December 18, 2000

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: NRC's INDIAN POINT 2 INSPECTION REPORT 05000247/2000-013

Dear Mr. Blind:

On November 18, 2000, the NRC completed an inspection at the Indian Point 2 reactor facility. The enclosed report presents the results of that inspection. The results of this inspection were discussed on November 21, 2000, with Mr. John Groth and members of your staff.

NRC inspectors examined numerous activities as they related to reactor safety and compliance with the Commission's rules and regulations, and with the conditions of your operating license. The inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, it involved seven weeks of resident and region-based inspections of engineering, operator requalification testing, operations and maintenance, radiation protection and the work involved with the steam generator replacement project.

The plant staff completed the steam generator replacement and addressed several degraded equipment issues while making preparations to restart the plant. Degraded conditions were noted on the #23 thermal sleeve, reactor protection system relays, support services in the Utility Tunnel, station battery 22, and 480 volt breakers. The NRC evaluated these issues for impact on plant operations. The issues were evaluated under the risk significance determination process and were determined to be of very low safety significance (green). These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report.

During an NRC-requested management meeting in Region I on October 25, 2000, which was held to discuss equipment reliability at the plant, you informed us of your plans to expedite the conduct of a thorough assessment of the 125 VDC system, and you committed to perform numerous system walkdowns/evaluations. The NRC noted some inconsistencies in the quality of these limited "system health" reviews; actions were taken to address the issues.

Our inspectors noted a high number of failures in the licensed operator requalification program and observed some deficiencies during the simulator drills. However, we noted that the failure rate was due to your efforts to improve the exams, and your action to remediate the failures was timely and appropriate.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be placed in the NRC Public Document Room and will be available on the NRC Public Electronic Reading Room (PERR) link at the NRC home page, http://www.nrc.gov/NRC/ADAMS/index.html. 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 05000247/2000-013

cc w/encl:
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REGION I

Docket No.:	05000247
License No.:	DPR-26
Report No.:	05000247/2000-013
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	October 1, 2000 to November 18, 2000
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector Todd Fish, Reactor Engineer Steve Dennis, Reactor Engineer Larry Briggs, Senior Reactor Engineer Paul Bissett, Reactor Engineer Ram Bhatia, Reactor Engineer Larry Scholl, Senior Reactor Engineer Steve Pindale, Reactor Engineer James Trapp, Senior Reactor Analyst George Morris, Reactor Engineer Harold Gray, Senior Reactor Engineer John McFadden, Senior Radiation Specialist
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

Indian Point 2 Nuclear Power Plant
NRC Inspection Report 05000247/2000-013

IR 05000247-00-13, on 10/01-11/18/2000; Con Edison; Indian Point 2 Nuclear Power Plant. Resident Operations Report, Public Radiation Safety, Occupational Radiation Safety and Steam Generator Replacement Project.

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 issues. No findings or issues were identified in the Reactor Safety, Public Radiation Safety and Occupational Radiation Safety Areas. The Licensed Operator Requalification Program inspection identified three No Color findings, one of which was a non-cited violation and one which is an unresolved item. The “no color” significance level indicates that the IMC 609 “Significance Determination Process” does not apply to these findings.

Cornerstone: Reactor Safety

(No Color). The NRC evaluated Con Edison’s actions to review plant systems prior to restart. No operability issues were identified during system walkdowns and status reviews. Most deficiencies were identified by Con Ed; an exception was a problem with a safety injection system pipe support. The NRC noted mixed quality with some reviews (e.g., system engineer preparation appeared inconsistent and some knowledge weaknesses were noted). Some improvements and procedure changes were made, and some systems were reviewed again. Management review of system health presentations met the intent of the administrative procedures. During the initial reviews weaknesses were noted, but were addressed and improvements were noted in later presentations. NRC review of system health issues continued at the conclusion of the inspection.

(Green) Con Edison implemented Modification FPX-00-12449-F to address degraded reactor protection system (RPS) relay conditions and eliminate a potential for multiple relay failures. The reactor protection system (RPS) was not required to be operable since the work was done while the reactor was in cold shutdown. Although the relays had remained functional, the replacement was deemed appropriate to assure the debris from degraded coils would not prevent proper relay operation. The inspector verified that the combination of work controls and post-modification-testing would provide assurance that the RPS would be operable for subsequent plant operations.

(Green) Con Edison completed actions to evaluate a degraded thermal sleeve in the #23 cold leg pipe of the reactor coolant system (RCS) and retrieve loose pieces. Thermal sleeves are internal to certain piping systems to help in minimizing thermal transients. The licensee had previously evaluated six thermal sleeves using radiography earlier in the 2000 refueling outage and incorrectly concluded that #23 sleeve was intact. The finding this period revealed that the radiographs had been incorrectly interpreted. The foreign object search and retrieval (FOSAR) completed after the reactor lower internals were removed recovered a majority of the remnants of the #23 thermal sleeve. Con Edison completed an evaluation for potential remaining piece(s) in the RCS.

(Green) Con Edison completed a risk significance evaluation of the components in the Utility Tunnel. The evaluation consisted of a functionality assessment of the mechanical and electrical components in the tunnel that were degraded due to inadequate supports and pipes corroded from ground water ingress into the tunnel. Portions of the fire protection header were replaced this period to address areas of substantial wall thinning. Long term corrective actions remained in progress to conduct additional engineering walkdowns to identify abandoned services.

(Green) Following replacement of Battery Bank 22, the battery failed a modified performance test when the capacity dropped below 90% (89.7%) prior to the end of the 4 hour test interval. The battery was installed during this period. The battery was considered functional because the capacity was greater than the design basis requirement to provide essential loads for two hours. However, the 22 Battery failed a capacity test on three previous tests during the present outage. Con Edison reported this matter to the NRC per 10 CFR Part 21 by letter dated November 16, 2000, based on a potential defect in the manufacture of the cell plate material. Con Edison continued to evaluate battery performance and finalize an operability determination.

(Green) The 23 auxiliary feedwater pump failed to start during a surveillance due to an electrical problem with the DB-50 supply breaker. The specific failure had low safety significance because only one of the redundant pumps of the system was affected, the breaker did close remotely on a second attempt, and further testing showed the failure did not affect manual closing capability. Corrective actions considered the extent of condition for other DB-50 breakers. This issue appears to be a missed opportunity for the corrective action and preventive maintenance programs to have identified high contact resistance in the breaker closing circuit prior to a demand failure of a safety related component.

Cross-cutting issues: Human Performance

(No Color). The facility has not maintained adequate records of licensed operator participation in the requalification program. Although the licensee has some records supporting proper attendance, daily records were not obtained. This finding is unresolved pending subsequent NRC review of additional information and evaluation of safety significance.

(No Color). The facility has experienced a higher failure rate on the Year 2000 requalification examinations. A high failure rate could indicate poor training and inadequate competence level. However, this did not appear to be the case because the facility had increased the level of difficulty of the written examinations for their Year 2000 exams, and exams administered in 1998 were adequate.

(No Color). The facility did not design their annual operating test such that all Senior Reactor Operators were fully evaluated on implementation of the emergency plan. The safety significance of this finding is low because emergency plan knowledge was tested on the written examination and sampled in the Year 2000 operational examinations as a result of this inspection. This is a non-cited violation of 10CFR55.59(a)(2).

Summary of Findings (cont'd)

Steam Generator Replacement

The activities of the IP2 steam generator replacement project (SGRP), including transport and storage of steam generators, the eddy current inspection of tubes in the replacement steam generators, in-progress radiography of welds, provision for reinstallation of components removed as part of the SGRP and control of work package closeout were noted to be well planned and conducted. Radiation surveys for interim storage of the old steam generators showed all measured radiation levels to be below regulatory limits.

Cornerstone: Occupational Radiation Safety

A) Inspector-Identified Findings

No significant findings or issues were identified.

B) Licensee-Identified Findings

A violation of very low significance which was identified by the licensee has been reviewed by the inspector. Corrective actions taken or planned by the licensee appear reasonable. This violation is listed in Section 4OA5 of this report.

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Report Details

SUMMARY OF PLANT STATUS

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspector verified that the design and implementation of the facility's adverse weather procedures would protect mitigating systems from effects of adverse weather. For this review, the inspector confirmed that the licensee had operational cold weather protection features in place, such as heat tracing, for the suction piping for the containment spray pumps and the suction piping for the safety injection pumps. The inspector also confirmed the licensee had completed adverse weather preparations in accordance with the applicable station directives and procedures, OAD-22, Freeze Protection, SOP30.1, Electric Heat Tracing System, and SOP-11.5, Space Heating and Winterization.

b. Issues and Findings

NRC inspection 05000247/2000-012 had additional findings in this area. No significant findings were identified.

1R04 Equipment Alignment

a. Inspection Scope

Tagouts

The inspectors observed operators cross-connect the 125 volt DC distribution buses 21 and 22 to support a battery load test on the 22 station battery.

System Health Reviews

Con Edison discussed plans to conduct system walkdowns and health reviews prior to plant restart at a public meeting with the NRC on October 25, 2000. The inspectors accompanied system engineers, operators, and maintenance personnel to observe system walkdowns. The system walkdowns reviewed included:

- auxiliary feedwater system
- service water system
- electric tunnel, control room and emergency diesel generator ventilation systems
- residual heat removal and safety injection systems
- isolation valve seal water system
- weld channel and penetration pressurization system
- steam generator blowdown system

- reactor protection, engineered safety features actuation, and nuclear instrumentation systems

Con Edison procedures SE-SQ-12.110, “System Reviews,” and SE-304, “System Health Reports/Presentations,” provided the guidance for the system walkdowns. The system walkdowns evaluated the current materiel condition of systems and the results were provided as an input to system health meetings. The inspectors also observed system engineering presentations to station management on the health of the following systems:

- plant electrical systems (125 volt DC, 120 volt AC, 440 volt AC, 480 volt AC, 6.9 KV, 13.8 KV, 138 KV, and 345 KV)
- Main steam and extraction systems

125 VDC SSFA

The inspectors reviewed plant documents and observed equipment to verify that the 125V DC distribution system and its critical components were correctly aligned per the established operating and licensing basis requirements.

b. Issues and Findings

The inspectors confirmed that no operability issues were identified during the walkdowns. One issue identified by the inspectors during the safety injection system walkdown was a disassembled suction piping support (SR-74). Condition report 200008675 was issued to document this discrepancy and evaluate past seismic qualification of the system. The inspectors confirmed that identified materiel deficiencies were adequately documented in the corrective action program.

The inspectors observed instances of poor preparation, pre-activity briefings, and adherence to management expectations for the system walkdowns. Some operators and maintenance personnel were notified less than one hour prior to some of the walkdown briefings. This was observed during the walkdowns for the safety injection, weld channel and penetration pressurization, service water, and auxiliary feedwater systems. No pre-walkdown briefing was observed on the weld channel and penetration pressurization system. The inspectors noted inconsistencies in the verification of system conditions within contaminated boundaries. Specifically, for the safety injection system one of the team members observed materiel conditions in a contaminated area; however for steam generator blowdown components in the same area the team viewed system condition from outside the contaminated boundary. Weaknesses in system engineering knowledge were evident, in that one system engineer could not define the system’s overall function. During the inspection period, Con Edison revised procedure SE-SQ-12.110 to add pre-walkdown expectations and perform post-walkdown evaluations.

Con Edison re-performed walkdowns on the weld channel and penetration pressurization system and steam generator blowdown systems. The inspector's observations of the system health meeting concluded that management reviews adhered to the expectations of procedure SE-304. However, work order and condition report backlogs were not always fully discussed, especially during initial meetings. For instance, the team did not question why deferral of a plant modification for the static inverters was acceptable during power operation (with a consequence of increasing maintenance rule unavailability). Overall, the results of the initial system health meetings followed the system engineers recommendation. Improvements were noted in subsequent meetings, and significant weaknesses were satisfactorily addressed. It is important to note that a significant maintenance and corrective action backlog for each of the systems was still evident.

No significant findings were observed by the inspectors during review of the licensee's assessment of the 125 volt DC system. The inspectors observed that operator aid 98-19 on the 22 DC power panel was recently outdated due to a plant modification. Con Edison initiated condition reports 200008921 and 200008975 to document this observation and address program controls to update operator aids following modifications.

1R05 Fire Protection

a. Inspection Scope

The inspector conducted tours of areas important to reactor safety, listed below, to evaluate conditions related to 1) licensee control of transient combustibles and ignition sources; 2) the materiel 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 also reviewed Con Edison actions to address deficiencies in the public address system (reference corrective actions for CRs 2000-6598, 6508, and 6504, Work Order 17570, and plans for permanent upgrades per engineering request RES 12327-00). Fire protection controls were reviewed for the following areas:

gas turbine 1
station auxiliary transformer
unit auxiliary transformer
21, 22, and 24 battery rooms

b. Issues and Findings

No significant findings were identified.

1R11 Licensed Operator Requalification

.1 Technical Training Program

a. Inspection Scope

The inspector reviewed a report regarding the facility's Technical Training Program. Findings in the report were the basis for Technical Training being placed on probation for 120 days, with another evaluation to be performed in January 2001. The inspector identified no immediate safety issues or concerns.

b. Issues and Findings

No significant findings were identified.

.2 Training Exam Results

a. Inspection Scope

The inspector reviewed licensee actions resulting from licensed operator requalification training exam failures. Additionally, the inspector reviewed the licensees examination quality as delineated in NUREG 1021 Rev.8, Operator Licensing Examination Standards for Power Reactors. The inspector also reviewed licensee actions to train operators relative to the operation on the new steam generators and plant restart.

b. Issues and Findings

No significant findings were identified.

.3 Requalification Program Review

a. Inspection Scope

A review was conducted of operating history documentation from a sample of NRC inspection reports, licensee condition reports (CRs), and the NRC plant issues matrix (PIM). The inspectors selected the specific events which reflected performance deficiencies and verified that they had been addressed in training by review of appropriate lesson plans and scenario exercises. These deficiencies included excessive discharge of number 24 battery (related to the August 1999 reactor trip), the improper implementation of emergency action levels (also related to the August 1999 reactor trip), an excessive cooldown (related to the February 2000 steam generator tube failure), and a failure to detect automatic rod insertion (related to a February 1999 event). The review also included Indian Point 2 PSA risk insights evaluation of training on the replacement steam generators (SG).

A sample of the written and operating exams for licensed personnel for the years of 1998, 1999 and 2000, including remedial training activities, were reviewed. These exams were compared for duplication of questions and differences in level of difficulty from year to year. Content of the examinations was reviewed against the requirements of 10 CFR55.59 and the NRC Examiner Standards.

Observations were made of operating test administration to one shift crew and the facility's evaluation of crew and individual operator performance. A review was conducted of response to training feedback by students and incorporation of plant and industry events into the training program for the two year training cycle. A sample of medical records, training attendance records, and documentation on maintaining an active license was reviewed.

b. Findings

No color. The inspector reviewed documentation related to three license reactivations. The records indicated the three operators met the watch standing requirements (at least 40 hours watch-standing under instruction, including a plant tour). Records for attendance at requalification, however, were not available for review, and therefore, the requirement to be current in requalification classes could not be verified. The facility reported being able to document attendance on a weekly basis by examination and evaluation results, but individual class attendance records are disorganized and incomplete. The facility does not know if records were lost or never accurately kept. This failure to keep complete attendance records is contrary to the requirements of the Indian Point 2 Licensed Operator Requalification Program description and Technical Specification 6.10.2.g, which require attendance records be kept for the life of the plant, and of 10 CFR 55.59(c)(5), which requires documentation of participation in the requalification program. This issue is unresolved pending verification of the weekly documentation and further assessment of safety significance. **(URI 05000247/2000-13-01)**

No color. The content of the operating examination was reviewed against the criteria of 10 CFR 55.59(a)(2)(ii), which references 10CFR55.45(a)(2) through (15). 10 CFR 55.45(a)(11) requires that senior reactor operators (SRO) be required to demonstrate their abilities in implementing the emergency plan. Contrary to this requirement, in the 1998 and 1999 annual operating tests, the facility did not evaluate SROs other than Shift Managers on the use of the emergency plan during the operating test. This implies that the SROs not qualified as Shift Manager would not be "at risk" for testing in a crew in an operating examination. This noncompliance was mitigated by the testing of these SROs on this topic on their written examinations. **(NCV 05000247/2000-13-02)**

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspector reviewed selected risk significant equipment problems that have occurred. Items checked included licensee evaluation of functional failures, maintenance preventable functional failures, repetitive failures, availability and reliability monitoring, and system specialist involvement. Additionally, the licensee's Maintenance Rule Basis Document and system condition reports were reviewed and system engineers were interviewed. The following system/component performance issues were assessed:

- 21 RHR pump oil sample and surveillance testing (CR 199908659)
- RHR valve packing leaks (CR 200002844)
- Preventative maintenance on motor control center 26A and disconnect switch 2D (CR 200002694)
- Fuel and Core Component Handling (CR 200004573, Significance Level 2)

c. Issues and Findings

(Green) Con Edison conducted a detailed review of fuel and core component handling (FCCH) system deficiencies identified during the Cycle 14/15 refueling outage and the steam generator replacement outage. The FCCH is in a Maintenance Rule A-1 status due to three maintenance preventable functional failures: manipulator crane undocumented jumper (reference Inspection 05000247/2000-005), the calibration of the cone drive torque setpoints, and the fuel storage building upender problems (reference Inspection 05000247/2000-009). The causes for the problems included inadequate preventive maintenance and training weaknesses. Short term and long term corrective actions addressed system performance and reliability. The licensee entered these problems in the corrective action program as condition reports 2000-2608, 2662, 2667, 2682, and 2692.

No significant findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspector evaluated the effectiveness of the risk assessments performed before maintenance was conducted and verified how the licensee managed the risk. The inspector verified that the licensee took the necessary steps to plan and control the resulting emergent work activities. Additionally, it was verified that the licensee had adequately identified and resolved maintenance risk assessments and emergent work problems. The following maintenance risk assessments and/or emergent performance issues were assessed:

- RHR System Heat Exchanger Gasket Replacement, WO 96-82852
- RPS Relay Replacement (WO 00-18345, 18691,CR20006714, FPX-00-12449-F)
- Isolation of fire protection supply lines associated with utility tunnel repairs (CR 200008065, WO 98-00739, WO 00-14667, WO 98-04803, WO 98-00637, WO 97-89781, WO 97-89780, WO 97-95462, WO 99-12938, WO 00-17745)

b. Issues and Findings

(Green) Con Edison implemented Modification FPX-00-12449-F to address degraded relays in the reactor protection system (RPS) and eliminate a potential for multiple relay failures. Con Edison replaced normally energized BFD relays in RPS racks E (Train A) and F (Train B). A total of 88 relays were replaced (44 in each train). The work was completed under work order 00-18345 (and others) and IC-SL-007, which provided the detailed instructions for the changeout. The RPS was not required to be operable since the work was done while the reactor was in cold shutdown. The relays were replaced after noting that the coil insulators had degraded over time due to heat generated by the normally energized relays (reference CR 2000-6714, 3111). Although the relays had remained functional, the wholesale replacement was deemed appropriate to assure the debris from a degraded coil would not fall into a relay at a lower level in the RPS racks and prevent proper relay operation. The new replacement relays have a heavier coil insulation to reduce the effects of thermal aging. The spacing between the relays inside the racks was also increased where possible to minimize relay-to-relay contact. The inspector verified that Con Edison had selected the appropriate scope for the relay replacement. The inspector observed the relay replacement in progress, independently verified the wiring on a sample of relays, and verified that the work controls were sufficient to maintain control of the RPS logic configuration. The inspector verified that the combination of work controls and post-work testing scope and method would verify the continuity of each termination disturbed by the replacement and thereby assure that the RPS would be operable for subsequent plant operations. NRC review of the relay replacement and post-maintenance testing activities continued at the end of the inspection period.

No significant findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspector reviewed the following operability evaluations to verify they were completed in accordance with licensee procedures and NRC requirements:

99-040, Gas Turbine Remote Start, Rev 1 May 24, 2000 (SE 00-378-EV, CR 20004559)
00-004, Safety Injection Accumulators
00-012, Reactor Vessel Foreign Objects
00-016, Reactor Thermal Sleeve Loose Parts (SE-SQ-12.317, CR 2000-6240, 7037)
Technical Report 00263-TR-001, Functionality and Risk Significance Evaluation of the Indian Point 1and 2 Mechanical and Electrical Systems in the Utility Tunnel

The inspector reviewed licensee evaluations and completed walkdowns of plant areas to independently evaluate licensee conclusions.

b. Issues and Findings

Thermal Sleeve

(Green) Condition report 200007037 describes Con Edison's findings for the failure and retrieval of the thermal sleeve in the #23 cold leg pipe of the reactor coolant system. The licensee had previously evaluated the thermal sleeves using radiography earlier in the 2000 refueling outage and incorrectly concluded that the #23 was intact. The findings this period revealed that the radiographs had been incorrectly interpreted. The inspector reviewed the radiographs to verify no other thermal sleeves were missing. The foreign object search and retrieval (FOSAR) completed after the lower internals were removed recovered the remnants of the #23 thermal sleeve. Con Edison estimated from the total surface area of pieces in hand that 95% of the thermal sleeve has been recovered. Safety evaluation SE-SQ-12.317 and OD 00-16 documents Con Edison's basis that IP2 can safety operate without a thermal sleeve and with the remaining piece(s) in the RCS. Con Edison planned further evaluations for loose parts in the reactor coolant system using the digital impact monitoring system during plant startup.

Utility Tunnel - Unit 2 Support Services

(Green) Con Edison completed a risk significance evaluation of the components in the Utility Tunnel. The evaluation consisted of a functionality assessment of the mechanical and electrical components in the tunnel that were degraded due to inadequate supports and pipes corroded from ground water ingress into the tunnel. The evaluation included the results of ultrasonic inspections of corroded piping. Although most utilities in the tunnel are Unit 1 services which have been abandoned in place, Con Edison identified several services that support Unit 2 operation, including: city water providing backup supply to the auxiliary feedwater system; fire protection water in 8, 10 and 12 inch headers; fuel oil; feeder cables for 13.8 kV breakers 52/B3-3 and 52/R1-4, and other non-safety related services. Con Edison concluded that all components evaluated remained functional with margins to allowable stress levels in piping affected by localized corrosion. Short term corrective actions were initiated and remained in progress at the conclusion of the inspection to address degraded fire water piping and supports. Long term corrective actions remained in progress to conduct additional engineering walkdowns to identify abandoned services that should be removed as a modification, and finalize long term repairs and upgrades.

No significant findings were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post-maintenance test procedures and test activities to verify the adequacy of system operability and functional capability. The inspectors witnessed tests and/or reviewed the test data to verify the equipment met the design/licensing bases requirements and commitments and to demonstrate that the equipment was capable of performing its intended safety functions. The effect of testing on the plant was reviewed. Additionally, the inspectors reviewed the testing to verify: it adequately addressed the scope of the maintenance work performed; acceptance criteria was clear and demonstrates operational readiness; and, that test equipment range and accuracy is consistent with the application. Finally, following the completion of testing, the inspectors reviewed whether equipment was returned to the positions/status required for the equipment to perform its safety function.

The inspector also reviewed the licensee identified surveillance testing problems to assess proper entry in the corrective action program, with appropriate corrective actions identified. The following system/component post maintenance tests were reviewed:

- 23 AFP per PMT PT-Q27B. The test failed due to the failure of the pump breaker to close on the first attempt. (CR20007576, 20007574)
- RHR System Heat Exchanger Leak Check, PMT# 82852 (WO 96-82852)
- Gas Turbine 1 per PMT 18601. The PMT failed due to a failure of the starting diesel generator (CR 200008907 and WO 00-18601)
- DB-50 480 volt breaker alarm switch intermittent failure (CR 200007618)
- PT-2M2, RPS Logic Testing (WO-NP-00-18358) following BFD relay replacement due to degraded coil insulators (reference modification FPX-00-12449-F).

b. Issues and Findings

(Green) The 23 auxiliary feedwater pump failed to start on the first attempt during a surveillance due to an electrical problem with the DB-50 supply breaker (reference CR 2000-7576, 7618). The specific failure had low safety significance because only one of the redundant pumps of the system was affected, the breaker did close remotely on a second attempt, and further testing showed the failure did not affect manual closing capability. The breaker problem was attributed to an alarm switch contact with intermittent high resistance in the breaker closing circuit. Corrective actions were appropriate to consider the extent of condition for other DB-50 breakers used in safety systems that receive an automatic signal to close following an accident. Several condition reports in the last year document problems with alarm switches having resistance values greater than acceptance criteria but which did not prevent breaker operation. It is not clear that critical attributes, such as alarm switch functioning, were properly identified in past preventive maintenance (PM) activities. This appears as a missed opportunity for the corrective action and the PM programs to have identified this issue prior to a demand failure on a safety related component. NRC review of Con Edison actions in this area continued at the end of the inspection period.

No significant findings were identified.

1R20 Refueling and Outage Activities

a Inspection Scope

The inspectors reviewed the following activities related to the Unit 2 refueling and maintenance outage for conformance to the applicable procedure, and witnessed selected activities associated with each evolution. Operations were reviewed to verify completeness within the technical specification and procedure requirements.

- spent fuel pool operations per SOP 4.3.1
- shutdown risk evaluations per OAD-38
- operation on the residual heat removal system per SOP 4.2.1
- SOP 17.31, Refueling Surveillance Checklist

Corrective actions were reviewed for issues described in condition reports (CR) and entered in the corrective action system.

b. Issues and Findings

No significant findings were identified.

1R21 Safety System Design and Performance Capability

a. Inspection Scope

The inspector assessed the performance of the licensee's team conducting the 125V DC system safety system functional assessment (SSFA) by attending daily SSFA team meetings, and by observing selected significant issue discussions of the SSFA team members with the licensee's staff personnel. The inspector also reviewed the SSFA team scope of work and, the list of questions generated. The inspector selected portions of the system and components to accompany walkdowns with SSFA team members and independently assess. The inspector discussed selected electrical issues with SSFA team members and licensee personnel.

The inspector also reviewed the 125V DC design and licensing basis documents, including the Updated Final Safety Analysis Report (UFSAR), plant Technical Specifications (TS) and design basis documents (DBD)s to determine whether the issues identified by the SSFA team affected the system and component functional requirements during normal and accident conditions.

b. Issues and Findings

There were no findings identified. Overall, the licensee's SSFA team performed a good quality assessment of the 125V DC system. The SSFA team was well organized with independent, knowledgeable and SSFA experienced contractors, and had developed a good inspection plan. Good technical issues were raised to assess the margin existing in the DC system.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed the surveillance tests and/or reviewed the test data of selected risk-significant SSCs listed below to assess, as appropriate, whether the SSCs met technical specification requirements, updated final safety analysis report, and licensee procedure requirements, and to determine if the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions.

The inspector also checked that the licensee identified surveillance testing problems at the appropriate threshold and entered them in the corrective action program and implemented appropriate corrective actions.

- PT-R76B, Station Battery 22 Load Test (CR 2000-6657, 5366, 4648, 4401)
- PT-Q25A, 21A Closed Cooling Water Pump
- PT-M38A, Gas Turbine No. 1 (CR 200008907 and CR 200008991)

b. Issues and Findings

(Green) The above CRs were placed in the licensee's corrective action program through the condition reports identified above. Con Edison tested the capacity of Battery Bank 22 using surveillance PT-R76B to verify it was operable. The battery failed the surveillance test when the capacity dropped below 90% (89.7%) prior to the end of the 4 hour test interval. The battery was installed during this period and thus there was no safety impact on past plant operations. The battery was considered functional because the capacity was greater than the design basis requirement to provide essential loads for two hours. However, the 22 Battery failed a capacity test on three previous tests during the present outage. Batteries 21, 23 and 24 have operated and tested satisfactorily. Con Edison worked with the battery vendor and an independent testing laboratory to identify the causes for the battery performance and appropriate corrective actions.
Con Edison reported this matter to the NRC per 10 CFR Part 21 by letter dated November 16, 2000 as a defect in the Model 2GN23 1800 amp-hour battery.
Con Edison concluded after microscopic analysis of battery cells that cracks had formed within the cell plate material. The cracks appear to have been produced during the manufacturing process. Con Edison actions continued to evaluate the battery performance and prepare an operability determination. NRC review of this matter continued at the end of the inspection period.

No significant findings were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspector reviewed Temporary Facility Change (TFC) 2000-014 regarding removal of mechanical stops on the Fuel Storage Building Crane. The TFC will allow the crane hook to travel over the spent fuel pool under specified conditions. Technical Specification (TS) compliance, design basis documentation, and tag placement were verified by the inspector.

b. Issues and Findings

There were no findings identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope (71121.01)

The inspector reviewed the following ongoing exposure significant activities and reviewed the following radiation work permits (RWPs), surveys, and reports to evaluate the effectiveness of the licensee’s access controls to radiologically significant areas.

- RWP 517 SGR-SGT and subcontractors to remove/install primary and secondary small bore interferences for steam generator (S/G) 21-24
- RWP 520 SGR-SGT and subcontractors to remove/install structural/electrical interferences, lateral supports, snubbers/support shoes, and temporary reactor coolant system (RCS) supports/restraints for S/G 21-24
- RWP 533 Application of instacote on surfaces of refueling cavity
- RWP 583 Remove fly ash from duct and place into crates
- RWP 633 Cleanup transfer canal for dose reduction
- Direct frisk and removable contamination surveys of duct expansion joint assemblies on September 27, 2000 for work on RWP 583
- Airborne activity survey log for work on RWP 583 on October 18, 2000
- Gamma spectroscopy report on October 6, 2000 at 0940 hours for fly ash for work on RWP 583
- Dose calculations for a right thumb contaminated on October 09, 2000
- Personnel contamination investigation report for particle contamination on November 16, 2000

The inspection included a review or 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 200007099, 200008040, 200008041, 200007860, 200008129, 200008112, 200008504, and 200009078.

Condition Report 200007860 addressed dose rates at the boundary of the interim storage area for the old steam generators. On November 13, 2000, the inspector verified the dose rates in the interim storage area surveys performed on October 28 and 30, 2000, and on November 2, 2000.

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. Findings

No significant findings or issues were identified during the inspection in this area.

2OS2 ALARA Planning and Control

a. Inspection Scope (71121.02)

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

- RWP packages for RWPs 533 and 633 including the ALARA reviews, surveys, and ALARA pre-job briefing records
- Daily ALARA report for November 13, 2000 (outage day 97)(Unit 2 steam generator replacement project)
- Steam Generator Replacement Project (SGRP) ALARA subcommittee meeting minutes on October 04, 10, 18, and 24 and November 01 and 09, 2000

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

b. Findings

No significant findings or issues were identified during the inspection in this area.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope (71121.03)

The inspector reviewed the following facilities and equipment and reviewed the following program documents to determine the effectiveness of radiation monitoring and survey instrumentation.

To check on the operability and calibration of installed radiation and radioactivity monitors, the inspector participated in plant walk downs on November 14 and 15, 2000. During these tours, the inspector identified and noted any accessible local response information on five of five Unit 1 area radiation monitors, five of six Unit 2 area radiation monitors, twenty-three process/effluent radiation monitors, and twelve of fourteen post-accident area/process/effluent monitors. The inspector also reviewed for compliance and adequacy the following calibration records and documents for installed monitors.

- Procedure PC-EM 4, Rev. 5, Non-VC Area Monitors R-1, R-4, R-5, R-6, R-8, R-5987, R-37-1, R-37-2, and R-37-3 Calibration, performed January 31 to March 6, 2000
- Procedure PC-EM 4, Rev. 6, Non-VC Area Monitors R-1, R-4, R-5, R-6, R-5987, R-37-1, R-37-2, and R-37-3 Calibration, not yet performed
- Procedure PC-R15B, Rev. 12, Area Radiation Monitors R-2 and R-7 Calibration, performed on October 20 and 21, 1999
- Procedure PC-R15B, Rev. 13, Area Radiation Monitors R-2 and R-7 Calibration, performed on October 24 to 27, 2000
- Procedure PC-R25, Rev. 7, Main Steam Line Radiation Monitors Radiation Calibration, performed on September 10 to 24, 1999
- Procedure PC-EM30, Rev. 5, Process Radiation Monitor R-41/42 Calibration, performed on January 5 to 11, 2000
- Procedure PC-R38, Rev. 6, High Range Containment Area Radiation Monitors Calibration, performed on May 2 to June 1, 2000
- Procedure HP-3.201, Rev. 5, Use of Calibration Sources
- Installation and Operation Manual for Series 10 Single Source Calibrator (Cesium-137, 100 millicuries on 09-15-94, SN 11071)
- Traceability certificates for Barium (Ba)-133 source model NES-9007 (SN 314F), Ba-133 source model NES-9007 (SN 314G), Ba-133 source model NES-9007 (SN 314H), Ba-133 source model NES-8060 (SN 372A), Cesium (Cs)-137 source model 848-8 (SN 179), Cs-137 source model 878-10 (SN 113), Cs-137 (SN D-550), Cs-137 source model NES-131S (SN 314B), Cs-137 source model NES-9017 (SN334A), Cs-137 source model GF-137D (SN 422-49-1), Cs-137 source model GF-137D (SN 422-49-2), Cs-137 source model GF-137D (SN 422-49-3), Chlorine (Cl)-36 source model BF-036 (SN 370-80-2), Cobalt (Co)-60

source model NES-9023 (SN 334B), Strontium (Sr)-90 source model NES-9073 (SN 314D), Strontium (Sr)-90 source model NES-9073 (SN 314J), and Strontium (Sr)-90 source model NES-9073 (3.9N4 microcuries nominal).
- Communications to staff for change to source certification value from 4.35N1 to 4.83N1 in procedures PC-EM28, PC-EM30, PC-EM31, PC-EM32, and PC-EM33
- 10 CFR 50.59 Safety evaluation for removal of requirements for radiation monitor RE-101
- 10 CFR 50.59 Safety evaluation for removal of area radiation monitor R-10 from functional service
- 10 CFR 50.59 Safety evaluation for installation of a 3-hour-rated fire door in fire zone 6A (El. 80' PAB) and removal of radiation monitor R-8

For health physics instrumentation, the inspector evaluated the following calibration records, procedures, documents, facilities, and equipment for regulatory compliance and adequacy.

- calibration records for Gamma 60 portal monitors, whole-body-personnel contamination monitors, RO2 and RO2A radiation survey meters, teletectors, RO7 survey meters, Triton Model III air samplers, AM 2 area monitors, lapel air samplers, low and high volume air samplers, and continuous air samplers
- Procedure HP-9.570 Calibration of Bicron RSO-5/RSO-50 and Eberline RO-2/RO-2A Ion Chambers
- Procedure HP-9.592 Operation of the NNC Model Gamma 40/60 Portal Monitor
- Procedure DOS-6.125 Calibration of Merlin-Gerin Electronic Dosimeter and CDM21 Calibrator Using Windows
- Procedure DOS-6.300 Calibration of Williston Elin WE 2001 Irradiator
- certification as a National Voluntary Laboratory Accreditation Program (NVLAP)-Accredited Ionizing Radiation Dosimetry Processing Facility effective through June 30, 2001
- facilities and equipment including an instrument repair facility, an electronic dosimeter calibration facility, a multiple source gamma calibrator for daily source checks on radiation survey meters, and a Shepard beam calibrator

The inspection in this area included a review or 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 200007524, 200008248, and 200008546.

The inspection included a review of the status and surveillance records of self-contained breathing apparatus (SCBA) staged and ready for use in the plant, the licensee's capability for refilling and transporting SCBA air bottles to and from the control room and operations support center during emergency conditions, and the status of training and qualification in the use of SCBA for appropriate personnel. The following documents and procedures were examined in the course of this review.

- Lesson plan for SCBA training for fire brigade, FBT-C-002, April 26, 1999
- Lesson plan for SCBA training for all personnel, FBT001002, January 6, 2000
- Handout for SCBA training for all personnel
- Lesson plan for lab for donning SCBA, FBT001002L, January 6, 2000

- Qualification status listing for respirator use
- Station Administrative Order (SAO)-300, Radiation Protection Plan, 4.6.1.4.c, clean-shaven policy prior to respirator use
- Operations Administrative Directive 15, Policy for conduct of operations, 4.1.25(2) clean-shaven policy for assuming the watch
- LER 99-008 Deficiency in respirator qualification
- EP-7.801 Inventory and control of SCBA units and spare tanks
- EP-7.802 Cleaning, sanitizing, and inspection of self-contained breathing apparatus
- EP-7.803 Calibration and repair of self-contained breathing apparatus (SCBA)
- EP-7.804 Filling self-contained breathing apparatus (SCBA) cylinders using the Eagle air system
- SCBA inspection record for October 2000
- SCBA cylinder inspection record for November 2000

The review was against criteria contained in 10 CFR 20.1501, 10 CFR 20 Subpart H, site Technical Specifications, and site procedures.

b. Findings

No significant findings or issues were identified during the inspection in this area.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Review

.1 Performance Indicator Data Collecting and Reporting

a. Inspection Scope (71151)

The inspector reviewed the licensee's performance indicator data collecting and reporting process as described in procedures 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, calculational methods definition of terms, and clarifying notes for the following indicators: safety system functional failures and safety system unavailability.

b. Issues and Findings

No significant Findings were identified.

.2 Performance Indicator Verification - Unplanned Power Changes

a. Inspection Scope (71151)

The inspector examined corrective action program records, control room logs, licensee event reports, and past NRC inspections reports for occurrences involving high pressure safety injection system unavailability. The inspector specifically reviewed data for the 2nd and 4th quarters of 1999. The inspector guidance in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0, was consulted to verify that plant data was properly identified within the published performance indicators.

- Safety System Functional Failures (4th quarter of 1999 and 1st quarter of 2000)
- Unplanned power changes per 7,000 critical hours (3rd quarter of 1999 and 2nd quarter of 2000)

b. Issues and Findings

No significant findings were identified.

.3 Performance Indicator Verification - Occupation Exposure Control Effectiveness

a. Inspection Scope

The inspector selectively examined records used by the licensee to identify occurrences involving high radiation areas, very high radiation areas, and unplanned personnel exposures for the time period from August 14, 2000 to October 17, 2000 against the applicable criteria specified in NEI 99-02, Regulatory Assessment Performance Indicator

Guideline, Revision 0, to verify that all conditions that met the NEI criteria were recognized and identified as Performance Indicators. The reviewed records included corrective action program records (Condition Reports) and radiologically controlled area (RCA) access control alarm reports. This examination, in conjunction with the reviews of Condition Reports documented in previous inspection reports (Report Nos. 05000247/2000-005, -009, and -011), did not find any problems with the PI accuracy or completeness and thus verified this performance indicator.

b. Issues and Findings

No significant findings or issues were identified.

4OA2 Identification and Resolution of Problems

a. Inspection Scope

A facility independent self-assessment of training was reviewed and corrective actions for specific deficiencies were evaluated. These deficiencies included poor quality written exam material, inadequate remediation for written exam failures, lack of a sample plan for operating tests, and lack of evaluation of the PSA (plant safety analysis) for training insights. The inspector reviewed facility resolution of these self-assessment findings. The inspector also reviewed facility corrective action and end of 1999 requalification exam review which identified weaknesses in emergency operating procedure (EOP) knowledge and use as a contributing factor in crew and individual failures.

b. Findings

No color. The facility has experienced a high failure rate on the Year 2000 requalification examinations. The inspectors reviewed an independent self-assessment performed for Indian Point 2 by personnel from two other utilities, which identified problems with quality of written examination questions. The facility also identified weaknesses in operator knowledge and use of emergency operating procedures (EOP). Con Edison responded to the finding concerning written exams by enhancing instructor training and upgrading the written examinations. Review of the written examinations for the Year 2000 exams shows a clear increase in difficulty level from prior years; this appears to be the main reason for the higher failure rate on the written exam this year as compared to prior years. The 1998 exams were reviewed and determined to have been adequate for the purpose of discriminating between competent and incompetent operators; the Year 2000 exams are an improvement. The overall facility results for this biennial written examination and annual operating test are as follows.

Summary of Requalification Test Results

	Written	Crew Simulator	Individual Simulator	Job Performance Measures	Overall
Number of Exams	52*	8	51	51	51
Number Unsat	10	2	2	3	14
Pass Rate	80.77%	75%	96.08%	94.12%	72.55%

* One operator resigned and did not complete the operating test.

While the number of failures overall (mostly due to failure on the written) was higher than expected, licensee representatives stated that the examinations were effective in uncovering knowledge weaknesses. Licensee representatives also identified a process "lesson learned" in examining the operators with the higher cognitive level questions on the biennial written without significant practice exposure to such questions.

The inspectors also reviewed a lesson plan developed in response to the evaluation result which noted EOP weaknesses. This lesson plan addressed the issues raised in the evaluation, which concluded that part of the problem was knowledge of high level mitigation strategies. The inspectors noted that the two crew failures in Year 2000 were in part due to poor decision making in EOPs. Although EOP related, this is not the same problem that resulted in failures in 1998 and 1999. The inspectors considered EOP knowledge and use to have been partially remediated, but it is still an area for improvement.

4OA3 Steam Generator Replacement Project

a. Inspection Scope (IP50001)

Inspections were performed to obtain an overview of current and planned work, radiography of welds, work control packages, related procedures, documentation, quality inputs and progress of the Indian Point Unit 2 steam generator replacement project (SGRP). This inspection included observations of welding in progress and preparations for related nondestructive examination (NDE) of piping and other components in the SGRP process; a sampling of the welding and NDE procedures; a review of seven of the sets of radiographs representing the reactor coolant system (RCS), main steam and feedwater welds; the weld documentation process; and the work control instruction packages. Additional areas of inspection included review of the planning to reinstall and test components other than the steam generators that had been removed as part of the SGRP; observation of conditions inside the containment building, construction of the storage building for the old steam generators, the involvement of Quality Assurance (QA) in project oversight, and current and planned steps to complete documentation of work and inspections done during the project by closeout of the work control packages.

During this inspection period the adequacy of the eddy current testing (ECT) of the tubes in the replacement steam generators was evaluated, in particular to confirm that the known types of degradation were considered in selecting the ECT scope and that the ECT data gathered will be useful as a reference during future ECT testing. Regional-based inspectors and a contractor ECT specialist participated in this evaluation.

The inspection also reviewed the transport of the old and new steam generators, the interim storage and shielding of the old steam generators, the control of the modification process through FMX-00-52429-D; the conduct of a 10 CFR 50.59 safety evaluation per SECL-00-0091; the construction of the old steam generator permanent storage building; and, the radiological controls for the interim storage of the old steam generators. The inspector reviewed Con Edison actions to address plant operating procedures as a result of the steam generator changeout, including the impact on the secondary side heat balance. The implementation of the following work packages for restoration of system impacted by the steam generator replacement project was reviewed while the work activities were in progress:

- WP 3534D, Install Miscellaneous Structural Interferences SG#24
- WP 3080B, Install Secondary Pipe Main Steam SG#22
- WP 3085B, Install Feedwater Pipe SG#22
- WP 3520A, Install Secondary Small Pipe Interferences SG#21
- WP 3520B, Install Secondary Small Pipe Interferences SG#22
- WP 5060, Remove Reactor Cavity Decking
- WP 3510, Install Electrical Interferences Inside Containment
- WP 2570B, Removal and Transport of SG #21

The inspector reviewed the process for status of restoration activities in accordance with Temporary Operating Administrative Directive #2, the Indian Point 2 Operational Readiness Review Plan, and the evaluations to review the restart punch list in concert with plant startup and entry into operational modes.

b. Observations and Findings

The activities of the IP2 steam generator replacement project (SGRP), including transport and storage of steam generators, the eddy current inspection of tubes in the replacement steam generators, in-progress radiography of welds, provision for reinstallation of components removed as part of the SGRP and control of work package closeout were noted to be well planned and conducted. Radiation surveys for interim storage of the old steam generators showed all measured radiation levels to be below regulatory limits.

No findings were identified.

4OA4 Previous Inspection Issues

.1 (Updated) AV 05000247/2000-006-01: Failure to augment the ERO in a timely manner - failure to meet planning standard 10 CFR 50.47(b)(2), Timely Augmentation of ERO. The inspector sampled completed corrective actions (described in licensee's Reply to the subject Notice) to verify the licensee had completed the actions. Sampled items - training on new ERO activation process and pager activation tests - were verified complete. Also, the licensee demonstrated its capability to augment the ERO within regulatory time limits during an April exercise, and in subsequent drills conducted in June and August.

.2 (Updated) AV 05000247/2000-006-02: Failure to complete accountability in a timely manner - failure to meet planning standard 10 CFR 50.47(b)(10), Protection of Radiation Workers. The inspector sampled completed corrective actions (described in licensee's Reply to the subject Notice) to verify the licensee had completed the actions. Sampled items - training on a revised accountability procedure, centralized accountability for non-essential personnel, and successful accountability drills - were verified complete.

.3 (Updated) AV 05000247/2000-006-03: Improper dissemination of information to public and local official - failure to meet planning standard 10 CFR 50.47(b)(7), Public information. Although the licensee implemented corrective actions to correct the cause of this violation, licensee-conducted drills in June, August, and September indicated further performance enhancements and additional training are desirable. These and other actions will be further reviewed during the 95003 inspection.

.4 (Updated) Planning standards extent of condition review: Con Edison's EP staff reviewed the IP2 Emergency Planning program to determine whether the plan met and implemented the sixteen EP planning standards of 10 CFR 50.47(b). The review was initiated because of performance issues related to three standards that were the subject of the white findings and associated violations. The review did not identify additional deficiencies and concluded that the IP2 Emergency Planning Program met and implemented fifteen standards; one standard, regarding dissemination of information to the public, was not being met. EP staff completed the corrective actions for this area by November 1 (see #3, above). In addition to the review of all planning standards, the EP manager considered a review of Appendix E and NUREG-0654 items to ensure the IP2 Emergency Planning Program also conformed with these documents.

.5 (Closed) Unresolved Item 05000247/2000-07-02 Gas Turbine Starting Capability: The inspectors performed an in-office review of safety evaluation (SE) 00-378-EV, Revision 1, "Removal of the Remote Start Capability of the Gas Turbines from the UFSAR." This SE was performed to support correcting the UFSAR which states that the gas turbines could be started locally or remotely from the central control room. The remote start feature is currently only available for the on-site gas turbine, GT-1. Following a modification to the GT-2 and GT-3 controls in 1994, Con Ed experienced problems with the remote start capability for these two units and decided to rely solely on local starting. This was based on Con Ed's determination that the gas turbines could be started locally within one hour to perform their functions as an alternate AC power source in the event of a station blackout and to provide power to operate safe shutdown equipment in the event of a fire. The SE also evaluated the potential effects of losing cooling to the reactor coolant pump seals for up to an hour and concluded that there would not be a catastrophic failure based, in part, on the resolution of NRC Generic Safety Issue 23, “Reactor Coolant Pump Seal Failure,” and that the enhanced design seals are installed in the Indian Point 2 reactor coolant pumps.

The inspectors found that the safety evaluation appropriately concluded that the changes to the GT controls, which eliminated the remote start capability, were acceptable and did not constitute an unreviewed safety question as defined in 10 CFR 50.59. This item is closed.

4OA5 Licensee-Identified Violation

The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violation (NCV).

NCV Tracking Number	Requirement Licensee Failed to Meet
(1) NCV 05000247/2000-013-03	Technical Specification 6.12.1

Technical Specification 6.12.1 requires that each high radiation area (HRA) shall be barricaded and conspicuously posted as a HRA. Contrary to the above, on September 21, 2000, the HRA posting on Recirc gate No. 2 was found reversed, and, on November 2, 2000, the HRA swing gate for Recirc gate No. 2 was found open. Reference Condition Reports 2000-07099 and 2000-08504, respectively.

b. Issues and Findings

No significant findings were identified.

4OA6 Management Meetings

a. Exit Meeting Summary

The inspector presented the licensed operator requalification inspection results to members of the licensee management in a pre-brief at the conclusion of the inspection on April 20, 2000, and in an exit on November 15, 2000.

On November 21, 2000, the inspector presented the overall findings to Mr. J. Groth and other Con Edison management. Con Edison acknowledged the findings and did not contest the conclusions. Additionally, none of the information reviewed by the inspectors was considered proprietary.

PARTIAL LIST OF PERSONS CONTACTED

A. Blind	VP Nuclear
M. Borkowski	HP Technician
T. Burns	Supervisor, Radiation Support
M. Cross	HP Technician
M. Dampf	Radiation Protection Special Projects
M. DiGenova	Senior System Engineer
M. Donegan	Health Physics/Radioactive Waste Manager
J. Ferrick	Operations Manager
H. Geisler	I&C Technician
G. Gross	Radiation Protection-Instrumentation
D. Maffei	Maintenance Supervisor
R. Majes	Radiation Support Health Physicist
R. Masse	Plant Manager
J. McCann	Nuclear Safety and Licensing
L. Menoscal	Radiation Support Health Physicist
L. Mettey	NEM Technician
D. Murphy	Nuclear Training Manager
M. Miele	Radiation Protection Manager
J. Nichols	Operator Training Manager
V. Nutter	Radiation Support Manager
J. Simmons	SGR HP Manager
M. Stroppel	Operations Trainer
G. Zolotas	HP Technician

ITEMS OPENED, CLOSED, AND DISCUSSED

Open

05000247/2000-13-01	URI	Inadequate records of licensed operator attendance at requalification training.

Updated

05000247/2000-006-01	AV	Failure to augment the ERO in a timely manner -
05000247/2000-006-02	AV	Failure to complete accountability in a timely manner
05000247/2000-006-03	AV	Improper dissemination of public information

Closed

05000247/2000-07-02	URI	Gas Turbine Starting Capability

Opened and Closed

05000247/2000-13-02	NCV	NRC identified that the licensee did not sample all Senior Reactor Operators on emergency plan implementation.
05000247/2000-013-03	NCV	Violation of Technical Specification 6.12.1.

LIST OF ACRONYMS USED

ALARA	as low as reasonably achievable
CFR	code of federal regulations
CR	condition report
ED	electronic dosimeter
EOP	emergency operating procedures
FME	foreign material exclusion
HP	health physics
HRA	High Radiation Area
LORT	licensed operator requalification training
MT	magnetic particle tests
NDE	nondestructive examination
NVLAP	National Voluntary Laboratory Accreditation Program
OAD	operation administrative directive
OS	Occupational Safety
PT	penetrant
QA	quality assurance
PSA	plant safety analysis
RCA	radiologically controlled area
RCS	Reactor Coolant System
RSG	replacement steam generator
RWP	radiation work permit
SAO	station administrative order
SCBA	Self-Contained Breathing Apparatus
SE	safety evaluation
SG	Steam Generator
SGRP	steam generator replacement project
SOP	system operating procedure
SQ	station qualified
TRAD 104	Implementation
UT	ultrasonic
VC	vapor containment

ATTACHMENT I

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revised its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents), radiation safety (protecting plant employees and the public), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety
- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety
- Occupational
- Public

Safeguards
- Physical Protection

To monitor these cornerstones of safety, the NRC uses inspections and performance indicators that generate information about the safety significance of plant operations. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the NRC can reach objective conclusions regarding overall plant performance. The NRC will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC actions in response to the significance of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.