded that the installation of the new RG 1.97 nuclear instrument is in compliance with the applicable requirements of RG 1.97. Installation is presently on-going and final acceptability is to be reviewed when installation and testing is complete. The unresolved item (289/87-11-05) remains open pending the above review.

4.5 Modifications/Engineering Support Summary

In general, licensee personnel adequately planned and implemented modification in accordance with established procedures. The inspector noted one exception with respect to the control room ventilation fan work.

In this case, personnel did not follow established administrative controls with respect to the use of vendor manuals and vendor-related information.

The licensee was in the process of resolving past inspection findings in this area.

5.0 Radiological Controls Program Implementation

5.1 Acceptance Criteria/Scope of Review

The inspector conducted a limited review of the licensee's radiation protection program implementation. The inspector accomplished this review during routine plant tours by observation of on-going maintenance and surveillance activities, review of Radiation Work Permits (RWP's), radiation surveys, discussion with licensee staff, and review of general radiological conditions in the radiologically-controlled areas (RCA). The inspector reviewed ALARA (as low as reasonably achievable) radiation exposure goals and skin contamination records. The inspector also conducted a walkdown in the RB with a licensee radiological engineer.

The findings/conclusions are as follows.

5.2 Radiation Work Permits

The inspector reviewed ten Radiation Work Permits (RWP's) against the licensee's procedure No. 9100-ADM-4110.04, Revision 2, Radiation Work Permit." The inspector determined that the RWP's had appropriate job descriptions, radiation/contamination levels, concentration of airborne radioactivity levels, respiratory/protective clothing-equipment requirements, dosimetry, special precautions, expiration dates, health physics coverage, and required approvals/reviews. Two discrepancies were identified.

(1) RWP No. 033568 for "Routine Sampling and Analysis" was issued with an expiration date of June 4, 1988; however, the date was extended to June 30, 1988, after it was already expired for two days. The inspector was advised that the computer data was updated prior to the expiration date, but the actual RWP was not updated in a timely manner. This was considered an isolated personnel error.

(2) The above-mentioned RWP procedure, Section 4.2.3, Block No. 9, stated that "if radiological conditions significantly change, updated survey information shall be included..." The inspector questioned the definition of "significant change." The licensee planned to provide additional guidance for significant changes in the radiological condition and revise the procedure accordingly. The inspector had no additional comments on this procedure.

(3) On July 7, 1988, the inspector noticed approximately eight outdated radiation surveys posted in the auxiliary building change room. The inspector was advised that current surveys were conducted and an error was made by not posting them in the change room only. The problem was corrected immediately by the Group Radiological Controls Supervisor (GRCS).

5.3 Control of Radioactive Material

The inspector observed licensee personnel entering and exiting RCA's at various times. All personnel complied with requirements to obtain appropriate dosimetry prior to working in the RCA.

Exit procedures from the RCA require whole body frisking. Presently, at TMI-1, the licensee utilizes the PCM-1, a computerized automated whole body frisking device that accomplishes the survey. Personnel routinely use the PCM-1 properly. Additional "friskers" (hand-held personnel survey devices) are throughout the RCA for use on an "as-required" basis.

The inspector noted that the licensee has in place an effective calibration program for various types of monitors, such as PCM, hand-held friskers, continuous air monitors, and area radiation monitors. The inspector found no outdated equipment calibrations.

The inspector, however, identified the following weakness in the control of radioactive material (see also Section 2.4 on housekeeping and industrial safety). The licensee's procedure No. 9100-ADM-4400.01, Revision 1, "Radioactive Material Identification and Handling Applicability/Scope," Section 4.2.2, required the radioactive material tag to be filled out on both sides. The inspector noticed that the backside of the tag was not filled out on the majority of materials inspected. The licensee reassessed the need for the information on the backside and determined that it was not needed and the procedure will be revised accordingly.

5.4 ALARA Planning

The inspector reviewed licensee's procedure 9100-ADM-4010.02, Revision 3, "ALARA Review Program," and also reviewed the job planning for "The Reactor Head Disassembly and Associated Work," ARN No. 88-05-18. He had the following observations.

-- The licensee has an effective ALARA review program. The overall job planning included breakdown of work elements, associated radiological conditions, establishment of manpower resources, and good interfaces among radiological, health physics, and maintenance personnel, emphasizing thorough work plans, pre-job briefing, training, adherence to the established radiological controls.

-- The ALARA planning also included post-job critique and lessons learned for future repeat jobs.

-- The ALARA goals were established for the current year based on actual past experience on the repeat jobs, as well as best estimates. The ALARA performance goals are tracked on a monthly basis for each department and overall performance was also posted for general information for the plant personnel.

-- The ALARA group engineers were assigned specific jobs to witness the work in progress, assess the effectiveness of exposure reducing controls, and initiate corrective actions as required.

The inspector did not find any weakness in this area. The overall radiation exposure was within the established annual goals. The ALARA group appeared to be well trained for their job function.

5.5 Skin Contamination

The inspector reviewed licensee's procedure 9100-ADM-4330.02, "Personnel Contamination Monitoring and Decontamination Applicability/Scope." The inspector also examined skin contamination reports, as well as the skin contamination log maintained a the control point and made the following observations.

-- The licensee had a good log, identifying all the skin contamination cases, their cause, location, etc.

-- The formal skin contamination report on each case included pertinent information, such as description, instrumentation used to detect the contamination, method of decontamination, information about whole body count if applicable, location of the contamination, review, and approvals.

-- The radiological engineering group provided trending to develop generic solutions.

-- The field radiological section took necessary immediate corrective actions following an analysis of each case.

The inspector determined that licensee had an effective skin contamination control program. The procedures were adequate and senior management involvement was evident.

5.6 Radiological Control Implementation Summary

Based upon this review, the inspector concludes that the licensee has a well-established radiological controls problem that, in general, is properly implemented. The organization is well-structured and well-staffed with well trained personnel.

6.0 Security Program Implementation

6.1 Acceptance Criteria/Scope of Review

The inspector conducted an implementation review of certain aspects of the physical security program. The following items were reviewed:

-- verification of minimum shift staffing;

-- verification of individual assigned to each shift for supervision;

-- review of surveillance test records for various search equipment;

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

-- conduct of visitor control badging and escort procedures.

6.2 Findings/Conclusions

The inspector conducted a walkdown of the PA during nighttime conditions to verify proper lighting availability. All areas of the site were observed to be properly lighted. Areas where temporary buildings were located were provided with additional lighting at various locations.

The inspector verified that appropriate armed security guards were present on two consecutive shifts or specified in the Physical Security Plan (PSP). The licensee assigned a sergeant for each shift to supervise activities. Personnel appeared knowledgeable of their responsibilities.

The inspector conducted a review of approximately four months of weekly operability surveillance checks for various search and detection equipment. The inspector noted no major discrepancies.

Through various tours of the plant, the inspector verified that the VA access was properly conducted. No personnel were noted misusing VA access controls.

Observations of Site Protection Officer (SPO) conduct of vehicle search for entry to the PA were observed. All searches were conducted properly and no problems were noted. Those actions were carried out properly during the present outage which was a busier than normal period of access to the PA.

Visitor access control was properly implemented. Appropriate permission was obtained for granting visitor access and personnel performing escort duties were adequately informed of their responsibilities.

6.3 Summary

The physical security and safeguards program was properly implemented. Personnel appeared knowledgeable and had proper concern for their duties. No major problems were noted and the inspector had no safety concerns for the site security plan implementation.

7.0 Exit Meetings

The inspectors discussed the inspection scope and findings with licensee management weekly and at a final exit meeting on July 15, 1988. Those personnel marked by an asterisk in paragraph 1.3 were present at the final exit meeting. In addition to the NRC inspectors, Messrs. C. Cowgill and L. Bettenhausen from NRC Region I attended the final exit meeting.

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

Unresolved Items are matters about which more information is required in order to ascertain whether they are acceptable, violations, or deviations. Unresolved items discussed during the exit meeting are addressed in Sections 2 and 4.

ATTACHMENT 1

NRC INSPECTION REPORT NO. 50-289/88-13

ACTIVITIES REVIEWED

Plant Operations

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

-- Areas outside the control room

-- Selected licensee planning meetings

-- 6/17/88 - Plant shutdown at 1 percent/minute

-- 6/17/88 - STP 1-88-0015, "Turbine Response to Loss of ATB Evaluation" to confirm expected EHC/turbine response to loss of external 120-volt a.c. source from ATB

-- 6/18/88 - RCS cooldown in accordance with OP 1102-11

-- 6/18/88 - Test of engineered safeguards "A" and "B" actuation and undervoltage test on "D" 4.16 kv bus per SP 1303-11.10

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

Day/Date	Time
6/17/88 (Friday)	10:00 p.m. - 11:00 p.m.
6/17/88 (Friday)	11:00 p.m. - 4:00 a.m.
6/18/88 (Saturday)	4:00 a.m. - 7:00 a.m.
6/18/88 (Saturday)	8:00 a.m. - 2:00 p.m.
6/18/88 (Saturday)	4:45 p.m. - 10:45 p.m.
6/19/88 (Sunday)	5:30 a.m. - 9:30 a.m.
6/19/88 (Sunday)	11:00 a.m. - 1:30 p.m.
6/19/88 (Sunday)	5:30 p.m. - 10:00 a.m.
7/02/88 (Saturday)	1:00 p.m. - 3:00 p.m.

Maintenance/Surveillance

-- Safety-Related Check Valve Inservice Testing - Valves: BS-V-52A, June 15, 1988; MS-V-9A/B, June 22, 1988; and, BS-V-30A, July 12, 1988

-- Low Pressure Injection Test witnessed on June 9, 1988

-- Steam Generator Eddy Current Testing periodically throughout the inspection.

-- 6/18/88 - Review of procedure for high pressure injection and low pressure injection core flooding tests

-- 6/18/88 - SP 1300-3T, "Pressure Isolation Test of CF-V4A/B and DH-V22A/B" and SP 12303-11.21, "Core Flooding System Valve Operability Test"

-- 6/18/88 - "A" side HPI injection test per SP 1303-11.9

-- 6/19/88 - SP 1303-11.54, "Low Pressure Injection"

Modifications

-- Replacement of Fan Blades on the Booster Fans for the Control Building Ventilation System

-- Appendix R Modification - Rewiring of Intermediate Closed-Cycle System Containment Isolation Valve