June 26, 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, New York 19511

SUBJECT: NRC's Indian Point 2 Inspection Report 05000247/2000-005

Dear Mr. Blind:

On May 20, 2000, the NRC completed an inspection at your Indian Point 2 reactor facility. The enclosed report presents the results of that inspection. The results of this inspection were discussed on May 23, 2000 with Mr. Robert Masse and other members of your staff, and on June 22, 2000, with Mr. Michael Miele 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 inspection and three region-based inspections of engineering, security and radiation protection.

The NRC noted design control problems related to the spent fuel pool storage racks and the manipulator crane controls. 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. The manipulator crane issue was determined to involve a violation of NRC requirements, but the violation is not cited because of the very low safety significance and because it has been entered into your corrective action program. If you contest this noncited violation, you should provide a response within
30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Documents Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United Stated Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Indian Point 2 facility.

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

Sincerely,

/RA by
Acting For/

A. Randolph Blough, Director
Division of Reactor Projects

Docket No. 05000247
License No. DPR-26

Enclosure: Inspection Report 05000247/2000-005

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REGION I

Docket No.:	05000247
License No.:	DPR-26
Report No.:	05000247/2000005
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	April 2, 2000 to May 20, 2000
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector Leanne Harrison, Project Engineer John McFadden, Health Physicist Paul Frechette, Physical Security Inspector Craig Smith, Resident Inspector Kathleen O'Donohue, Resident Inspector Scott Freeman, Resident Inspector Gerald McCoy, Resident Inspector James Trapp, Senior Risk Analyst (Regional Assistance)
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 05000247/2000-005

The report covered a seven-week period of resident inspection. The significance of issues is indicated by their color (GREEN, WHITE, YELLOW, RED) and was determined by the Significance Determination Process (SDP) in draft Inspection Manual Chapter 0609 (see Attachment 1).

Mitigating Systems

Green - The licensee failed to maintain adequate control of the manipulator crane control circuits. The circuit wiring was not in accordance with controlled drawings. A jumper bypassed a safety feature in the manipulator crane control circuit. With the jumper installed, the manipulator crane gripper could have been released prior to the fuel assembly being fully lowered into the core. The manipulator crane load cell interlock was not affected. The circuit would have prevented the operator from releasing the gripper under load and dropping a fuel assembly. The event was reviewed with the regional Senior Reactor Analyst (SRA), who evaluated the safety significance as very low (Green) based on the fact that the load cell remained operable and the procedural requirement for the operator to verify the location of the fuel assembly prior to releasing the gripper. The failure to maintain adequate design controls was determined to be a non-cited violation of 10 CFR 50, Appendix B, Criterion III. This inadequate control did not have an actual impact on safety. (Section R20.2)

Green - The licensee identified a degradation in the boraflex panels in the spent fuel storage racks, which resulted in a plant condition outside the design basis. Con Edison monitored degradation in boraflex panels in spent fuel pool racks using surveillance coupons, pool chemical analyses and analytical simulations using a computer program. On April 6, 2000, the results of boron-10 areal density measurements showed that thinning had occurred and gaps up to 7 inches had formed in the boraflex panels. Conservative criticality analyses assuming worst case gap size and geometry showed that the design requirement established in the technical specifications could not be met. Technical specification (TS) 5.4.2.B requires that the storage racks be designed such that the effective multiplication factor (Keff) is less than 0.95 without soluble boron in the pool water. The NRC Safety Evaluation for License Amendment No. 158 described the use of administrative controls such as fuel assembly relocation to compensate for boraflex degradation. Con Edison used additional controls on soluble poison concentration and spent fuel loading patterns to assure the Keff requirements were satisfied. This issue was considered to have a very low risk significance (Green) using the Significance Determination Process (SDP) phase 3 evaluation, because the storage rack Keff remained below 0.95 during past periods when a checkerboard pattern was not used but soluble boron concentration was at least 1500 ppm. (Section R21)

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Report Details

SUMMARY OF PLANT STATUS

During the inspection period the plant was in cold shutdown to inspect steam generators, conduct refueling, and complete maintenance. Reactor assembly and plant startup preparations were in progress at the end of the inspection period.

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

1R04 Equipment Alignment

a. Inspection Scope

The inspectors reviewed plant documents and observed equipment to verify that the emergency diesel generator air start systems were correctly aligned during a GT-1 auto transformer replacement and a station auxiliary transformer outage. The inspectors also verified that the spent fuel pool cooling system was correctly aligned when the reactor core was relocated to the spent fuel pool.

b. Issues and Findings

There were no findings identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors conducted tours of areas important to reactor safety, listed below, to evaluate, as appropriate, 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.

c. Emergency diesel generator building

a. Vapor containment

a. Primary auxiliary building pipe penetration area

b. Issues and Findings

There were no findings identified.

1R17 Permanent Plant Modifications

.1 Reactor Coolant Pump Oil Collection System

a. Inspection Scope

The inspectors reviewed Con Edison Modification Procedure No. FPX-00-12334-F, “Upgrade RCP Oil Collection System,” Revision 0. The inspector examined the installed enhancements to verify they were in accordance with the modification package.

b. Issues and Findings

Upon completion of the modification (FPX-00-12334-F) the inspector verified the adequacy of the installed enhancements on the 21 and 24 reactor coolant pumps (RCPs) to collect external oil leakage picked up by the motor cooling air vents and subsequently sprayed out to surrounding areas. Although the inspector found that (1) the drain tubing was properly installed to return oil to its reservoir and (2) the metal trough was properly installed around each oil collection valve at the lower oil pot assembly, the inspector noted that sealant had not been applied between the support flange and flywheel cover flange on the 24 RCP, as required by the modification package. Without this sealant applied, the potential existed for oil to leak out and not be captured by the oil collection system. Con Edison initiated CR 200003621 to determine the root cause for the failure to have the sealant application included in the installation work package.

There were no findings identified.

.2 Isolation Valve Seal Water System

a. Inspection Scope

The inspector reviewed Modification No. FMX-00-12338-M, “IVSW System Flow Control Upgrade,” Revision 0. The inspector examined the installation to verify it was in accordance with the modification package.

b. Issues and Findings

The modification was completed to assure the seal water tank for the isolation valve seal water system (IVSWS) meets the 24-hour supply requirements for the assumed containment isolation valve leakage described in Section 6.5.2.1 of the Updated Final Safety Analysis Report (UFSAR). The IVSWS was modified to limit the outflow of inventory by installing a solenoid valve and flow orifices in lines that serve four reactor coolant pump seal water and cooling water process lines. The modifications also assured there would be no loss of IVSWS inventory to the “essential” penetrations following the generation of a phase A isolation signal and prior to the generation of a phase B isolation signal. The inspector confirmed the modified design would address the IVSWS performance deficiencies identified during the February 15, 2000 steam generator tube rupture event, as described in Condition Report 2000011026 and Inspection 05000247/2000002.

There were no findings 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. Surveillance tests and inspections were reviewed to verify completeness within the technical specification and procedure requirements.

b. reactor coolant system cooling on residual heat removal system
c. refueling operations from April 12 - 23, 2000
d. spent fuel shuffle into checkerboard pattern for criticality control
e. spent fuel pool activities
f. shutdown risk evaluations
g. electrical lineup during GT-1 transformer replacement
h. electrical lineup during station auxiliary transformer outage
i. containment contingency closure for Orange Risk condition
j. fuel and top nozzle spring inspections
k. outage-related inspections and surveillance tests
    a. Refueling System Interlocks and Associated Bypass Tests, PT-R8
    b. Unit 2 Fuel Top Nozzle Clamp inspection, RPE-S-16.206
    c. Fuel Assembly Top Nozzle Inspection, STD-FP-1998-8251
    d. Fuel Handling Data Sheet and Sequence, FP-IPP-R15
    e. Eddy current certifications per SNT-TC-1A, CR 200002583,2584, 2585

b. Issues and Findings

Several problems with the refueling machine and fuel transfer system prolonged the core offload sequence and revealed inadequacies in preventative maintenance practices for the refueling equipment. The core defueling sequence was also delayed due to a broken spring screw on fuel assembly Q67, and a broken fuel plate alignment pin in the top nozzle of fuel assembly P54. These problems were entered into the corrective action system and were resolved.

There were no findings identified.

.1 Steam Generator Tube Plugging

a. Inspection Scope

The inspector reviewed Con Edison's response to the discovery of additional steam generator tubes that had to be plugged.

b. Issues and Findings

On May 8, 2000 Con Ed identified six tubes that should have been removed from service based on the results of the current steam generator inspections that had not been plugged prior to installing the steam generator primary manways. Plugging the six tubes required Con Ed to remove the manways, manually insert plugs in the six tubes, and then reinstall the manways. A total dose of approximately 2 man-rem was expended for the entire operation. At the time of the discovery, the reactor was defueled with the core offloaded to the spent fuel pool and the reactor coolant system drained down for planned maintenance. The tube plugging was completed during this inspection period. This item was entered in the condition report system as CR 200003331, and highlighted problems in the systems used to manage the database from the steam generator eddy current inspection program.

In response to this event, Con Ed employed an independent contractor to review the steam generator inspection data base to ensure that all of the tubes identified to be removed from service had been plugged. No additional missed tubes were found.

There were no findings identified.

.2 Manipulator Crane Wiring Errors

a. Inspection Scope

The inspector reviewed Con Ed's response to CR 200002608 which identified that an indicating light for the manipulator crane gripper came on prematurely as a fuel assembly was being lowered into the core on April 13, 2000. The light should not have come on until the gripper was fully inserted into the core.

b. Issues and Findings

Con Edison discovered that an unmarked jumper installed in the manipulator crane control circuit caused the erroneous indication. In addition, the wiring of the control circuit was not in accordance with controlled drawings for the system. Drawings on file did not reflect the field conditions nor the latest design. The load cell and limit switches were changed by design changes DOE-00156-PGI and DOE-00157-PGI in 1991 but the drawings and vendor manual were not updated. These drawings show that the gripper cannot be opened unless the gripper is in the down position over the basket or the core and that the weight on the load cell is less than 1200 lbs. The design agrees with the safety interlocks described in Section 9.5.2.2.4 of the UFSAR. The inspector was able to verify all interlocks except for number two which prevents simultaneous bridge and trolley movement. These are shown in the inching circuit but not the normal circuit. The missing drawing update and interlock number two were described on CR 199809090. The corrective actions of that CR were still outstanding as of May 2, 2000.

The inspector identified that the unmarked jumper, in addition to providing an erroneous indication, bypassed a safety feature in the manipulator crane control circuit. With the jumper installed, the manipulator crane gripper could have been released prior to the fuel assembly being fully lowered into the core. The manipulator crane load cell interlock was not affected by the jumper. Therefore, the control circuit would have prevented the operator from releasing the gripper under load and dropping a fuel

assembly. However, the jumper would not have prevented the operator from releasing the gripper on a fuel assembly that hung up while being inserted into the core. The procedure for operating the manipulator crane required the operator to verify the physical location of the fuel assembly prior to releasing the gripper. Prior to discovery of the installed jumper,
Con Ed relied on operator action to prevent opening the gripper in the event a fuel assembly stuck while being inserted into the core, instead of the manipulator crane safety interlock.

This event was reviewed with the regional Senior Risk Analyst (SRA), who evaluated the safety significance as very low (Green) based on the fact that the load cell interlock remained operable and the procedural requirement for operators to verify the physical location of the fuel assembly prior to releasing the gripper. The failure to adequately control the manipulator crane circuits and refueling interlocks was a violation of 10 CFR 50, Appendix B, Criterion III (**NCV 05000247/2000-005-01**). This issue is in the Indian Point 2 corrective action program as Condition Report 200002608.

.3 Spent Fuel Rack Boraflex Degradation and Fuel Shuffle

a. Inspection Scope (37551)

The inspector reviewed Con Ed's actions to assure the spent fuel rack design requirements of Technical Specification 5.4.2.B were satisfied. This issue was addressed in licensee event report (LER) 05000247/2000004 and was open pending the review of the impact of boraflex degradation on the spent fuel rack safety (UNR 05000247/2000-001-01).

b. Issues and Findings

Con Ed monitored degradation in boraflex panels in spent fuel pool racks using surveillance coupons, pool chemical analyses and analytical simulations in the RACKLIFE computer program. On April 6, 2000, the results of boron-10 areal density measurements (BADGER testing) showed that thinning had occurred and gaps up to seven inches had formed in the boraflex panels. The result of conservative criticality analyses assuming worst case gap size and geometry showed that the design requirement established in the technical specifications (License Amendment No. 158) could not be met. Technical specification (TS) 5.4.2.B requires that the storage racks be designed to keep the effective multiplication factor (Keff) less than 0.95 without soluble boron in the pool water. The design requirement is to allow full core offloads for the term of the license with all storage cells full.

The inspector reviewed the spent fuel rack testing and criticality analyses, as described in contractor's report NET-160-01 and Operability Determination 00-009. The criticality analysis was very conservative and was based on assumptions that would bound the worst case conditions noted by the test measurements. Actual spent fuel rack conditions were less severe than assumed in the analysis. The criticality analysis assumed the racks were in the end point configuration with all cells full with fuel containing the maximum allowable loading of Uranium-235. The storage racks have been less than completely full and contain fuel with the lower concentrations of Uranium-235. The licensee concluded that an analysis of the racks using realistic assumptions for boraflex gap size and distribution would show that the Keff requirements would be met. However, there is no analysis for the actual, interim fuel storage configuration. Thus, Con Edison reported this matter to the NRC on April 6, 2000 as a condition outside the design basis of the plant.

The inspector reviewed the corrective actions to assure the design requirements will be maintained. The use of a checkerboard loading pattern would restore Keff below 0.95 without crediting soluble boron in the pool. Specifically, Keff would be 0.9497 using a 15-out-of-16 checkerboard pattern in Region I, and 0.923 using a 3-out-of-4 checkerboard pattern in Region II. Con Ed shuffled fuel in the storage pool and implemented a core off load sequence that met the aforementioned checkerboard patterns established in the criticality analysis. The inspector independently verified that the fuel movement procedures and final spent fuel storage configuration met the requirements in the criticality analysis (reference FP-IPP-R15 and Work Step List NP-99-12730). The NRC Safety Evaluation dated April 19, 1990 in support of License Amendment No. 158 recognized that corrective actions in response to the discovery of degraded boraflex could include the use of administrative controls such as fuel assembly relocation to control Keff. The licensee had plans to obtain NRC approval to change the technical specifications to take credit for soluble boron in the spent fuel pool water.

The spent fuel soluble boron concentration is maintained at least 1500 parts per million (ppm) per Technical Specification 3.8.D.2, and is normally about 2200 ppm. This issue was considered to have a very low risk significance (Green) using the Significance Determination Process (SDP) phase 3 evaluation, because the storage rack Keff remained below 0.95 in the past when a checkerboard pattern was not used but soluble boron concentration was at least 1500 ppm. This issue is in the Indian Point Unit 2 corrective action program as Condition Reports (CRs) 199904943 and 200002451. Inspection Item UNR 05000247/2000-001-01 is closed.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspector evaluated temporary facility change (TFC) 2000-134, “Alternate Feed To 21 and 22 Spent Fuel Pool Pumps.” The purpose of the temporary facility change was to provide temporary electrical power to spent fuel cooling pumps with the reactor core fully offloaded into the spent fuel pool. The inspector verified the TFC alignment and evaluated the 10 CFR 50.59 analysis against system design bases documentation.

b. Issues and Findings

There were no findings identified.

2. RADIATION SAFETY

Public Radiation Safety [PS]

Cornerstone: Occupational Radiation Safety (OS)
Occupation Radiation Safety [OS]

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope

The inspector reviewed the effectiveness of access control to radiological areas:

a. Radiation permits (RWPs) for exposure significant work within radiation areas and high radiation areas were identified, and the associated postings and barricades for these areas were reviewed for acceptability: eddy current testing on all steam generators (RWP 236); and, valve inspections and repairs inside containment (RWP 366). The above-cited areas were walked down with a survey meter to verify the accuracy of postings.

b. The RWPs used for access to the above-cited areas were examined to verify whether they contained appropriate work control instructions; the availability and use of electronic alarming pocket dosimeters were observed; and the use of different sets of integrated dose and dose rate set points, determined by survey indications, were reviewed. Radiation protection job coverage for the above-cited areas was reviewed for adequacy.

c. During tours of the above-cited work areas, the inspector evaluated radiation workers and radiation protection technician awareness of the significant radiological conditions in their workplace and of the controls/limits of the applicable radiation work permits.

d. The adequacy of posting, barricading, and locking, in compliance with regulatory requirements and Technical Specifications, for all entrances to high radiation areas, locked high radiation areas, and very high radiation areas in Unit 2, were examined during a walkdown of each of these locations.

e. Condition Reports 199903122, 199908678, 199909141, and 199909179 were reviewed which addressed worker and/or radiation protection technician performance errors, occurring between April 16, 1999 and December 9, 1999. Associated cause evaluations and corrective actions were examined. Included in this review were problem reports addressing lack of proper RWP compliance, a worker not appropriately responding to a electronic dosimeter alarm, and lack of procedurally required documentation.

b. Issues and Findings

There were no findings identified.

2OS2 ALARA Planning and Control

a. Inspection Scope

The following activities were conducted to determine the effectiveness of ALARA planning and control:

- Based on scheduled work activities and associated exposure estimates, the following high exposure or high radiation area active job locations and their associated RWPs were selected, and the use of ALARA controls in those locations were evaluated.

  - RWP 236 eddy current testing on all steam generators
  - RWP 263 reactor coolant pump 21 and 23 seal and motor maintenance
  - RWP 269 sludge lancing
  - RWP 314 reactor head disassembly
  - RWP 366 valve inspections and repairs throughout containment

- Several of the above-cited locations were surveyed to evaluate whether postings for low dose rate waiting areas and for elevated dose rate areas were accurate and adequate.

- The pre-job ALARA reviews for several of the above-cited jobs were evaluated for use of engineering controls to achieve dose reductions.

- The effective use of low dose rate waiting areas by radiation workers was reviewed.

- The adequacy of job coverage by radiation protection technicians at the above-cited job locations was observed.

- The licensee's Source Term Reduction Plan documentation was examined to determine what specific sources have been identified for exposure reduction action and what results were achieved since the last refueling outage.

- The assumptions and basis for the current annual exposure estimate and annual exposure goal were reviewed.

- The exposure tracking system, its level of exposure tracking detail, exposure report timeliness, and exposure report distribution were investigated.

b. Issues and Findings

No significant findings or observations were identified.

The steam generator tube rupture event in February 2000, resulted in emergent steam generator eddy-current testing and repair activities. Consequently, as of April 2000, the licensee's ALARA organization anticipated that more personnel exposure than originally estimated (29.3 person-rem) would be required to perform the emerging work. The licensee's re-estimate and evaluation of the required work indicates that about 66 person-rem is expected to be expended.

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

Several portable dose rate survey meters being used in the field were examined for operability and for current calibration and source check status.

b. Issues and Findings

No significant findings or observations were identified.

3. SAFEGUARDS (Cornerstone Physical Protection)

3PP1 Access Authorization Program

i. Inspection Scope

Verify that the licensee is properly implementing the behavior observation portion of their personnel screening and fitness-for-duty program. Five representatives of licensee management and escort personnel were interviewed concerning their understanding of their behavior observation responsibilities and ability to recognize aberrant behavior traits. Access authorization and fitness-for-duty self-assessments, event reports, audits and loggable events were also reviewed.

j. Issues and Findings

No findings were identified.

3PP2 Access Control

a. Inspection Scope

Verify that the licensee has effective access controls and equipment in place designed to detect and prevent the introduction of contraband (firearms, explosives, incendiary devices) into the protected area that could be used to commit radiological sabotage. Verify that the identification and authorization process is used to confirm that only those who have been properly screened are granted unescorted access to the protected and vital areas. Access control activities were observed, including multiple observations of personnel processing through the search equipment during peak ingress periods and testing of all access control equipment. Access control event logs, audits and maintenance work requests were also reviewed.

b. Issues and Findings

No findings were identified.

4. OTHER ACTIVITIES [OA]

4OA2 Performance Indicator Verification

Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector examined corrective action program records for occurrences involving high radiation areas, very high radiation areas, and unplanned personnel exposures for the past four quarters 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 inspector reviewed the corrective action program records for an occurrence involving a Locked High Radiation Area that was identified by the licensee in April 1999 and subsequently recognized as a performance indicator. This occurrence involved a failure to comply with a procedurally-required record keeping item for documenting a Locked High Radiation Area entry. The inspector confirmed that the procedural noncompliance involved very low risk significance and that the occurrence was identified and effectively resolved.

b. Issues and Findings

No significant findings or observations were identified.

4OA3 Human Performance

a. Inspection Scope

On April 20, 2000, the inspector observed inattentive steam generator repair and health physics workers inside the vapor containment. The inspector spoke with the workers and discussed the issue with Con Ed personnel. The area inside containment where the workers were inattentive was posted as a high radiation area.

b. Issues and Findings

Con Edison initiated condition report 200002815 to document the inspector's observations. The actual radiation field where the workers were found was less than 0.5 millirem/hour and no significant additional occupational dose was incurred by the workers. Con Edison stopped work for all workers to reemphasize site expectations on attentiveness and to discuss past industry events. The inspector confirmed that corrective actions were appropriate and that no other incidents involving inattentive workers were observed during the inspection period.

4OA3 Event Follow-up

a. Inspection Scope

The inspector reviewed Con Ed's actions to correct deficiencies identified as a result of the steam generator tube rupture event on February 15, 2000.

a. SOP 4.1.2 was revised to coordinate operation of reactor coolant and residual heat removal pumps during a plant cooldown. ES 3.1 was revised to agree with SOP 4.12.

b. SOP 4.1.2 now contains the correct guidance to close normal spray valve when using auxiliary spray. Adequate guidance had already been included in SOP 3.3, Rev. 21. (CR 200001015)

c. Procedure guidance on the use of 10 CFR 50.54(x) was detailed in three procedures: OAD 15 - Conduct of Operations; OAD 33 - procedure adherence/use; OAD 26 - EOP Users Guide. The procedures were revised to ensure accurate interpretation and specific licensee requirements now meet the requirements and intent of 50.54x.

   Training was provided in three tiers: shift advocate discussions with shift personnel; required reading; and, simulator and formal class room training during the requalification program. The 50.54x training was completed for all shifts. The training package for operators states the NRC should be notified before invoking 50.54x if time permits, but if not do it as soon as possible. The procedure guidance implements the present requirements in 10 CFR 50.72, which states the notification is to be made within 1 hour. This discrepancy was discussed with the licensee.

d. Procedure POP 3.3, Plant Cooldown, Rev 45, was changed to address boration during emergency actions to cooldown the plant. The procedure requires a calculation of the required boron concentration for the final reactor coolant system (RCS) temperature. It then directs the cooldown to designated temperature plateaus where boron concentration is verified to be adequate prior to cooling down to the next plateau.

e. Extent of Condition (EOC) reviews were performed to verify that emergency operating procedures (EOPs) properly transition to POPs, SOPs, and abnormal operating instructions (AOIs) (CR200002186). Con Edison identified all the EOP transition points and verified that the EOP transition to the correct procedure and that the conditions in the procedure matched the conditions in the EOP. Discrepancies were identified and corrected.

f. Extent of Condition (EOC) on procedures - manual operation of equipment. Con Edison completed a review to identify all automatic actions replaced by operator manual action. The identified activities were evaluated to determine if any manual activity impacts emergency response. The items were reviewed from tracking systems such as operator work arounds, temporary modifications, work orders and condition reports (CRS). All items were reviewed with 91-18 criteria per the management procedures SAO204 and 112. CR 200002190 documents the review and conclusions.

g. Extent of Condition for December 1998 “administrative” EOP changes. Con Edison did not properly evaluate procedure changes when revising the RHR operating point from 450 psig to 300 psig. When making the change, the preparer checked that the change was “administrative” and thus no validation and verification (V&V) was completed. The EOC reviews of other procedures confirmed changes designated administrative were correct; and, if a change was not administrative, Con Ed confirmed the procedure received a proper V&V. Several instances were identified where at least tabletop reviews were required but not performed, although the licensee believes the reviews were completed but not documented. Con Edison performed simulator verifications whereever possible and table-top reviews were completed. CR200002456 documents the issue and tracks the table top reviews.

h. Abnormal Operating Instructions for Steam Generator Tube Leak - Compared to EPRI Guidelines. Procedure AOI 1.2, Revision 19 did not include the 15 minute monitoring recommendation itemized in Action Level 1 of the EPRI document. Con Edison will address this issue as part of the Operational Assessment and is considering the use of more conservative limits than specified in the EPRI document, with leak rate actions levels at 75 gpd and 30 gpd. Additionally, an action level for leak rate increase of 30 gpd/hr will be used. Licensee reviews were in progress at the end of the inspection period to review and approve the procedure guidelines.

i. Con Ed actions to address the following equipment issues were satisfactory: control circuit blowing fuses (CR 200001023 and WO 00-14172); reactor coolant pump fire zone alarms (CR 200001028 and WO 00-14273); and, leaking main steam isolation valve MS-1-24 (CR 200001118, WO 99-12149, and WO 00-13362); sluggish operation of the steam jet air ejector (SJAE) PCV-1230 discharge line to the containment (WO 99-10373); and pressurizer master controller, PC-455K (WO 00-14295, CRs 200001137 and CR 200001214).

j. SJAE pressure control valve PCV-1132. Modification FIX-00-12331 was issued to install upgraded pneumatic controllers and air relays in both valves PCV-1132 (pressure control) and PCV-1222 (high pressure shutoff). It also installs an upgraded positioner on PCV-1132. The licensee also planned to inspect the internals of each valve.

b. Issues and Findings

There were no findings identified. NRC review of this area is also described in Inspection 05000247/2000007.

4OA4 Other

.1 (Closed) URI 2000-001-01: Spent Fuel Rack Degradation. This item was closed as described in Section 1R21.1 of this report.

.2 (Closed) LER 200-004: Design Basis Compliance Failure due to Spent Fuel Storage Rack Boraflex Degradation. This item was a minor issue and was closed as described in Section 1R21.1 of this report.

4OA5 Management Meetings

a. Exit Meeting Summary

The inspector met with licensee representatives on April 6, 2000 to present the preliminary findings in physical security. The licensee acknowledged the preliminary findings.

On May 23, 2000, the inspector presented the preliminary findings to Mr. Robert Masse of Con Edison management. The inspector presented the final conclusions to Mr. Michael Miele on June 22, 2000. Con Edison acknowledged the findings and did not contest the conclusions. Additionally, none of the information reviewed by the inspectors was considered proprietary.

During the exit, the non-cited violation was discussed. The inspector informed Con Edison that should they elect to contest the NCV, that a response should be provided within 30 days of the date of this inspection report, with the basis for their denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Indian Point 2 facility.

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee:</u>

K. Cullen	Health Physics Technician
M. Dampf	Radiation Protection Special Projects
M. Donegan	Health Physics/Radioactive Waste Manager
J. Gallagher	Health Physics Technician
L. Gander	Radiation Support Health Physicist
R. Mages	Radiation Support Health Physicist
L. Menoscal	Radiation Support Health Physicist
M. Miele	Radiation Protection Manager
V. Nutter	Radiation Support Manager
R. Masse	Plant Manager
G. Cullen	Security Manager
T. Elsroth	Security Supervisor
J. McCann	Licensing Manager
Bill O'Toole	Manager QA Programs
Mark Entenberg	Manager Facilities Engineering
Patrick Russell	Manager Corrective Action Program
Anthony Spaziani	Nuclear Safety and Licensing Engineer
Curtis Ingram	System Engineering
Peter DeStefano	System Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened and Closed</u>
NCV 05000247/2000005-001: Failure to Maintain Design Control of Manipulator Crane

<u>Closed</u>
URI 05000247/2000-001-01: Evaluation of Spent Fuel Rack Boraflex Degradation
LER 05000247/2000-004: Spent Fuel Storage Rack Design Basis Not Met

LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
AOI	abnormal operating instruction
CFR	Code of Federal Regulations
CR	condition report
EDG	emergency diesel generator
EOC	extent of condition
EOP	emergency operating procedure
FP	fire protection
FSAR	Final Safety Analysis Report
GT	gas turbine
I&C	Instrument & Control
IP2	Indian Point 2
IVSW	isolation valve steam water
Keff	effective multiplication factor
LER	licensee event report
MOD	modification
PCV	pressure control valve
ppm	parts per million
QC	quality control
RCP	reactor coolant pump
RCS	reactor coolant system
RES	request for engineering services
RWP	Radiation Work Permit
SAO	station administrative order
SDP	significance determination process
SE	safety evaluation
SRA	senior risk analyst
TFC	temporary facility change
TS	technical specification
UFSAR	updated final safety analysis report
UNR	unresolved item
V&V	validation and verification

ATTACHMENT I

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped 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 if they occur), radiation safety (protecting plant employees and the public during routine operations), 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 seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. 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 agency can reach objective conclusions regarding overall plant performance. The agency 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's actions in response to the significance (as represented by the color) 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.