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: January 9 - February 6, 1988

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Approved by: ______________________ 3/10/88
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Inspection Summary: The review was a routine safety inspection (162 inspection hours) assessing performance primarily in the plant operations and maintenance areas. In the plant operations area, the inspector reviewed: various plant transients; verification of correct activities; and, partial loss of the instrument air control system. In the maintenance area, the inspector reviewed: the installation of expansion joints into the reactor building emergency cooling water system; battery terminal cleaning; repair of a liquid waste transfer pump; differences in the diesel generator injector assemblies; liquid waste disposal transfer line repair; and other surveillances and maintenance-related activities. A modification was also reviewed dealing with revised setpoints for the low level interlock on the reactor building sump isolation valve. Also, reviewed were Safety Issue Management System (SIMS) Nos. A-26, "Reactor Vessel Overpressure Protection," and MPA-B-66, "Natural Circulation Cooldown." The review of past inspection findings focused on licensee actions related to past violations in the plant operations, surveillance, and maintenance areas.

Inspection Results: Overall, operational performance remained at a high level. Operational mistakes were made and/or equipment malfunctions challenged licensed and non-licensed operators. However, the mistakes were licensee self-identified and operators were responsive to the challenges they received by taking immediate appropriate corrective actions. Weaknesses in the implementation of the independent verification program continued to be noted. Management attention and involvement in these and other facets of operations were noteworthy.

The continued trouble-free operation of the plant without any unplanned maintenance outages is indicative of an overall effective maintenance program. The negative findings identified by the inspector were indicative of certain procedural weaknesses in the areas of control of applicable and non-applicable work instructions and of control of procurement and replacement of parts. In particular, several examples of maintenance-related errors collectively represented a violation of regulatory requirements (paragraph 3.3). The most significant example was the installation of a wrong model (pressure rating) expansion joint in the reactor building emergency cooling water system. Overall, the licensee's initial and immediate corrective actions were responsive to the inspector's concerns. However, measures to prevent recurrence may need enhancement.

For licensing actions, in particular SIMS Nos. A-26 and MPA-B-66, the licensee effectively translated general licensing actions into plant specific actions with one exception. Also, the licensee took acceptable corrective actions and/or fulfilled commitments made in response to the violations reviewed. Licensee personnel were cooperative in providing additional information to resolve regulatory issues such as for the fastener sampling required by NRC Bulletin No. 87-02.

DETAILS

1.0 Introduction and Overview

1.1 NRC Staff Activities

The overall purpose of this inspection was to assess licensee activities during the power operations mode as they related to reactor safety, safeguards, and radiation protection. Within each area, the inspectors documented the specific purpose of the area under review, acceptance criteria and scope of 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.

1.2 Licensee Activities

During this period, the licensee operated the plant at essentially full power. There were several plant transients as noted below and as addressed in paragraph 2.2.1 of this report.

-- On January 18, 1983, at 4:46 a.m. there was a transient from 100 to 101 percent power, apparently due to a malfunction in the integrated control system (ICS). The operators manually stopped the transient.

-- On January 22, 1988, between 5:19 p.m. and 5:33 p.m. there was a transient between 101 and 90 percent power, which was apparently operator induced while switching between manual and automatic ICS control during maintenance work on the ICS.

-- On January 28, 1988, between 1:10 p.m. and 1:20 p.m. there was a transient from 100 to 97 percent power, apparently due to ICS malfunction in feedwater demand signal processing circuits.

Also, between 3:57 p.m., January 31, 1988, and 2:10 a.m., February 1, 1988, the licensee operated at reduced power (approximately 77 percent) in order to make repairs to a feedwater pump (FW-P-1A). Coupling grease had leaked out of the feedwater pump coupling causing excessive vibrations on the pump.

2.0 Plant Operations

2.1 Criteria/Scope of Review

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

-- review of selected plant parameters for abnormal trends;

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

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

-- control of documents, including logkeeping practices;

-- implementation of radiological controls; and,

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

The inspectors focused on the areas listed in Attachment 1.

2.2 Findings/Conclusions

2.2.1 Power Transients

Paragraph 1.2 of this report summarized the various power transients that occurred during this period. Since the Cycle 6 startup, the licensee continued to observe minor power oscillations apparently due to control system "hunting" in certain sections of the ICS. The licensee continued to troubleshoot the problem leading to the installation of an electronic filter circuit into the suspected problem section of the ICS to dampen these oscillations. These actions were not completely successful in giving the licensee desired results.

During this period, the power transients were correlatable to maintenance on selected operational activities with the ICS troubleshooting. Although root causes could not be clearly identified, they were generally due to an equipment malfunctioning and/or were operator induced. Appropriate equipment corrective actions were taken, such as replacement of selected ICS components. The operator-induced events were overshadowed by the positive operator response to these events, especially those actions that mitigated power increases over 100 percent. No plant trip resulted and there were no challenges to safety-related equipment. The inspector identified no concerns on these events.

2.2.2 Verification of Correct Activities

On January 18, 1988, at 9:20 p.m., control room operators observed that the open and close lights were out for reactor building emergency cooling valve RR-V4A, "Supply Isolation for the "A" Ventilation Cooler in the Reactor Building (RB)." After further review,

the operators determined that the breaker for the motor-operated valve at the "1A" engineered safeguards (ES) Motor Control Center (MCC) switchgear in the control building was open.

The valve and its MCC breaker are normally closed, but an engineered safeguards actuation system (ESAS) is supposed to open the valve for RB emergency cooling water flow. With the breaker open, the safety functions would not have been fulfilled. There are two other coolers that could be used in the RB.

Licensee operations personnel attempted to identify the root cause. There were workers in the area at the time of event identification, but they provided no additional information on how the breaker was opened. The inspector noted the breaker was located at about foot level on the switchgear panel, but it was not in a main traffic pathway. No cause was determined; however, inadvertent bumping could not be ruled out.

The licensee concluded that the ESAS checklist was successful in performing its intended function, since it prompted operators to look at the open/close status for this valve (once per shift). The inspector similarly concluded on the licensee's findings.

Further, on January 23, 1988, at 11:05 p.m., on-coming control room operators noted that Core Flood Containment Isolation Valves CF-V-19A and B were open and should be closed. The previous shift operated these valves to maintain proper pressure and level in the Core Flood Tanks. However, the shift did not properly restore this lineup to normal. The on-going operator verified no current activities with the subject valves and closed them.

The valves remained operable in that a containment isolation signal would have closed the valves. Further, the system was still sufficiently isolated to prevent Core Flood Tank pressure and level from being out of technical specification range requirements.

The inspector noted that these valves are not listed as "critical valves" (as defined by the licensee in their Independent Verification Program (IVP) (Administrative Procedure (AP) 1067). Only critical valves require independent verification. The applicable Operating Procedure (OP) called for independent verification on restoration to normal for these valves. This obviously was not completed. The licensee intends to make OP consistent with the AP 1067 program. These program and implementation weaknesses were addressed in NRC Inspection Report No. 50-289/87-19.

The inspector considered it noteworthy that the shift personnel alertly identified the incorrect positions for the subject valves. At the exit interview, licensee management reiterated their active

Institute of Nuclear Power Operations (INPO) sponsored program to thoroughly review human performance errors. The inspector had no additional comments in response.

2.2.3 Partial Loss of Instrument Air Control System

At 3:00 a.m. on January 20, 1988, while workers were working on WDL-V61 pipe repairs (see paragraph 3.2.5) in the auxiliary building, an instrument air pipe coupling in the same area broke causing low air pressure in the system. Control room operators responded by assuring back-up instrument and service air compressors (four total) were operating to make up to the system. Auxiliary operators responded by locally isolating the leak. As a result of this local isolation, the inlet valves to parallel letdown filters went closed causing a loss of letdown flow; and, as expected, the letdown relief valve lifted to the Miscellaneous Waste Storage Tank. This resulted in a slight increase in the effluent monitor noble gas channel for the auxiliary/fuel handling building ventilation system. This release was well within technical specifications limits. By 3:12 a.m., instrument air system pressure was within normal range and the reactor plant was relatively unaffected, except as noted above.

As a part of the interference removal work for WDL-V61, workers removed the air operator for WDL-V61 while instrument air (IA) remained lined up to the valve operator. This was intentional because isolating IA to the valve would have also isolated IA to the letdown isolation valves (see above) eventually causing these valves to go shut. Apparently, as the valve operator for WDL-V61 was moved, a coupling came loose and blew open. No personnel injuries occurred.

The auxiliary operator (AO) on watch was alert to effectively isolate the leak with a minimal impact on the plant. The inspector had no further questions on this matter.

2.3 Plant Operations Summary

Overall, operational performance remained at a high level. Operational mistakes were made and/or equipment malfunctions challenged licensed and non-licensed operators. However, these mistakes were licensee self-identified and operators were responsive to the challenges they received by taking immediate appropriate corrective action. Weaknesses in the implementation of the IVP program continued to be noted. Management attention and involvement in these and other facets of operations were noteworthy.

3.0 Maintenance/Surveillance - Operability Review

3.1 Criteria/Scope of Review

The inspectors reviewed selected activities to verify proper imple of the applicable portions of the maintenance and surveillance programs. The inspector used the general criteria listed under the plant operations section of the report. Specific areas of review are listed in Attachment 1. A more detailed review of equipment operability is also addressed.

3.2 Findings/Conclusions

3.2.1 Expansion Joints for the Reactor Building Emergency Cooling System Pumps

3.2.1.1 Background

An improper expansion joint for the "B" pump of the reactor building emergency cooling system (RR-P1B) was installed in September 1987. This problem was identified in NRC Inspection Report No. 50-289/87-24. The licensee prepared a satisfactory evaluation to justify continued operation; and, subsequently, they initiated a detailed review on how this occurred. The licensee provided the results of their review on January 13, 1988, as planned. The inspector also looked into the other relevant areas for the conduct of the particular mainten nce activity. The details are as follows.

3.2.1.2 Licensee Findings

The licensee's findings were documented in an internal memorandum dated January 12, 1988, from the Manager of Quality Assurance (QA) Modifications/Operations to Operations and Maintenance Director, TMI-1. A summary of licensee findings is addressed below.

(1) The expansion joint was procured from a different vendor as a replacement-in-kind. The plant engineering evaluation on this replacement-in-kind, as required per procedure PEP-2, Revision 3, dated March 30, 1987, "Plant Modification and Replacement-in-Kind Applicability/Scope," was not properly implemented prior to the installation. The review was completed on December 11, 1987, following identification of the problem.

(2) The end use for the several joints specified in the purchase requisition was not transferred on the purchase order. Therefore, the receipt inspection tags, which are based on the information in the purchase order, did not identify the specific application either. The lic-

ensee stated that this was a repetitive problem identified previously by the Quality Assurance Department (QAD) (Quality Deficiency report (QDR) No. DLL-037-87, May 1987). The corrective action was to implement an increased level of review by Materials Management and to provide for a computerized system that automatically transfers information from the purchase requisition to the purchase order.

(3) The receipt inspection did not discover the item which was mislabeled on the outer wrapping and, also, it was incorrectly tagged by the vendor. The licensee plans to instruct the Quality Control (QC) receipt inspectors to remove the outer wrapping when identification of the item is questionable.

(4) The job ticket (CK-212) indicated the incorrect model number of the expansion joint because the JT was prepared using an unapproved copy of the purchase requisition which did not include the end use (see above). The corrective maintenance (CM) procedure 1407-1, Revision 30, does not provide detailed direction for selecting replacement parts. QAD is working with maintenance to upgrade the applicable part of this procedure.

(5) The final phase of the installation was witnessed by a QC inspector and the documentation was appropriately verified. The QC inspector did not catch the error. He did not witness the removal of the old joint nor did he compare the new joint with other pump joints. The licensee is still reviewing different ways of enhancing the inspection process.

(6) Following the NRC inspection, engineering provided a justification for continued operation. However, no Materials Nonconformance Report (MNCR) was prepared as required. The MNCR was issued by QAD to document the condition.

3.2.1.3 <u>NRC Findings</u>

The inspector concurred with the above-noted licensee findings; but, based on this review, the inspector had the following additional observations. The inspector also reviewed JT No. CK-211 for a similar replacement on the "A" pump (RR-P1A).

(1) Plant Engineering Review (PEP-2) of replacement-in-kind was not performed as required per procedure PEP-2. This was identified by the licensee as an oversight. Based

upon past NRC experience, such reviews are performed as necessary and, therefore, this incident is considered an isolated case.

(2) JT No. CK-212 for the "B" expansion joint did not include the current revision of the installation procedure as required by CM 1407-1, Revision 30. There are a few blanks on the job ticket which are not filled in per the above-mentioned procedure. For example, employee number, the cause of failure, regulatory agency, outage codes, etc. Maintenance Procedure (MP) 1407-1 does not clarify as to the use of "N/A" and initials of an authorized person for the data which is not applicable.

The job ticket did not specify part number, model number, purchase order, etc. anywhere, except on the inspection tag, (which specifies the wrong model). On the work planning form, page 2, under part 2.1, "Materials," the purchase order number is not filled in. This incomplete information appears to have contributed to the wrong installation. JT No. CK-211 for the "A" pump had similar errors. (The final installation of the expansion joint on RR-P1B under JT No. CP-683 did include all the pertinent data.)

(3) The original Bill of Material RN-220 for the expansion joints specifies 160 psig as the design working pressure. The expansion joints installed on RR-P1A and 1B are rated at 150 psig. While the difference is insignificant and since the maximum discharge pressure is well within the limits, there is no safety significance; however, the discrepancy should have been addressed in the documentation review.

(4) The MNCR no. 0176-87, dated December 14, 1987, could have provided more specific information about the component identification, such as model, part number, etc. The corporate administrative procedure 1000-ADM-7215.01, Revision 1, "GPUN Material Nonconformance Reports and Receipt Deficiency Notices," as well as the Operational Quality Assurance Plan 1000-PLN-7200.01, Revision 1-00, emphasizes the importance of component identification.

(5) Job Ticket (JT) No. CK-211 for the installation of the joint on the pump RR-P1A included an inspection tag, which shows Model No. 500R, even though the actual installed joint is Model No. 150R. After the inspector identified this discrepancy to maintenance personnel,

the correct tag was then included in the job ticket. The normal work/document review process did not identify and correct this error.

(6) The work planning sheets of JT No. CK-211 specifies a torquing procedure (1410-Y-72). The procedure was voided on September 10, 1987, after the joint apparently was overtorqued during installation on September 8, 1987. The use of this procedure means excessive pressure on the rubber flange. On January 15, 1988, the licensee corrected this problem by loosening the bolts as required. (NOTE: The joint for the "B" pump was replaced with the correct model on January 13, 1988, without using the torquing procedure. The installation was witnessed by the NRC inspector.)

(7) For JT No. CP-683 conducted on January 13, 1988, the replacement joint on the RR-P1B showed 150MR model number embossed on the outside of the joint. (The discrepancy was identified by licensee QC inspectors witnessing the replacement.) The job ticket and other documents indicate 150R. Based on the discussion with the engineering and QC personnel, it was noted that the letter "M" stands for "Mercer," the supplier, and does not have any significance. The clarification of the model number discrepancy, if documented, would have been an enhancement.

The licensee management involvement and responsiveness was quite noteworthy. As soon as the problem was identified, the licensee took immediate actions to perform an engineering analysis to justify continued operation and then followed up with the detailed report. The licensee's review was sufficiently thorough in identifying several areas that need improvement as a result of the improper installation of the expansion joints.

3.2.2 Cleaning of the Station Battery Terminals

On January 14, 1988, during the mid-shift (11:00 p.m. to 7:00 a.m.), the inspector witnessed the subject activity being performed per JT No. CP-603, dated December 12, 1987. The specific purpose was to assess the quality of work, documentation, adequacy of planning, procedure, as well as procedural compliance. The scope of the activity involved cleaning of minor corrosion products and applying a thin layer of non-ox-id grease on the affected terminal. The following observations were made.

(1) The battery terminal activity was being performed by two craftsmen; however, the JT was not at the worksite. When questioned by the inspector, the JT was brought to the worksite shortly thereafter. Based on the inspector's review, worker activities were in accordance with prescribed instructions.

(2) The JT specified a CM 1420-EL-1, Revision 5, "Troubleshooting Unit 1 Battery Chargers." Revision 5 was not specified on the JT as required by MP 1407-1. The craftsman had marked "N/A" on this procedure since that it was obvious that it was a wrong procedure for the job. The JT was reviewed and formally approved this way prior to starting the job. The planning process, however, did not identify the wrong procedure specified on the ticket. The licensee reported that this was due to an administrative oversight.

(3) The JT package included a Preventive Maintenance (PM) Procedure E-72, Revision 6, "Station Batteries Terminal Connection Inspection." The JT did not specify the use of this procedure. This condition was discussed by the inspector with the shift maintenance supervisor. The copy of the JT received on January 20, 1988, by the inspector showed the correct procedure and was signed by the maintenance supervisor. The work activity, when witnessed on January 14, 1988, was in progress without formal review and approvals of the steps that were marked "N/A".

(4) The craftsman did not have the protective clothing as recommended in the E-72 procedure. Also, grease heating was completed in the battery room. The procedure did not include any grease heating requirements. This is indicative of weak implementation of occupational safety and health measures.

(5) The terminal clean-up activity was sufficiently defined in JT so as to preclude a safety issue. It was quite evident that the craftsmen had a very good knowledge of the scope of the work.

3.2.3 Liquid Waste Transfer Pump Repair

The corrective maintenance activity for JT No. CP-727 on WDL-P7B was to correct an oil seal leak and, at the same time, to perform preventive maintenance. The activity was witnessed by the inspector on January 29, 1988, during the mid-shift. The following observations were made.

(1) The work was in progress without JT No. CP-727 being at the worksite (Auxiliary Building - 281-foot elevation). The work was supervised by a maintenance supervisor.

(2) After the JT was located in the maintenance shop, the inspector's review of the JT indicated that none of the four procedures included in the work package had "verified" stamp assuring verification of the controlled work procedures and this was contrary to Administrative Procedure (AP) 1001G requirements.

(3) The JT did not indicate current revisions of any of the four procedures as required by MP 1407-1.

(4) One of the procedures required recording some data, such as pump coupling alignment, as well as certification of the torque wrench. Since the work package was not at the worksite, it was not clear how this data would be documented. The inspector reviewed the final package. The inspector was informed that the required data will be transferred from the working documents to the controlled copy following completion of the job.

3.2.4 Emergency Diesel Injector Assemblies

The emergency diesels have two injectors on each cylinder (24 on each diesel). The inspector observed two different types of injector assemblies on both diesels. The controlled technical manual shows only one type. The licensee was informed of this discrepancy immediately. Based on licensee's discussion with the vendor, it became clear that both types of injectors are acceptable; however, the type identified in the technical manual is not the latest model. Licensee also has requested the vendor to provide a formal change notice to the manual.

The diesels have been tested satisfactorily with both types of injectors installed. However, the apparent discrepancy was not identified during the recent inspection or during the installation. The injectors were accepted during the recent inspection based on vendor's certificate of compliance, as well as the part number identified on the purchase order. Physical verification of the parts against the approved applicable drawings or comparison against the installed units would be an enhancement to identify needed changes to vendor documents. The inspector had no additional comments on this matter.

3.2.5 Waste Disposal Pipe Repair

During the week of January 19-22, 1988, the inspector reviewed control room activities and the job ticket record associated with the repair to a leaking pipe associated with WDL-V61, Boric Acid Make-Up System Isolation Valve. Two job tickets were involved. one (JT No. CP-133) was for the pipe replacement and associated cutting and welding; and, the other (JT No. CP-685) was to set up special plant conditions and for restoration to normal. The special plant condi-

tions were needed because of the difficulty in isolating this section of pipe. A temporary modification was implemented to use a check valve as an isolation valve. Further a seventy-two hour action statement of the Technical Specifications (TS) (Section 3.2.2.c) needed to be entered because both alternate sources of boric acid make-up (in addition to the Borated Water Storage Tank) to the reactor coolant system had to be isolated.

Overall, licensee performance in implementing this work was quite good. The JT CP-685 methodologically set up the isolation boundaries, leak checked the affected pipe sections for effective isolation, and provided necessary precautions and limitations to operators and workers before actual replacement work was started. Leakage was detected and the tagout isolation was properly revised to reflect additional valve isolation. No TS action statements were violated. There was substantial quality control involvement and documentation on these activities with no significant findings. The safety evaluation for the temporary modification (TM No. 23, dated January 19, 1988) addressed appropriate potential material concerns with the temporary use of a bonnet for the check valve isolation function. Plant conditions were restored to normal.

The above-noted job ticket packages were still in "the review for completeness phase" two weeks after completion of work. The situation reflected similar administrative control discrepancies as noted elsewhere in this report. The operability/test block was not signed off by the operations department. This verification was accomplished by other means upon tag clearance immediately after job completion. A licensee management representative acknowledged the inspector's observation and indicated that the sign-off should have occurred when operations accepted the components back and relied on that action for operability. The licensee representative agreed to review this matter. The inspector had no additional comments.

3.2.6 Surveillance Observation

On February 5, 1988, the inspector witnessed surveillance and testing of the emergency feedwater pump per Surveillance Procedure (SP) 1300-3G A/B. The activity was being performed in accordance with the procedure and the communication among the personnel at the pump location, as well as the control room, was effective. Also, testing of the check valve MS-V-9 was witnessed per JT Nos. CP-856 and CP-869. Both activities were well done.

3.2.7 Secondary Plant Drain Cooler Foundation

During the routine plant tour, the inspector found a cracked foundations supporting drain cooler "A" in the turbine building, 305-foot elevation. On further examination, it was noticed that all

the baseplate bolts for that foundation were loose. The equipment is not safety-related, but its failure could result in a secondary plant transient. The matter was reported to the licensee representatives. The licensee representative initiated a review of the matter by forwarding it to plant engineering. No further NRC staff action is required on this matter.

3.3 Summary of Maintenance Observations

Paragraph 3.2.1.3(2) identified the installation of the wrong expansion joint on the reactor building emergency cooling system pump RR-P1B. Paragraphs 3.2.1.3(5) and (6) identified improper component identification, as well as improper torquing of the expansion joint, although installation of the joint was correct on the pump (RR-P1A). Paragraph 3.2.2(1) and (2) identified the cleaning of the station battery terminals without the job package at the site and without proper procedure change control. Paragraphs 3.2.3(1) and (2) identified the repair on the liquid waste transfer pump (WDL-P7B) again without the job package at the site and, also, without verified controlled procedures attached to the JT. These examples collectively indicate an apparent violation of 10 CFR Part 50, Appendix B, Criteria V and of licensee's NRC-approved Quality Assurance Plan, Section 3.0 and ANSI 18.7-1976, Section 5.3 (289/88-01-01).

At the interim exit meeting of February 1, 1988, the inspector discussed proposed licensee corrective actions on their findings (paragraph 3.2.1.2). Planned action at that time appeared not to be tangible in that most of the actions were oriented toward counseling personnel. As a result of the inspectors' observation and follow-up review, there appeared to be weak areas within related maintenance/administrative control procedures. These areas dealt with the control of applicable/non-applicable work instructions and on the control of procurement and replacement of parts in the plant to preclude such errors as noted above. Licensee management acknowledged the inspectors' comments.

The other inspector observations addressed in this section deal with attention to detail in strictly adhering to related administrative control in the maintenance area. No additional NRC staff action on these matters is warranted at this time.

3.4 Maintenance Summary

The licensee management involvement and responsiveness was noteworthy. The continued trouble-free operation of the plant without any unplanned maintenance outages is indicative of an overall effective maintenance program. The negative findings described above indicate procedural weaknesses in certain sub-areas of the maintenance program. However, the licensee, in addition to their own actions, plan to utilize independent resources to further strengthen their current maintenance program.

4.0 Reactor Building Sump Low Level Interlock Modification

4.1 Background/Existing Design

The inspector reviewed applicable documents that addressed the licensee's action on the reactor building sump low level interlock modification (BA No. 413916). This modification installed the necessary hardware for lowering the reactor building (RB) sump minimum level.

Valves WDL-V534/535, isolate or permit flow from the RB sump to the auxiliary building sump. The low level interlock closes this valve at a preset level and it was reset to 15 inches (previously 45 inches). The low level setpoint alarm (B-3-3 in the control room) was reset to 18 inches (previously 48 inches). The low level setpoint interlock function has been removed from level switches LS-116 D&E and are now provided by the LT-804 instrument channel. The basis for this change was to provide an instrument channel which could accommodate a variable setpoint capability. The level transmitters are safety grade and provide indication from 0-90 inches.

The purpose for this modification is to allow more effective use of the miscellaneous waste evaporator by allowing additional water to be processed, thereby reducing the cycling of the miscellaneous waste evaporator. The basis for the 15-inch setpoint is to maintain the 6-inch drain line from the RB sump to the auxiliary building sump covered with water. This will prevent RB gaseous atmosphere from entering the auxiliary building and causing a radioactive gas hazard during normal or emergency operations while still maintaining a Net Positive Suction Head for decay heat removal pumps on post-accident long-term recirculation.

4.2 Acceptance Criteria/Scope of Review

The purpose of this review was to:

- -- ensure changes have been reviewed and approved in accordance with 10 CFR 50.59;
- -- verify that design changes were reviewed and approved in accordance with Technical Specifications and established QC/QA controls;
- -- verify that design changes were controlled by approved procedures;
- -- verify that the licensee conducted a review and evaluation of test results in a reasonable time frame;
- -- verify operating procedure modifications were made and approved in reasonable time frame.

-- verify operator training programs were revised prior to the modification being declareo operable; and,

-- verify that as-built drawings were changed prior to the modifications being declared operable.

The inspector reviewed the following documentation during this inspection:

-- Final Safety Analysis Report, Sections 7.3.2.1 (c), 11(b), 6.4.2(e), and 6.2.2;

-- Technical Specifications Section 4.1;

-- Operating Procedure (OP) 1104-40, Revision 21, dated June 12, 1987, "Plant Sump and Drainage System;"

-- Surveillance Procedure (SP) 1302-5.25, Revision 10, dated May 6, 1986, "RB Sump Level;"

-- OP 1101-4, Revision 58, dated November 16, 1987, "Sumps and Drains;"

-- Administrative Procedure (AP) 1043, Revision 12, dated March 12, 1987, "Control of Plant Modifications;"

-- Operations Plant Manual, Section P-1, Revision 10;

-- Training Handout, dated December 9, 1987, "RB Sump Lo Level Setpoint Change;" and,

-- Installation Specification for WDL-V535 Low Level Interlock Modification, Revision 0, dated July 2, 1987.

The inspector verified that the control room annunciator and procedures were changed to reflect the setpoint change and that the operators were aware of the change. The inspector also inspected the RB sump monitoring cabinet, which houses the LT-804 instrument channel alarm and interlock module.

4.3 Findings/Conclusions

The inspector found all documents to be adequately reviewed and approved, as well as technically correct. Test data was found to be within the acceptance criteria. The written basis for the 10 CFR 50.59 safety evaluation was found to be technically correct and the questions necessary to determine whether the change constitutes an unreviewed safety question have been considered by the licensee. The design change was incorporated into applicable procedures and the FSAR. As-built drawings were verified to be updated prior to the modification being declared

operable. Training was completed in a timely manner. The Fire Hazard Analysis Report (FHAR) was found to be complete and accurate for this modification.

In conclusion, this modification was performed according to licensee administrative procedures, as well as Technical Specifications and QA/QC controls. Appropriate reviews and approvals were evident. Procedures reflected the setpoint change in a timely manner. The installation and turnover of this modification was found to be in accordance with procedures.

5.0 Safety Issue Management System Item Verification

5.1 Introduction

The inspector verified proper implementation, on a sampling basis, of licensee actions related to the below-listed NRC Safety Issue Management System (SIMS) items. The generic inspection approach for a SIMS item was:

-- research various licensee and NRC correspondence, including safety evaluation reports (SER's) to identify key assumptions, commitments, or other licensee actions to be taken to resolve the safety issue;

-- identify any additional items which need to be verified as delineated in the related NRC Temporary Instruction or other inspection procedures;

-- verify proper implementation of the items planned above; and,

-- assess licensee performance related to that implementation and related to dissemination of the issue and its resolution to licensee personnel who need to know, such as by procedural upgrading and training.

5.2 Reactor Vessel Overpressure Protection (SIMS No. A-26)

5.2.1 Background

The "Technical Report on Vessel Pressure Transients" in NUREG 0138, November 1976, summarizes the technical considerations and, also addresses the safety concerns related to the overpressure protection of pressurized water reactor (PWR) vessels at low temperature. The NRC letter dated August 11, 1976, requested the licensee (then Met-Ed) to design and install necessary modifications to mitigate the consequences of pressure transients at low temperatures. The letter also required the licensee to implement proper administrative controls as an interim measure until the hardware changes were completed. The licensee submitted a proposed action plan on October 15, 1976. Following NRC reviews and in response to additional

safety concerns, the licensee upgraded their action plan and also prepared other relevant documentation, including Technical Specification Change Request No. 74, dated March 13, 1978. The technical specification was revised through Licensee Amendment No. 56, dated March 28, 1980.

The safety evaluation of licensee's actions on this safety issue was performed by NRR on July 28, 1980, and licensee actions were found to be adequate, complete, and in compliance with the design criteria specified in 10 CFR Part 50, Appendix G. The design criteria for the system performance included: (1) necessary operator action; (2) single failure susceptibility of components; (3) testability of the system; and, (4) seismic and other safety-grade criteria (IEEE Standard 279 requirements).

The NRC safety evaluation also assessed licensee's evaluation of design basis events and mitigating controls.

5.2.3 Findings/Conclusions

The inspector verified that the licensee's design documents reflect all the hardware and procedural changes as identified in the above referenced safety evaluation. Key design aspects were: (1) an alarm if the system pressure exceeds 485 psig (manually enabled) when the temperature is less than 275 F; (2) an alarm if high pressure injection (HPI) valves are not racked-out at temperature less than 275 F; (3) an alarm associated with pressure levels and the system pressure; (4) an interlock on the core flood tank discharge valve so that they will not open until the system pressure is reduced to 600 psig; (5) an alarm indicating position of the pressurizer relief block valve; and, (6) testing provision for these functions. The cooldown and start-up procedures include the necessary steps to reflect the above-noted design.

The inspector also verified that the licensee's training program indicated that appropriate training was completed. Records on this matter were quite extensive.

All the requirements related to the overpressure mitigating system were satisfactorily completed.

5.3 Natural Circulation Cooldown (SIMS No. MPA-B-66)

5.3.1 Background

While St. Lucie Unit 1 was cooling down under natural circulation conditions on June 11, 1980, flashing of coolant produced a void in the reactor vessel upper head, forcing water into the pressurizer. The reactor was successfully brought to cold shutdown. Based on the NRC review of the event, MPA item B-66 was initiated. This MPA

requires that all pressurized water reactor's (PWR's) implement procedures and training programs to ensure the capability to deal with such events. Licensees were requested by Generic Letter (GL) No. 81-21 to provide an assessment of their facility procedures and training program, including:

-- a demonstration (e.g., analysis and/or test) that controlled natural circulation cooldown from operating conditions to cold shutdown conditions, conducted in accordance with plant procedures, should not result in reactor vessel voiding;

-- verification that supplies of safety-grade auxiliary feedwater are sufficient to support plant cooldown methods; and,

-- a description of plant training programs and the provisions of emergency procedures (e.g., limited cooldown rate, response to rapid change in pressurizer level) that deals with prevention or mitigation of reactor vessel voiding.

Licensee responses to this issue are embodied in the following letters: December 7, 1981; July 20, 1983; April 4, 1984; June 26, 1984; and, July 24, 1985. The NRC safety evaluation report (SER) was issued June 5, 1984. Previous inspections (NRC Report Nos. 50-289/84-31 and 85-17) confirmed licensee procedural commitments in response to this issue. However, both the NRC SER and inspection reports noted that additional analysis was needed in order to determine a maximum cooldown rate (beyond 10 degrees F/hr) for which a void would not form at the reactor vessel upper head (RVUH). The additional analysis was completed and documented in the latest licensee letter (July 24, 1985).

The results of this review are addressed below.

In conjunction with this inspection, the following procedures were reviewed on a sampling basis:

-- Operating Procedure (OP) 1102-11, Revision 69, effective August 18, 1987, "Plant Cooldown;" and,

-- The Abnormal Transient Procedure (ATP) Series 1210-1 to 10.

For further guidance to the inspector, NRC staff issued Temporary Instruction No. 2515/86, dated April 7, 1987. This inspection documents the review required by that TI.

5.3.2 Natural Circulation Demonstration

The licensee opted to address this concern by computer analysis, rather than actual plant test. Licensee preliminary analysis was reviewed and found to be acceptable as noted in the NRC scaff SER.

The revised analysis (letter, dated July 24, 1985) identified that a void would not form at the RVUH area provided that the cooldown was limited to no greater than 50 degrees F/hr and reactor coolant system (RCS) pressure and temperature were held above a minimum pressure versus temperature curve (MPTC). This curve is a combination of the natural circulation cooldown curve and fuel pin compression curve for natural circulation which already existed as a plant limiting condition. There is a section of the pressure versus temperature domain that the natural circulation curve is more restrictive than the fuel pin compression curve.

The inspector noted that OP 1102-16, "Natural Circulation Cooldown," was cancelled and its control measures were properly incorporated into a revised OP 1102-11, "Plant Cooldown." In particular, the latest MPTC was properly reflected in OP 1102-11.

However, it was noted that the Abnormal Transient Operating Guide (ATOG) procedure 1210-10, Revision 15, effective January 13, 1988, "Abnormal Transients Rules, Guides, and Graphs," did not address the MPTC for natural circulation, nor did it address the special measures to prevent a void formation in the RVUH. Upon further review, the licensee's letter of June 4, 1984, introduced terminology that the natural circulation restrictions established by analysis apply to non-emergency situations. Non-emergency situations are not clearly defined. Further, this appears to conflict with emergency procedure scope in that a loss of reactor coolant pumps (RCP's) is covered by emergency procedure; namely, 1202-14, "Loss of RC Flow/RC Pump Trip," along with the ATOG series. Also, the staff's SER does not make the above-noted distinction. It would appear that the reality of how the licensee would manage all design basis events involving natural circulation (NC) is not accurately reflected in the staff's SER.

Further, the following ATOG procedure would direct the operator to a natural circulation cooldown and/or the use of Operating Procedure (OP) 1102-11, Revision 70, effective December 11, 1987, "Plant Cooldown:" 1210-5, Revision 12, effective December 11, 1987, "OTSG Tube Leak/Rupture;" 1210-6, Revision 9, effective December 30, 1987, "Small Break LOCA Cooldown;" and, 1210-9, Revision 11, effective December 30, 1987, "HPI Cooling-Recovery from Solid Operations." Paragraph 3.18 of ATP 1210-5 directs the use of OP 1102-11 but, also, indicates "do not limit cooldown to 50 F/hour." Paragraph 2.13 of ATP 1210-6 assumes reactor coolant pumps (RCP's) are off, does not reference OP 1101-11, and requires that a cooldown of 100 F/hour be established. Paragraph 2.10 of ATP 1210-9 directs the use of OP 1102-11 without exceptions or restrictions. It would appear that the actions of ATP's 1210-5 and 1210-6 would lead to a void formation at the RVUH during a design basis event in light of the licensee's analysis on natural circulation cooldown. That void formation is not a safety issue as long as it does not interfere with decay

heat removal. More importantly, no notes/cautions exist in ATP 1210-5 and 1210-6, nor do general rules exist in 1210-10 that would caution the operator about this situation and/or provide the guidance of OP 1102-11 on detecting and collapsing the void before it interfered with natural circulation for ATOG actions.

The inspector consulted with the NRR project manager and lead technical reviewer on this issue for NRR. The draft SER recognizes that certain design basis events may cause voids to form in the RVUH. This information apparently was not transcribed into the final NRC staff SER. Beyond this administrative oversight, the NRR representatives indicated that sufficient guidance must exist for all design basis events in order for the operator to recognize and take action to collapse the void in the RVUH if it were to form on a natural circulation cooldown.

The inspector concluded that the ATOG procedures do not adequately provide this guidance either directly or by reference in ATP 1210-5, "OTSG Tube Leak/Rupture," and 1210-6, "Small Break LOCA Cooldown."

The inspector also agreed to further meet with licensee representatives on this issue in the next inspection period. The results of that meeting will also be documented. Accordingly, SIMS No. MPA-B-66 will remain open but all actions of the related Temporary Instruction (TI) 2515/86 were completed and the TI will therefore be closed.

5.3.3 Safety-Grade Sources of Water for Natural Circulation Cooldown

The NRC SER accepted the licensee response to this concern. Essentially, the emergency feedwater system (EFW) (now safety grade) would be used for natural circulation cooldown. Several sources of water are available. Condensate storage tanks (CST's), condenser hotwell, on-site one-million gallon storage tank, and an unlimited supply of river water from the reactor building emergency cooling water system (last resort). The inspector independently confirmed plant configuration sources as noted in the licensee's response to this item.

5.3.4 Training Program and Implementation

The licensee's training program description in their letter response, dated December 7, 1981, is minimal. Essentially, it states that operators would be made aware of the St. Lucie event, procedural actions would be reviewed at the 1982 simulator sessions, and training on the then applicable procedure would be accomplished. This was acceptable by the NRC staff's SER.

The inspector discussed this item with training department personnel involved in both classroom and simulator training. The inspector also interviewed four licensed operators associated with two shifts to ascertain their knowledge of the event and licensee procedures for natural circulation cooldown.

In addition to the above, the inspector reviewed selected sections of the following documents.

-- Exhibit 2 to 7811-PGD-2613, Revision 4, "Licensed Operator Requalification Training"

-- 7811-PGD-2611, Revision 5, "Replacement Operator Training Program"

-- Lesson Plan No. 11.2.01.281, Revision 0, "Natural Circulation"

-- Lesson Exercise 11.7.01.015, Revision 0, "Loss of Off-Site Power"

-- Lesson Exercise 11.7.03.020, Revision 0 (Replacement Operator Training Program), "Loss of Off-Site Power"

Based on this review, the inspector concluded that the licensee provides appropriate training to their operators on this event but, more importantly, the training was substantial on how to conduct a natural circulation cooldown with respect to the concerns and restrictions imposed subsequent to the St. Lucie event. Operator interviews confirmed general familiarity with the St. Lucie event, but it also confirmed familiarity with the licensee's procedural requirements. The in-plant training on natural circulation was extensively reviewed by NRC staff during the TMI-1 Restart process of October-December 1985. The inspector had additional observations in this area.

Certain lesson plans/exercises still reflected the cancelled procedure OP 1102-16. The training representatives were aware of that and plans exist to revise these documents to reflect the current procedure.

The in-classroom lesson plan for operator requalification program was not set on a specified frequency. Plans exist to do so on a frequency yet to be determined by the licensee. A new cycle of training on this plan is set for 1988 Cycle 1. It was last completed in 1984 and during the 1985 in-plant training. Simulator training on natural circulation is usually provided during the loss of off-site power drills.

The inspector had no residual concerns in the training area for this SIMS item.

5.4 SIMS Issues Summary

For the two SIMS issues reviewed above, the licensee appropriately translated generic licensing actions into plant specific actions. Specifically, they properly incorporated requirements and commitments into procedural requirements and/or training plan/exercise elements, except as noted above for the residual issue dealing with natural circulation as described above.

6.0 Licensee Actions on Previous Inspection Findings

6.1 Introduction

For these items listed below that were previously identified violations, the inspector reviewed the licensee's response and corrective to:

-- verify the licensee responded in a timely manner;

-- verify measures taken to correct the item and avoid recurrence were completed and within the specified time frame; and,

-- verify licensee commitments were completed.

6.2 (Closed) Violation (289/86-06-01): Failure to Properly Implement Facility Procedures

The licensee responded to this violation on August 11, 1986, and provided a supplemental response on August 26, 1986. Details of NRC evaluation of the licensee's responses are documented in NRC Inspection Report No. 289/86-17. The licensee further responded to this violation in a letter dated March 5, 1987. A meeting between the licensee and NRC was held on June 8, 1987, to discuss the licensee's responses to this violation. The details of this meeting are provided in NRC Inspection Report No. 289/87-08.

Licensee corrective actions have been verified in these previous inspection reports. Due to an oversight in Inspection Report No. 289/87-08, this item was not closed. This item is, therefore, being closed in this inspection report.

6.3 (Closed) Violation (289/86-12-02): Inadequate Safety Evaluation for Change to Procedures Described in FSAR and Modifications and Single Failure Analysis for Back-Up Instrument Air

The comprehensive single failure analysis on the air supply system to the emergency feedwater valves is discussed in NRC Inspection Report No. 289/87-06 and currently remains as an open item (289/87-06-08).

The emergency feedwater pump Surveillance Procedure (SP) 1301.11.42 has been revised and reviewed in NRC Inspection Report No. 289/87-09. Additional review of safety evaluations and design verification was completed in NRC Inspection Report Nos. 289/87-08 and 87-19.

The inspector reviewed the revised procedure for installation of temporary shielding (9100-IMP-3282.01, Revision 2, dated November 3, 1986) and performed an in-plant review to inspect temporary shielding for conformance to this procedure. This revision has incorporated all the corrective measures discussed in the licensee's response to the violation.

The inspector verified that temporary shielding was in conformance with Radiological Controls Procedure 9100-IMP-3282.01. Upon discussions with cognizant licensee personnel, it was determined that temporary shielding is reviewed/approved on a semi-annual basis. Approved methods exist for determining the acceptability for continued use of existing temporary shielding. It is noted that the use of temporary shielding is not excessive.

The determination to maintain a temporary shielding installation is made by the radiological engineer who is responsible for the installation. Technical Functions Work Requests are written for installation existing beyond one refueling outage, currently eighteen months, and a cost/benefit analysis is done and the installation is dispositioned in accordance with the results of this analysis. The inspector had no further questions.

This item is closed.

6.4 (Closed) Violation (289/86-12-08): Failure to Take Prompt Corrective Action on Known Conditions Adverse to Quality

The conditions adverse to quality were repetitive examples of out-of-specification log readings and of drawing deficiencies. The inspector reviewed auxiliary operator (AO) log sheets for the period December 28, 1987, to January 9, 1988. Out-of-specification data for important-to-safety systems were properly identified on the log sheets. There was not an excessive amount of log entries which were out-of-specification. Appropriate reviews of the log sheets were evident. This portion of the violation is closed.

The control room drawing deficiencies and GPUN drawing procedure revision have been reviewed and closed in NRC Inspection Report No. 289/87-25.

Corrective measures taken are adequate and have been implemented in a timely manner. This item is closed.

6.5 (Closed) Violation (289/86-17-02): Failure to Follow Procedures Associated with Engineered Safeguards Actuation System Testing

In a meeting of June 8, 1987, this item was discussed. The details of this meeting and topic were documented in NRC Inspection Report No. 289/87-08.

The inspector verified that crew members were briefed and understood the minimum requirements for independent verification. Corrective actions were completed in a timely manner and adequately addressed the violation.

6.6 (Closed) Violation (289/86-17-11): Source Range Instrumentation Inoperable Without a Proper Safety Evaluation

The licensee provided a response to this violation in a letter dated March 5, 1987. The corrective actions provided in that letter were determined acceptable in NRC Inspection Report No. 289/87-08.

The inspector verified that the corrective actions discussed in the licensee's response have been implemented. The individuals involved have been instructed in the proper methods of altering plant systems. All Instrument and Control (I&C) technicians have been instructed in the proper use of AP 1013, Revision 25, effective date October 28, 1987, "Bypass of Safety Functions and Jumper Control." A similar problem with the improper implementation of AP1013 was identified in NRC Inspection Report No. 50-289/87-19. The licensee is currently preparing a training seminar on the general use of AP 1013 which will provide several days of classroom instruction. This seminar should be provided within the next month. Follow-up actions will be reviewed under violation 289/87-19-01.

Difficulty in verification of training records was experienced. The licensee could not locate the training records of the I&C technicians for AP 1013 (specific to this violation). The inspector verified training records for past violations were readily available. The I&C supervisor then agreed to re-instruct the I&C technicians on the proper use of AP 1013 and provide verification of such training. This has been verified.

The corrective actions taken are adequate and have been verified to be complete. This item is closed.

6.7 (Closed) Violation (289/87-09-04): Pressurizer Platform Installation Design Modification Failure to Properly Review and Verify Design

The inspector verified GPUN Specification SP 1101-32-022, Revision 1, "Nuclear Grade Fiberglass Insulation Systems," incorporated the necessary changes to avoid further occurrences of this type. The corrective actions were incorporated in a timely manner and all commitments were completed.

The changes incorporated included sections to clarify thermal growth and obstruction clearance requirements. An evaluation is now required prior to installation of any replacement insulation which reduces clearances between adjoining or adjacent components. Quality control verification is also required. Drawings showing items to be insulated that could produce interferences due to thermal movement or growth must also be provided.

These measures are adequate. This item is closed.

6.8 (Closed) Unresolved Item (289/87-09-05): Licensee Incorporate Four Channel Level Checks into Site Procedures

This issue concerned the fact that not all four channels of Once-Through Steam Generator (OTSG) start-up and operating range level transmitters for the new Heat Sink Protection System (HSPS) are provided with indication in the control room. The licensee was considering discontinuing the four channel comparison. However, there were several level transmitters problems since initial installation. Further, NRC staff gave the licensee a position that four channel checks were needed to meet applicable TS. Accordingly, the level comparison was incorporated into a formal Surveillance Procedure (SP) 1301.4.1, Revision 39, "Weekly Checks."

The inspector had reviewed the data compiled via this document in NRC Inspection Report No. 50-289/87-24 and it was satisfactory. The licensee plans to continue recording and comparing the "blind" OTSG level transmitters until permanent corrective action is taken. At present, the inspector's concern about the "blind" OTSG level transmitter channel checks has been resolved and this item is closed.

6.9 (Closed) Unresolved Item (289/87-13-01): Licensee to Resolve Improper Installation of Anchor Bolts in the Chill Water System for Control Building Ventilation System

Licensee actions related to this item were reviewed in a region-based inspection (Report No. 50-289/87-18). However, due to an administrative oversight, this item was not closed in the report. Accordingly, the item is closed administratively by this report.

6.10 (Open) NRC Temporary Instruction 2500/26 (289/25-00-26): NRC Bulletin 87-02, Fastener Testing

The licensee completed selection and testing of a representative sample of fasteners (studs, bolts, and nuts) to comply with the requirements of NRC Bulletin 87-02.

The inspector participated in the initial selection process, but the licensee later determined that some bolts/nuts selected for the "not important to safety" classification were not traceable. Another selec-

tion of bolts/nuts was made where material identity was known. The licensee subsequently informed the inspector of this action and also noted this in the initial response to the bulletin in a letter dated January 15, 1988.

The inspector reviewed the types of bolts/nuts selected for the alternate sample and concluded that the sample selection was made in an appropriate manner. This item remains open pending review of the licensee warehouse program and NRR comment on the test results.

6.11 NRC Information Notice (IN) No. 85-38 (289/85-IN-38): Loose Parts Obstruct Control Rod Drive Mechanism

This IN was issued to alert Babcock & Wilcox (B&W) licensees of a potential problem with control rod binding due to failure of a leaf spring in one control rod drive mechanism (CRDM) at Davis-Besse. The potential safety issue was also addressed in an NRC letter dated December 4, 1985 (DD 85-19) to a 10 CFR 2.206 petitioner who called for a shutdown of all B&W-designed reactors. The petitioner's request was denied based on the NRC staff's SER attached to the above-noted letter. The licensee at TMI had used a different procedure/process for control rod unlatching and initially determined that the inspection recommended by the IN was not necessary. However, as noted in the subject SER, the licensee committed to verify the position of their leaf spring at the next refueling outage.

Subsequently, the licensee, with B&W assistance, performed the recommended inspection of the leaf spring assembly. This was performed in April 1986 during the "5M" eddy current outage. The inspector reviewed the results of the inspection and discussed the evolution with licensee engineering personnel. The work was completed in accordance with JT No. CH-254. There were no problems similar to that for the Davis-Besse facility. The inspector concluded that the licensee had taken adequate corrective action in response to IN No. 85-38 and the NRC staff's request as noted in the subject SER.

6.12 Past Inspection Findings Summary

The licensee took appropriate corrective action and/or fulfilled commitments made in response to the above-noted violations. Generic licensing actions were effectively translated into plant specific action, along with meeting commitment dates. Licensee representatives cooperated fully with NRC staff on the bolt sample selection process for NRC Bulletin No. 87-02. The licensee provided necessary information to resolve the subject unresolved items.

7.0 Exit Interview

The inspectors discussed the inspection scope and findings with licensee management at an interim exit meeting on February 1, 1988, on the maintenance area and at a final exit meeting conducted February 9, 1988. Senior licensee personnel attending the final exit meeting included the following:

T. Broughton, Operations and Maintenance Director, TMI-1
J. Colitz, Plant Engineering Director, TMI-1
J. Fornicola, Manager, Quality Assurance Modification/Operations
H. Hukill, Director, TMI-1
D. Shovlin, Plant Material Director, TMI-1
C. Smyth, TMI-1 Licensing Manager
P. Snyder, Manager, Material Assessment

A representative of the Commonwealth of Pennsylvania, Ajit K. Bhattachayyra, also attended the 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 Section 6.

ATTACHMENT 1

NRC INSPECTION REPORT NO. 50-289/88-01

ACTIVITIES REVIEWED

Plant Operations

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

-- Areas outside the control room

-- Selected licensee planning meetings

-- Plant transients of January 18, 22, and 28, 1988

-- Partial loss of instrument air on January 20, 1988

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

Day/Date	Time
Wednesday, 1/13/88	3:00 a.m. - 7:00 a.m.
Week of 1/18-22/88 2.5 hours between	6:00 a.m. - 7:00 a.m.
Sunday, 1/24/88	12:30 p.m. - 2:00 p.m. 8:00 p.m. - 10:30 p.m.
Friday, 1/29/88	3:00 a.m. - 7:00 a.m.

Maintenance/Surveillance

-- Expansion Joints for the Reactor Building Emergency Cooling System Pumps - JT Nos. CK-211, CK-212, and CP-683

-- Cleaning of the Station Battery Terminals - JT No. CP-603

-- Liquid Waste Transfer Pump (WDL-P7B) Repair - JT No. CP-727

-- Waste Disposal Pipe Repair - JT Nos. CP-133 and CP-685

-- Surveillance Observations - JT Nos. CP-856 and CP-869

Reactor Coolant System (RCS) Leak Rate

The inspector selectively reviewed RCS leak rate data for the past inspection period. The inspector independently calculated certain RCS leak rate data reviewed using licensee input data and a generic NRC "BASIC" computer program "RCSLK9" as specified in NUREG 1107. Licensee (L) and NRC (N) data are tabulated below.

TABLE

RCS LEAK RATE DATE
All Values GPM

DATE/TIME DURATION	L_G	N_G	(NUREG 1107) N_U	CORRECTED N_U	L_U
1/14/88 0220 2 Hours	0.4279	0.43	-0.07	0.03	0.0365
1/17/88 0030 2 Hours	0.4071	0.41	0.03	0.13	0.1346
*1/19/88 0917 2 Hours	-0.1845	-0.18	-0.93	-0.83	-0.8232
1/21/88 1803 2 Hours	0.3631	0.36	0.12	-0.02	-0.0143
1/22/88 1542 2 Hours	0.4616	0.48	0.27	0.37	0.3581
1/26/88 1530 2 Hours	0.1817	0.18	0.01	0.11	0.1187
2/4/88 0048 2 Hours	0.4707	0.47	0.01	0.11	0.1151

G = Identified gross leakage
L - Licensee calculated
U = Unidentified leakage
N = NRC calculated

*Declared invalid by licensee due to water addition to make-up tank.

Columns 2 and 3, 5 and 6 correlate ± 0.2 gpm in accordance with NUREG 1107. N_u is corrected by adding 0.1044 gpm to the NUREG 1107 N_u due to total purge flow through the No. 3 seal from RCP's.