June 21, 2001

Mr. A. Alan Blind
Vice President - Nuclear Power
Consolidated Edison Company of
New York, Inc.
Indian Point 2 Station
Broadway and Bleakley Avenue
Buchanan, NY 10511

SUBJECT: INDIAN POINT 2 - NRC INSPECTION REPORT NO. 05000247/2001-004

Dear Mr Blind:

On May 19, 2001, the NRC completed an inspection at the Indian Point 2 nuclear power plant. The enclosed report presents the results of that inspection. The results were discussed on May 24, 2001, with you and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations, and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the inspectors identified four findings of very low safety significance regarding the retention of operator training records, the completion of post maintenance testing, and two issues involving testing of the main steam safety valves that were determined to be violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating the issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Indian Point 2 Nuclear Power Plant.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room **or** from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room). Should you have any questions regarding this report, please contact Mr. Peter Eselgroth at 610-337-5234.

Sincerely,

/RA/

Brian E. Holian, Deputy Director
Division of Reactor Safety

Docket No.05000247
License No. DPR-26

Enclosure: Inspection Report No. 05000247/2001-004

Attachment 1 - Supplemental Information

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	05000247
License No.	DPR-26
Report No.	05000247/2001-004
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	April 1 - May 19, 2001
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector John R. McFadden, Health Physicist Joseph M. D'Antonio, Operations Engineer David Silk, Emergency Preparedness Inspector Greg Smith, Senior Physical Security Inspector Ken Jenison, Project Engineer
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000247/2001-004, on 4/1/01 - 5/19/01; Consolidated Edison; Indian Point 2 Nuclear Power Plant. Licensed Operator Requalification, Maintenance, Post-Maintenance Testing, and Surveillance.

The inspection was conducted by resident and region-based inspectors. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP). This inspection identified all green or no color issues. The “no color” significance level indicates that the Manual Chapter 609 “Significance Determination Process” does no apply to these findings.

Cornerstone: Mitigating Systems

No Color. Con Edison did not have attendance records for an average of 30% of the licensed operator training classes for the years 1998-2000. This issue has minimal safety significance because the facility was able to provide examination/evaluation records of program participation. Con Edison verified operator attendance through written and simulator evaluation records. Corrective actions were addressed in Condition Report 200008293. The failure to have complete records of licensed operator training was contrary to the 10 CFR 55.59(c)(5) and the record retention requirements of Technical Specification 6.19.2.g. This item is being treated as a non-cited violation. (Section 1R11)

Green. Gas turbine #1 (GT-1) failed during a test on May 3, 2000. Con Edison identified degradation in the turbine and compressor sections, and noted significant cracking in the first stage stationary blades. A preliminary assessment concluded the degradation was significant and questioned whether GT-1 could have operated for its design basis mission time. The plant risk associated with all three gas turbines potentially inoperable for a 24 hour period in March 2001 was reviewed using the Significance Determination Process and had a very low safety significance. GT-1 remained out of service pending disassembly, inspection, repair assessment, and a formal operability assessment. (Section 1R13)

No Color. Con Edison identified that corrective actions were not effective to correct a violation related to the completion of post-maintenance testing (PMTs). There were no operability or safety issues related to the outstanding PMTs for safety related equipment that had been returned to service. This matter was a repetitive, licensee-identified violation of TS 6.8.1 having minimal safety significance for the failure to have documented assessment of the outstanding PMTs. This item is being treated as a non-cited violation. (Section 1R19)

Green. The NRC identified that Indian Point Unit 2 failed to take adequate corrective actions to address the effect of ambient temperature on the setpoint of main steam code safety valves, in response to a prior NRC violation, related to pressurizer code safety valve setpoint testing. Because there was no indication that an actual loss of safety function occurred, the Significance Determination Process screened this condition as one of very low safety significance. This violation of Criterion XVI, “Corrective Action,” of 10 CFR Part 50, Appendix B, has been entered in Con Ed’s corrective action system and is being treated as a non-cited violation. (Section 1R22)

Green. The NRC identified that Indian Point Unit 2 (IP2) failed to establish measures to ensure that main steam code safety testing requirements were implemented, while making use of a lift assist device. Because there was no indication that an actual loss of safety function occurred, the Significance Determination Process screened this condition as one of very low safety significance. This violation of IP2 technical specification 4.2.1, Inservice Testing, has been entered in Con Ed's corrective action system and is being treated as a non-cited violation. (Section 1R22)

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<u>Report Details</u>

SUMMARY OF PLANT STATUS

The plant operated at full power during the inspection period, except for a load reduction to 60% on May 6-8, 2001, to repair the 22 main boiler feedwater pump speed control circuit.

1. REACTOR SAFETY
(Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness)

1R04 Equipment Alignments

.1 Partial System Walkdown

a. Inspection Scope (71111.04S)

On April 9, 2001, the inspector performed a partial walkdown of gas turbine (GT) No. 1. At the time, the licensee was performing a major overhaul on gas turbine No. 2 to address ignition basket failures. The references used included check-off list (COL) 31.1, "Gas Turbine 1," Revision 4, emergency operating procedure ECA 0.0, "Loss of All AC Power," Revision 36, abnormal operating procedure (AOI) 27.1.9, "Control Room Inaccessibility Safe Shutdown Control," Revision 31, and AOI 31.1, "Gas Turbine 1," Revision 3.

On April 20, 2001, the inspector performed a partial walkdown of auxiliary feedwater suction path alignment including the condensate storage tank. At the time a temporary plant modification was installed on the city water tank (backup supply to auxiliary feedwater system). The reference used was COL 21.3, "Steam Generator Water Level and Auxiliary Boiler Feedwater," Revision 19.

On April 23, 2001, the inspector performed a partial walkdown of the boric acid system. The licensee had drained and isolated the 21 boric acid storage tank. The references used included COL 3.1, "Chemical and Volume Control System," Revision 30, system operating procedure (SOP) 3.2, "Reactor Coolant System Boron Concentration Control," Revision 17, operator aid TC-3, "Boric Acid Storage Tanks," Revision 4, and technical specification requirements 3.2, "Chemical and Volume Control System."

The partial system reviews were conducted to verify support systems and component alignments were proper, and that both licensee and NRC identified deficiencies (CR 200104059) did not impact system function or operability of GT1, auxiliary feedwater, or the boric acid system.

b. Issues and Findings

No significant findings were identified.

1R05 Fire Protection

.1 Fire Zone Tours

a. Inspection Scope (71111.05Q)

The inspector toured the areas important to plant safety and risk listed below to evaluate conditions related to (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and (3) the fire barriers used to prevent fire damage or fire propagation. The inspector reviewed a sample of fire protection issues within various fire zones entered in the corrective action program over the last 12 months (reference CR 200004285, 200004290, 200006396, 200010401, 200103071). A number of minor material condition issues were identified by the inspector that did not impact fire protection, mitigation, or initiation. The observations were entered into the corrective action program as CRs 200103475, 200103442, 200104152, 200104149, and 200104150.

- Fire Zone 23, Auxiliary Boiler Feed Pump Room
- Fire Zone 2A, Primary Water make Up Pump Area
- Fire Zone 22, Intake Structure Area and Alternate Safe Shutdown Equipment
- Fire Zone 1, Component Cooling Pump Room
- Fire Zone 27A, Primary Auxiliary Building 98 ft elevation

b. Issues and Findings

No significant findings were identified.

1R11 Licensed Operator Requalification

.1 Observation of Simulator Training

a. Inspection Scope (71111.11)

The inspector reviewed training conducted per Lesson Plan SS.406.013 for licensed operators on April 26, 2001, to assess the adequacy of the training, licensed operator performance, emergency plan implementation, and the adequacy of the licensee’s critique. The training considered lessons learned from industry experiences and included instruction and simulator drills on responding to a loss of reactor coolant outside the containment using procedures E-0, E-1, ECA-1.2 and IP-1024.

b. Issues and Findings

No significant findings were identified.

.2 Operator Training Records

a. Inspection Scope

The inspector performed an in-office review of Con Edison's actions in response to unresolved item (URI) 2000-13-01, which was opened because the licensee could not produce full attendance records for a sample of operators whose licenses had recently been renewed. This review included corrective actions stated in CR 200008293, which was written to address the URI.

b. Issues and Findings

(No Color) Con Edison was unable to produce attendance records for an average of 30% of daily classes. Con Edison demonstrated attendance on a weekly basis through written and simulator evaluation records. This loss of records was attributed to a lack of staff, a lack of procedural guidance concerning record keeping requirements, and lack of assigned accountability for attendance records. Corrective actions included assignment of a records' custodian, development of a training administrative procedure describing record keeping requirements, and the use of computer rather than paper attendance tracking.

10 CFR 55.59(c)(5) requires documentation of participation in the requalification program. Technical Specification 6.10.2.g requires that attendance records be kept for the life of the plant. Contrary to the above, the licensee failed to maintain complete attendance records for the years 1998-2000. This issue potentially impacts the ability of the NRC to perform its regulatory function of inspecting compliance with the requalification program. The safety significance of this issue is low because the facility was able to provide examination/evaluation records as documentation of operator participation in the program. This issue is being treated as a non-cited Violation. **(NCV 05000247/2001-04-01)**

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspector reviewed risk significant equipment problems and Con Edison followup actions to assess the effectiveness of maintenance activities. Issues selected for review included licensee evaluation of functional failures, maintenance preventable functional failures, repetitive failures, availability and reliability monitoring, and system engineering involvement. Additionally, the licensee's Maintenance Rule documents and system condition reports (CRs) were reviewed and system engineers were interviewed. The following performance issues are associated with the 13.8 kilovolt (kV) system, auxiliary feedwater system (AFW), and component cooling water system.

- CR 200101361, Loss of 13W92 (13.8kv supply) due to capacitor failure at Buchanan Switchyard (February 3, 2001)
- CR 200101298, Incorrect tap setting on gas turbine auto transformer (February 3, 2001)

- CR 200102173, Air leak on 22 auxiliary boiler feedwater pump hand control valve (HCV)-1118 (March 3, 2001)
- CR 200102753, Valve leak-by from a city water suction valve to 23 auxiliary boiler feedwater pump (PCV-1189) (March 21, 2001)
- CR 200010261, Component cooling water pump 23 oil leak (December 13, 2000)

b. Issues and Findings

No significant findings were identified.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope (71111.13)

The inspectors reviewed and observed the maintenance risk assessments and corrective maintenance work packages for the following emergent work, and discussed the deficient conditions with cognizant personnel (system engineers, maintenance technicians, etc.). The inspector reviewed Con Edison actions to minimize the probability of initiating events and maintain the functionality of mitigating systems. The inspector reviewed work control and equipment restorations to ensure the plant was not placed in an unacceptable configuration.

- Generex Inverter #14 Temporary Fuse per TFC 2001-037, WO 01-20467, and Safety Evaluation 01-195-TM
- Control Rod Fuses work order (WO) 01-20069, Work Step and Plan, April 24-25, 2001 (CR 200104126)
- Replace 10 ampere stationary gripper fuses for control rod D8 per WO 01-20271, April 6, 2001
- 22 MBFP Speed Control WO 01-19915, 01-19682, (CRs 20010701, 1002, and 4458)
- GT-1 Inspections following Test PT-M38A Failure per WOs 99-06506, 01-21601 (CR 200104337)
- 21 EDG Governor servo air lines plugged, WO 01-21703, 21704, 21706, 21 EDG Troubleshooting Plan, CR 200104845

b. Issues and Findings

(Green) Gas turbine #1 (GT-1) failed a monthly test on May 3, 2001 due to burner flame failure. The failure occurred despite previous attempts to restore operability by mitigating air flow disturbances in the compressor section (TFC 2001-009). Con Edison identified degradation in the turbine while conducting visual inspections to identify potential causes for flame irregularities. Con Edison noted significant cracking and gaps at the base of the first stage stationary blades that likely contributed to burner flow disturbances. The inspections also noted other degradation in the compressor and turbine outlet sections. A preliminary vendor representative and site engineering assessment concluded the degradation was sufficient to likely result in turbine failure after 12 to 36 hours of base load operation (CR 200104337).

The degradation in the first stage stationary blades was normal wear for the turbines with extended service time. Blade repairs are a normal part of gas turbine overhauls. GT-1 was last overhauled in 1985; GT-2 was last overhauled in 2001; and, GT-3 was overhauled in 1996. Con Edison noted cracks in the GT-1 blades in December 1997 and determined the cracks did not impact gas turbine operability at that time. Site engineering concluded the GT-1 issues were a Repeat Maintenance Preventable Functional Failure, and the overhaul of GT-1 was a required corrective action to return the gas turbine to a Maintenance Rule a(2) status.

GT-1 remained out of service for the remainder of the inspection period as Con Edison disassembled the turbine to allow full inspection and repair assessments. After performing a damage assessment, Con Edison planned to evaluate the gas turbine's capability to operate for the design mission time, and decide on repair options.

GT-2 and GT-3 remained operable during this inspection period to satisfy the Technical Specification 3.7.C requirements that at least on gas turbine be operable at all times. The risk associated with all three gas turbines being potentially inoperable for a 24 hour period in March 2001 (the previous inspection period) was evaluated using the Significance Determination Process. The inoperability of all three gas turbines was an issue potentially having more than minor significance in that a degraded 13.8KV power supply affects the ability to mitigate a loss of normal power event. This issue affects a Mitigating System cornerstone which affects the ability to mitigate a station blackout. This condition had low risk significance since the three emergency diesel generators were available and the 24 hour potential inoperability of all three gas turbines did not exceed the seven day technical specification allowed outage time.

1R14 Personnel Performance During Non-Routine Plant Evolutions and Events

a. Inspection Scope (71111.14)

During the inspection period, the licensee responded to conditions that required operator actions using special or abnormal procedures. The inspectors reviewed operator performance, reviewed operator logs, reviewed plant data, evaluated procedure adherence, and verified adherence to technical specification limiting conditions for operation. The inspectors reviewed the licensee actions for the events listed below:

- Failure of Control Room Annunciator Circuit Fuses, AOI 27.1.4, (CR 200104626)
- Power Reduction to 60% due to 22 MBFP Speed Control Problems on May 6-8, 2001 (AOI 21.1.2, (CRs 200104458, 4460, 4461, 4463)
- Station Auxiliary Transformer Tap Changer Hangup, SOP 27.1.4, CR 200104543

b. Issues and Findings

No significant findings were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspector reviewed various CRs on degraded or non-conforming conditions that raised questions on equipment operability. The inspector reviewed the resulting operability determinations (ODs) for technical adequacy, whether or not continued operability was warranted, and to what extent other existing degraded systems adversely impacted the affected system or compensatory actions. The following CRs and operability evaluations were evaluated:

- CR 200103901, Diesel Fuel Oil Tank Level Switch USI-A46 Evaluation
- CR 200104543, Station Auxiliary Transformer Tap Changer, TS 3.7.B.3
- CR 200101394, Emergency Diesel Control Power Transfer Switches EDD7, EDD5 (OD 01-01)

b. Issues and Findings

No significant findings were identified.

1R16 Operator Work-arounds

a. Inspection Scope (71111.16)

The inspector reviewed the licensee's list of operator work-arounds. The inspector selected operator work-around 20783, manual operation of the 21 heater drain pump quench water controller (FC-1204), for further review. The selection of this work-around was based upon the potential trip risk and secondary plant transient if quench water to the suction of the 21 heater drain pump was to be isolated. The inspector evaluated if an adverse impact existed on the plant operator's ability to implement abnormal operating procedures or emergency operating procedures with this operator work-around. The inspector verified the condition of FC-1204, talked to nuclear plant operators, and reviewed a two year corrective action history on FC-1204 in the condition reporting system.

The inspector reviewed Con Edison's actions to reduce operator burdens (work-arounds and central control room deficiencies). Con Edison reduced operator burdens by about half (to 24 as of May 16, 2001), and correspondingly reduced operator burden time, a measure of the impact work-arounds have on operators. However, after progress to reduce this backlog leveled off, Con Edison had to develop additional strategies to continue to address operator burdens.

b. Issues and Findings

No significant findings were identified.

The licensee identified that the 22 heater drain pump quench spray controller, FC-1205, was in manual and required compensatory operator actions, yet had not identified the deficiency as an operator work-around. Controller FC-1205 was caution tagged on January 20, 2001, due to failure of the controller in the full open position. The response to the deficiency on FC-1205 was inconsistent with the treatment of a similar deficiency on FC-1204. The licensee documented this minor discrepancy in condition report (CR) 200104060.

1R19 Post Maintenance Testing

.1 Post Maintenance Testing

a. Inspection Scope (71111.19)

The inspectors reviewed post-maintenance test procedures and associated testing activities to assess whether 1) the effect of testing in the plant had been adequately addressed by control room personnel, 2) testing was adequate for maintenance performed, 3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing documents, 4) test instrumentation had current calibrations, range, and accuracy for the application, and 5) test equipment was removed following testing. The following surveillance activities were evaluated:

- Control Rod Checks and Tests per WO 01-20271 and PT-M70, April 24, 2001
- 22 Main Boiler Feedwater Pump Test per PMT-19682, May 8, 2001
- 21 EDG Relay Replacement per PT-M21A and NP-00-17943, 17944, 17945
- 21 EDG Load Test per PT-M21A on May 16, 2001 (CR 200104845)

b. Issues and Findings

No significant findings were identified.

.2 Post-Maintenance Test Backlogs

a. Inspection Scope (71111.19)

The inspector evaluated the effectiveness of licensee corrective actions associated with an inspection finding in report 05000247/2000-01-01. A non-cited violation was issued due to the failure to complete post-maintenance tests (PMTs) on safety-related equipment prior to returning the equipment to service, or to otherwise document an assessment to show that the equipment could perform its intended safety function without performance of the test.

Recent findings by the Con Edison Quality Assurance department identified that the previous corrective actions were not effective and that process issues remain. The corrective actions were not effective due to a lack of rigor within the work control process to ensure that PMTs were completed prior to equipment restoration. The inspector reviewed the current list of outstanding PMTs for safety related equipment that had been returned to service without a documented assessment.

b. Issues and Findings

(No Color) No operability or safety significant issues were identified. The failure to complete PMTs or to have a documented assessment prior to restoring safety-related components back to service is a violation of TS 6.8.1.a. This ConEd identified, repetitive violation has minimal safety significance. This issue is being treated as a non-cited violation **(NCV 05000247/2001-04-02)** consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368) in that it was entered into the licensee's corrective action program as Condition Report 200102569.

1R22 Surveillance Testing

.1 Surveillance Observations

a. Inspection Scope (71111.22)

The inspector reviewed surveillance test procedures and associated testing activities to assess whether 1) the test preconditioned the component(s) tested, 2) the effect of testing was adequately addressed in the control room, 3) the acceptance criteria demonstrated operational readiness consistent with design calculations and licensing documents, 4) the test equipment range and accuracy was adequate and the equipment was properly calibrated, 5) the test was performed in the proper sequence, 6) the test equipment was removed following testing, and 7) test discrepancies were appropriately evaluated and dispositioned.

The inspector reviewed/observed portions of the following surveillance tests and performed a review of related historical data and surveillance performance.

- PT-M70, Control Rod Exercise
- PT-M38A, Gas Turbine #1 Testing and Flame Failure, CR 200104337
- PT-Q28B, 22 Residual Heat Removal Pump (April 19, 2001)
- PT-M21A, 21EDG Load Test, May 18, 2001, CR 200104845 and 4846

b. Issues and Findings

No significant findings were identified.

.2 Main Steam Code Safety Valve-Normal Operating and Testing Environments

a. Inspection Scope

Selected portions of surveillance test, PT-R6, Main Steam (MS) Safety Valve Test Methodology and Leakage, and supporting Indian Point Unit 2 (IP2) documentation were observed and/or reviewed. The processes implemented by IP2, to measure MS code safety valve body and vault temperatures, were observed and evaluated. The actions initiated by IP2 to ventilate the valve vault and the impact of those actions on the established testing environment temperature were also observed and inspected. IP2's corrective actions for a previous, similar NRC violation (247/97-008-02) were inspected. The previous violation involved the normal valve operating environment, a laboratory test environment and an established pressurizer code safety valve (PSV) in as-found testing environment. IP2's violation response, in a letter dated November 6, 1997 and a corresponding licensee event report (LER) 97-013 were also reviewed to verify that IP2 reported its root cause conclusion that removing the roof hatch over the PSV lowered the test environment air temperature surrounding the PSVs and was a significant factor in the as-found, vendor performed, test exceeding its acceptance criteria.

IP2 technical specification (TS) 4.2.1, Inservice Testing, was reviewed to verify that it implemented ASME OM-1 sections I-1.2, I-1.4.1, I-8.1.1, I-8.3.2 and I-8.3.3. Section I-8.1.1 states that the ambient temperature of the operating environment shall be simulated during the set pressure test. If the effect of ambient temperature on set pressure can be established for a particular valve type, then the valve may be set pressure tested using an ambient temperature different from the operating ambient temperature. Section I-8.1.1 further states that correlations between the operating and testing ambient temperatures shall comply with the requirements of sections I-8.3.2 and I-8.3.3. Sections I-8.3.2 and I-8.3.3 state that the owner shall ensure that the correlation established will include acceptance criteria, a written procedure that specifies all test parameters that affect correlation including specific requirements for lift assist equipment, operating conditions, fluid temperature and device temperature. Section I-1.2 describes the terms “ambient temperature” and “normal system operating conditions”. Ambient temperature is the temperature of the environment surrounding a pressure relief device at its installed plant location during the phase of plant operation for which the device is required for overpressure protection. “Normal system operating conditions” are the normal system fluid pressure and temperature during the phase of plant operation for which that system is intended to function.

b. Issues and Findings

The inspectors determined that the IP2 main steam (MS) vault test environment temperature, which existed during the observed portions of PT-R6, was affected by IP2 actions to ventilate the vault and was different than the service operating environment (ambient temperature) normally experienced by the tested code safety valves. OM-1 requires that if the effect of ambient temperature on set pressure can be established for a particular valve type, then the valve may be set pressure tested using an ambient temperature different from the operating ambient temperature if a correlation between the operating and testing ambient temperatures complies with the OM-1. The inspectors determined that impact of the temperature difference on code safety testing was not correlated by IP2, in accordance with OM-1, prior to accepting the MS code safety valves as operable. The inspectors concluded that a temperature difference between operating and test environment temperatures existed within the vault. The existence of a temperature difference between the operating and test environments was determined to not be a function of where temperature was measured. It was further determined that a similar issue was identified as an NRC violation and reported by IP2 in an LER. The inspectors found the corrective actions implemented in response to the violation, which included a review of the MS safety valves for similar issues, were inadequate to prevent the installation of an equivalent ventilation scheme during the December 2000, MS safety valve testing.

This issue has a credible impact on safety, since the effect of environmental conditions on safety relief valve performance is an established phenomenon. If left uncorrected, this would become a more significant concern and could cause an increase in the frequency of an initiating event by affecting the main steam code safety valve lift setpoint. Based on the NRC Significance Determination Process (SDP) group 2 logic, this finding is considered to be of very low safety significance (GREEN) in that there was no indication that a safety function was lost. The failure of IP2 to implement adequate corrective actions to address the impact of the difference between normal operating and testing environment temperatures on MS safety valve setpoint testing is a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions." This violation is being treated as a non-cited violation **(NCV 50-247/2001-04-03)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368) in that it was of very low safety significance and was entered into the licensee's corrective action program as condition report (CR) 200010766.

.3 Main Steam Safety Valve - Lift Assist Testing Device use during Setpoint Surveillance Testing

a. Inspection Scope

The inspectors observed and/or reviewed selected portions of surveillance test, PT-R6, Main Steam Safety Valve Test Methodology and Leakage and a sample of associated test instrument calibration documentation. The documentation included pressure transducer data, linear variable differential transducer (LVDT) calibration data and lift assist set pressure verification device (SPVD) calibration data. SPVD calibration data were inspected to evaluate repeatability error, testing environment parameters, and the development of a SPVD test correlation factor, termed the valve effective seat area (EA). In addition, NRC unresolved item (UNR 05000247/2000-14-06), NRC Information Notice (IN) 94-56, IP2 technical specifications (TS), and the American Society of Mechanical (ASME) Code for Operation and Maintenance of Nuclear Power Plants, Section XI, appendix 1, 1987 (OM-1) were reviewed.

The inspectors observed the measures established by IP2 to test the lift setpoint of selected MS safety valves through the use of a SPVD, that was operated and controlled by an IP2 vendor. Laboratory instrumentation calibration test data for the SPVD were reviewed to evaluate the vendor's calibration methodology. The calibration methodology included several repetitions that compared lift pressure and assist device determined lift pressure for one valve of the type HA-6R10. The inspectors evaluated the process used by the vendor to compare lift pressure to assist device determined lift pressure and develop the EA correlation factor.

IP2 TS 4.2.1, Inservice Testing, was reviewed to verify that it implemented ASME OM-1 sections I-1.4.1 (b) and I-1.8.1.1. Section I-1.4.1 (a) states that all test equipment shall be calibrated to standards traceable to the National Institute of Standards and Technology, and Section 1-1.4.1(b) stated that test equipment used to determine valve set pressure shall have an overall combined accuracy not to exceed +/-1% of the indicated set pressure. Section I-1.8.1.1 states that assist devices may be used for set pressure testing provided their accuracy complies with the requirements of OM-1 section I-1.4.1.

A sample of SPVD laboratory calibration documentation was evaluated to determine if the overall combined loop accuracy included the error contributions of the lift assist device as required by OM-1, Section 1-1.4.1.

b. Issues and Findings

The inspectors reviewed the instrument loop calculations and determined that the overall instrument loop calibration data did not include nor did it reconcile two sources of SPVD error. Therefore, the calibration of the SPVD used by IP2 in the performance of surveillance test PT-R6 did not meet the requirements of OM-1. The first source of error was a SPVD repeatability error. The inspectors determined that a SPVD repeatability error of approximately +/- 0.67% was associated with a 95% assurance, two sigma variance as presented in vendor test data. The second source of SPVD related error resulted from the use of a one point statistical correlation between the single HA-6R10 valve tested in the laboratory and the population of HA-6R10 MS code safety valves installed and tested at IP2. The one point statistical correlation does not account for the physical variance in valve seat effective area between the single valve tested in the laboratory and the population of code safety valves installed at IP2.

This issue has a credible impact on safety since the use of the induced effective area (EA) conversion factor affects code safety valve setpoint determination and is an established phenomenon. If left uncorrected, this issue could become a more significant concern and could cause an increase in the frequency of an initiating event by affecting the main steam code safety valve lift setpoint. Based on NRC Significance Determination Process (SDP) group 2 logic, this finding is considered to be of very low safety significance (GREEN) in that there was no indication that a safety function was lost. Failure to adequately establish an adequate main steam code safety valve test is a violation of IP2 TS 4.2.1, Inservice Testing. This violation is being treated as a Non-Cited Violation **(NCV 50-247/2001-04-04)**, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368) in that it was of very low safety significance and was entered into the licensee's corrective action program as condition report CR 200010766.

The portion of unresolved item (UNR 05000247/2000-14-06) that addresses main steam code safety valve testing is closed. The effectiveness of follow-up corrective actions will be sampled in a future inspection.

.4 Auxiliary Feedwater (AFW) Pump Testing

a. Inspection Scope

The inspectors observed/reviewed selected portions of PT-Q27A, Quarterly Flow Surveillance of the 21 AFW Pump, and supporting materials including engineering calculations, the IP2 Final Safety Analysis Report (FSAR), system descriptions, control drawings and design basis documentation, to ensure that the AFW system and its components were capable of performing their safety function. The inspectors verified whether a new or reconfirmed reference pump performance value was established prior to declaring a pump operable (as required by IP2 Technical Specification 4.2.1, Inservice Testing, which implements Section XI of the ASME Boiler and Pressure Vessel Code). The inspectors reviewed IP2's AFW pump test acceptance criteria, which establish "minimum engineering values" for pump operability and its implementation of ASME Section XI. NRC unresolved item, UNR 247/2000-14-06, was also reviewed.

The normal AFW system configuration, which includes two motor driven auxiliary feedwater (MDAFW) pumps that automatically respond to an engineered safeguards (ESF) signal and a turbine driven auxiliary feedwater (TDAFW) pump that is valved out and must be manually aligned to respond to a design basis accident (DBA), was evaluated. IP2 design basis and AFW system calculations were inspected to determine if DBA assumptions were consistent with the credit taken for TDAFW pump performance in the IP2 FSAR. Based on the documentation provided by IP2, the inspectors verified that the IP2 FSAR does not take credit for flow provided by the TDAFW pump.

The methodology used by the licensee to maintain its design basis and its implemented process for the technical review and approval of vendor supplied calculations, tests and data were reviewed to consider the required AFW flow for a DBA feedline rupture, accompanied by the normal AFW system alignment, the minimum FSAR rated MDAFW pump flow, and a single active failure of the non-associated MDAFW pump.

b. Issues and Findings

1. The inspectors determined that Surveillance PT-Q27A does not establish a new reference pump performance value nor does it reconfirm previous values prior to declaring the pump operable. This is a failure to meet the requirements of IP2 Technical Specification (TS) 4.2.1, Inservice Testing. Because the test results combined with minimum engineering calculation values bound the required reference values and ensured that the pump testing results met the intent of IP2 TS, this is a minor violation of NRC requirements and the AFW related portion of UNR 05000247/2000-14-06 is closed.

2. Although the inspector verified that operation of the TDAFW pump was in accordance with the UFSAR and other supporting documentation, additional NRC assessment was ongoing at the end of the inspection period. For example, although the MDAFW pump, as tested, provides adequate flow, based on the information provided the inspector was not able to determine that the AFW system could automatically provide sufficient cooling of post accident decay and sensible heat while delivering the <u>minimum</u> <u>rated</u> MDAFW

pump flow indicated in the UFSAR. Further NRC review is required to determine the adequacy of the normal AFW system alignment with respect to its response to a feedline rupture. This issue is unresolved. (**UNR05000247/2001-04-05).**

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23)

The inspector reviewed the temporary facility changes (TFCs) and associated safety evaluations listed below to verify the facility changes did not impact safety system operability and the license requirements. The inspection verified the activities were completed in accordance with Con Edison controls for installation and deficiencies were entered in the corrective action system (reference CR 200104149). The following TFCs were reviewed:

- 2001-032, Service Water Temporary Chlorination (SE 01-179-MD)
- 2001-025, 38, 39, City Water Header Repair (SE 01-177-TM)

b. Issues and Findings

No significant findings were identified.

EMERGENCY PREPAREDNESS [EP]

EP4 Emergency Plan Reviews

a. Inspection Scope

The inspector conducted an in-office review of licensee submitted changes for several emergency preparedness documents to determine if the changes decreased the effectiveness of the plan. The review assessed changes to emergency plan implementing procedures related to the risk significant planning standards in 10 CFR 50.47(b) (event classification, notification, radiological assessment and protective action recommendations). Implementing procedures not directly related to the risk significant planning standard were reviewed on a sampling basis. The reviewed documents are as followed:

IP-1007, Dose Assessment (Rev 11)
IP-1011, Joint news Center (Rev 1)
IP-1015, Radiological Surveys Outside the Protected Area (Title Change) (Rev 9)
IP-1018, Media Relations (CANCELED)
IP-1033, Modular Emergency Assessment & Notification System (MEANS) (Rev 0)

b. Issues and Findings

No significant findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope (71121.01)

The inspector reviewed radiological work activities and practices and procedural implementation during observations and tours of the facilities, and inspected procedures, records, and other program documents to evaluate the effectiveness of the licensee's access controls to radiologically significant areas.

The inspector observed activities at the routine radiologically-controlled-area (RCA) control points on a daily basis to verify compliance with requirements for RCA entry and exit, dosimetry placement, and issuance and use of electronic dosimeters. On April 17, the inspector attended the pre-job briefing for a containment entry at power on radiation work permit (RWP) 01-0007. The inspector accompanied a work group which included a radiation protection technician into containment and observed the radiological controls used. Also, the inspector verified the dose rates communicated at the pre-job briefing for this entry by using a radiation survey meter. On April 18, the inspector toured the maintenance and outage building, the fuel storage building, the boric acid building, and the primary auxiliary building and inspected the radioactive material storage locations (within the protected area) and the external boundary of the radiologically-controlled area (RCA) to verify the proper application of radiological controls. On April 19, the inspector attended the pre-job briefing for the changing of a reactor coolant filter (RWP 01-0207) and observed the evolution. During these observations and tours, the inspector reviewed the posting, labeling, barricading, and level of radiological access control for locked high radiation areas (LHRAs), high radiation areas (HRAs), radiation and contamination areas, and radioactive material areas.

The inspector selectively examined the following procedures, records, and other program documents.

- Procedure HP-SQ-3.008, Rev. 19, Radiation Work Permit
- Procedure HP-SQ-3.903, Rev. 10, Contaminated underwater diving operations
- Procedure SAO-107, Rev. 0, Heat stress procedure
- RWP 01-0007, Rev. 1, Inspection of Unit 2 vapor containment to include operation of air locks and first aid support
- RWP 01-0207, Rev. 0, Remove and replace reactor coolant filter as required
- NRC Form 5, Occupational Exposure Record for a Monitoring Period, 09/01/2000 to 12/12/2000, which included a dose calculation for a radiologically-contaminated thumb
- Calculation Number PGI-00461-00 (2000 Thumb Dose Calculation, January 3,

2001)

- 10 CFR Part 50/61 Analysis Report for smear of steam generator piping with a sample reference date of November 2, 2000
- Documented chronology of an event involving the radiological contamination of a thumb
- Results of blind testing of the Panasonic personnel dosimetry system for the fourth quarter of 2000 dated March 10, 2001
- Radiological Assessor Report - February 2001
- Radiological Assessor Report - March 2001
- Radiological Surveys for the EDG Alleyway and the Owner Controlled Area Storage Facility after the Discovery of Soil Contamination During Storm Sewer Excavation in the EDG Alleyway

The inspection included a review of the following Condition Reports for the appropriateness and adequacy of event categorization, immediate corrective action, corrective action to prevent recurrence, and timeliness of corrective action: Condition Reports 2000-03684, 2001-01263, 2001-01481, 2001-02135, 2001-02189, 2001-02349, 2001-02369, 2001-03691, 2001-4222, 2001-4566, and 2001-4571.

The review was against criteria contained in 10 CFR 19.12, 10 CFR 20 (Subparts D, F, G, H, I, and J), site Technical Specifications, and site procedures.

b. Issues and Findings

No significant findings were identified.

2OS2 ALARA Planning and Control

a. Inspection Scope (71121.02)

The inspector selectively reviewed the following procedures, records, and program documents to determine the effectiveness of ALARA (As Low As Reasonably Achievable) planning and control.

- Procedure SAO-303, Rev. 10, ALARA Program
- Procedure SAO-305, Rev. 9, Station ALARA Committee
- Procedure AD-S-2.203, Rev. 6, ALARA Mock-Up Requirements
- Procedure HP-SQ-3.008, Rev. 19, Radiation Work Permit
- Procedure RS-S-8.005, Rev. 4, ALARA Cost-Benefit Analysis Methodologies
- Procedure RS-SQ-8.006, Rev. 5, Radiological Support ALARA Design Review
- Procedure RS-SQ-8.101, Rev. 4, Temporary Shielding Program
- Letter dated December 21, 2000 and titled Person-Rem Exposure Goals for Year 2001 (non-outage Unit 2 goal, 17 person-rem; non-outage Unit 1 goal, 0.7 person-rem) which included a goal breakdown by individual work groups
- Forced Steam Generator Outage 2000 ALARA Report
- Steam Generator Replacement 2000 ALARA Report
- Post Steam Generator Replacement/Refueling Outage 2000 ALARA Report
- Condition Reports 2000-03518, 2000-03684, 2000-04573, and 2000-09603

- Agenda for the Station ALARA Committee Quarterly Meeting on March 21, 2001

During this inspection, the inspector noted that the actual collective dose for the year of 2001 was tracking closely to the projected dose estimate.

The licensee stated that the actual person-rem total for the year 2000 was 588 person-rem which would result in a three-calendar-year-rolling average of 311 person-rem (1998 - 2000). In order to further review the effectiveness of ALARA (As Low As Reasonably Achievable) planning and control for the year 2000, the inspector identified any RWPs which resulted in the accumulation of greater than five rem of actual collective dose in the year of 2000 and for which the actual collective dose exceeded the original estimate by greater than fifty percent. There were eleven RWPs which met this description. The inspector reviewed three of these (i.e., RWPs 0415, 0508, and 0528) to investigate: 1) the possible causes for (and their relative contributions to) the actual dose exceeding the original estimate, and 2) the degree of involvement of the ALARA program in any attempts to minimize the additional dose. Based on this review, the inspector noted that the ALARA program was effectively implemented during the performance of these evolutions.

The review was against criteria contained in 10 CFR 20.1101, 10 CFR 20.1702, site Technical Specifications, and site procedures.

b. Issues and Findings

No significant findings were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope (71121.03)

The inspection included several activities to determine the accuracy and operability of radiation monitoring instruments that are used for the protection of occupational workers, and to determine program adequacy for issuance of self-contained breathing apparatus (SCBA) to occupational workers. The inspector reviewed field instrumentation utilized by health physics technicians and plant workers to measure radioactivity and radiation levels, including portable field survey instruments, hand-held contamination frisking instruments, continuous air monitors, and beta-activity counters. The inspector conducted a review of the instruments located in the toured areas, which included verification of current calibration, and proper function, and certification of appropriate source checks.

The inspection also included a review of Condition Reports addressing SCBA issues. The following Condition Reports were reviewed for the appropriateness and adequacy of event categorization, immediate corrective action, corrective action to prevent recurrence, and timeliness of corrective action: Condition Reports 2001-01590, 2001-1933, and 2001-1945.

b. Issues and Findings

No significant findings were identified.

3. SAFEGUARDS

3PP4 Security Plan Changes

a. Inspection Scope (711130.04)

An in-office review was conducted of the Revision 20 changes to the Physical Security Plan submitted to the NRC in accordance with the provisions of 10 CFR 50.54(p). The review was conducted to confirm that the changes were made in accordance with 10 CFR 50.54(p), and did not decrease the effectiveness of the plan.

b. Issues and Findings

No significant findings were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification

a. Inspection Scope (71151)

.1 Performance Indicator Data Collecting and Reporting

a. Inspection Scope (TI 2515/114)

The inspector reviewed the licensee's performance indicator data collecting and reporting process as described in procedure SAO-114, "Preparation of NRC and WANO Performance Indicators." The purpose of the review was to determine whether the methods for reporting PI data are consistent with the guidance contained in NEI 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guidelines." The inspection included a review of the indicator definitions, data reporting elements, calculation methods, definition of terms, and clarifying notes for the performance indicators. The inspector reviewed licensee actions to address discrepancies in the performance indicator measurements to verify problems were satisfactorily resolved.

Safety System Unavailability-Emergency AC Power System

The inspector reviewed the program for the AC Power System Performance Indicator, and included a review of the data from the operating logs for the 1st quarter of 2001, the 4th quarter of 2000, the 3rd quarter of 2000 and the 3rd quarter of 1999. This PI had been in the White band due to fault exposure hours in 1999 associated with the unavailability of the 23 emergency diesel generator (EDG). The licensee addressed problems with EDG unavailability in the corrective action program and took actions to address the causes for the fault exposure hours. Con Edison recalculated the PI per NEI 99-02 as 1.4% and reported the PI in the Green band for the 1st quarter 2001.

b. Issues and Findings

No significant findings were identified.

4OA4 Licensee Event Report and Open Item Reviews

.1 (Closed) LER 05000247/2001-01: Turbine Trip During Startup. The inspector reviewed the information the licensee provided to analyze this event. The corrective actions for this event were reviewed in NRC Inspection 05000247/2000-15. The LER accurately described the event. This LER is closed.

.2 (Closed) LER 05000247/2000-09: Accumulator Pressure Limits. The inspector reviewed the information the licensee provided to analyze this event. The corrective actions for this issue were reviewed in NRC Inspection 05000247/2000-14. The LER accurately described the event. This LER is closed.

.3 (Closed) URI 0500247/2000-13-01: Operator Training Records. The NRC identified that the licensee did not maintain adequate records of requalification attendance. This matter is discussed in Section 1R11.

.4 (Closed) URI 05000247/2000-14-06: This unresolved item concerned the actions to establish new reference values in PT-Q27A following an overhaul and major maintenance of the auxiliary feedwater pump, and to review the adequacy of main steam safety valve testing. This matter was reviewed in Section 1R22.4 of this report. This item is closed.

.5 (Closed) URI 05000247/2000-14-01: The unresolved item was open pending further NRC review of Con Edison actions to implement 10 CFR 50.65 a(4). Specifically, the inspector identified on December 14, 2000, that Con Edison did not initially perform a risk assessment when gas turbine No. 1 failed during a surveillance test and gas turbine 2 was already out of service for maintenance. The failure to perform a timely risk assessment in accordance with 10 CFR 50.65(a)(4) is considered a minor violation that is not subject to enforcement actions in accordance with Section IV of the NRC Enforcement Policy, because subsequent risk assessments did not require additional actions to manage risk. The inspector confirmed that Con Edison approved station administrative order (SAO)-161, “Operational Risk Assessment,” was effective on January 30, 2001. The inspector reviewed procedural expectations and noted they adequately addressed implementation of 10 CFR 50.65a(4). The inspector reviewed the corrective action program since January, 2001 and no issues were identified that indicated a failure to perform a risk assessment prior to planned maintenance or testing evolutions. This item is closed.

4OA6 Meetings

Exit Meeting Summary

On May 24, 2001, the inspectors presented the inspection results to Mr. A. Blind and other Consolidated Edison staff members who noted the findings. No proprietary information examined during the inspection was included in the inspection findings.

ATTACHMENT 1

LIST OF DOCUMENTS REVIEWED
(not listed in the body of the inspection report)

TRAD 201 Scheduling, Attendance & Classroom Conduct
TRAD 203 Course Documentation/Training Records Requirement

PARTIAL LIST OF PERSONS CONTACTED

T. Burns	Supervisor, Radiation Support
K. Cullen	Health Physics Technician
M. Donegan	Health Physics/Radioactive Waste Manager
R. Fucheck	Health Physics Supervisor
R. Majes	Radiation Support Health Physicist
R. Masse	Plant Manager
L. Menoscal	Radiation Support Health Physicist
L. Mettey	NEM Technician
M. Miele	Radiation Protection Manager
J. Nichols	Operations Training Manager
V. Nutter	Radiation Support Manager
W. Osmin	Reactor Engineer
T. Poirier	Work Control Manager
E. Salisbury	Radiation Support Health Physicist
G. Schwartz	Chief Engineer
W. Smith	Operations Manager
C. Tippin	Reactor Engineer
T. Waddell	Maintenance Manager
E. Woody	I&C Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

05000247/2001-04-01	NCV	failure to maintain adequate records of requalification attendance
05000247/2001-04-02	NCV	failure to complete post-maintenance testing
05000247/2001-04-03	NCV	failure to take adequate corrective actions to address the effect of ambient temperature on the setpoint of main steam code safety valves
05000247/2001-04-04	NCV	failure to ensure main steam code safety testing was adequate while using a lift assist device

Open

05000247/2001-040-05	URI	Auxiliary Feedwater System Design Basis

Closed

05000247/2000-14-01	URI	Failure to Perform a Timely Risk Assessment
05000247/2000-14-06	URI	Reference Values for AFW Pump Testing
05000247/2001-01	LER	Turbine Trip During Startup
05000247/2001-09	LER	Accumulator Pressure Limits

LIST OF ACRONYMS USED

AFW	auxiliary feedwater
ALARA	As Low As Reasonably Achievable
AOI	abnormal operating instructions
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
COL	checkoff list
CR	condition report
DBA	design basis accident
EA	effective area
EDG	emergency diesel generator
EP	emergency preparedness
ESF	engineered safeguard
GT	gas turbine
HRA	high radiation area
IN	information notice
IP2	Indian Point 2
IR	inspection report
kv	kilovolt
LER	licensee event report
LHRA	locked high radiation area
LVDT	linear variable differential transducer
MEANS	modular emergency assessment & notification system
MBFP	main boiler feed pump
MDAFW	motor-driven auxiliary feedwater
MS	main steam
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OD	operability determination
OS	Occupational Safety
PARS	Publicly Available Records
PSV	pressurizer code safety valve
RCA	radiologically controlled area
RWP	radiation work permit
SAO	station administrative order
SCBA	self-contained breathing apparatus
SDP	significance determination process
SOP	system operating procedure
TDAFW	turbine-driven auxiliary feedwater
TFC	temporary field change
TI	Temporary Instruction
TS	technical specifications
URI	unresolved item
WANO	World Association of Nuclear Operators
WO	work order