October 30, 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 05000247/2000-011

Dear Mr. Blind:

On September 30, 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 October 5, 2000, with Mr. John Groth and other 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, operations and maintenance, including work involved with the steam generator replacement project. All findings were determined to be of very low safety significance (Green). This report documents the resident inspector's concern regarding the need for improved attention to central control room "operator burdens". This topic was one of several issues discussed during a meeting on engineering support held in Region I on October 25, 2000.

In this report we identified three violations that are being treated as non-cited violations and are discussed further in the report.

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 05000247/2000-011

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	05000247
License No.:	DPR-26
Report No.:	05000247/2000-011
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	August 20, 2000 to September 30, 2000
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector E. Harold Gray, Senior Reactor Inspector Joseph Furia, Senior Health Physicist John McFadden, Health Physicist Steven Dennis, Operations Engineer Thomas Burns, Reactor Inspector Douglas Dempsey, Reactor Inspector Gregory Smith, Senior Physical Security Inspector
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 05000247/2000-011

IR 05000247-00-11, on 08/20-09/30/2000; 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. This inspection identified all green issues. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP). None of the conditions reviewed during the inspection required assessment using the SDP. No findings or issues were identified in the Reactor Safety, Public Radiation Safety and Occupational Radiation Safety Areas.

Cornerstone: Initiating Events

Green - A minor fire inside containment occurred on September 3, 2000, when sparks from a grinding evolution landed on a combustible foreign material exclusion (FME) tarp during work controlled under work permit 1060, "Install Reactor Cavity Decking." The fire occurred due to the failure to properly evaluate and control transient combustibles. This issue had very low safety significance because the fire was isolated to the area of the tarp, the location of the fire did not impact safe shutdown equipment, and the fire was extinguished in 25 minutes. The failure to control transient combustibles in accordance with station administrative orders is being treated as a non-cited violation of license condition 2.K. (Section 4OA2)

Cornerstone: Mitigating Systems

Green - An inadequate fire fighting instruction existed to align fire suppression water to the containment. The deficiency impacted the efforts to suppress the fire inside containment on September 3, 2000. This issue had very low risk significance because safe shutdown equipment was not impacted by the fire. A violation of license condition 2.K is being treated as a non-cited violation**.** (Section 1RO5)

Cornerstone: Other

Green - The licensee issued a modification to reroute the nitrogen piping to the reactor coolant drain tank. During implementation of the modification, workers failed to review drawings, perform a work area walkdown, and conduct a pre-job brief. The workers failed to locate the correct pipe and cut the nitrogen supply line to the safety injection accumulators and the power operated relief valves. This issue had very low safety significance because the safety injection accumulators and the power operated relief valves were not required to be operable at the time. The failure to implement maintenance procedures pursuant to technical specification 6.8.1 is being treated as a non-cited violation. (Section 4OA2)

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Report Details

SUMMARY OF PLANT STATUS

At the start of the inspection period, the plant was in cold shutdown to inspect steam generators, conduct refueling, and complete outage activities. Following the decision to replace the steam generators (SG), the reactor was defueled from August 19 - 23; the containment was turned over to the steam generator replacement project on August 25; and, the first steam generator, SG#22, was removed on September 29 in preparation for transport to the on-site, interim storage area. An Unusual Event was declared on September 3, 2000, when a minor fire occurred inside containment. The fire did not affect plant safety systems.

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

1R01 Adverse Weather Protection

a. Inspection Scope

The inspector conducted a walkdown of the service water and 480 volt vital buses to verify risk significant systems would be protected from adverse weather. The inspection included a review of the following procedures to verify the licensee provided instructions to the operators needed to maintain readiness of essential systems during adverse weather: AOI 29\8.0.7, Hurricane/Tornado/High Winds/Severe Thunderstorm; AOI 28.0.4, Plant Flooding; AOI 28.0.5, Containment Building Flooding; and, AOI 28.0.6, Nuclear Side (Outside Containment) Flooding. The inspector reviewed Final Safety Analysis Sections 2.0, 6.4, 8.1, and 14 to verify design basis conditions were met. The inspector also reviewed licensee actions to correct deficiencies that might impact system performance during adverse weather (reference Condition reports 199901038, 199907062, and 199907424).

b. Findings

There were no findings identified.

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 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.

- Electrical Penetration Area (Fire Zone 74A)
- No. 23 Reactor Coolant Pump Area (Fire Zone 70A)
- Electrical and Piping Tunnel, Piping Penetration Area (Fire Zone 1A)
- Containment Outer Annulus (Fire Zone 87A)

- Main Steam and Feedwater Valve Area (Fire Zone 65A)

The inspector also evaluated fire brigade response to a minor fire in the vapor containment on September 3, 2000.

b. Issues and Findings

Degraded Fire Barrier

The inspector observed an unsealed conduit from a manhole in the transformer yard to the electrical penetration area outer wall in fire zone 74A. Con Edison initiated condition report (CR) 200006981 to document this observation. Con Edison plans to evaluate the impact this deficiency had on past operability of the fire barrier, and take corrective actions prior to plant startup.

Other fire protection deficiencies identified by the inspector included errors in the fire hazards analysis drawings depicting the location of emergency lights in fire zones 74A and 1A. Correction of these deficiencies was incorporated in a revision of the fire hazards analysis drawings.

Fire Inside Containment - Unusual Event Declaration

The inspector evaluated the fire brigade response during a minor containment fire on September 3, 2000. The fire occurred when a reactor cavity foreign material exclusion (FME) barrier ignited from sparks generated by welding and grinding activities. Although minor, the fire could not be extinguished quickly because it was underneath steel decking, which had to be removed to put the fire out. Con Edison appropriately declared an Unusual Event pursuant to emergency action level 8.2.1. The emergency plan requires that an Unusual Event be declared when a fire has occurred within the vapor containment and has not been extinguished in less than 15 minutes. The fire was extinguished in 25 minutes. Con Edison issued Condition Report 200006504 to evaluate the causes of the fire and the impact on plant equipment, and identify corrective actions. There was no damage to plant equipment or structures.

Performance issues revealed during the fire brigade response included poor quality control room announcements following the fire alarm (CR 200006501), and inadequate pre-fire plan instructions for fire zone 85A-2 (CR 200006504, 200006507, and 200006512). The pre-fire plan instructions are provided to the fire brigade leader to align fire suppression to containment. The poor quality announcements resulted in untimely evacuation of non-essential personnel from the containment. Corrective actions were taken during the inspection period to improve announcements.

Inadequate Pre-Fire Plan Instructions

The inadequate pre-fire plan instructions involved the alignment of fire suppression water to containment. The design of the fire water system allows the high pressure fire header inside the primary auxiliary building to be cross-connected with the city water supply normally connected to the containment fire hoses. This would provide the necessary pressure in accordance with a class II hose station per NFPA 14. During the

September 3 fire, the fire brigade leader's pre-fire plan did not have these instructions. Further, the pre-fire plan had been incorrect since it was originally issued.

License condition 2.K requires, in part, that Con Edison implement and maintain all provisions of the NRC approved fire protection program as described in the NRC safety evaluation report dated January 31, 1979. This safety evaluation report documented that manual hose stations will be provided inside containment for suppressing fires that may occur. Sufficient hose capacity will be provided to reach all areas containing electrical cables and the reactor coolant pumps with an effective fire fighting hose stream. Contrary to the above, Con Edison did not have adequate system alignment consistent with plant modifications to provide the design capacity of the fire hose inside containment. The inspector determined that the improper instructions to align fire suppression water to containment had very low risk significance because safe shutdown equipment and areas adjacent to safe shutdown equipment were not impacted by the fire. This was a Green finding. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A. of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). **(NCV 05000247/2000-011-01)**

1R06 Flood Protection

a. Inspection Scope

The inspector conducted a walkdown inspection of the containment, turbine and switchgear buildings and the service water intake area, and examined the condensate, service water, and fire protection systems to verify that the equipment was not subject to damage resulting from internal flooding (e.g. from pipe breaks). The inspector reviewed the internal flooding analysis design calculations performed to demonstrate that the safety-related equipment was not vulnerable to internal flooding and also reviewed the design basis for the plant site to verify that the intake and service water areas were not vulnerable to external flooding events. The following documents were used as criteria for this inspection: IPEE Section 5.5 in Internal and External Floods; and UFSAR Sections 1.11.8. 2.0, 8.1.1, 6.4.2, and 14. The inspector reviewed operator actions in response to various potential floods in abnormal operating instructions (AOIs) 28.0.4, "Plant Flooding," AOI 28.0.6, "Nuclear Side (Outside Containment) Flooding," and AOI 28.0.5, "Containment Building Flooding." The inspector reviewed licensee actions to correct deficiencies that might impact system performance during flooding (reference Condition reports 199810863, 199901038, 199901591, 199903800, 199907062, 200001019, 200006001).

b. Findings

There were no findings identified.

1R08 Inservice Inspection Activities

a. Inspection Scope

The inspector selected samples of inservice inspection (ISI) activities for inspections based on a review of the inspection procedure objectives, Indian Point 2 system ranking of risk significant systems, and nondestructive examinations (NDE) performed during this refueling outage.

The inspector performed a review of three types of NDE activities including volumetric, surface and visual examinations to verify the effectiveness in monitoring the conditions of risk significant systems, structures and components. The results of ultrasonic (UT) and magnetic particle tests (MT) performed on the reactor vessel head-to-flange weld were reviewed. The evaluation and disposition by the licensee of indications identified in this examination were evaluated by the inspector for compliance with the requirements of ASME Section XI. Test results of MT examination of the replacement of four pipe welds (field welds 1AA, 2AA, 3AA and 4AA on drawing 9321-2571-AZ) in the service water system and penetrant testing (PT) results of welds in the residual heat removal, and safety injection and main steam systems were reviewed and evaluated for compliance with the ASME boiler and pressure vessel code requirements. Radiographs of welding activities were reviewed for welds made during piping replacement (field welds 1367, 1409 and 1446) on the service water system and component replacement (field welds 3.2AA, 3.5AA, 9.2AA and 9.5AA) in the chemical volume control system to verify the welding activities and acceptance were performed in accordance with ASME Section IX and Section XI Code requirements. The disposition of inspection results identified by visual test (VT) of component supports and restraints (supports SIH 188, SR 895, RSR8 and SR 700) were evaluated to verify the justification for continued service was appropriate, and that action had been initiated to enter the components into the corrective action program.

A sample of ASME Section XI Code replacements performed in the residual heat removal (min flow line #337) and service water (modification FFX-91-07049-M) systems was selected for review to verify that the replacement activities were accomplished in accordance with Code requirements.

The inspector reviewed condition reports CR 200006793 (welder qualification), 200006794 (service water pipe wall loss) and 200006882 (vessel head to flange weld indications) which were initiated during this inspection period for resolution of inspector observations.

Issues and Findings

No findings identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspector observed licensed operator annual requalification examinations administered for Indian Point 2 licensed operators on September 19, 2000, to assess the adequacy of training, licensed operator performance, and the adequacy of the licensee's evaluations. The examinations were conducted on the IP2 simulator and included responding to a steam generator tube leak using abnormal and emergency operating procedures (EOPs). The licensee evaluation of operator performance on the examinations determined that an operating crew failure had occurred due to improper implementation of EOPs. The inspector reviewed the licensee evaluations and associated corrective actions.

b. Issues and Findings

Based on independent evaluation the inspector/examiner concurred with the crew failure and verified the licensee's process that operators and crews would be remediated and re-evaluated prior to returning to licensed duties. The inspector also noted a number of failures on the written test portion of the requalification exam.

No findings were identified.

Additional NRC review of these issues was subsequently conducted during the scheduled baseline requalification inspection during the week of October 16, 2000. The next NRC resident inspection report will document this inspection.

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 issue was assessed:

- Starting Diesel for Gas Turbine 2 (CR20006998, WO 99-12797)

b. Issues and Findings

No findings were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors evaluated conditions within the central control room to identify if any additional items were considered operator burdens as described in operations administrative directive (OAD)-41, “Operator Burden.” The inspectors sampled 20 central control room deficiencies (CCRDI’s) and seven operator workarounds (OWAs) that had been previously corrected in the last three years to evaluate the effectiveness of the corrective action process. The inspector also accompanied the primary and conventional nuclear plant operators on plant tours to evaluate if degraded conditions should have been considered either a CCRDI or a OWA.

b. Issues and Findings

A number of degraded deficiencies in the central control room were not initially identified as CCRDIs or OWA until presented by the inspector to operations management. Degraded deficiencies subsequently classified as CCRDIs or OWAs included: failure to reset the main and auxiliary transformer deluge panel, accumulator gas supply valve (891C) leaking by seat, 23 condensate pump temperature indication failure, unit 1 riverwater pump bearing Hi/Low alarm inoperable, reactor coolant pump seal return flow alarm, and low nitrogen pressure to the 24 accumulator. The inspector learned that these degraded issues were not identified by the operations department during weekly preventative maintenance, or highlighted on the condition report, or identified by the corrective action screening committee as a potential operator burden. The inspector confirmed that each of the issues were reclassified as either CCRDIs or OWAs. The inspector identified that a number of deficiencies still had deficiency identification (DI) tags on the components when the maintenance had been completed. This is contrary to station administrative order (SAO)-204, “Work Control,” addendum II which states that the work crew shall ensure that the DI tag be removed once the work activity has been completed.

The inspector concluded that approximately half of the closed CCRDIs and OWAs had reoccurred since they were initially identified as a degraded condition. Some recurrent degraded conditions included the 21 traveling screen differential pressure alarms, the inboard bearing leak on the 22 component cooling water pump, leakage past accumulator fill valve 891C requiring operators to periodically pressurize the accumulator, and numerous unexpected boric acid heat trace alarms (circuit 23). At the end of the inspection, Con Edison management provided higher station visibility to operator burdens by publishing work down curves, established new goals for open CCRDIs and OWAs, and published schedules for correcting various operator burdens.

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. The following activities were reviewed to verify completeness within the technical specification and procedure requirements.

- reactor operation on residual heat removal system
- refueling operations from August 19 to August 23
- shutdown risk evaluations
- foreign object retrieval from top of core plate
- criticality controls during reactor offload
- plant drain down for steam generator replacement
- containment turnover for steam generator replacement

Corrective actions were reviewed for issues described in condition reports (CR) and entered in the corrective action system. See Section 4AO2 below.

b. Issues and Findings

No significant findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas (71121)

a. Inspection Scope

The inspector reviewed the following procedures, records, and 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 point (HP-1) and at the steam generator replacement project RCA control point (HP-2) on September 5, 6, and 7 to verify compliance with requirements for RCA entry and exit, wearing of record dosimetry, and issuance and use of electronic dosimeters. During a tour of elevations 46, 68, and 95 of reactor containment on September 6, the inspector reviewed radiological control technician activities in support of erection of scaffolding, installation of temporary shielding, and asbestos removal. Also, during this tour, the inspector witnessed materials being moved into and out of the equipment hatch on elevation 95 to observe the use of proper contamination boundary controls.

- Indian Point (IP) 2 Daily Radiation Work Permit (RWP) Reports for 09/02/00, 09/03/00, and 09/05/00

- Investigation Planning Guidelines/Condition Report 200006332
- RWP/Pre-Job Briefing Highlights Form
- Radiological surveys for various containment elevations for August 2000
- Procedure Station Administrative Order (SAO)-112, Corrective Action Program
- Procedure Dosimetry (DOS)-6.109, Dosimetry Evaluations
- Procedure DOS-6.103, Secondary Dosimetry Issue, Control, and Tracking
- Procedure DOS-6.118, Operation of the Merlin Gerin Automatic Dosimetry System
- Procedure Health Physics (HP)-Station Qualified (SQ)-3.011, Radiation and Contamination Survey Techniques
- Procedure HP-3.101, Radiological Access Control Point
- Procedure HP-SQ-3.008, Radiation Work Permit
- Technical Specification 6.12, High Radiation Area
- Health Physics Activity Plan (HPAP)-06, ALARA Support Plan
- HPAP-08, Pipe Cutting, Machining and Welding
- HPAP-11, Moving Original Steam Generators to Original Steam Generator Storage Facility
- HPAP-14, Radioactive Material Handling Plan

The inspector evaluated the adequacy of the characterization and immediate corrective actions identified for the following Condition Reports (CRs) covering a period from August 22, 2000 to September 3, 2000, by review of the detailed documentation and by discussions with the individuals involved with the investigations: CRs 2000-06200, 2000-06225, 2000-06290, 2000-06328, 2000-06332, 2000-06333, 2000-06398, and 2000-06502. The inspector also reviewed and verified licensee corrective actions for the following CRs by observation of activities at each RCA control point: CR 2000-06332 and CR 2000-06290.

The review was against criteria contained in 10 Code of Federal Regulations (CFR) 19.12 (Instruction to workers), 10 CFR 20.1301 (Dose limits for individual members of the public), Subpart F (Surveys and monitoring), 20.1601 (Control of access to high radiation areas), 20.1902 (Posting requirements), site Technical Specification 6.12 (High radiation area), and site procedures (cited above in this section).

b. Issues and Findings

No significant findings were identified.

2OS2 ALARA Planning and Control (71121)

a. Inspection Scope

The inspector reviewed the following program documents and records to determine the effectiveness of ALARA (As Low As Reasonably Achievable) planning and control. The inspector toured the containment elevations and observed the amount of temporary shielding installed and the use of signs to identify low dose waiting areas and to identify higher dose areas where access time should be minimized. The inspector also examined the following RWP packages to evaluate the adequacy of the work permit request, surveys, ALARA review, approved RWP, and any pre-job briefing records.

- Health Physics Activity Plan (HPAP)-06, ALARA Support Plan (Steam Generator Replacement Project)(SGRP)
- HPAP-07, Ventilation Plan (SGRP)
- ALARA Review Packages for the following RWPs
- RWP 508, Install/remove scaffolding
- RWP 510, Reactor Coolant System pipe templating/cuts /machining/ welding
- RWP 511, Perform pipe end decontamination

The review was against criteria contained in 10 CFR 20.1101 (Radiation protection programs), 10 CFR 20.1701(Use of process or other engineering controls), and site procedures (cited above in this section).

b. Issues and Findings

No significant findings were identified.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

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

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

b. Issues and Findings

No significant findings were identified.

2PS2 Radioactive Material Control Program

a. Inspection Scope

The inspector reviewed the following documents and licensee activities to ensure that the licensee's surveys and controls were adequate to prevent the inadvertent release of licensed material to the public domain.

b. Issues and Findings

No significant findings were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP4 Security Plan Changes

1. Inspection Scope(711130.04)

Security Plan Changes. A review was conducted of changes to the Indian Point Security Plan identified as Revision 20, submitted to the NRC on August 20, 2000, in accordance with the provisions of 10 CFR 50.54(p).

Based on a review of a sampling of the changes, as described in the plan revision, no NRC approval of this change is required.

2. Issues and Findings

No significant findings were identified.

4. Other Activities (OA)

4OA1 Performance Indicator Verifications

.1 Scrams with Loss of Normal Heat Removal

a. Inspection Scope

The inspector examined corrective action program records, control room logs, licensee event reports, and past NRC inspection reports for occurrences involving scrams with loss of normal heat removal. The inspector also reviewed data for the 12 quarters dating back to July 1997. 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.

b. Issues and Findings

No significant findings were identified.

.2 Safety System Unavailability, High Pressure Safety Injection System (HPSI)

a. Inspection Scope

The inspector examined corrective action program records, control room logs, licensee event reports, and past NRC inspection reports for occurrences involving high pressure safety injection system unavailability. The inspector specifically reviewed data for the 2^{nd} and 4^{th} 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.

b. Issues and Findings

No significant findings were identified.

.3 Safety System Unavailability, Residual Heat Removal System

a. Inspection Scope

The inspector examined corrective action program records, control room logs, licensee event reports, and past NRC inspections reports for occurrences involving residual heat removal system unavailability. The inspector specifically reviewed data for the 3rd quarter of 1999 and the 1st and 2nd quarters of 2000. 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.

b. Issues and Findings

No significant findings were identified. The inspector did identify that Con Edison failed to properly account for 217 required hours during the 2nd quarter of 2000 when shutdown cooling was in operation. The consequence of this error was minimal and, when corrected the performance indicator was maintained within the Green threshold.

4OA2 Cross Cutting Issues

a. Inspection Scope

The inspection reviewed issues involving several cross-cutting issues, including concerns related to human performance and problem identification and resolution.

b. Issues and Findings

Control Of Outage Work

Several lapses in control of outage activities were noted: (a) on August 24 - inadequate control of reactor water level while landing the upper internals in the vessel (no fuel in the reactor) due to poor coordination between the control room and refueling personnel (Condition Report 200006271); (b) on August 30 - inadequate instructions to workers completing a modification on the pressurizer relief tank resulted in the inadvertent cutting of the nitrogen line to the accumulators (Condition Report 200006462); (c) on September 3 - inadequate control of fire hazards during welding on cavity deck plates resulting in a containment fire and Notification of Unusual Event declaration (Condition Report 200006504); (d) on September 14 to 15 - lapses in control and delayed identification of the source of asbestos contamination resulting in a cessation of containment work on two occasions (Condition Report 200006852, 6854, 6861, 6874, 6894); and, (e) on September 27 - inadvertent release of hydrazine to the floor and storm drains while draining steam generators due to failure to track the status of chemical additions to the secondary side during the outage (Condition Report 200007249). No significant findings were identified.

Inadequate Contractor Oversight

The licensee issued a modification to reroute the nitrogen piping to the reactor coolant drain tank. During implementation of the modification on August 31, 2000, workers failed to review drawings, perform a work area walkdown, and conduct a pre-job brief. The workers failed to locate the correct pipe and cut the one-inch, stainless steel, nitrogen supply pipe to the safety injection accumulators and the power operated relief valves. Con Edison initiated condition report 200006462 and assigned this problem as a significance level 1 for root cause analysis. Appropriate short-term corrective actions were taken that included a stop work order.

Technical specification 6.8.1.a requires written procedures to be implemented for activities referenced in Appendix “A” of Regulatory Guide 1.33, rev. 2. Appendix A includes Item 9.e. “General Procedures for the Control of Modification Work.” Station Administrative Order (SAO)-405, “Modifications to Indian Point Facilities,” requires that the project managing authority (PMA) shall be overall responsible for implementation of the modification including quality. Contrary to the above, the PMA did not provide quality in the implementation of the modification by performing a walkdown, pre-job brief, or review drawings. This issue was determined to be of very low safety significance because the modification error occurred when operability of the safety injection accumulators and power operated relief valves were not required, and resulted in a Green finding. This violation is being treated as a Non-cited Violation, consistent with Section VI.A. of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). **(NCV 05000247/2000-011-02)**

As previously documented in section 1RO5, a minor fire in containment occurred on September 3, 2000. The primary cause of the fire was failure to properly evaluate and control transient combustibles during welding, cutting, and grinding evolutions. Specifically, sparks from a grinding evolution landed on a combustible foreign material exclusion (FME) tarp. The work area was controlled under work permit 1060, “Install Reactor Cavity Decking.” License condition 2.K. requires, in part, that Con Edison implement and maintain all provisions of the NRC-approved fire protection plan as approved in various safety evaluation reports. Station Administrative Order (SAO)-702, “Control of Ignition Sources,” implements aspects of the NRC-approved Fire Protection Program in accordance with license condition 2.K. SAO-702 requires prior to authorization of hot work that combustible material within a radius of 35 feet from the perimeter of the work area be either removed or covered with fire blankets. This preventative measure did not occur prior to the conduct of hot work under work permit 1060. This issue was determined to be of very low safety significance because the location of the fire did not impact safe shutdown equipment, and resulted in a Green finding. This violation is being treated as a Non-cited Violation, consistent with Section VI.A. of the Enforcement Policy, issued on May 1, 2000 (65 FR 25368). **(NCV 05000247/2000-011-03)** No significant findings identified.

Business Plan Actions -

Safety System Functional Assessment

The inspector reviewed Con Edison Report No. 99-11-A, “Auxiliary Feedwater System Safety System Functional Assessment,” to confirm that the system was capable of performing its design safety functions. The assessment was a vertical slice technical review, similar to an NRC Safety System Functional Inspection, that satisfied the licensee’s commitment to the NRC to perform such an assessment annually. The report identified 33 items which Con Edison consolidated into twelve “significant issues” and four conditions adverse to quality. The inspector reviewed the corrective action program condition reports that were initiated by the licensee for the report’s findings and discussed with licensee engineers the corresponding corrective actions that were planned or had been implemented. No significant findings were identified. However, a meeting was planned in Region I to discuss extent of condition reviews as a result of this SSFA (subsequently held October 25, 2000).

Corrective Action Effectiveness Reviews

During this period, Con Edison completed an effectiveness review of the corrective actions for the Business Plan areas. The Nuclear Quality Assurance (NQA) assessment was presented to the plant staff in September and the report is scheduled to be issued in October, 2000. The NQA review concluded the station made progress in some areas (business plan implementation, modification process, training programs, emergency preparedness, configuration management, corrective action program, and log keeping), but has yet to fix some problems (in the areas of degraded equipment, awareness of plant conditions, station labeling, implementation of corrective actions, condition reporting system, human performance training, performance metrics, station administrative orders, and work control). Con Edison staff and NQA identified areas where additional reviews were needed to assess the adequacy of the corrective actions, and determine whether process weaknesses have been corrected to preclude another event. An additional effectiveness review per SAO-112 was in progress under the purview of the Corrective Action Review Committee. The Plant Manager and NQA were assessing how to integrate the findings of both efforts, while at the same time strengthening the process to define the methods to conduct the effectiveness reviews. NRC review of this area continued at the end of the inspection period to assess the effectiveness of the Business Plan initiatives to improve station performance.

4OA3 Steam Generator Replacement

a. Inspection Scope (IP 50001, 7111102, 7111117, 7111119, 7111123)

The inspectors reviewed the following activities related to the replacement of the Unit 2 steam generators. Inspection Procedure 500001 was used as a guide to review activities for conformance with applicable licensee and regulatory requirements.

.1 Replacement Project Overview

Inspections were performed to obtain an overview of current and planned work, work control packages, related procedures, documentation, quality inputs and progress of the Indian Point Unit 2 steam generator replacement project (SGRP).

This inspection included a review of the preparations for welding and related nondestructive examination (NDE) of piping and other components in the SGRP process; welder performance qualification and training; a sampling of the welding and NDE procedures; the weld documentation process; and the work control instruction packages. Additional areas of inspection included observation of the replacement steam generator (RSG) staging area and work in progress; walking the haul routes for both the RSGs and original steam generators (OSGs), and haul route proof load testing per Work Package 2090; observation of conditions inside the containment building; the involvement of Quality Assurance (QA) in project oversight; construction of the OSG storage building and actions to protect 13.8 KV line 13W92 near the building foundation; fire prevention and mitigation preparations; review of the Readiness Inspection Team comments on health physics (HP), welding/cutting, and rigging/hauling; load testing of rigging components; RSG preservation and pre-installation activities; temporary OSG storage preparations; polar crane modifications; and modifications and operational plant water chemistry controls for the RSGs. NRC review of the modification packages to support steam generator replacement, along with the associated safety evaluations to demonstrate the work could be completed without prior NRC approval, continued at the end of the inspection period as ConEd finalized its review and approval of these efforts.

.2 Lifting and Rigging

The inspector reviewed Safety Evaluation (SE) No. IP2-SGR-SE-30, Revision 0 for the temporary lifting device (TLD) used to lift the original and replacement steam generators inside the containment building. This SE was prepared in support of steam generator replacement work package 1035 for the installation of the TLD.

The pertinent drawings and calculations associated with the SE, including calculation No. SGRP-C-019, Revision 0 were reviewed and discussed with the project engineering staff. The TLD was analyzed and designed using the GTSTRUDL computer program. The inspector reviewed the accuracy of the mathematical model representing the TLD and the loads imposed on the model. The inspector verified that the resultant stresses were within the allowable stresses.

.3 Haul Route Evaluation and Steam Generator Transport

The inspector reviewed the proposed SG transportation system and haul route to verify they were adequately evaluated. This included the self-propelled modular transporter (SPMT) which will be used to transport the original steam generator (OSG) from the hatch transfer system (HTS) outside the Containment Building along the designated haul path to an interim storage location in the lay-down yard. The SPMT will later transport the OSG from the interim storage location to the final storage location in the original steam generator storage facility. The SPMT will also be used to transport the replacement steam generator (RSG) from the steam generator storage building, along the same designated haul route to the HTS outside the containment building.

The inspector assessed the haul route path for the transportation of the steam generators. The inspector reviewed the pertinent calculation and had discussions with engineering personnel to verify the evaluation of the haul route path adequacy captured technical details of the haul route. The inspector also walked down key segments of the haul path and examined the transporter.

The inspector verified that the subsurface utilities along the pathway were analyzed for the effects of the applied surcharge loads from the SPMT. This included a review of Calculation No. SGRP-C-003, Revision 1 to verify that the haul route, with the pertinent upgrades to protect the underground commodities, was acceptable for the transportation of the OSG and RSG by the SPMT. The inspector also verified that this calculation showed that the temporary storage of the OSG in the interim storage location was acceptable.

.4 Radiological Controls

The inspector performed the following activities and reviewed the following documents to determine the effectiveness of: 1) access control to radiologically significant areas, and 2) HP planning and control for the SGRP. On September 6, 2000, the inspector toured portions of the transport path from the reactor containment to the interim and to the permanent storage locations for the original steam generators after their removal through the equipment hatch. The inspector observed the location and the on-going erection of the shielding barriers at the interim storage location. The inspector examined the dose calculations for the interim and permanent storage provisions for the original steam generators to verify that the calculations used appropriate dose criteria.

- IP2 SGRP Daily Dose Summary for 09/06/00
- IP2 SGRP Daily ALARA Report for 09/06 and 07/00
- HPAP-02, Containment Access Facility Plan-SGRP
- HPAP-11, Moving the Original Steam Generators to the OSG (Original Steam Generator) Interim Storage Area (SGRP)
- Radiological surveys for steam generators Numbers 21, 22, 23, and 24

- SGT Calculation No. SGRP-N-014, Radiological Analysis of a Steam Generator Drop Accident
- SGT Calculation No. SGRP-N-005, IP2 On-Site Steam Generator Storage Facility Radiation Shielding Calculation
- Con Edison Calculation No. FCX-00364-00-00, IP2 Steam Generators Characterization and Shielding Analysis Report WMG 20020-20085
- SGRP RWP Listing
- SGRP person-rem estimates by task number
- Procedure HP-SQ-3.011, Radiation and Contamination Survey Techniques
- Procedure HP-SQ-3.008, Radiation Work Permit

During the inspection periods of September 13-15 and September 18-21, 2000, the inspector reviewed work performance during the steam generator replacement project (SGRP). Numerous entries into the radiologically controlled area (RCA), especially the Unit 2 vapor containment (VC) were conducted during both the day and night shifts. Significant work activities observed included: (1) asbestos abatement on the 23 and 24 steam generators, (2) welding activities on the 21 and 22 steam generators, and (3) erection of the temporary lifting device. The inspector reviewed the following eleven radiation work permits (RWPs) to verify that work being performed in support of the steam generator replacement project was in accordance with the controls and conditions set forth in the RWPs and that the radiological controls provided were adequate for the described tasks.

- RWP-19, Health Physics Duties
- RWP-20, Walkdowns and Inspections by the Steam Generator Team
- RWP-21, Install/Remove Temporary Power/Water/Communications
- RWP-22, Steam Generator Replacement Project Supervisory Tours
- RWP-24, Steam Generator Replacement Decon Support
- RWP-25, Support Activities for Steam Generator Replacement Project
- RWP-506, Insulation Removal
- RWP 507, Temporary shielding
- RWP-508, Install/Remove Scaffolding
- RWP-509, Prep Hatch, Transfer System, Lift System
- RWP 520, Remove/install structural/electrical interferences

The review was against criteria contained in 10 CFR 19.12 (Instruction to workers), 10 CFR 20.1301 (Dose limits for individual members of the public), Subpart F (Surveys and monitoring), 20.1902 (Posting requirements), and site procedures (cited above in this section).

.5 Problem Identification and Resolution

The inspectors selectively reviewed the following documents to assess the effectiveness of problem identification and resolution of issues associated with the replacement of steam generators. Documents reviewed by the inspectors consisted of station administrative order (SAO)-112, “Corrective Action Program,” SAO-185, “Indian Point Station Steam Generator Replacement Policy,” and SGRP-SQ-21.003, “Steam Generator Replacement Project Review for NonConformances.”

The inspector attended various corrective action screening committee meetings and meetings between steam generator team (SGT) and SGRP corrective action team (CAT). Various condition reports were independently reviewed by the inspector that had been analyzed by the CAT. The inspector also compared SGT QA surveillance log and those issues entered into the condition reporting system. QA surveillance log items and SGRP QA observations were properly identified and evaluated in the CR system.

b. Issues and Findings

Inspections of the current and planned work for the IP2 steam generator replacement project, including welding and nondestructive examination preparations; related procedures, documentation, quality inputs and progress of the SGRP found generally good preplanning. Where the project oversight group and others have identified problems, these were documented and reviewed for corrective action. Examples include actions taken to address QA surveillance findings and SGRP Condition Report items.

No significant findings were identified.

4OA4 Plant Material Condition

a. Inspection Scope

The inspectors conducted tours in the Unit 1 utility tunnel. The utility tunnel contains piping for both high and low pressure fire water suppression lines, city water supply to the auxiliary feedwater pumps and various electrical cables and conduits. The inspection scope was to review the inspection history, and evaluate past Con Edison corrective actions and proposed restoration activities.

b. Issues and Findings

The inspectors found very poor material conditions within the Unit 1 utility tunnel, including degraded piping and supports, and degraded conduits and cable trays. The poor conditions resulted from exterior degradation of piping and electrical conduits from many years of surface water intrusion into the tunnel. The NRC had documented concerns on the conditions within the tunnel since 1982 (reference report 05000247/1982-019). Con Edison corrective actions to address ground water intrusion and overall component restoration have had limited success and have been untimely.

NRC reviews continued at the end of the inspection to evaluate Con Edison's actions to identify all degraded conditions within the tunnel, confirm system operability, complete risk assessments of the degraded conditions, and take short-term actions to assess the impact on Unit 2 and Unit 3 operations, if any.

No significant findings were identified.

4OA5 Management Meetings

a. Exit Meeting Summary

On October 5, 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

Licensee:

M. Dampf	Radiation Protection Special Projects
M. Donegan	Health Physics/Radioactive Waste Manager
D. Loope	SGRP HP Day Shift Manager
M. Miele	Radiation Protection Manager
V. Nutter	Radiation Support Manager
J. Simmons	SGRP HP Manager
T. Vehec	Training Technology Manager
P. Falciano	Site Access Training Administrator
G. Cullen	Security Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

NCV 05000247/2000-011-01	An inadequate fire fighting strategy instruction existed to align fire suppression water to containment.
NCV 05000247/2000-011-02	During implementation of a plant modification workers failed to perform a work area walkdown, pre-job brief, and review of removal drawings.
NCV 05000247/2000-011-03	A minor fire inside containment occurred due to a failure to properly evaluate and control transient combustibles during a grinding evolution.

LIST OF ACRONYMS USED

ALARA	as low as reasonably achievable
AOI	abnormal operating instructions
CAT	corrective action team
CCRDI	central control room deficiencies
CFR	code of federal regulations
CR	condition report
DI	deficiency identification
ED	electronic dosimeter
EOP	emergency operating procedures
FME	foreign material exclusion
HP	health physics
HPAP	health physics activity plan
HPSI	high pressure injection system
HTS	hatch transfer system
ISI	inservice inspection
MT	magnetic particle tests
NDE	nondestructive examination
NQA	Nuclear Quality Assurance
OAD	operation administrative directive
OSG	original steam generator
OWA	operator work around
PMA	project managing authority
PT	penetrant
QA	quality assurance
RCA	radiologically controlled area
RSG	replacement steam generator
RWP	radiation work permit
SAO	station administrative order
SE	safety evaluation
SGRP	steam generator replacement project
SOP	system operating procedure
SPMT	self-propelled modular transporter
SQ	station qualified
TLD	temporary lifting device
UT	ultrasonic
VC	vapor containment
VT	visual test

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.

DOCUMENTATION REVIEW

Component Replacement

WO 97-94058	Replacement of Relief Valve Discharge Piping
WO 95-81311	Replace Non-Regenerative HX outlet drain stop valve
WO 99-12149	Replace main steam isolation valve (vent valve)
WO 98-02990	Replacement of RHR min flow line #337
SGDA-00-199	Technical Justification of Helicoil Repair on Secondary Manway
FMX-00-52429-D	Component Evaluation of Replacement Steam Generators
SECL 00-091	Replacement Steam Generator Safety Evaluation
SECL 00-115	Secondary Manway Bolt Hole Helicoil

Condition Reports

CR200003172	SIS Component Support SIH 188, Missing Grout at Corner of Support
CR200003160	CVCS Component Hanger SR 895Hanger rod misalignment
CR200003054	SSD Spherical bearings bound and corroded, clamp misaligned
CR200003156	SSD CVC Component Support RSR8, bent rod
CR200004534	Service Water pipe wall loss 0f >20%
CR199701972	Safety Injection spring hanger movement (IPPVT89-3)

In-service Inspection Reports

NP00-16282	Magnetic Particle Inspection Service Water welds 1AA,2AA,3AA and 4AA
IPP-97-MTI-007	Magnetic particle test of reactor head to flange weld
WO98-02990	RHR penetrant inspection of FW 337-1,6 and 7
IPP-97-UTS-035	Ultrasonic test of reactor head to flange weld
IPP-97-PTI-003,89-1	Penetrant test of Safety Injection weld iso B206682 and B206903
IPP-VT-89-3	Visual Inspection of spring hanger -safety injection system

Weld Procedure

WPS 8118 R1	Manual Gas Tungsten Arc, P1-P1 (PQR 7211-C)
WPS 7216 R2	Manual Shielded Metal Arc, P1-P1 (PQR 7216, 7314)
WPS 7211 R3	Manual Gas Tungsten/Shielded Metal Arc, P1-P1 (PQR 7211,12 16)

Weld Information Form

WIF FW 1AA, 2AA, 3AA and 4AA	Weld Process sequence and hold points for the field welds indicated Service water pipe replacement

Radiograph Review

WO 98-03196	Radiograph work order for CVCS 3656-005 (Cl A)
FW 9.2,9.5,3.5	Charging pump/pipe field welds (and 3.2 AA)
WO 16282	Replacement of corroded service water pipe

Work Order

WO 00-15717	Redesign of support 42-H-3 (close of CR200003156)
WO 00-15718	Alteration of support CH41-H8 (close of CR 200003156)
WO 00-15799	Repair misaligned CVCS hanger (close CR 200003160)
WO 00-15777Y	Adjust clamp, lubricate bearings (close of CR 200003054)

WO 98-02990	Remove and replace defective pipe and fittings
WO 00-16282	Remove and replace defective pipe (close of CR 200004534-wall loss)
WP 3030A	Haul Route Proof Load Test

Quality Records

QA 08-SR-2K-13	Design Control
QA 01-SR-2K-010	Rigging and Hauling
QEP 11.04	Support and Structural Steel Installation
QEP 11.07	Electrical Conduit and Cable Removal and Installation
QEP 11.08	Performance of As-Built Verifications
FN-SGRP- 003	Audit of Design Activities
FN-SGRP-015	Review of Work Package Preparation
FN-SGRP-014	Eddy Current Testing for New Steam Generators
FN-SGRP-009	Replacement Steam Generator Preparation Activities
FN-SGRP-039	Replacement Steam Generator Preparation Activities
SAO-185	Indian Point Station Steam Generator Replacement Policy

Engineering Evaluations

ECR 057	Test Load for UP/DOWN Ending Device Test Plan
ECR 054	Steam Generator Replacement Haul Route Evaluation - Manhole #4,5
SGRP-C-003	Algonquin Gas Line Evaluation
SGRP-C-015	Hatch Transfer System Support Calculation