U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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Licensee:	New York Power Authority
Facility:	Indian Point 3 Nuclear Power Plant
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EXECUTIVE SUMMARY

Indian Point 3 Nuclear Power Plant
NRC Integrated Inspection Report No. 50-286/99-08

This integrated inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covered a six-week period of resident inspections, and included inspections by region-based specialists in radiation protection, in-service inspection, and physical security.

Operations:

Operators correctly diagnosed and stopped an uncontrolled loss of approximately 1100 gallons of reactor coolant system (RCS) inventory, recovered the normal RCS level, and prevented a rise in the bulk RCS temperature. The failure to implement adequate system test and configuration controls while the residual heat removal system was in operation is a violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control" (NCV 50-286/99-08-01) (Section O1.1).

Operator response was well focused and effective in diagnosing and isolating a high pressure leak in a failed reactor coolant system flow transmitter inside containment. They also made timely entries into the appropriate response procedures and recognized the correct conditions for classifying an Unusual Event. Emergency Plan responders made the required local, state, and NRC notifications properly and on time. Operators promptly diagnosed control room indications and directed auxiliary operators inside containment to isolate the leak before a significant volume of RCS inventory was lost. The resulting consequences to personnel and equipment inside containment were minimized, and the subsequent analysis by the post-transient review group was timely and comprehensive (Section O1.2).

The licensee established a good program for performing outage risk assessments. The pre-outage schedule review was thorough and resulted in the incorporation of numerous schedule modifications to reduce risk. Appropriate contingency planning was performed for conditions where system redundancies would be reduced. In one instance, the operations department did not perform a daily risk assessment following a significant plant condition change, as recommended by the procedure (Section O2.1).

The licensee's policy for excess overtime was appropriately managed and restricted for operators during the refueling outage. The excess overtime authorized during the last two weeks of the outage was appropriately justified and controlled by station management prior to operations personnel exceeding the normal overtime limits (Section O7.1).

Maintenance:

Maintenance activities observed were conducted satisfactorily and in accordance with applicable maintenance and administrative procedures. The licensee appropriately monitored performance of equipment within the scope of the maintenance rule (Section M1.1).

Executive Summary (cont'd)

Routine surveillance tests were conducted appropriately and in accordance with procedural and administrative requirements. Test and performance monitoring personnel maintained a good level of communication and coordination with control room operators during observed surveillance tests. Test instrumentation was observed to be within the required calibration periods and all test acceptance criteria for operability were met (Section M1.2).

The pre-job briefing for the Safety Injection/Station Blackout integrated functional test was thorough, addressed industry events, and emphasized self-checking and peer checking techniques. Performance of the test had the appropriate level of management oversight. The test coordinator made a good configuration control effort during the pre-job briefing by ensuring that an isolation boundary was maintained between the reactor coolant system and the refueling water storage tank to preclude a potential leak path (Section M1.3).

The second ten-year interval in-service inspection program was satisfactorily completed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI. No indications of structural defects were found during visual inspection of the reactor vessel and removable internals. Three steam generator tubes inspected revealed indications due to loose parts wear, but the indications were considered arrested at their current acceptable depth, without a potential for further growth (Sections M1.4 - M1.8)

The inspector identified several discrepancies during walkdown inspections of plant equipment during the refueling outage. The discrepancies appeared to be minor in nature, but were referred to the licensee for evaluation and resolution prior to the end of the outage (Section M2.1).

Poor communications and inadequate work control guidance led to a condition prohibited by the plant technical specifications. In loosening the pressurizer safety relief valve body-to-bonnet bolts, the licensee compromised the operability of all three valves prior to establishing a suitable pressure relief path as required by technical specification 3.1.A.2. This failure to comply with the technical specification is a violation of NRC requirements **(NCV 50-286/99-08-02)** (Section M8.1).

Engineering:

The licensee did not verify that a reactor coolant system flow transmitter procured as commercial-grade material could satisfactorily perform the pressure boundary safety function. The flow transmitter subsequently failed at high system pressure and caused an RCS leak that required operators to declare an Unusual Event. This is a Severity Level IV Violation of 10 CFR 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services." **(NCV 50-286/99-08-03)** (Section E2.1).

Plant Support

The licensee effectively implemented radiological controls during the current refueling outage. Access controls to radiologically controlled areas were applied effectively, and appropriate occupational exposure monitoring devices were provided and used. Personnel occupational exposure was maintained within applicable regulatory limits and as-low-as-is-reasonably-achievable. Also, the radiation work permit program was properly implemented (Section R1.1).

In response to unanticipated reactor containment building conditions, the radiation protection department did not provide effective guidance to preclude excessive personnel facial contaminations. Emergent issues associated with the reactor coolant temperature and the unavailability of a containment fan cooling unit were not well integrated in the radiation protection plan or the ongoing outage radiation protection work (Section R1.2).

Overall, the licensee implemented effective surveys, monitoring, and control of radioactive materials and contamination. Occupational exposure was maintained as-low-as-is-reasonably-achievable, self-assessment and corrective action processes in the area of radiation protection were effective (Section R1.3 - R7.1).

Security and safeguards activities with respect to alarm station controls, communications, and protected area access controls were effectively implemented and met licensee commitments and NRC requirements. The level of management support was adequate to ensure implementation of the security program, and the security audit and self assessments were effectively implemented (Sections S1 - S7).

The licensee's security audit program indicated that the program was being properly administered. In addition, a review of the documentation applicable to the self-assessment program was being effectively implemented to identify and resolve potential weaknesses (Section S7).

TABLE OF CONTENTS

Table of Contents (cont'd)

ATTACHMENTS

Report Details

SUMMARY OF PLANT STATUS

At the beginning of the inspection period on September 14, 1999, the Indian Point 3 plant was in cold shutdown, and at the beginning of the tenth refueling outage (RO-10). On September 17, when the reactor vessel head was de-tensioned, the plant entered the refueling mode. The plant entered the defueled mode after all fuel was removed from the reactor vessel on September 21. On October 5, the licensee commenced refueling operations, and all fuel was reinstalled in the vessel by October 8. The plant entered cold shutdown on October 10 and hot shutdown on October 17. The reactor achieved criticality on October 20, and low power core physics and main turbine testing began. The main generator was synchronized and latched to the grid on October 21, and a steady power ascension was commenced. At the end of the inspection period on October 24, plant power was at approximately 80% and increasing.

I. OPERATIONS

O1 Conduct of Operations

O1.1 Inadvertent Loss of Reactor Coolant During Residual Heat Removal Operations (NCV 50-286/99-08-01)

a. Inspection Scope (71707, 61726)

The inspectors reviewed the licensee's immediate response and short term corrective actions associated with an inadvertent loss of reactor coolant while the residual heat removal (RHR) system was in operation for core cooling. The loss of reactor coolant occurred while the licensee was performing a refueling surveillance test of the safety injection system.

b. Observations and Findings

At 9:40 p.m. on October 10, 1999, while performing test procedure 3PT-R003A, "Safety Injection Test of Recirculation Switches," (Revision 16), the licensee inadvertently drained approximately 1100 gallons of reactor coolant into the recirculation sump inside containment. The loss occurred when operators opened manual valve V-1803 that directed the recirculation pump minimum flow line to the recirculation sump. Prior to opening V-1803, operators did not close two motor-operated valves (MOV-1802A and MOV-1802B) which would have isolated the RHR system from the recirculation pump discharge piping. With all three valves open, a flow path was created from the RHR system to the recirculation sump. Reactor coolant system (RCS) level dropped approximately 1-1/2 feet before the leak was isolated and the level recovered. RCS temperature remained constant at 123°F throughout the incident.

At the time of the incident, the primary plant was in cold shutdown, RHR was in operation, all four 480 volt safeguards buses were in service, and all three emergency diesels were available in standby. The reactor vessel head was landed, and containment integrity was relaxed with the equipment hatch removed. Control room operators had been aware of the on-going test activities, and immediately investigated the RCS level decrease. Some confusion existed when the level started to drop because operators had just secured the containment purge system for the test, and they initially thought that securing the purge caused the level perturbation. However, as RCS level continued to decrease unexpectedly, operators reviewed all on-going plant activities in an effort to diagnosed the situation. Within approximately 10 minutes the operators concluded that a drain path was created when V-1803 was opened. Consequently, they immediately closed both MOVs from the control room and initiated make-up to the RCS using the charging system. Normal RCS level was recovered at 67ft - 2in within approximately one hour.

The licensee initiated DER 99-02254 to address this event and to document corrective actions. The following day, a Post-Transient Review Group (PTRG) was formed to analyze and determine the cause of the incident, to identify the sequence of related events, and to make recommendations to station management on short term corrective actions needed to safely proceed with planned evolutions. The PTRG reviewed the 3PT-R003A procedure and determined that a series of previous revisions had changed the test sequence for the recirculation switches, and the order of valve manipulations without specifying or maintaining sufficient configuration controls to prevent leaving the RHR isolation boundary open to the drain path that was created when V-1803 was opened.

3PT-R003A was an integrated functional test that followed multiple maintenance activities completed during the outage. The test was primarily intended to verify the correct actuation functions of eight switches on the main control board that control room operators would use to transition from the injection cooling phase of a design basis accident to the recirculation phase. In previous outages, the switches were tested in numerical sequence. However, for the current outage, the test sequence was revised by separating the test into two segments so that recirculation switches #3 (removes unnecessary loads from the emergency diesels and trips both RHR pumps) and #5 (initiates low head recirculation flow) could be tested early when the reactor was completely defueled, and when RHR cooling would not be temporarily disabled by the test. However, the revised test sequence also changed the order and timing of valve manipulations. The previous sequence opened MOVs-1802A/B for the switch #4 test (initiates internal recirculation flow), and closed them before opening V-1803. However, Revision 16 to the procedure did not direct closure of the MOVs before opening V-1803.

The inspector noted that the potential for a direct flow path from the RCS through the RHR piping and into the recirculation sump had existed since Revision 10 to 3PT-R003A was issued in 1995. That revision removed the step from the procedure which verified that the two MOVs were closed, and relocated it to a table (Attachment 3) which listed multiple valves repositioned in the post-test restoration lineup for the #4 switch test. The step to perform the Attachment 3 lineup was initiated by the procedure immediately after the step to throttle open V-1803 (its pre-test position for switch #5), and required that control room operators close the MOVs from the main control board. Since V-1803 was operated locally, the control room operators were able to close the MOVs sooner than an auxiliary operator was able open V-1803. That delay had previously prevented the open drain path. However, Revision 16 to the procedure also added several post-test checks after opening V-1803 and prior to initiating the Attachment 3 line-up. The additional steps allowed sufficient time for an auxiliary operator to open V-1803 before control room operators closed the MOVs.

Prior to refueling outage RO-10, portions of the 3PT-R003A test were performed as a special evolution under administrative procedure AP-19.1, "Infrequently Performed Tests and Evolutions." The AP-19.1 requirements imposed an additional level of oversight and configuration controls designed to assure that special evolutions were properly performed. However, when the safety injection (SI) test sequence was altered for the current outage, the AP-19.1 controls were made non-applicable since switches #3 & #5 were tested when RHR was not in operation. Also, approximately 4-1/2 hours before the event, the licensee officially suspended the defense-in-depth controls for RCS inventory (DID-R10-013). At the time of the event, plant operations procedure POP-4.2, "Operation Below 10% Pressurizer Level with Fuel in the Reactor," was in use. It contained contingency actions for a loss of RCS inventory, but its instructions were only generally aimed at isolating letdown flow and initiating make-up. Although operators took the correct actions to isolate the leak and initiate RCS make-up, they did not have an off-normal operating procedure (ONOP) available that covered a loss of RCS inventory from above the mid-loop level and below 200°F with RHR in operation. Also, since no alarms occurred during the draindown, the operators were not directed to an alarm response procedure for guidance.

This event could have been prevented, and resulted from an inadequate test procedure and inadequate test controls to prevent leaving the RHR system boundary open when a drain path was created directly to the recirculation sump. Inadequate test reviews by operations and engineering personnel were also significant contributors. The technical review requirements of administrative procedure AP-3, "IP3 Procedure Preparation, Review, and Approval," were not adhered to in that the scope of the changes to the latest revisions of 3PT-R003A were not correctly analyzed to determine the appropriate type of technical review necessary. The reviewers did not use a piping and instrumentation diagram to verify the adequacy of the test boundary or the detailed sequence of valve manipulations.

Following this event, operations placed all further SI functional tests during the outage under AP-19.1 controls as special evolutions to assure that the proper system configuration and boundary controls were in place. The licensee also performed plant simulator procedure validations to identify potential configuration or test sequence problems, and initiated an ACTS (Action/Commitment Tracking System) item to develop specific operator response instructions for a loss of RHR/RCS inventory in cold shutdown.

c. <u>Conclusions</u>

Operators correctly diagnosed and stopped an uncontrolled loss of approximately 1100 gallons of reactor coolant system (RCS) inventory, recovered the normal RCS level, and prevented a rise in the bulk RCS temperature. The failure to implement adequate system test and configuration controls while the RHR system was in operation is a violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control." However, because this incident was promptly identified by the licensee, and appropriately entered into the corrective action system, this Severity Level IV Violation will be treated as a Non-Cited Violation in accordance with Appendix C of the NRC Enforcement Policy **(NCV 50-286/99-08-01)**.

O1.2 Unusual Event Due to Reactor Coolant System Leak

a. Inspection Scope (71707)

Plant operators declared an unusual event after a high pressure leak of primary coolant in excess of 10 gallons per minute (gpm) was identified. The inspector responded to the event and evaluated the licensee's actions to diagnose and isolate the leak, and the subsequent actions to diagnose its root cause.

b. Observations and Findings

On October 17, 1999, the licensee was performing a post-outage RCS leak test. The system had just achieved its normal operating temperature (547°F), and system pressure was at approximately 2100 psig. The reactor was shut down with all control rods fully inserted. All four reactor coolant pumps and two auxiliary boiler feed pumps were in operation. RCS temperature was stable, but pressure was still increasing toward normal operating pressure (2235 psig), and other plant parameters, e.g., RCS dissolved gases and containment humidity, had also not stabilized.

At approximately 7:10 p.m., workers inside containment observed steam rising through the floor grating on the upper (95ft) elevation, and a health physics technician notified the control room of a steam leak inside containment. Based upon an increasing containment sump level and increasing containment humidity, operators entered off-normal procedure ONOP-RCS-7, "Excessive RCS Leakage," and performed an RCS inventory balance using the difference between charging and letdown flows, and level changes in the volume control tank and pressurizer to determine the leak rate. Two operators independently estimated the leak rate to be 15 gpm. Based on the calculated leak rate, operators evacuated all non-essential personnel from containment at 7:45 p.m. Operators also entered emergency action level (EAL) 3.1.1 (Notice of Unusual Event) at 7:54 p.m. based upon unidentified or pressure boundary leakage >10 gpm. The inspector responded to the control room to observe operator actions and to assess the actions of emergency plan personnel who responded to the Unusual Event. Senior plant managers also responded to the control room to review plant conditions, to oversee operator and response team actions, and to assure that the response of emergency plan personnel was timely and complete. The licensee entered Emergency Plan Procedure IP-2001, "Emergency Director (ED), Plant Operations Manager (POM), Shift Manager (SM) Procedure," and commenced the required notification to the NRC, and to local and New York State authorities. All notifications were made in a timely manner and were completed by 8:21 p.m.

Operators diagnosed the source of leakage based on control room indications and reports from individuals inside containment. They observed that the control board instrument for the 33 RCS intermediate loop flow transmitter (FT-435) was pegged high, and determined that the leak was in the lower pressure side of the flow transmitter. None of the radiation detectors inside containment alarmed, since the new core had not yet been critical and the RCS did not contain any significant activity. Control room operators immediately dispatched a response team to the lowest level of containment to close the root valve (RC-507B) on the lower pressure side of FT-435. The response team entered the containment to close RC-507B and to open the equalizing line across the transmitter. Within 10 minutes, the team had isolated the transmitter from the RCS and stopped the leak approximately 1 hour and 20 minutes after its initial discovery. At 8:50 p.m., operators officially declared the Unusual Event terminated after confirming that the leak had been stopped and that all required actions under the emergency plan were satisfied.

The licensee formed a Post-Transient Review Group (PTRG) the same evening to interview operators and the response team, and to collect factual information related to the leak. The response team indicated that the leak was in a flanged joint between the transmitter body and a tubing adapter. The team also indicated that both bolts in the flange joint could be easily moved by hand, and that a white gasket material (Teflon) was protruding from the gap in the flange connection. After making an initial assessment of the situation, the PTRG recommended that the response team re-enter containment to double isolate the flow transmitter so it could be removed for a detailed failure analysis. The failed transmitter was subsequently isolated and removed, and the licensee prepared a second replacement transmitter from the site warehouse. Further review by the PTRG revealed that the failed transmitter was installed during the current refueling outage to replace a transmitter the licensee considered to be overly sensitive. The PTRG also determined that the replacement transmitters had been procured commercially and dedicated for safety-related service by the licensee (see section E2.1). FT-435 was the only safety-related transmitter replaced during the outage.

The PTRG also collected data to analyze plant conditions related to the leak. Most of the available data permitted a timely review of the incident, and permitted the team to develop an accurate time line of the associated plant conditions. The PTRG used plant data records to account for all of the RCS inventory that leaked, and concluded that a total volume of 228 gallons were released inside containment. Their analysis of containment humidity was hampered slightly by the dewpoint recorder which had its chart paper installed in the reverse direction and had marked gradations different from actual dewpoint values. However, the overall trend of containment humidity was clear from the chart record, and permitted the PTRG to identify the point where containment humidity rapidly increased. The dewpoint recorder indicated that the leak rate was not constant, and had existed for approximately 2-1/2 hours. The leak was initially very small and was not observed by workers in containment for approximately an hour before the gasket ruptured. FT-435 was located in an instrument rack on the intermediate level, and inside a partly enclosed space and not directly observable from most containment areas. Subsequent evaluation of the failed gasket revealed that it had been compressed inside the flange more than once, but had not been subsequently replaced in accordance with the manufacturer's (Foxboro) recommendation. Both the response team and the maintenance personnel who installed the transmitter during the outage stated that the flange connection had not been loosened or tightened during shop testing or installation.

The PTRG completed its evaluation the following day, and compiled a report that was submitted to senior plant managers. The report provided a comprehensive root cause analysis of the leak and enabled the plant manager to direct a resumption of scheduled activities. The licensee changed the Teflon gasket in a second replacement transmitter with a stainless steel gasket in accordance with the manufacturer's current specifications for safety-related applications. The licensee performed a calibration and hydrostatic test of the second replacement transmitter at 3800 psig, and then installed it in the plant. That transmitter was subsequently placed into service and observed to be leak free.

c. Conclusions

Operator response was well focused and effective in diagnosing and isolating a high pressure leak in a failed reactor coolant system flow transmitter inside containment. They also made timely entries into the appropriate response procedures and recognized the correct conditions for classifying an Unusual Event. Emergency Plan responders made the required local, state, and NRC notifications properly and on time. Operators promptly diagnosed control room indications and directed auxiliary operators inside containment to isolate the leak before a significant volume of RCS inventory was lost. The resulting consequences to personnel and equipment inside containment were minimized, and the subsequent analysis by the post-transient review group was timely and comprehensive.

O2 Operational Status of Facilities and Equipment

O2.1 Outage Risk Assessment Review

a. Inspection Scope (71707, 62707)

The inspector reviewed the licensee's program for outage risk assessment to assess the adequacy of pre-outage review, implementation of recommendations that resulted from this review, and day-to-day program implementation.

b. Observations and Findings

The inspector reviewed procedure, AP-9.2, "Outage Risk Assessment," which was the governing procedure designed to assess the outage schedule to determine that primary and backup means to satisfy key safety functions existed, and to identify and develop contingency plans for conditions which could result in a reduction in the margin of safety. The procedure also discussed use of the risk assessment during the outage, and the process for incorporating emergent work.

The schedule assessment was performed by a Risk Assessment Team, consisting of at least three knowledgeable individuals (at least one of which is a currently licensed senior reactor operator) who were not involved in development of the schedule. The key safety functions that are evaluated were 1) core cooling, 2) reactor coolant system inventory, 3) power availability, 4) reactivity control, and 5) containment integrity. The amount of defense-in-depth provided by a given configuration was calculated (based on numeric values contained in a table in AP-9.2), and then translated to a color designator as follows:

Green - A primary and backup means of satisfying the safety function are available.

Yellow - A reduction in the redundancy that is used to satisfy either the primary or backup means has occurred.

Red - Only a primary or a backup means of satisfying the safety function is available.

Overall, AP-9.2 provided good guidance for assessing outage risk based on the scheduled outages for safety equipment, and for ensuring that backup equipment was available for the key safety functions. However, the inspector noted that the procedure did not establish a process for assessing the cumulative effect of the key safety functions on plant safety, and it did not develop an overall risk indicator. Consequently, the licensee did not assess the overall risk of a particular plant configuration. For example, the assessment did not indicate whether a "yellow" for a particular safety function (with the other functions "green") was better or worse for a given plant configuration than any other combination of one "yellow" and four "green." Although no regulatory requirement existed, such an indicator could be useful to plant operators and outage management for better assessing the impact of emergent work upon scheduled and ongoing activities. The licensee considered that quantifying the total risk for each daily assessment would be worthwhile

The inspector also reviewed the licensee's initial risk assessment document dated July 1, 1999, for the current refueling outage (R0-10). The assessment resulted in 38 recommendations to re-schedule outage activities and to revise procedures to cover conditions that were expected to occur during the outage. Through discussions with risk assessment personnel, the inspector confirmed that the recommendations that resulted from the review had either been incorporated into the outage schedule or were being tracked as action items that were keyed for completion prior performing the item at issue. In addition, the assessment identified 17 risk conditions that required contingency plans for conditions which result in a reduction in the margin of safety. The inspector verified that the operations department had developed procedures to implement these contingencies. The inspector concluded that the licensee performed a thorough assessment of outage risk in accordance with AP-9.2.

During the refueling outage, the licensee assigned two individuals whose primary function was to perform shutdown risk assessments. They reviewed the daily shutdown risk assessment against the scheduled activities for the day to determine if any conflicts existed. They also reviewed emergent work and coordinated with the scheduling department to minimize its risk impact (by scheduling the work during existing equipment unavailability windows, or identifying and scheduling contingency plans).

The operations department was responsible for performing the daily shutdown risk assessments. AP-9.2 stated that additional daily shutdown risk assessments should be performed for major changes in plant equipment. The inspector noted during the day shift on September 15 that the reactor vessel head was de-tensioned. This changed the core cooling and power availability functions from green to yellow. However, operations department personnel did not perform another daily risk assessment. From discussions with risk assessment personnel, the inspector determined that the applicable contingency procedures had been scheduled on the daily schedule and had been instituted prior to head de-tensioning. As such, performing an additional daily risk assessment would not have identified any new compensatory actions. However, performance of a risk assessment on the basis of events, rather than on a regular time basis (i.e., daily), would be important for emergent plant condition changes. The inspector concluded that, while there was no procedure violation, the operations department missed an opportunity to back up the risk assessment group by performing a daily risk assessment after the vessel head was de-tensioned.

c. <u>Conclusions</u>

The licensee established a good program for performing outage risk assessments. The pre-outage schedule review was thorough and resulted in the incorporation of numerous schedule modifications to reduce risk. Appropriate contingency planning was performed for conditions where system redundancies would be reduced. In one instance, the operations department did not perform a daily risk assessment following a significant plant condition change, as recommended by the procedure.

O7 Quality Assurance in Operations

O7.1 Licensed Operator Overtime

a. <u>Inspection Scope</u> (71707)

The inspector reviewed the licensee's policy and administrative procedure on licensed operator overtime, and evaluated the process for management approval to exceed the administrative limits.

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

The licensee used administrative procedure AP-36, "Overtime Restrictions," to implement the technical specification requirement (6.2.2g) to maintain adequate shift coverage without the routine heavy use of overtime. The procedural requirements essentially repeated the technical specifications for "regulatory required" personnel, and outlined their daily and weekly limitations. "Regulatory Required" individuals (licensed reactor operators, senior reactor operators, auxiliary operators, shift technical advisors, and shift contingency personnel) were restricted to work no more than 16 hours in any 24 hour period, no more than 24 hours in any 48 hour period, and no more than 72 hours in any week (not including time required for turnover). If it became necessary to exceed these restrictions for any individual, special management authorization and justification was required.

While the plant was at power, licensed operating personnel were normally scheduled to work a 40 hour week on a rotating shift basis. However, during the refueling outage, operators were assigned to work 5 straight 12-hour days and then had one day off, so as to comply with the 72 hour maximum over any single week. During the last two weeks of the refueling outage, the operators were allowed to work additional overtime hours beyond their scheduled shifts. The operations manager published a short-term revised overtime policy in the daily shift orders, and reiterated that the AP-36 rules still applied. Any individual expecting to exceed the 72 hour per week limit was directed to obtain management approval beforehand. In addition, individuals and their supervisors were held responsible for ensuring their mental and physical fitness for duty without undue risk of injury or mental error. Individuals were told they were responsible to track their own hours, and that they would have to get management approval to exceed the administrative limits before they performed the work. Excess overtime beyond 72 hours in a week was allowed for operators only to perform off-shift work in direct support of the critical work path, and with prior management approval.

The inspector sampled the management authorization forms from AP-36 that were used to justify excess overtime for operators, and interviewed several operators to evaluate the nature of their overtime work. No licensed operator working excess overtime was observed to perform on-shift duties. The inspector also reviewed the DERs written on overtime for all of 1999, and noted that none had been written during the current refueling outage. The last DER on excess overtime for operations personnel was written March 22, 1999 for a Control Room Supervisor who worked 27 hours in 48, 2 of which were for turnover. One other DER on excess operations overtime was on March 6 for an NPO who worked ½ hr over the limit. There were no other DERs related to operator overtime written in 1999.

c. <u>Conclusions</u>

The licensee's policy for excess overtime was appropriately managed and restricted for operators during the refueling outage. The excess overtime authorized during the last two weeks of the outage was appropriately justified and controlled by station management prior to operations personnel exceeding the normal overtime limits.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 Maintenance General Comments

a. Inspection Scope (62707)

The inspectors reviewed selected maintenance work activities and supporting work documentation. Activities were selected based on the systems, structures, or components contained within the scope of the maintenance rule.

b. Observations and Findings

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

- WR 93-03719-00, Replacement of the 32 Auxiliary Boiler Feedwater Pump Steam Admission Valve
- WR 92-03483-28, Replacement of the 33 Safety Injection Pump
- WR 97-06966-00, Repair of the Oil Lift System for the 33 Reactor Coolant Pump Motor

c. Conclusions

Maintenance activities observed were conducted satisfactorily and in accordance with applicable maintenance and administrative procedures. The licensee appropriately monitored performance of equipment within the scope of the maintenance rule.

M1.2 Surveillance General Comments (61726)

a. Inspection Scope (61726)

The inspectors reviewed selected surveillance test activities and supporting documentation. The activities were selected based on the systems, structures, or components contained within the scope of the maintenance rule.

b. Observations and Findings

The inspectors observed all or portions of the following surveillances:

- 3PT-R003A, "Safety Injection Test of Recirculation Switches,"
- 3PT-R007B, "32 Auxiliary Boiler Feed Water Pump Full Flow Test,"
- 3PT-R172C, "Station Battery #33 Modified Performance Test,"
- 3PT-R003D, "Safety Injection Test,"
- 3PT-CSO4, " Low Head Injection Check Valve Test,"
- 3PT-R85, "RHR Valves 730 and 731 Disk Integrity Test."

c. Conclusions

Routine surveillance tests were conducted appropriately and in accordance with procedural and administrative requirements. Test and performance monitoring personnel maintained a good level of communication and coordination with control room operators during observed surveillance tests. Test instrumentation was observed to be within the required calibration periods and all test acceptance criteria for operability were met.

M1.3 Safety Injection/Station Blackout Surveillance Test 3PT-R003D

a. Inspection Scope (61726, 71707, 37551)

The inspector reviewed the licensee's development and implementation of 3PT-R003D, "Safety Injection Test." The inspector observed the performance of the test including the pre-job brief, the special evolution brief, verified portions of the valve line-ups, and observed the performance of the blackout step sequences.

b. Observations and Findings

Overall, the pre-job briefing for 3PT-R003D was thorough, addressed industry events, and emphasized self-checking and peer-checking techniques. Performance of the test had the appropriate level of management oversight. The inspector noted several equipment failures during the execution of the test. These included the failure of the 31 containment spray to start upon an actuation signal, the failure of the 31 and 33 component cooling water pumps to strip off the 480 volt bus as a result of the undervoltage condition, and the failure of the control room SI reset lights to illuminate when the SI signal was reset.

Instrumentation and controls personnel evaluated the failures, identified their root causes, and subsequently repaired the deficiencies. The failure of the containment spray pump to initiate was due to a bad contact on the control switch in the control room. This contact was in series or immediately in line with the automatic initiation contact and therefore impeded the start signal. This contact was replaced and the pumps ability to automatically initiate was re-tested as satisfactory.

During the pre-job brief, the inspector noted that the test coordinator emphasized to auxiliary operators performing the pre-test valve line-ups that four SI discharge header valves (MOV-856C/E/J/H) should be verified closed before verifying or manipulating the position of any other valves on the line-up sheet. He considered this necessary in order to prevent a possible flow path between the RCS and the refueling water storage tank (RWST) through the 32 SI pump suction valve V-898 if the MOV-856 valves were open when V-898 was opened. The MOV-856 valves were normally closed during a refueling outage; however, the test coordinator wanted to ensure that an isolation boundary always existed between the RCS and the RWST during the pre-test line-up. The licensee later considered that a permanent change to procedure 3PT-R003D may be warranted since the valve line-up step did not specify a particular sequence. The MOV-

856 valves were not the first valves on the list and would not be verified first without specific instructions to do so.

c. Conclusions

The pre-job briefing for the Safety Injection/Station Blackout integrated functional test was thorough, addressed industry events, and emphasized self-checking and peer checking techniques. Performance of the test had the appropriate level of management oversight. The test coordinator made a good configuration control effort during the pre-job briefing by ensuring that an isolation boundary was maintained between the reactor coolant system and the refueling water storage tank to preclude a potential leak path.

M1.4 In-service Inspection (ISI) Program Review

a. Inspection Scope (73753)

The inspector reviewed the licensee's implementation of the in-service inspection (ISI) program for the last refueling outage (RO-10) in the third period of the second ten-year interval. The review included a verification that the licensee completed the second ten-year interval in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Article IWB 2000, Inspection Program B requirements. The review also included the results of volumetric and visual inspection of the reactor vessel shell and nozzle welds, and eddy-current testing of the u-tubes in steam generators 31 and 32.

b. Observations and Findings

The inspector found that the licensee's documentation of the "Indian Point 3 Ten Year In-service Inspection Program - Second Interval 8/30/86 to 7/20/00 Program - Westinghouse Nuclear Services Division" accurately represented the progress of the program up to the present refueling outage (RO-10). Supplementing this documentation, the "Indian Point 3 Examination Plan - Second Interval/Third Period/Second Outage - Westinghouse Nuclear Services Division" also effectively represented the inspections completing the second interval program during RO-10. The inspector verified that the inspections not completed in the total second interval document were included in the RO-10 inspection, thereby completing the required scope of the second ten-year interval.

The inspector found that the division of inspections during the three periods of the inspection elements over the ten-year interval were consistent with the extent and frequency of examinations in the ASME B&PV Code, Section XI, Tables IWB 2500-1. Furthermore, the inspector found acceptable documentation of Code relief request approvals and denials throughout the second interval, with acceptable identification and documentation of the actions taken for denied relief requests.

c. Conclusions

The second ten-year interval in-service inspection program was satisfactorily completed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI.

M1.5 Reactor Vessel Examination

a. Inspection Scope (73753)

The inspector observed portions of the volumetric examination of the reactor vessel circumferential, meridional, and nozzle welds. In addition, the inspector observed the examination of the inner surface of the reactor vessel shell and the removed reactor internals. The inspector also reviewed the results of these examinations.

b. Observations and Findings

Volumetric Examination of Reactor Vessel (RV) Welds

The inspector observed portions of the volumetric examination of the reactor vessel and nozzle welds and found the examinations to be in accordance with the IP-3 Plant Technical Specifications, the ASME B&PV Code, Section XI and Section V, 1983 Edition including Summer 1983 Addenda, U.S. NRC Regulatory Guide 1.150, Revision 1, and IP-3 procedure INT-ISI-254, "Instructions for Calibrations, Examination, and Assessment of Recorded Data," Revision 0. The licensee performed inspections of the welds with ultrasonic transducers mounted on the remote operated service arm (ROSA) system using the Westinghouse Submersible Platform with ROSA End Effector Motion (SUPREME) RV inspection tool, and the Ultrasonic Data Recording and Processing System - 2 (UDRPS-2) for data retrieval and interpretation.

The inspector observed the examinations from the remote data facility outside the containment through televised pictures of the examination process, and with a computerized transducer orientation display model that precisely demonstrated the three-dimension locations of the reactor vessel shell and nozzles. NDE technicians demonstrated precise movement of the transducers to reactor shell and nozzle locations within the reactor.

The reactor vessel volumetric inspections included the following:

- Circumferential and Longitudinal Shell Welds
- Shell to Bottom Head Circumferential Weld
- Bottom Head Meridional Weld
- Shell to Flange Welds
- Nozzle to Vessel Welds and Inside Radius Section
- Reactor Flange Threads

The inspector reviewed documentation of the results of six reactor vessel volumetric inspections. Thirteen recordable indications were found as a result of the inspections. The inspector found that these indications were correctly evaluated in accordance with ASME B&PV Code, Section XI, Paragraph IWB 3500, and were within acceptable limits. The inspector reviewed several preliminary reactor vessel ultrasonic test (UT) calibration and examination data sheets, and found them to properly reflect calibration, results, and primary and secondary verifications of the examinations.

The licensee reported several areas having obstructions that limited the ability of the transducer to examine 90% of the expected volume. These included the lower head to lower shell weld (66%), the upper shell course longitudinal seam at 7 degrees relative to vessel axis location (76%), and four outer shell-to-weld, tangential seams (coverage estimated from 43% to 100%). The licensee stated that these inspection limitations will be identified and reported to the NRC.

Visual Examination of Reactor Vessel Internal Surfaces

The inspector observed the video tape records of the visual examinations of the reactor vessel inner surface and other removed internal structures that were made from a camera housed in the hull of a Westinghouse Mini-Rover submarine. The video images were transmitted via cable to a remote video display, from which the reactor internal images were observed, evaluated, and recorded. The inspector reviewed portions of the video-taped inspections as follows:

Tape 1: Reactor Vessel Lower Internals, Flange, and Upper Internals
Tape 2: Reactor Vessel Interior Surface
Tape 3: Reactor Vessel Lower Internals

The inspector found the high resolution of the video images recorded on the internal reactor vessel surfaces allowed excellent viewing. During the review, the inspector observed no indications of structural defects. However, the inspector noted a considerable number of loose particles laying on vertically-faced surfaces, which required removal. The licensee indicated the particles were expected to be found after an extended operating period. The licensee stated that these particles would be removed before restart after evaluating any possible effects on the core.

c. Conclusions

The licensee satisfactorily performed volumetric examinations of the reactor vessel and nozzle welds in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, U.S. NRC Regulatory Guide 1.150, and Indian Point 3 procedures. The licensee correctly evaluated recordable indications. No indications of structural defects were found during visual inspection of the reactor vessel and removable internals.

M1.6 Steam Generator (SG) Tube Inspections

a. Inspection Scope (73753)

The inspector reviewed implementation of eddy current examination of SG 31 and SG 32 tubes. The inspector reviewed the results of these examinations.

b. Observations and Findings

The inspector observed the eddy current examination of the 31 and 32 steam generator tubes. The examinations included: 1) Full length bobbin probe inspection of 100% of the tubes in both SGs except U-bend tubes that were inspected by rotating pancake coils (RPCs); 2) 40% RPC TTS (+/- 3 inches from top of tubesheet) inspection of both SGs using a plus point (PPT) probe with a pan cake coil, plus point coil, and a high frequency coil; 3) 40% U-bend PPT RPC inspection of U-bend tubes (74 tubes) in each SG; 4) Special interest inspection program which included all "dents and dings" with probe indication signals greater than 5 volts in the hot leg straight sections (these included 3 tubes in SG 31 and 0 tubes in SG 32); 5) Special interest program to include probe signals dispositioned as possible loose parts from bobbin probe tests; and 6) expansion of inspection scope to include 30 tubes tested by RPC at the TTS in areas were possible loose part wear indications were reported.

The inspector observed the examination procedure from a remote location outside the containment from information relayed by cable from inside the containment. The inspector observed the remote-controlled entrance and exit of the eddy current bobbin into and out-of the steam generator tubes with dual probe pushers in four legs of the two SGs, using the eddy-current inspection procedure and acquisition technique sheets (ATSs). Data analysis was performed using Westinghouse software. Primary analysis was performed by remote electronic connection to Waltz Mill, Pennsylvania, and secondary analysis was performed by Corestar in Pittsburgh. The inspector observed that data management and resolution of eddy current signal problems was performed at the IP-3 site in accordance with IP-3 steam generator data analysis technique procedures and data analysis technique sheets.

As a result of the SG 31 and SG 32 tube inspection, the licensee found no tubes to be classified as defective and no tubes required plugging. However, the licensee reported results of supplementary tests and evaluations as follows: 1) Licensee historical review of bobbin free-span indications in comparison with test results from the previous inspection found no change in free-span indications in 51 tubes in SG 31 and 27 tubes of SG 32. No changes were found in manufacturing buff mark indications in 49 tubes in SG 31 and 38 tubes in SG 32; 2) Two signals at support plates formerly reported as having distorted tube (DSI) indications were tested with PPT probes during the current inspection and revealed no tube degradation.

TTS RPC inspections found volumetric indications in three tubes of SG 32 that formerly had no reported indications. These were believed to be caused by wear resulting from impingement of foreign objects. Foreign material had been removed from SG 32 at a location near these tubes during this inspection. Based on current RPC examination, no loose parts remained within SG 32. These indications were reported by the licensee in Deviation/Event Report (DER) 99-02164 and corrective actions were initiated to inspect the integrity of the affected tubes. The licensee determined that the reported indications could be considered arrested at their current acceptable depth, without potential for further growth.

c. Conclusions

The licensee satisfactorily completed tube inspections of steam generators No. 31 and No. 32 in accordance with procedures. Most tubes inspected were found not to have indications of cracks, wall thickness reduction, dents, or dings beyond acceptable levels. Three tubes inspected revealed indications due to loose parts wear, but the indications were considered arrested at their current acceptable depth, without a potential for further growth.

M1.7 Code Repairs and Replacements

a. Inspection Scope (73753)

The inspector reviewed implementation of selected repairs and replacements performed under ASME B&PV Code Section XI rules.

b. Observations and Findings

The inspector selected the following three work packages for review from a list of work packages performed under the rules of ASME B&PV Code Section XI for repair or replacement:

Pressurizer Code Safety Valve Mechanical Maintenance Testing

The licensee removed pressurizer code safety valve PCV-464 and shipped it to Wylie Laboratories for testing as a part of their planned mechanical maintenance. The valve was satisfactorily removed, shipped, disassembled, tested, return shipped, re-assembled and then re-installed at IP-3 in accordance with ASME B&PV Code, Section XI.

Service Water Line Socket Weld Repair

The licensee repaired a weeping Seismic Class 1 bimetallic socket weld in the containment fan cooler unit (FCU) service water supply line drain. The socket weld repair was implemented satisfactorily in accordance with American National Standards Institute (ANSI) B31.1 - 1967 Edition.

Tube Bundle Replacement on Jacket Water and Lubrication Oil Heat Exchanger

As a result of tube-side abnormal wear found in the 32 emergency diesel generator (EDG) jacket water and lubrication oil heat exchangers during planned preventive maintenance (PM), the licensee replaced the Seismic Class 1 heat exchanger tube bundle with like-kind material. The licensee also investigated possible causes related to galvanic, nodular, pitting, or microbiological corrosion and found no evidence of these types of degradation. Nevertheless, the licensee decided to replace the tube bundle. The inspector verified that the tube bundle was correctly replaced in accordance with IP-3 Technical Specification and ASME B&PV Code Section XI.

c. Conclusions

The licensee properly followed the rules of the American Society of Mechanical Engineer's Boiler and Pressure Vessel Code for the repair or replacement of emergency diesel generator jacket water and lubrication oil cooler components.

M1.8 Contractor Personnel Certification and Qualification

a. Inspection Scope

The inspector reviewed contractor personnel qualifications and certification documentation for reactor vessel inspectors.

b. Observations and Findings

The inspector reviewed documentation of the qualifications and certifications of selected contractor non-destructive examination (NDE) personnel performing the examinations during RO-10. The inspector found the examiner qualifications to be consistent with the American Society for Non-destructive Testing (ASNT) Standard for Qualification and Certification of Non-destructive Testing Personnel. The records reviewed indicated the Level of Qualification of each examiner achieved in each of the examinations (ultrasonic, magnetic particle, dye penetrant, radiographic, and visual). The inspector checked the inspector qualifications for several of the RO-10 examinations performed, and found them to be satisfactory.

c. Conclusions

Review of selected contractor personnel qualification and certification documents for RO-10 non-destructive examinations indicated personnel performing the examinations satisfied American Society for Non-destructive Testing standards for qualification and certification.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Plant Material Conditions

a. Inspection Scope (62707)

Throughout the inspection period, the inspector performed plant walkdowns to observe the material condition of equipment and to identify any conditions that could impact equipment operability or reliability.

b. Observations and Findings

Emergency Diesel Generators (EDGs)

On the 32 EDG, a flex hose between the lube oil (LO) strainer drain and engine block was wearing against a flex hose from the warmup LO pump discharge, which appeared to be melting the outer rubber. Rupture of this hose during engine operation would drain the LO system. The licensee evaluated the condition and determined that it would support operation until the next scheduled EDG outage, which was approximately a week away. The licensee placed a piece of rubber between the two hoses to prevent further wear.

On three of the four steam generator atmospheric relief valves (ARVs), an inadequate fastener was used to connect the linkage between the stem and the air-operated positioner. Specifically, a simple hex nut with marginal thread engagement was in use, rather than a self-locking nut or a nut with a lock washer. These valves were subject to vibration when in use, and separation of the linkage from the air positioner would render remote control of the valve inoperable. Nuclear industry experience has shown this type of failure to be a problem in some air-operated valves. The inspector discussed the observation with the licensee, and was informed that the condition would be corrected during the current outage.

Safety Injection Pumps

Two one-inch pipes in a vertical run on the wall near the 31 SI pump (labeled 505-11 and 194-13) - each had a u-bolt clamp with a loose nut, 194-13 also had a loose nut.

SI Valve MOV-842 (SI pump recirculation isolation) had a relief on the actuator casing; however, MOV-843 (identical valve) did not have a relief valve installed.

Valve position indicators for SI valves MOV-842, -843, and -1810 (RWST outlet isolation) were attached to the MOV using 2 of the 6 screws that held the cap to the actuator casing. They appeared to be the same screws, so there may have been insufficient length going into the MOV casing.

The 3/4" line between the inboard pump bearing and the cooler mounted at the front of the pump. The 32 and 33 pumps had a support bracket welded to the inboard pedestal, but the 31 pump did not.

Containment Material Conditions

Two instrument air (IA) system pipe supports in the lower level near stanchion 15 had u-bolt supports that were threaded steel rod. No other material was used to prevent fretting of the copper IA piping.

One IA pipe support in the lower level overhead near stanchion 16 was not properly made up (the hex nuts on the u-bolt were not tight).

The bottom support for an instrument line drain in the lower level near stanchion 18 was not made up.

c. Conclusions

The inspector identified numerous discrepancies during several walkdown inspections of plant equipment during the refueling outage. The discrepancies appeared to be minor in nature, but were referred to the licensee for evaluation and resolution prior to the end of the outage.

M8 Miscellaneous Maintenance Issues

M8.1 (Closed) Licensee Event Report (LER) 1999-011, "Technical Specification Violation Due to Inoperable Pressurizer Safety Valves" **(NCV 50-286/99-08-02)**

a. Inspection Scope (71707, 62707, 92700)

The inspector reviewed the licensees actions and work planning activities which led to three inoperable pressurizer safety relief valves while the reactor head was still on the reactor vessel.

b. Observations and Findings

During reactor disassembly on September 16, 1999, the licensee discovered that the pressurizer code safety relief valves (SRVs) had all but two of their bolts removed from their flanges prior to removing the reactor vessel head. Technical specification limiting condition for operation (LCO) 3.1.A.2 required that at least one pressurizer code SRV be operable or that there be an opening in the reactor coolant system greater than or equal to the size of one code SRV flange to allow for pressure relief while the reactor head is on the vessel.

This condition was a result of inadequate communication between maintenance personnel and work control personnel. Specifically, maintenance personnel requested permission to move ahead of the written outage work schedule to begin de-tensioning the bolts on the pressurizer SRVs. The licensee's work control process did not confirm the specifics of this request and permission was given to proceed with the activity mistakenly thinking that the activity was to remove the restraints on the pressurizer safety relief valves. This was a violation of technical specification limiting condition for operation (LCO) 3.1.A.2, "Safety Valves"

On October 15, 1999, the licensee submitted LER 1999-011 to the NRC. The inspector performed an in-office review of the report and noted that it contained an adequate description of the root cause and corrective actions associated with this event as required by 10 CFR 50.73. This LER is closed.

c. Conclusions

Poor communications and inadequate work control guidance led to a condition prohibited by the plant technical specifications. In loosening the pressurizer safety relief valve body-to-bonnet bolts, the licensee compromised the operability of all three valves prior to establishing a suitable pressure relief path as required by technical specification 3.1.A.2. This failure to comply with the technical specification is a violation of NRC requirements. However, because this was identified and corrected by the licensee, this Severity Level IV Violation will be treated as non-cited in accordance with Appendix C of the NRC Enforcement Policy (NCV 50-286/99-08-02).

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 Commercial-Grade Dedication of Reactor Coolant System Flow Transmitters (NCV 50-286/99-08-03)

a. Inspection Scope (38703)

The inspector reviewed the licensee's commercial-grade dedication of two safety-related flow transmitters. One was installed in the reactor coolant system (RCS) during the current refueling outage, and subsequently failed at high system pressure.

b. Observations and Findings

In 1996, NYPA procured two Foxboro flow transmitters (Model No. E13DH-HSAH1) as commercial-grade material from the D. C. Cook Nuclear Plant. That licensee (Indiana Michigan Power Company) had purchased the transmitters directly from Foxboro as commercial-grade material in 1989. NYPA had initially planned to install them in the 1996 refueling outage; however, they were not installed during that outage and remained in the IP-3 warehouse for future use. Following a reactor trip in April 1999, NYPA determined that the 33 RCS intermediate loop flow transmitter (FT-435) was overly sensitive to minor flow perturbations, and that FT-435 should be replaced during the next (1999) refueling outage. NYPA considered that both of the transmitters procured from

Indiana Michigan Power Co. were suitable for the FT-435 application since they had the same model number as FT-435 and other safety-related flow transmitters installed at IP-3 during original plant construction. The transmitters do not perform a reactor protection or accident mitigation function; however, they do form part of the RCS pressure boundary and must be able to withstand the high operating pressure of the RCS (2235 psig) in order to maintain pressure boundary integrity. Consequently, the transmitters needed to be upgraded (dedicated) to safety-related (Category I) material from commercial-grade (non-Category I) material in order to use them in the reactor coolant system.

In preparation for the 1999 refueling outage (RO-10), the licensee prepared a dedication package that contained Technical Evaluation 99-000635 to support the upgrade of both transmitters to Category I material. The upgrade required that the licensee identify the critical safety functions provided by the transmitters, and to assure the functional adequacy of the upgraded material. The technical evaluation listed the manufacturer's model number, and the transmitter's calibration and operation as the three critical characteristics that had to be verified for the upgrade. The licensee elected to dedicate the transmitters using a combination of receipt inspection points and documentation provided by the manufacturer and seller. NYPA's purchase order to Indiana Michigan Power Co. specified that the transmitters were to be "new and not refurbished material." They were also to be supplied with the manufacturer's certificate of compliance, and all available reports of material tests, hydrostatic tests, non-destructive examinations, non-conformances, and with the original purchase order documents that could trace the transmitters to the manufacturer. Indiana Michigan Power Co. only provided the original purchase order document since the transmitters had never been subjected to the quality assurance controls that would have applied to safety-related equipment under a 10 CFR 50, Appendix B procurement program. The transmitters had never been classified by The Indiana Michigan Power Co. as safety-related equipment and were not maintained or stored under Appendix B requirements.

NYPA's receipt inspection requirements included visual observations for cleanliness, shipping damage, and vendor identification. The transmitters were not disassembled for an inspection of its internal components since the soft seals and gaskets would have to be replaced. Since the technical evaluation did not specify that the pressure boundary function was a critical characteristic, the transmitters were not subjected to a hydrostatic test after arriving at IP-3. Prior to installation, the transmitter designated to replace FT-435 was calibrated in the instrumentation and controls (I&C) shop, but the calibration was performed at a relatively low pressure and was not an adequate test of the pressure boundary function.

After the transmitter was installed, the licensee declared it operable for safety-related service prior to initiating the RCS post-outage leak test (3PT-R131). During the leak test, the gasket in the flange connection between the transmitter body and a tubing adapter failed at approximately 2100 psig and caused a loss of approximately 230 gallons of RCS inventory into containment before the transmitter could be isolated and caused the operators to declare an Unusual Event (see Section O1.2). A post-failure evaluation revealed that the gasket had been compressed more than one time inside the flange connection, and had not been subsequently replaced. This was contrary to the manufacturer's repair instructions which indicated that the gasket must be replaced whenever the flange connection was broken.

This event resulted from the failure of the licensee's commercial dedication process to identify and verify the pressure boundary safety function of the replacement transmitter. This item was entered into the licensee's corrective action system, and final resolution was still pending at the end of the inspection period; however, the licensee had not yet identified the need for a procurement program change to prevent a recurrence. The failure to specify, and to subsequently verify, the adequacy of the pressure boundary safety function as part of the dedication of a commercial-grade flow transmitter is a violation of 10 CFR 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," **(NCV 50-286/99-08-03).**

c. <u>Conclusions</u>

The licensee did not verify that a reactor coolant system flow transmitter procured as commercial-grade material could satisfactorily perform the pressure boundary safety function. The flow transmitter subsequently failed at high system pressure and caused an RCS leak that required operators to declare an Unusual Event. This is a Severity Level IV Violation of 10 CFR 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services." However, this item was identified and corrected by the licensee, and has been appropriately entered into the corrective action system. Therefore, this is a Non-cited Violation in accordance with Appendix C of the NRC Enforcement Policy **(NCV 50-286/99-08-03).**

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Controls - External and Internal Exposure

a. <u>Inspection Scope</u> (83750-02)

The inspector evaluated the effectiveness of selected aspects of the applied radiological controls program during the outage. The evaluation included a selective review of the adequacy and implementation of the following radiological controls program elements:

- access controls to radiologically controlled areas
- use and adequacy of personnel occupational exposure monitoring devices
- maintenance of personnel occupational radiation exposures (external and internal) within applicable regulatory limits and as low as reasonably achievable (ALARA)
- implementation of the radiation work permit program including the effectiveness of work planning

The inspector evaluated licensee performance in the above selected areas via observation of activities, tours of the radiologically controlled area (RCA), discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The licensee implemented effective access controls to the radiologically controlled areas of the station. The licensee's use of radiation work permits (RWPs), bar code readers, and computerized log-in stations was good. The main health physics (HP) control point to the RCA had been enlarged and reconfigured to allow a clearer view of workers' entries and exits, and to eliminate the separate HP control for contractors. HP control points on different elevations inside the vapor containment provided effective onsite radiological protection coverage. No access control deficiencies were identified.

A new remote and centralized radiological monitoring station with closed-circuit television (CCTV), wireless headset communication, and teledosimetry was in use for the first time at Indian Point 3, and was being evaluated for its potential to improve HP job coverage.

Appropriate personnel monitoring devices for access to the RCA were supplied and used. Thermoluminescent dosimeters (TLDs) and personnel alarming dosimeters were observed to be properly worn to measure external doses. Access controls for high radiation areas (HRAs) were effective. Radiological postings and labels throughout the areas toured provided additional administrative controls and information to the worker. There were no areas requiring posting for airborne radioactivity. Survey maps with radiological data were posted at the main HP control point and also with selected posted RWPs.

The licensee maintained personnel occupational radiation exposures (external and internal) within applicable regulatory limits and as low as reasonably achievable (ALARA). A review of personnel exposure data for 1999 (year to date) identified that individual exposure results for total effective dose equivalent (TEDE), lens of the eye dose equivalent (LDE), shallow-dose equivalent (SDE), and extremity dose equivalent were well below regulatory requirements. Further, the maximum individual committed effective dose equivalent (CEDE) for any one individual was well within applicable NRC limits. The occupational exposure of declared pregnant women and the dose to the embryo/fetus were controlled in accordance with Title 10 Part 20.1208 of the Code of Federal Regulations (10 CFR 20.1208).

A review of the RWPs for the removal/replacement of the reactor lower internals (RWP 99-323), for the steam generator handhole opening and sludge lancing (RWP 99-345), and for the reactor head lift (RWP 99-322) showed that they contained appropriate work descriptions, radiological survey data, protective clothing and dosimetry requirements, and instructions and precautions.

c. Conclusions

The licensee effectively implemented radiological controls during the current refueling outage. Access controls to radiologically controlled areas were applied effectively, and appropriate occupational exposure monitoring devices were provided and used. Personnel occupational exposure was maintained within applicable regulatory limits and as-low-as-is-reasonably-achievable. Also, the radiation work permit program was properly implemented.

R1.2 Personnel Contaminations

a. Inspections Scope (71750)

The inspector reviewed the licensee's response to several personnel contaminations during the tenth refueling outage (RO-10).

b. Observations and Finding

During RO-10, the licensee noted several unanticipated personnel contaminations. The majority of these contaminations occurred in the facial area and could be attributed to the excessive humidity and warmer than expected temperatures in the containment. Two maintenance activities, 1) the steam generator eddy current testing and 2) the reactor cavity decontamination work were the most significant contributors to the contaminations. During the eddy current testing the area under the steam generator where most of the testing equipment is staged became contaminated. As a result, the breakdown and removal of equipment in that area resulted in 15 personnel contaminations. The licensee recognized the excessive number of contaminations and in some cases accepted some additional risk in order to prevent more severe problems such as heat stress. As a result of the contaminations from the eddy current testing, the licensee attempted to take corrective actions to minimize additional contaminations in other areas. Consequently, prior to the reactor cavity decontamination, additional guidance was given to the health physics personnel to direct and maintain better control over that activity.

The inspector reviewed the outage contamination data, observed portions of the work activities, and interviewed licensee personnel regarding outage planning and preparation for radiation protection. Overall, the radiation protection department set aggressive goals for radiation and contamination exposure limits for the outage. Health physics personnel had active roles in the planning and implementation of outage work. However, the unavailability of the fan cooler units due to emergent service water repairs and the higher humidity and reactor coolant temperature from the less efficient backup spent fuel pool cooling system were not accounted for in the outage planning, or were not well integrated into the on-going radiation protection work activities. The overall containment building air temperature was much higher than initially expected and caused personnel to wipe excessive perspiration from their facial areas while working. Also, the high temperature imposed heat stress avoidance requirements on some work. In order to avoid employee heat stress, the health physics department had to make a conscious choice to relax some of the protective clothing requirements (eg,. allowing single layer rather than double layer clothing in highly contaminated areas). The inspector noted that the licensee's anticipation and response to the higher temperatures in the containment building was slow and at times ineffective, as evidenced by the high number of facial contaminations. Also, the licensee's corrective action guidance after the first large number of contaminations did not prevent additional contaminations during the reactor cavity decontamination.

c. Conclusions

In response to unanticipated reactor containment building conditions, the radiation protection department did not provide effective guidance to preclude excessive personnel facial contaminations. Emergent issues associated with the reactor coolant temperature and the unavailability of a containment fan cooling unit were not well integrated in the radiation protection plan or the ongoing outage radiation protection work.

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

a. Inspection Scope (83750-02)

The inspector evaluated the effectiveness of the licensee's surveys, monitoring and control of radioactive materials and contamination. The evaluation included a selective review of the adequacy and effectiveness of the following radioactive material and contamination control program elements:

- surveys and monitoring of radioactive material and contamination
- the calibration status of survey and monitoring equipment
- the proper use of personal contamination monitors and friskers
- the tracking of personnel contamination events and goals

The inspector evaluated performance in the above selected areas via observation of activities, tours of the RCA, discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The licensee implemented an effective radioactive material and contamination control program. Hand-held contamination monitors (friskers) and radiation survey meters exhibited current calibration stickers and were appropriately used by personnel. Survey records in the RWP packages and those posted contained appropriate radiological information. Radioactive material and radioactive waste were clearly labeled, segregated, and stored in an orderly manner. Receptacles for used anti-contamination clothing, radiologically contaminated trash, and radiologically clean trash were available in the RCA and were clearly labeled.

Personnel were properly frisking at the RCA exit using whole body contamination monitors. Small article monitors (SAMs) at the RCA exit were used to frisk hand-held items. The whole body contamination monitors and SAMs had current calibration stickers. The calibration records for all the whole body contamination monitors and SAMs in use at the RCA exit, and their calibration and use procedures were reviewed and found to be satisfactory.

Personnel contamination records and the procedure for personnel decontamination were reviewed and were appropriate. Adequate personnel decontamination facilities were available adjacent to the RCA exit. Goals for controlling and minimizing personnel contaminations continued to be established and used. The annual personnel contamination rate goal was set as equal to or less than 10 per 10,000 RWP entries with an additional goal of equal to or less than 99 personnel contaminations for the outage. Performance at the time of the inspection met the established goals.

c. <u>Conclusions</u>

Overall, the licensee implemented effective surveys, monitoring, and control of radioactive materials and contamination. Proper surveys were performed, and properly documented survey results were available. The personnel contamination rate was tracked and trended against a goal. The radiological surveys, monitoring, and controls were implemented with calibrated and properly used devices.

R1.4 Radiological Controls-As-Low-As-Is-Reasonably-Achievable (ALARA)

a. <u>Inspection Scope</u> (83750-02)

The inspector evaluated the effectiveness of the licensee's program to maintain occupational radiation exposure as-low-as-is-reasonably-achievable for a refueling outage. The evaluation included a selective review of the adequacy and effectiveness of the following ALARA program elements/documents:

- RWP dosage report for 1999 (year to date)
- RE-REA-4-1, Rev. 15, Attachment 4, Flowpath for Job Specific RWP/ALARA Reviews
- RWP Package 99-323, Removal/replacement of the reactor lower internals package
- RWP Package 99-345, Steam generator handhole opening and sludge lancing
- RWP Package 99-322, Reactor head lift

The inspector evaluated performance in the above selected areas via observation of activities, tours of the RCA, discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The inspector reviewed RWP packages 99-323, 99-345, and 99-322. The packages contained task descriptions, historical data, lessons learned, work scope comparisons, and dose estimates for past and current work in addition to the current RWPs, HP pre-job briefing papers, briefing attendance sheets, and job log sheets. Also, contingencies for a larger than normal crud burst (a possibility due to the longer fuel cycle in use) were evaluated and documented. The RWP package contents and crud burst contingency planning showed that significant planning and preparation for the HP aspects of outage work was performed.

The licensee had established annual goals for person-rem exposure. This outage was a refueling outage and included the 10-year in-service inspection (ISI) work. The annual person-rem goal was determined based on achieving an overall improvement in dose savings over past results for the tasks identified in the preliminary outage scope. The person-rem goal for 1999 was 150. The person-rem goal for the current outage was 130 with an additional challenge goal of 99. At the time of this inspection, the actual person-rem for the outage was approximately 36 which was 7% less than that projected.

c. Conclusions

The licensee implemented an effective program to maintain occupational radiation exposure as-low-as-is-reasonably-achievable based on the planning and preparation accomplished for the HP aspects of outage work and on the person-rem goal results achieved up to the current point in the refueling outage.

R5 Staff Training and Qualification in RP&C

R5.1 Health Physics Personnel Site-Specific Training and Qualifications

a. Inspection Scope (83750-02)

The inspector reviewed the qualifications and site-specific training of selected contracted HP technicians. Information was gathered through observation of activities, discussions with cognizant personnel, and review and evaluation of documents.

b. Observations and Findings

The documented qualifications of selected contracted senior HP technicians were reviewed and were found to exceed the technical specification requirements (i.e., two years experience required by technical specifications while three years experience required by contract). The site-specific training of senior and junior contracted HP technicians was done in conformance with site procedures. Again, formalized job performance measures (JPMs) were used for training, and the records for the conduct of the JPMs had been made more easily auditable. Again, selected senior contracted HP technicians were provided several days of training on how to conduct on-the-job training and then were tasked with providing on-the-job training for junior contracted HP technicians. More consistent HP practices and better performance by the junior contracted HP technicians were reasons for conducting these latter two efforts.

c. Conclusions

The qualifications of selected senior contracted health physics technicians exceeded technical specification requirements, and the site-specific training was well done and well documented.

R7 Quality Assurance In RP&C Activities

R7.1 Self-Assessments, Self-Identification, and Corrective Action Processes

a. Inspection Scope (83750-02)

The inspector evaluated the effectiveness of the licensee's self-assessment, self-identification, and corrective action processes. The evaluation included a selective review of the adequacy and effectiveness of the following program elements and documents:

- Radiation Protection Program Monitoring Team
- self-assessments by the radiation protection organization
- corrective action program

The inspector evaluated the performance in the above area via observation of activities, tours of the RCA, discussions with cognizant personnel, review of applicable documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The IP-3 Radiation Protection Program Monitoring Team was established by the Radiation Protection Manager (RPM) as a means of self-assessing the radiation protection program during outages by monitoring the implementation of established standards during activities and tasks performed by both plant radiation workers and the radiation protection staff. This team gathered information through direct observations of work performed in the RCA. The team included members from inside and outside the radiation protection group. On a weekly basis, the team coordinator provided a written report to the RPM summarizing the observations by the team members. A review of a report of weekly observations by a team member and of a weekly summary report by the team coordinator indicated that numerous instances of both good and poor performance and other deficiencies were being identified and corrected and that significant deficiencies were being submitted as Deviation/Event Reports (DERs) in the established corrective action system. For example, the weekly summary report for the first week of the outage stated that approximately thirty-five hours of observation occurred, that four DERs were generated, and that resolution of posting discrepancies, coaching of radiation workers on the wearing of proper protective equipment, and resolution of industrial safety and housekeeping concerns were accomplished.

Self-assessments by the radiation protection organization addressed benchmarking of HP facilities and equipment, participation in the development of and assessment by an Electric Power Research Institute (EPRI) ALARA program assessment methodology, and a comparison of air sample counting results and derived air concentrations (DACs) produced by different counting equipment. Overall, the self-assessments evidenced generally effective efforts to identify procedural compliance issues, strengths, and weaknesses. For example, the self-assessment of facilities and equipment and resultant Action/Commitment Tracking System (ACTS) items eventually resulted in several significant actions including the establishment of the remote centralized radiological monitoring station and the reconfiguration of the main HP control point.

The corrective action program used the DER and ACTS items systems for implementation. The HP-related DERs for 1999 were reviewed including DER 99-124 (RCA entry through unauthorized access point), DER 99-515 (post-job ALARA review not completed), DER-1860 (discrepancy in SNM inventory of movable in-core detectors), DER-1872 (hot particle contamination event - 357 millirem), and DER-1949 (individual removed CH-203 without HP present). The review included evaluating the adequacy and timeliness of immediate corrective actions and actions to prevent recurrence. In general, this program was effective in identifying deficiencies and in providing timely and effective actions.

c. Conclusions

Overall, the licensee's self-assessment and corrective action processes in the area of radiation protection were effective. The Radiation Protection Program Monitoring Team was effective in resolving minor deficiencies in the field and generating DERs for more significant issues. The HP department's own self-assessments were instrumental in making significant improvements to their facilities and equipment. In general, the corrective action program continued to be effective in identifying deficiencies at a low threshold. In general, the licensee implemented timely and effective corrective actions for findings.

S1 Conduct of Security and Safeguards Activities

a. Inspection Scope (81700)

The inspector reviewed the conduct of security and safeguards activities to determine if they met the licensee's commitments in the NRC-approved security plan (the Plan) and NRC regulatory requirements. The security program was inspected during the period of October 18-21, 1999. Areas inspected included: alarm stations; communications; protected area (PA) access control of personnel, packages and vehicles.

b. Observations and Findings

Alarm Stations Multiple observations of operations in the Central Alarm Station (CAS), and the Secondary Alarm Station (SAS) provided verification that the alarm stations were equipped with appropriate alarms, surveillance and communications capabilities. The inspector's interviews with the alarm station operators found them knowledgeable of their duties and responsibilities. The inspector also verified, through observations and interviews, that the alarm stations were continuously manned, independent and diverse so that no single act could remove the plants capability for detecting a threat and calling for assistance and the alarm stations did not contain any operational activities that could interfere with the execution of the detection, assessment and response functions.

Communications The inspector's document reviews and discussions with alarm station operators, demonstrated that the alarm stations were capable of maintaining continuous intercommunications, communications with each security force member (SFM) on duty, and were exercising communication methods with the local law enforcement agencies as committed to in the Plan.

PA Access Control of Personnel, Vehicles, and Hand-Carried Packages and Material On October 19 and 20, 1999, the inspector observed personnel and package search activities at the personnel access portals. The inspector determined that positive controls were in place to ensure only authorized individuals were granted access to the PA, that all personnel and hand-carried items entering the PA were properly searched, and that vehicles entering the PA were properly controlled and searched.

c. Conclusions

Security and safeguards activities with respect to alarm station controls, communications, and protected area access control of personnel, packages and vehicles were effectively implemented and met licensee commitments and NRC requirements.

S2 Status of Security Facilities and Equipment

a. Inspection Scope (81700)

The inspector reviewed PA assessment aids, PA detection aids, personnel search equipment and testing, maintenance and compensatory measures.

b. Observations and Findings

PA Assessment Aids On October 19, 1999, the inspector evaluated the effectiveness of the assessment aids, by observing on closed-circuit television, two SFMs conducting a walkdown of the perimeter of the PA. The assessment aids had generally good picture quality and zone overlap. However, five PTZ cameras were being used as compensatory measures for minor fixed-camera overlap issues. The licensee initiated maintenance actions to rectify this condition during the inspection period. Additionally, to ensure Plan commitments were satisfied, the licensee had procedures in place requiring the implementation of compensatory measures in the event the alarm station operators are unable to properly assess the cause of an alarm.

Personnel and Package Search Equipment On October 19 and 20, 1999, the inspector observed both routine use and performance testing of the licensee's personnel and package search equipment. Observations and procedural reviews indicated that the search equipment performed in accordance with licensee procedures and Plan commitments.

PA Detection Aids The inspector conducted multiple observations of an SFM conducting performance testing of the perimeter intrusion detection system (PIDS). The testing consisted of intrusion attempts in numerous randomly selected zones, during the camera walkdown. The appropriate alarms were generated in each attempt. The equipment was functional and effective and met the requirements of the Plan.

c. Conclusions

The licensee's security facilities and equipment adequately met the licensee's commitments and NRC requirements.

S3 Security and Safeguards Procedures and Documentation

a. Inspection Scope (81700)

The inspector reviewed the licensee's security program implementing procedures and security event logs.

b. Observations and Findings

Security Program Procedures Verification that the procedures were consistent with the Plan commitments, and were properly implemented was accomplished by reviewing selected implementing procedures associated with PA access control of personnel, packages and materials, testing and maintenance of personnel search equipment and performance testing of PA detection aids.

Security Event Logs The inspector reviewed Security Event Logs for the previous nine months. Based on this review, and discussions with security management, it was determined that the licensee appropriately analyzed, tracked, resolved and documented safeguards events that the licensee determined did not require a report to the NRC within 1 hour.

c. Conclusions

Security and safeguards procedures and documentation were being properly implemented. Event Logs were being properly maintained and effectively used to analyze, track, and resolve safeguards events.

S4 Security and Safeguards Staff Knowledge and Performance

a. Inspection Scope (81700)

The inspector evaluated the licensee's security staff requisite knowledge.

b. Observations and Findings

Security Force Requisite Knowledge The inspector conducted observations of a number of SFMs in the performance of their routine duties during the inspection period. These observations included alarm station operations, personnel, vehicle and package searches, and performance testing of the PIDS. Additionally, interviews of SFMs were conducted. Based on the responses, the inspector determined that the SFMs were knowledgeable of their responsibilities and duties, and could effectively carry out their assignments.

c. Conclusions

The SFMs adequately demonstrated that they had the requisite knowledge to effectively implement the duties and responsibilities associated with their position.

S5 Security and Safeguards Staff Training and Qualification

a. Inspection Scope (81700)

The inspector reviewed security training and qualifications, and individual training records.

b. Observations and Findings

Security Training and Qualifications (T&Q) On October 20, 1999, the inspector reviewed seven randomly selected T&Q records of SFMs. Physical and requalification records were inspected for armed and supervisory personnel. The results of the review indicated that the security force was being trained in accordance with the approved T&Q plan.

Training Records The inspector's review of training records indicated that the records were properly maintained, accurate and reflected the current qualifications of the SFMs.

c. Conclusions

Security force personnel were being trained in accordance with the requirements of the Training &Qualification Plan. Training documentation was properly maintained and accurate and the training provided by the training staff was effective.

S6 Security Organization and Administration

a. Inspection Scope (81700)

The inspector reviewed the effectiveness of the licensee's security management support, and security staffing levels.

b. Observations and Findings

Management Support The inspector's review of the security program implementation since the last program inspection disclosed that adequate support and resources continued to be available to ensure program implementation.

Staffing Levels. The total number of trained SFMs immediately available on shift met the minimum requirements specified in the Plan and implementing procedures. The impact of training, sick time, vacation time and outage support with this minimal level of staffing is manifested by unusually high overtime burden on the security force. The level of staffing and the associated overtime needed to accomplish activities were documented as observations in the last internal QA audit (A98-2OI) in December of 1998. Since that audit, the staffing levels had been further reduced by eight SFMs, resulting in an increased overtime burden. During the current outage, routine exceptions were required to the Indian Point 3 administrative procedure that set overtime hours to be worked, in order to man all regulatory required posts. With the exception of the overtime issue, no performance issues were noted in the areas inspected.

c. Conclusions

The level of management support was adequate to ensure implementation of the security program, and was evidenced by the allocation of resources to support programmatic needs.

S7 Quality Assurance (QA) in Security and Safeguards Activities

a. Inspection Scope (81700)

The inspector reviewed security program audits, problem analyses, corrective actions and the effectiveness of management controls.

b. Observations and Findings

Audits The inspector conducted a review of the annual physical security audit (A98-20I). The audit was thorough and in-depth, and identified 12 deficiencies. The deficiencies were mostly related to administrative controls, and minor documentation errors. None of the audit findings were indicative of programmatic issues.

Problem Analyses The inspector accomplished a review of data derived from the security department's self-assessment program. Potential weaknesses were properly identified, tracked, and trended.

Corrective Actions The inspector's review of the corrective actions implemented by the licensee in response to the 1998 QA audit and self-assessment program indicated that the corrective actions were technically sound and were performed in a timely manner.

Effectiveness of Management Controls The licensee had programs in place for identifying, analyzing and resolving problems. They included the performance of annual QA audits, a departmental self-assessment program and the use of industry data such as violations of regulatory requirements identified by the NRC at other facilities, as a criterion for self-assessment.

c. Conclusions

The licensee's security audit program indicated that the program was being properly administered. In addition, a review of the documentation applicable to the self-assessment program was being effectively implemented to identify and resolve potential weaknesses.

X1 Exit Meeting Summary

The health physics inspector presented inspection findings and results to NYPA management on September 24, 1999; the materials inspector presented results on October 1; and the security inspector presented results on October 21, 1999. On November 10, 1999, the resident inspectors presented the integrated results for the entire inspection period. The licensee acknowledged the findings presented, and did not identify any materials examined during the inspection that were considered proprietary.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

R. Barrett	Site Executive Officer
F. Dacimo	Plant Manager
J. Comiotes	General Manager-Operations
R. Cullen	Senior Chemical Engineer
L. Davis	Project Manager
J. DeRoy	Director, IP-3 Engineering
T. Dougherty	Director - Nuclear Engineering
R. Deschamps	HP General Supervisor
N. Heddle	Senior Engineer Quality Assurance
K. Kingsley	Acting Manager, Licensing
M. Leonard	Security Manager, Corporate
D. Mayer	Acting General Manager-Support Services
J. Odendahl	Security Manager
R. Patch	Director - Quality Assurance
J. Perrotta	Quality Assurance Manager
R. Powers	Manager - Inspection Programs
J. Russell	General Manager-Maintenance
H. Salmon	Vice President - Engineering

INSPECTION PROCEDURES USED

IP 37551:	On-site Engineering
IP 38703:	Commercial Grade Dedication
IP 40500	Corrective Action Program
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 73753:	In-service Inspection
IP 81700:	Physical Security Program for Power Reactors
IP 83750:	Occupational Radiation Exposure
IP 92700:	Event Reports
IP 92901:	Followup - Operations
IP 92902:	Followup - Maintenance
IP 92903:	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

NCV 50-286/99008-01	Inadequate Test Controls Resulted in a Loss of Reactor Coolant Inventory During Residual Heat Removal Operations
NCV 50-286/99008-02	Pressurizer Safety Valves Inoperable, Condition Prohibited by Technical Specifications
NCV 50-286/99008-03	Inadequate Commercial Dedication of Purchased Material, Equipment, and Services Resulted in a Breach of the Reactor Coolant System Pressure Boundary at Normal Operating Pressure and the Declaration of an Unusual Event

Closed

LER 1999-011	"Pressurizer Safety Valves Inoperable with the Reactor Vessel Head On Without an Equivalent Opening of One Valve Flange Established Due to Inadequate Communications; A Condition Prohibited by Technical Specifications"

LIST OF ACRONYMS USED

ACTS	Action Commitment Tracking System
ALARA	As-Low-As-Is-Reasonably-Achievable
ANSI	American Nuclear Standards Institute
AP	Administrative Procedure
ARV	atmospheric relief valve
ASME	American Society of Mechanical Engineers
ASNT	American Society for Nondestructive Testing
ATS	Acquisition Technique Sheets
B&PV	Boiler and Pressure Vessel
CAS	Central Alarm Station
CCTV	closed circuit television
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
DAC	derived air concentration
DER	deficiency/event report
EDG	emergency diesel generator
EPRI	Electric Power Research Institute
FCU	fan cooler unit
FT	flow transmitter
gpm	gallons per minute
HP	Health Physics
HRA	High Radiation Area
IA	instrument air
I&C	instrumentation and controls
IP-3	Indian Point Nuclear Power Plant Unit 3
ISI	In-service Inspection
JPM	job performance measure
LCO	limiting condition for operations
LDE	Lens of the Eye Dose Equivalent
LER	Licensee Event Report
LO	lube oil
MOV	motor-operated valve
NCV	Non-cited Violation
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
NUE	Notice of Unusual Event
NYPA	New York Power Authority
ONOP	off-normal operating procedure
PA	protected area
PCV	pressure control valve
PDR	Public Document Room
PIDS	perimeter intrusion detection system
PM	preventive maintenance
PPT	Plus Point Probe
PTRG	Post-Transient Review Group

QA	Quality Assurance
RCA	Radiologically Controlled Area
RCS	reactor coolant system
RHR	residual heat removal
RO	Refueling Outage
RPC	Rotating Pancake Coil
RP&C	Radiological Protection and Chemistry
RPM	Radiation Protection Manager
RWP	Radiation Work Permit
SAM	small article monitor
SAS	Secondary Alarm Station
SDE	Shallow Dose Equivalent
SFM	security force member
SG	steam generator
SI	safety injection
SRV	safety relief valve
SUPREME	Submersible Platform with ROSA End Defector Motion
T&Q	training and qualification
TEDE	Total Effective Dose Equivalent
the Plan	NRC-approved physical security plan
TLD	Thermoluminescent Dosimeter
TTS	Top of Tubesheet
UDRPS	Ultrasonic Data Recording and Processing System
USNRC	United States Nuclear Regulatory Commission
UT	Ultrasonic Test
VC	Vapor Containment
VCT	volume control tank
WR	work request