March 23, 2001

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

SUBJECT: INDIAN POINT 2 - NRC INSPECTION REPORT 05000247/2000-015

Dear Mr. Blind:

On February 17, 2001, 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 February 23, 2001, with you and members of your staff.

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

Several human performance errors occurred during the inspection, some of which impacted plant equipment or affected plant operation. Human performance errors resulted in a turbine trip on January 2, and the loss of safety Bus 3A on February 14, 2001. Weaknesses were noted in the initial review of the January 2 turbine trip. The correct performance of operating activities in accordance with procedures and management expectations warrants continued attention. Four violations of NRC requirements were noted in the areas of procedure use and adherence, and event reporting. Because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating the four issues as non-cited violations, in accordance with Section VI.A of the NRC's Enforcement Policy. If you deny these non-cited violations, 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: 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 Unit 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 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 05000247/2000-015

Attachment: NRC Revised Reactor Oversight Process

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REGION I

Docket No.:	05000247
License No.:	DPR-26
Report No.:	05000247/2000-015
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York 10511
Dates:	December 31, 2000 to February 17, 2001
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector Frank Arner, Reactor Engineer Joseph D'Antonio, Reactor Engineer Todd Fish, Reactor Engineer David Silk, Senior Emergency Preparedness Specialist Alfred Lohmeier, Reactor Engineer Laurie Peluso, Radiation Specialist Beth Sienel, Resident Inspector John McFadden, Health Physicist
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000247-00-15, on 12/31/2000 - 2/17/2001; Con Edison; Indian Point 2 Nuclear Power Plant. Personnel Performance During Nonroutine Plant Evolutions and Events, Cross-Cutting Issues. The inspection was conducted by resident and region-based inspectors. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP). This inspection identified all green or no color issues. The “no color” significance level indicates that the IMC 609 “Significance Determination Process” does not apply to these findings.

Cornerstone: Mitigating Systems

Green. On January 2, 2001, with the unit at 6.5% power, a main turbine trip signal was generated by a high level in the 21 steam generator. The high steam generator level tripped the main boiler feed pump and actuated the auxiliary feedwater system. Three operator or crew performance problems were identified and consisted of the following: the failure to adequately control steam generator level; operator control of rod insertion without a complete understanding of reactor conditions; and, operator communication errors, which resulted in an unnecessary period of plant cooldown and the simultaneous insertion of reactivity by two means. The issue was evaluated using the NRC’s significance determination process as having very low safety and risk significance. The failure to respond in accordance with procedures for reactivity management and controlling reactor temperature was a **non-cited violation** of Technical Specification 6.8.1.a.

No Color. The licensee did not make timely notification to the NRC of an actuation of the auxiliary feedwater system. This issue was determined to be of minimal safety significance because the report was provided to the NRC approximately 3 hours later than required. It is listed as "No Color" because the significance determination process does not apply to this finding. This was a **non-cited violation** of 10 CFR 50.72(b)(2).

No Color. Review of the January 2 event to evaluate performance and procedure adherence was hampered by poor log-keeping practices, untimely and undocumented operator interview information, and poor plant data retrievability. The initial management response to the event was not thorough and allowed power escalation to continue after taking two short term actions. The initial licensee reviews did not identity the procedure adherence and reactivity control issues. Subsequent review by the event review team identified that startup pressures potentially impacted operating activities. Followup actions to address this concern were appropriate. The failure to implement procedure requirements for log keeping was a **non-cited violation** of Technical Specification 6.8.1.a. The log keeping violation was considered more than minor because corrective actions from August 31, 1999, and February 15, 2000, events were not completely effective.

Green. With the plant operating at 100% full power on February 14, 2001, power was lost to 480 volt Bus 3A during a test of safety bus undervoltage relays. The event was caused by technician error in failing to follow the test procedure. This issue had very low safety significance because the loss of safety Bus 3A was of short duration and the remaining multi-train systems were available. The failure to follow procedures was a **non-cited violation** of Technical Specification 6.8.1.a.

No Color. Cross-Cutting Issues. The inspection findings this period, and other issues documented in the corrective action process, indicated a number of human performance issues, some of which had significance relative to personnel safety, plant operation or plant equipment. NRC concerns with the number and significance of human performance errors were discussed with the Plant Manager in a meeting on February 16, 2001. The licensee described actions and plans to address this issue.

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Report Details

SUMMARY OF PLANT STATUS

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

1R04 Equipment Alignments

a. Inspection Scope (71111.04S)

The inspector performed a full system walkdown of the component cooling water (CCW) system to verify correct system and component alignment using check off list (COL) 4.1.1, "Component Cooling System." The inspector reviewed system operating procedure (SOP) 4.1.2, "Component Cooling System Operation," abnormal operating procedure (AOI) 4.1.1, "Loss of Component Cooling Water," and various control room annunciator response procedures. The inspector reviewed the temporary facility change log, operator work around list, and open corrective actions from condition reports in the last two years to identify any deficiencies that may impact CCW system operability.

b. Issues and Findings

No significant findings were identified.

The inspector identified three deficiencies during the system walkdown: a mis-positioned spent fuel pool heat exchanger outlet valve (804); the lack of a seal for the component cooling water surge tank outlet valve (832); and, missing bolts on support ACH-60 for the 23 component cooling water pump. The deficiencies in support ACH-60 did not impact system operability. Con Edison initiated condition reports 200101568, 200101462, and 200101670 for the above discrepancies. Minor equipment deficiencies that were immediately resolved or evaluated included packing leaks, valve label tag deficiencies, poor material condition for suction temperature indicator TI-627, and a missing alignment pin on the inboard bearing cover for the 22 component cooling water pump (CR 200101509). Previously identified degraded components within the component cooling water system were adequately evaluated and characterized by Con Edison. None of the degraded conditions impacted system operability or resulted in operator work arounds, central control room deficiencies, or temporary facility changes.

Valve 804 was required to be open in accordance with check-off list (COL) 4.1.1. The inspector noted that the valve was throttled to control spent fuel pool temperature. Upon identification, Con Edison immediately repositioned valve 804 to the full open position. Spent fuel pool temperature remained within the acceptable range prior to and after the valve was repositioned. Valve 832 is a common suction source for all three component cooling water pumps, and is required to be sealed in the open position. The valve was confirmed open but lacked a seal. Con Edison's preliminary investigation identified that in June, 2000, operators inappropriately initialed the seal as "not applicable" in the COL. The failure to adhere to COL 4.1.1 is considered a minor violation that is not subject to enforcement actions in accordance with Section IV of the NRC Enforcement Policy, because there was no impact on plant safety or system operability. These issues involved a failure to maintain configuration controls for a mitigating system.

1R05 Fire Protection

.1 Fire Zone Tours and Drills

a. Inspection Scope (71111.05)

The inspector toured the areas important to plant safety and risk listed below to evaluate conditions related to (1) licensee control of transient combustibles and ignition sources; (2) the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and (3) the fire barriers used to prevent fire damage or fire propagation. The inspector reviewed a sample of fire protection issues entered in the corrective action program over the last 12 months to verify the licensee took appropriate corrective actions.

- Unit 1 Spent Fuel Area - Fire Zone 700
- Turbine Building-Main Boiler Feedwater Pump Lube Oil - Fire Zone 20
- Turbine Building - Turbine Lube Oil - Fire Zone 18
- Unit 1 Turbine Building Lower Level - Fire Zone 370
- Technical Support Center Diesel Generator Room - Fire Zone 274
- Fire Pump House #12 - Fire Zone 371

The inspector observed a fire brigade drill on February 14, 2001. The inspector evaluated the fire brigade readiness to fight fires with properly donned protective clothing and self-contained breathing apparatus (SCBA). The inspector confirmed the brigade entered the fire area in a controlled manner and brought sufficient fire fighting equipment. The inspector confirmed that the pre-planned drill scenario was followed, the drill objectives were met, and the licensee critiqued the training session. Deficiencies identified during the drill were entered into the licensee's corrective action program as Condition Report 200101644.

b. Issues and Findings

No significant findings were identified.

The inspector noted that the fire protection program plan drawings did not match in-plant conditions. Portable fire extinguishers were either of a different type or in a different location than shown in the drawings. The deficiencies were not significant since the minimum number and type of fire extinguishers within a fire zone agreed with the fire protection program description. Con Edison entered this deficiency in the corrective action program as condition report 200101625.

.2 Fire System Operability During Planned Maintenance

a. Inspection Scope (71111.05)

The inspector reviewed licensee actions to maintain the fire system operable per procedure SAO-703 during periods of planned maintenance to repair fire system valves, including FP-192.

b. Issues and Findings

No significant findings were identified.

The maintenance work required the isolation and tagout of a section of the inner fire header and the #12 fire main booster pump (FMBP). Freeze seals were used to establish the isolation boundary. The fire system remained functional and Con Edison implemented contingency measures to assure adequate fire protection was maintained for sections of the plant potentially impacted by the tagout. Con Edison made notifications to the NRC as required by SAO-703. Con Edison tried to conduct the maintenance on two occasions, but the intended repairs were not completed due to problems in establishing an adequate isolation boundary. Further, human performance issues were noted during this work which resulted in operation of the fire system in configurations not established by procedure SOP 29.6, as described below.

Con Edison operated the #11 FMBP at shutoff head for an extended period of time on February 14, 2001, resulting in pump overheating and damage to the packing seal (CR 200101645). This event was caused, in part, by an operational alignment for the #11 FMBP that was not covered by system operating procedure, and which relied on a relief valve to provide minimum flow recirculation protection for the pump. While restoring from maintenance on February 15, 2001, leakage through valve FP-192 with the diesel fire pump in service resulted in pumping water from the fire water storage tank (FWST) to the city water tank (CR 200101691). Operators included FP-192 on the operational lineup even though the valve was known to leak. The event resulted in the involuntary entry into the 72 hour limiting condition for operation per SAO-703 for having less than the minimum required level in the FWST. The low FWST tank level occurred because the automatic makeup capability through MOV-700 was not functioning. The failure to follow SOP 29.6 is considered a **minor violation** of Technical Specification 6.8.1.a, that is not subject to enforcement action in accordance with Section IV of the NRC Enforcement Policy, because there was minimal impact on plant safety.

1R11 Licensed Operator Requalification

a. Inspection Scope (71111.11)

The inspector reviewed training conducted per Lesson Plan SS.401.011 for Indian Point 2 licensed operators on February 2, 2001, to assess the adequacy of the training, licensed operator performance, emergency plan implementation, and the adequacy of the licensee’s critique. The training considered lessons learned from industry experiences in

SOER 92-1 and included instruction and simulator drills on responding to a small break loss of coolant accident using procedures E-0, E-1, ES 1.2, AOI 28.5, AOI 27.1.1 and SAO-124.

b. Issues and Findings

No significant findings were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspector reviewed Con Edison's follow-up actions for issues on selected structures, systems, and components (SSC) to assess the effectiveness of IP2's maintenance activities. The inspector reviewed selected risk significant equipment problems that occurred. Items checked included licensee evaluation of functional failures, maintenance preventable functional failures, repetitive failures, availability and reliability monitoring, and system specialist involvement. Additionally, the licensee's Maintenance Rule documents and system condition reports were reviewed and system engineers were interviewed. The following system/component performance issues associated with the gas turbines were assessed:

- CR 199900303: Failure of black start diesel generator for Gas Turbine-3
- CR 199905320: Failure of inlet louvers for Gas Turbine-1
- CR 199905519: Gas Turbine-3 combustor high temperature trip
- CR 199906466: Gas Turbine-1 upper turning gear box plugged
- CR 199908022: Failure of starting diesel generator for Gas Turbine-1
- CR 199906466: Failure of Gas Turbine-2 on low bearing oil pressure
- CR 200008051: Gas Turbine-2 trip on high vibrations
- CR 200008974: Failure of starting diesel generator for Gas Turbine-1

b. Issues and Findings

No significant findings were identified.

The inspector identified that the licensee failed to properly account for unavailability hours associated with the Gas Turbine-2 (GT-2) failure on low bearing oil pressure (CR 199906466). Con Edison accounted for 48 hours of unavailability, however GT- 2 was out of service for 65 hours. This deficiency was discussed with the system engineer, who initiated actions to revise the calculation of the performance goal. The significance of this error was minimal since Gas Turbine-2 is currently in (a)(1) status for not meeting unavailability goals. Gas turbine performance and the timeliness of preventive maintenance activities associated with the gas turbines was briefly discussed at the public exit for the 95003 inspection (March 2, 2001) and will be documented in NRC inspection report 05000247/2001002.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope (71111.13)

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 activity. The following maintenance issue was assessed:

- 21 Residual heat removal upper motor oil sample (WO 01-19565, CR 200100756, 200100848)

b. Issues and Findings

No significant findings were identified.

The inspector questioned Con Edison whether the residual heat removal (RHR) pump motor lubricating oil was changed every 2 years as required in the environmental qualification program. Con Edison could not confirm that the preventive maintenance program was implemented correctly and initiated Condition Report 200100848. Con Edison concluded that the motors were operable, but replaced the oil in the upper reservoirs for both residual heat removal pump motors as an emergent action. Con Edison prioritized CR 200100848 as a significance level 2 report to evaluate the root cause and extent of condition associated with the preventive maintenance program for environmentally qualified motors. The inspector confirmed that Con Edison adequately evaluated the extent of condition on other safety-related motors.

Con Edison had previously identified a weakness in the equipment reliability and predictive maintenance program in October 2000 (CR 200007616). One outstanding corrective action for this CR with a due date of December 28, 2001, was to formalize a program to track and trend oil analyses for safety-related motors. The failure to implement timely corrective actions for CR 200007616 resulted in an unnecessary emergent work activity on the RHR pump motors and minimal increase in plant risk. This was an example of a failure to complete corrective actions commensurate with the potential risk significance of the degraded condition. NRC concerns regarding the effectiveness of corrective actions were documented in NRC report 05000247/2000-012.

1R14 Personnel Performance During Nonroutine Plant Evolutions and Events (71111.14)

a. Inspection Scope (71111.14)

During the inspection period, the licensee responded to conditions that required operator actions using special or abnormal procedures. The inspectors observed operator performance, reviewed operator logs, reviewed plant computer data, evaluated operator procedure adherence, conducted operator interviews, and evaluated Con Edison's event response findings. The inspectors reviewed the licensee actions for the transients listed below.

- January 2 Turbine Trip on High Steam Generator Level (AOI 26.4.6)
- February 7 Load Decrease to 65% for Heater Drain System Leak (POP 3.1)
- February 14, Loss of 480 volt Emergency Bus 3A (AOI 27.1.13)

b. Issues and Findings

No significant findings were identified.

.1 Turbine Trip on January 2, 2001

On January 2, 2001, during preparations for a main turbine start-up with the unit at approximately 6.5% power, the main turbine tripped due to high level in the 21 steam generator. By design, the high steam generator level resulted in a main boiler feed pump trip and the actuation of the auxiliary feedwater system.

Operator Performance

Three operator or crew performance problems were noted: 1) the failure to adequately control steam generator level causing the turbine trip and loss of feedwater; 2) operator control of rod insertion without a complete understanding of reactor conditions; and, 3) operator communication errors that resulted in an unnecessary plant cooldown.

With respect to the first problem, the operator noticed an unexpected reduction in main boiler feedwater pump suction pressure and the initial decrease in steam generator levels. The plant operating procedure allowed for starting of a second condensate pump, however the operations crew failed to adequately evaluate the anomalous pressure prior to starting the pump. Further, the operator did not anticipate the effects of swell while rapidly feeding the affected steam generator, and therefore, did not take action to control rising level in a timely manner. The inspector noted that the operator had sufficient information to anticipate the level transient due to the condensate pump start. Con Edison's event response team review concluded that this steam generator (SG) level control problem was the root cause of the turbine trip on high SG level.

For the second problem, the operator inserted control rods in response to rising reactor coolant system average temperature (Tave) and to reduce reactor power to be within the auxiliary feedwater capability. The operator action to drive rods in was, by licensee procedure, a significant control manipulation, and reactor parameters were adequately monitored at the end of rod insertion. However, the operator inserted control rods approximately 7 - 8 steps more than needed which resulted in reactor power going to a lower value than expected (the lowest reactor power was 0.6%). The information on how far to insert rods was based on previous rod positions noted during reactor operation at 2% power a couple of days prior to the event. The senior reactor operator did not provide clear direction to the reactor operator on how much the rods should be inserted or the desired reactor power. The reactor operator did not accurately determine how far the control rods should be inserted. There was no reactivity brief conducted at the beginning of the shift contrary to the licensee's performance standards (not proceduralized). Inserting rods farther than necessary was a crew knowledge/ability weakness. Con

Edison's corrective actions were to ensure core reactivity issues are discussed at shift crew briefings and to change abnormal operating procedures to require a reactor trip when power levels are greater than 4%. The inspectors verified implementation of the corrective actions.

With respect to the third problem, the operators improperly operated the high pressure steam dumps and atmospheric steam dumps due to a communication error between the senior reactor operator and the reactor operator. The abnormal operating procedure in effect was AOI 26.4.6, Main Turbine Trip Without a Reactor Trip. AOI 26.4.6 requires that the operator to stop steaming when RCS Tave is less than 547 F, and to place the high pressure steam dumps in automatic pressure mode at a setpoint of 965-1005 psig. The senior reactor operator told the reactor operator to stop steaming. This was interpreted by the reactor operator to take manual control of the high pressure steam dumps and close the valves. Steam generator pressure increased as expected due to this operator action, and the atmospheric dumps valves automatically opened at the 1020 psig setpoint. The senior reactor operator directed the reactor operator to shift the atmospheric steam dump controller from automatic to manual and open the steam dump valves. This sequence of actions resulted in an unnecessary challenge to the atmospheric steam dumps and a unnecessary reactor coolant system cooldown of approximately 6 degrees to 542 degrees Fahrenheit. At the time of the cooldown, another operator was also withdrawing control rods to restore reactor coolant system temperature. This was contrary to general reactivity controls as stated in station administrative order (SAO)-442, Reactivity Management.

(Green) Technical Specification 6.8.1.a. requires, in part, written procedures to be implemented for activities referenced in Appendix "A" of Regulatory Guide 1.33, Rev. 2. Appendix A includes the requirement for items "1d.","Procedure Adherence," and "6q", "Turbine and Generator Trip." Station Administrative Order (SAO)-133, "Procedures, TS, and License Adherence and Use Policy," step 4.1 requires that procedures shall be followed. AOI 26.4.1 step 4.6 states that if RCS Tave is less than 547F and decreasing, stop dumping steam. AOI 26.4.1 step 4.12, requires operators to ensure that the steam dump control switch is in pressure control and set to maintain 965 - 1005 psig. Contrary to AOI 26.4.1, the operator placed the high pressure steam dump controls in manual and closed the valves, and the operator opened the atmospheric steam dump valves in manual control. Contrary to SAO-442, the operator changed reactivity simultaneously by two means. This is a Green finding due to the low risk and safety significance. 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). This is an example of an operator procedure usage problem. **(NCV 05000247/2000-015-01)**

<u>Equipment Response</u>

Safety equipment responded properly to the steam generator high level trip signal. The 21 boiler feedwater pump tripped, the associated discharge isolation valve closed, both auxiliary feedwater pumps started, and the steam generator blowdown valves closed.

As noted above, the operations crew failed to adequately evaluate the anomalous feedwater pump suction pressure prior to starting a condensate pump. Con Edison's event response team did not identify the specific cause of the low suction pressure. However, one contributor to low suction pressure was flow control valve (FCV)-1113, the gland exhaust condenser return to the main condenser, which was in manual bypass instead of automatic control due to controller deficiencies. The consequence of this condition was to direct approximately 3,000 gallons per minute of condensate flow to the main condenser instead of the suction of the main boiler feedwater pumps. Con Edison prepared CR 200100361 to investigate the low suction pressure.

Quality of Procedures

Con Edison's event response team identified examples of inconsistent and unclear procedural guidance. The inspector reviewed changes made to operations administrative directives (OADs), abnormal operating instructions (AOIs), plant operating procedures (POPs), and system operating procedures (SOPs). The inspector observed various crew briefings on the procedural changes prior to plant power ascension. The procedure changes were adequately implemented.

Con Edison's Event Response

(No color) Con Edison failed to perform a timely evaluation of the turbine trip for reportability. 10 CFR 50.72(b)(2) requires, in part, that the licensee notify the NRC within four hours of the occurrence of the following: any event that results in an automatic actuation of any engineered safety feature. Contrary to the above, on January 2, 2001 at approximately 3:24 p.m., the main turbine tripped resulting in an automatic initiation of the 21 and 23 auxiliary feedwater pumps. Con Edison reported this engineered safety feature automatic actuation to the NRC at 10:05 p.m. This issue was determined to be of minimal safety significance because the report was provided to the NRC approximately 3 hours later than required. It is listed as "No Color" because the "Significance Determination Process" does not apply to this finding. Con Edison initiated condition report 200100200 to document the untimely report. **(NCV 05000247/2000-15-02)**

(No Color) Reconstruction of the events to determine performance problems and procedure adherence issues was hampered by poor log-keeping practices, untimely and undocumented operator interview information, and poor plant data retrievability. Operators failed to implement requirements within Operations Administrative Directive (OAD)-3, "Plant Surveillances and Logkeeping." Specifically, OAD 3 sections 4.4.6(1) requires , in part, that subsequent chronological entries shall be made of operational problems or events. Some examples of log entries shall be changes in reactor status or power. Contrary to the above, operators failed to log a decrease in reactor power from the turbine trip on January 2, 2001. Condition report 200100364 documented that logkeeping did not meet OAD3. 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 the requirement for "Log Entries, Record Retention, and Review Procedures." The repetitive failure to follow procedures and enter the required data into control room logs indicates a performance trend which resulted in a violation. This violation was considered more than minor because corrective actions from August 31, 1999, and February 15, 2000 events were not completely effective. It is listed as "No

Color" because the "Significance Determination Process" does not apply to this 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 05000247/2000-015-03)**

The post-event debriefs with the operating crew and Con Edison management were not thorough. The conclusions from the initial debriefs were that it was acceptable to continue the plant startup following the turbine trip with two short-term actions. The first action was to ensure that the second condensate pump continued to operate to minimize any potential feedwater flow transient. Secondly it was stressed to ensure that steam generator levels were maintained and controlled in accordance with the applicable procedure. The initial debriefs did not challenge operator actions relative to procedure adherence and appropriate operator reactivity manipulations to lower reactor power. Line management and senior management reviews of the performance problems surrounding the event were untimely relative to proceeding with plant power escalation.

Con Ed follow-up actions were more thorough and balanced. Con Edison identified, based upon subsequent operator interviews, that plant startup pressures potentially influenced operator actions within the control room. The inspectors observed numerous all-hands meetings between Con Edison senior management and middle management, and then subsequent meetings between managers and employees to discuss startup pressures. The meetings addressed the perceptions on schedular pressures to perform.

.2 February 7 Load Decrease to 65% for Heater Drain System Leak (POP 3.1)

The operators reduced plant load to 65% full power on February 7, 2001 in response to a leak from the impulse line for flow controller FC 1117S on the discharge of the 21 heater drain tank pump. The leak was repaired by replacing a fitting on the impulse line and the plant was restored to full power operation. Further NRC review of this event and concerns regarding the use of plant operating procedures will be described in NRC Inspection Report 05000247/2001002.

.3 February 14, Loss of 480 volt Emergency Bus 3A (AOI 27.1.13)

Event Overview

With the plant operating at 100% full power on February 14, 2001, power was lost to 480 volt safety Bus 3A during a test of the safety bus undervoltage relays. The event was caused by technician error in failing to follow a step in the test procedure. Operators responded to the event per AOI to re-energize the bus and to restore loads. The electrical system and plant responded as designed. This human performance event had minor impact on plant operations and had low risk significance. Con Edison's followup reviews and actions were appropriate.

Personnel Performance

The loss of Bus 3A event was caused by a technician error in the failure to follow procedure PT-M48. Procedure Step 3.3.3 required that the undervoltage trip signal be blocked prior to inserting the trip signal to test the undervoltage relays. The trip signal was inserted without blocking the trip signal. Although procedure PT-M48 was adequate as written, Con Edison identified several ways to enhance the procedure format and test implementation (component labeling and agreement between prints, procedure and controls). The pre-job briefings with operators and amongst the technicians performing the test was short and informal.

Upon loss of Bus 3A, the operators entered the technical specification (TS) limiting condition for operation (LCO) per TS 3.7.A.3 and TS 3.0.1 for the loss of safety related equipment; however, the operators failed to enter the 72 hr LCO per TS 3.5-3 Item 3.b, Column 5, regarding the inoperable undervoltage relays during the conduct of PT-M48 (CR 200101646).

(Green) The failure to follow surveillance procedure PT-M48 was a violation of Technical Specification 6.8.1.a, which requires, in part, that written procedures shall be implemented covering the activities referenced in Appendix A of NRC Regulatory Guide 1.33, revision 2. NRC Regulatory Guide 1.33, Revision 2, Appendix A 8.b. requires specific implementing procedures for each surveillance listed in the technical specifications. Surveillance procedure PT-M48 was prepared to implement the TS Table 3.5-3 requirements. The failure to follow PT-M48 had very low safety significance because of the short duration associated with loss of safeguards bus 3A and availability of the remaining multi-train systems. This finding affects the Mitigating Systems Cornerstone because it relates to a loss of a safety function (480 volt supply to Bus 3A) for greater than the technical specification allowed outage time. This violation was treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65 FR 25368).
(NCV 05000247/2000-015-04)

Equipment Response

Upon loss of Bus 3A, all three emergency diesel generators (EDGs) started and operated in standby, including EDG 22. The operator manually energized Bus 3A loads from EDG 22 per AOI 27.1.13 within about 25 minutes. The AOI directs that the diesel be used to load the bus even if the 6.9KV bus is available. The time to energize the bus was reasonable given the sequence of actions in the control room and the switchgear room to verify the status of the electrical system, start backup equipment, ensure normal reactor functions are being maintained, realign the switches and breakers for loads on the affected bus, and then energize and load the affected bus from the diesel generator.
The electrical system responded per design, in that a loss of power on any 480 volt safety bus will start all three emergency generators. The diesel breaker closure logic requires that in addition to an undervoltage condition, a blackout or safety injection (SI) signal also be present to energize the bus.

One anomalous equipment condition noted during the event was that the operator had to manually start the crankcase exhauster for EDG 22. This issue was addressed in the corrective action program (CR 200101624). Once started, the operators ran each diesel generator loaded per normal operating procedures prior to restoring them to a standby condition.

Con Edison's Event Response

Con Edison initiated an investigation for the event, which included a review of human factors in the test method. As interim corrective actions, Con Edison had a stand-down with I&C personnel on February 15 to review the event, lessons learned and self-checking techniques. Con Edison identified several minor improvements for AOI 21.1.13 (CRs 200101658, 200101660 and 200101661). Con Edison's actions in this area remained in progress at the conclusion of the inspection.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

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

- CR 200101442, Flow Assisted Corrosion of Main Steam Piping
- CR 200100848, 21 Residual Heat Removal Pump Motor Oil Sampling
- OD 00-20, Pressurizer Safety Valve Leakage, CR 200010860
- OD 01-02, RPS Discrepancies and Operability, CRs 200010688, 200100327

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

b. Issues and Findings

No significant findings were identified.

Con Edison noted leakage from two of the three pressurizer safety valves during the plant startup, with most leakage coming from valve PSV-466 (CR 200010860). The leakage was identified and is being monitored using temperature indicators on the valve tail pipes and by using thermography. The leakage is captured in the pressurizer relief tank (PRT) and assessed by monitoring changes in PRT temperature, pressure and level. The leak rate increased slightly during the inspection period, but remained below 0.1 gallons per minute (gpm), which was well within the Technical Specification 3.1.F.3.c limit of 10 gpm for "identified" RCS leakage. The minor leakage did not affect plant safety. Con Edison completed Operability Determination OD 00-20 and established an administrative limit of 1.0 gpm for continued plant operation with the code safety valve leakage. Con Edison continued to monitor the leak rate each shift and initiated plans to replace and/or repair the safety valves during the next outage in which the plant was taken to cold shutdown.

Con Edison issued OD 01-02 to address the reactor protection system (RPS) wiring discrepancies identified in Condition Report 200100327, which related to the system wiring configuration, and the loss of quality control over system modifications. Con Edison concluded the RPS remained operable based on the reviews and corrective actions for the individual deficiencies, the RPS surveillance test program, the evaluation conducted in response to Generic Letter 96-01, and the operability evaluation described in OD 00-18. NRC inspection did not identify anything that would compromise functionality of the RPS. NRC inspections in this area continue.

Con Edison identified erosion in the #23 main steam line piping outside containment (Condition Report 200101442). Con Edison's technical evaluation concluded there was assurance that the main steam pipe minimum wall would remain above the ASME code allowable value until approximately July, 2001. Con Edison plans to either update the engineering calculation utilizing methodology identified in ASME Code Case N597, or perform and evaluate additional non-destructive examinations. NRC inspections in this area continue.

1R20 Startup and Power Operations

.1 Sustained Control Room and Plant Startup Observations

a. Inspection Scope (71111.20, 71715)

The inspector implemented an augmented inspection plan during plant startup to permit long-term, heightened observations of control room and plant activities (which included periods of three shift staffing) from December 31 to January 12, 2001. Inspectors observed and monitored activities that implemented a mode change from low power to full power operations. Operator actions were observed to ensure that the operators, control room supervision and shift management maintained cognizance of system and component conditions, work activities in the field, and expected off normal plant conditions associated with maintenance and testing activities. Operator actions were observed to ensure that operator response was effected in a timely manner to unexpected plant conditions.

Shift turnovers and shift briefings were observed to verify adequate and correct information was communicated to the operators. Communications between operators and testing personnel before and during testing were observed to verify clear identification of the planned effect on plant equipment and impact on the control room and to verify appropriate control room indications resulting from in-field testing activities. The inspector verified control room operators were aware of and effectively communicated equipment conditions caused by testing or other field evolutions. The inspector also confirmed appropriate operator response to changing equipment conditions with the use of alarm response procedures, as necessary. In addition, the inspector questioned operators on various control room indications and reasons for lit annunciators. Shift logs were reviewed to confirm proper entry into and exit from technical specification limiting conditions for operation for inoperable equipment and documentation of control room activities. Compliance with selected plant technical specifications (TS), plant procedures, and final safety analysis report (FSAR) assumptions were verified. The following activities were reviewed:

- Shutdown risk evaluations per OAD-37

- Zero Power Physics Testing per RFE-S-16.032
- NI Power Range Alignment per SOP 13.1
- GT-1 Testing per PMT 19229 and PT-A36A (CR 200010756)
- Flux Mapping during Startup - Map 15FC02 and 15FC01
- Core Power Distribution Limits Evaluation on January 5 and 23, 32001
- Plant Startup and Power Operation per POP 1.3 and 2.1
- Action per AOI 29.2 for Offsite Power Perturbations
- Actions per AOI 20.2 for Hotwell Chemistry
- Actions to Repair 21 MBFP High Pressure Stop Valve Leak
- Actions to Repair Excessive Condenser Air In-leakage (CR 200100471)
- Simulator Validation of Power Ascension Procedures
- Control Rod Stationary Coil Indicator Fuse Failure (CRs 200101265, 200101359)
- Steam Generator Blowdown Radiation Monitor Alarm (CR 200101057)
- Inoperable IRPI for Control Bank ‘A’ Rod L-3 (CR 200100916)

Plant configurations, equipment status, procedural controls, and compensatory measures consistent with applicable license requirements, design drawings and utility commitments were verified. Control room panel deficiencies, tagouts, work requests, abandoned equipment, alarmed conditions and alternate plant configurations were inspected, validated and discussed with control room operators and supervision. The inspectors also evaluated the cumulative effects of identified conditions on the ability of operators to respond to transients.

b. Issues and Findings

No significant findings were identified.

Inspector observations of performance deficiencies having minor safety significance were discussed with plant management. Exceptions to acceptable performance in the areas of procedure use, communication and briefings are described in Section 1R14 of this report.

The inspector evaluated a temporary procedure change made to the Annunciator Response Procedure (ARP), revision 21, “Reactor Coolant System,” to ensure that the procedural modification had provided operational guidance which was consistent with operability determination (OD 00-020) recommendations. The procedure change provided guidance for plant shutdown on pressurizer safety valve leakage consistent with the OD which stated that pressurizer safety valves were operable with minor amount of leakage provided it was less than 1.0 gpm. The inspector also reviewed changes to standard operating procedure 1.7, "Reactor Coolant System Leakage Surveillance," to ensure consistency between the procedures relative to actions regarding increased leakage. The inspector concluded the safety valves remained operable and Con Edison’s actions were acceptable.

The inspector evaluated the operator response to a steam generator blowdown radiation monitor alarm on January 28, 2001 (CR 200101057). The inspector verified the automatic isolation functions performed as designed, no primary to secondary leakage existed, and that operators implemented abnormal operating instruction (AOI) 1.2, “Steam Generator Tube Leak.” The cause of the alarm was radioactive hideout within the monitor sampling lines from the tube leak on February 15, 2000. Previously, on January 4, 2001, a nitrogen-16 radiation monitor alarm occurred (CR 200100112) due to electrical spiking on the system. No primary to secondary leakage existed throughout the plant startup, as shown by the process radiation monitors and by analysis of the steam generator blowdown and the air ejector offgas samples.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The tests listed below were observed in the control room and in the field to confirm performance in accordance with approved procedures. The test results were reviewed to verify the equipment met procedural acceptance criteria and was operable consistent with technical specification requirements.

- PT-Q29B 22 Safety Injection Pump Test
- PT-Q59 Containment Pressure Bistable Logic Testing
- PT-Q13W 22 Fan Cooler Service Water Containment Isolation Valve Test

The inspector witnessed the test of the 22 Safety Injection pump to assess whether the system met technical specification requirements, updated final safety analysis report assumptions and licensee procedural requirements. The inspector also reviewed the test methodology to ensure that it effectively demonstrated that the 22 SI pump was capable of performing its intended safety function. The inspector also verified that appropriate instrumentation was utilized by the licensee and walked down the pump, piping, and associated components to ensure that previous identified system leakage had not increased beyond acceptable limits.

b. Issues and Findings

No significant findings were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23)

The inspector reviewed the temporary plant modifications and associated safety evaluations listed below to verify the facility changes did not impact safety system operability and the license requirements, and did not create an unreviewed safety question per 10 CFR 50.59. The inspection verified the activities were completed in accordance with Con Edison controls for installation and testing, and that conditions associated with the changes were addressed in the corrective action system. The following TFCs were reviewed:

- 2001-013, IRPI C-9 Removal From Proteus (SE 91-032-TM)
- 2001-004, Remove L03 IRPI from Proteus (SE 91-032-TM)
- 2001-011, R-49 Removed From Service (WO 01-19967)
- 2000-286, Core Exit Thermocouples Removed from Scan (SE 95-204-PR)
- 2000-294, Pressurizer Safety Valve 466 Temperature Alarm (WO 00-19499)
- 2000-293, GT-1 Fuel Injection

b. Issues and Findings

No significant findings were identified.

EMERGENCY PREPAREDNESS (EP)

EP4 Emergency Plan Reviews

a. Inspection Scope (71114.04)

The inspector reviewed licensee submitted changes for the following emergency plan implementing procedures to determine if the changes decreased the effectiveness of the plan:

IP-1002, Emergency Notification and Communications (Rev 21)
IP-1011, Joint News Center (Rev 0)
IP-1013, Protective Action Recommendations (Rev 8)
IP-1018, Media Relations (Rev 8)

b. Observations and Findings

No findings of significance were identified in this area.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas

a. Inspection Scope (71121.01)

The inspector witnessed the activities listed below and selectively examined the records and reports to evaluate the effectiveness of the licensee's access controls to radiologically significant areas. The inspector reviewed the following:

- Observation of soil sampling and collection of split samples on January 30, 2001 inside the former interim steam generator storage area
- Condition Report 200100451 - removal of small amounts of radioactive material from cribbing and soils in the interim steam generator storage area

- Personnel contamination investigation CR 20007698 generated on October 9, 2000; Attachment 8.2, HP-SQ-3.801, Personnel Decontamination, Rev. 16)
- Skin dose initial evaluation (October 9, 2000, Attachment 8.3, HP-SQ-3.801, Personnel Decontamination, Rev. 16)
- Calculation Number PGI-00461-00 (2000 Thumb Dose, January 3, 2001)
- Removable contamination surveys for external surfaces of replaced steam generators (body and bowl) in disintegrations per minute per 100 square centimeters (September 2000) and the isotopic sample log for the same surveys
- Gamma spectroscopy results for soil samples taken on January 15, January 18, January 30, and January 31, 2001
- Qualification status listing of individuals for respirator use (both current and expired) (January 29 and January 30, 2001)
- Expire-soon reports (for the weeks of January 22 and of January 29, 2001) (list of persons for whom one or more requirements to perform work in the RCA are about to expire on dates listed)
- Quality Assurance Audit Report No. 00-03-C Radiation Protection Program (October 16 - 27, 2000)
- Radiological Protection Self-Assessment, Indian Point Station, November 28 - December 22, 2000
- Letter Report, Review of Phase 2 Environmental Due Diligence Radiation Survey Results and Recommended Site Specific Derived Concentration Guideline Limits for Soil Characterization
- Condition Report 200101481 - retrieval of tools and valve parts with small amounts of radioactive material from offsite storage areas

The inspector also reviewed selected sections of the following procedures to evaluate the effectiveness of the licensee's access controls to radiologically significant areas.

- HP-SQ-3.008, Rev. 19, Radiation Work Permit
- HP-SQ-3.304, Rev. 6, Hot Particle/Particle Monitoring and Control
- HP-SQ-3.701, Rev. 11, Health Physics Count Room Standard Practices
- RS-8.002, Rev. 6, Skin Dose Assessment
- RS-Q-8.301, Rev. 8, Radiological Parameters Data Collection and Trending
- Attachment 7.1 Review of Daily Individual and Radiation Work Permit (RWP) Exposures
- RS-10.001, Rev. 9, Selection of Respiratory Protection Devices
- Attachment 7.1 Total Effective Dose Equivalent (TEDE) ALARA Evaluation for Use of Respiratory Protection Devices
- Addendum 8.2 TEDE ALARA Evaluation Guidelines

The inspection included a review of the following Condition Reports for the appropriateness and adequacy of event categorization, immediate corrective action, corrective action to prevent recurrence, and timeliness of corrective action: Condition Reports 2000-07698, 2000-09188, 2000-09461, 2000-09593, 2000-10347, 2000-10579, 2001-00451, 2001-00451, 2001-01481 and 2001-00650.

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

b. Findings

No significant findings were identified.

Con Edison received notification from Indian Point 3 that four pressure switches removed from storage in Unit 1 had minor amounts of radioactive contamination (CRs 200101347 and 200101481). Con Edison completed radiation surveys of materials and areas where the switches were stored in Unit 1 and at other locations. Con Edison also identified 14 other slightly contaminated items in Unit 1, at a warehouse in the owner controlled area, and at a Con Edison warehouse in Astoria, NY. The materials were recovered for proper processing. In each case, the radiation levels caused by the fixed contamination was very low and did not create a significant radiological hazard. Con Edison began a Significance Level 2 investigation to identify the causes for this lapse in the control of radiological materials. This performance issue involved a failure to maintain radiological controls for the public safety cornerstone area. The failure to follow controls for release of materials from the radiological controls area is considered a **minor violation**, that is not subject to enforcement actions in accordance with Section IV of the NRC Enforcement Policy, because there was no impact on public safety.

.1 Old Steam Generator Removal and Site Remediation

a. Inspection Scope (71121.01, 50001)

The inspector reviewed Con Edison actions to move the old steam generators into storage in the permanent building constructed on site for that purpose, and to restore and decontaminate the interim steam generator storage area. The inspection included independent NRC radiation surveys (survey instruments 8669 and 17119) and independent NRC sampling and confirmatory analysis of remediated soils in the interim steam generator storage area.

The inspector observed the licensee's collection of soil samples from pits #03 and #04 at the steam generator (SG) interim storage area on February 8, 2001. The inspector reviewed procedure NEM 5.106, "Collection, Preparation, and Analysis of Soil Samples", Rev. 6.; and, discussed the sample plan with the Health Physics Manager.

b. Issues and Findings

No significant findings were identified.

After moving the old steam generators to the permanent storage area, Con Edison identified minor amounts of radioactive material in the wood cribbing and soils below the locations where the steam generators were stored in the interim storage area (Condition Report 200100451). Con Edison and Independent NRC radiation surveys confirmed that the radiation levels on the timbers were well below regulatory limits and contamination levels in the soil were very low.

The radioactive material was cesium-137 which leached through the encapsulating material and was transported with small amounts of encapsulating material that fell below the steam generator primary bowl on to the storage pads. The initial soil concentrations of plant-related cesium-137 were comparable to environmental levels of cesium. Con Edison surveyed the interim storage area and the transport path to the permanent storage building. No other contamination was found. Con Edison removed the contaminated timbers and soils and remediated the interim storage area for unrestricted access. After the completion of remediation activities, soil samples were taken for independent NRC analysis.

On February 8, 2001, the inspector observed the licensee collect soil samples from pits #03 and #04. The soil samples were split with the NRC. A HP technician collected the samples under the guidance of Nuclear Environmental Monitoring (NEM) personnel, who routinely collect samples for the Radiological Environmental Monitoring Program (REMP). The soil sampling procedure, NEM 5.106, “Collection, Preparation, and Analysis of Soil Samples”, Rev. 6 contained the necessary information and steps to perform adequate sample collection, but was not utilized to collect the samples from the SG interim storage location. The samples were not always collected per the NEM guidance. The inspector specifically observed the licensee’s soil sampling techniques to prevent cross-contamination. The licensee wiped the shovel and removed loose material to prevent cross-contamination. The licensee used a separate plastic bag inside the bucket for each sample, and frisked the equipment prior to commencing soil collection at the next location. Overall, the licensee efforts to collect the soil samples were acceptable, however, by not using the procedure the potential existed for inconsistent sample collection.

The inspector discussed the sample plan to remediate the interim storage area with the Health Physics Manager and determined that the plan did not contain specific details to assure samples were collected per the NEM guidance. Con Ed acknowledged this observation.

2OS2 ALARA Planning and Control

a. Inspection Scope (71121.02)

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

- Letter dated December 21, 2000 and titled Person-Rem Exposure Goals for Year 2001 which included a goal breakdown by individual work groups
- Forced Steam Generator Outage 2000 ALARA Report
- Steam Generator Replacement 2000 ALARA Report
- Post Steam Generator Replacement/Refueling Outage 2000 ALARA Report

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

b. Findings

No significant findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Occupation Exposure Control Effectiveness

a. Inspection Scope

The inspector selectively examined records used by the licensee to identify occurrences involving high radiation areas, very high radiation areas, and unplanned personnel exposures for the fourth quarter of 2000 and for early January 2001 against the applicable criteria specified in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 0, to verify that all conditions that met the criteria were recognized and identified as Performance Indicators. The reviewed records included corrective action program records (Condition Reports) and Reviews of Daily Individual and RWP Exposures (Attachment 7.1 of RS-Q-8.301, Rev. 8, Radiological Parameters Data Collection and Trending).

b. Findings

No significant findings were identified.

.2 Reactor Coolant System Activity

a. Inspection Scope (71751)

The inspector examined corrective action program records, chemistry records and logs, and isotopic analyses of reactor coolant for occurrences involving specific activity and compliance with the limits of Technical Specification 3.1D, Maximum Coolant Activity". The inspector also reviewed data for the 12 quarters dating back to January 1998. The guidance in NEI 99-02, was consulted to verify that plant data was properly identified within the published performance indicators. The inspection included a review of the indicator definitions, data reporting elements, calculational methods, definition of terms, and clarifying notes. The inspector reviewed the collection, analysis and reporting of reactor coolant samples per procedures IPC-S009, IPC-A-294, and IPC-A-299.

b. Issues and Findings

No significant findings were identified.

Indian Point 2 does not have a technical specification limit for reactor coolant dose-equivalent Iodine-131 (as addressed in Condition Report 200009540 along with other issues). Con Edison used the Westinghouse Owner Group Standard Technical Specification Value of 1.0 micro-Ci/gram to compute the performance indicator in a manner similar to the NEI 99-02 guidance. Con Edison intends to include a limit for dose equivalent Iodine-131 in a subsequent revision to the IP2 license. The PI was in the green band for plant operations up to February 15, 2000, and was reported as not applicable during the remainder of 2000 due to the plant shutdown for the steam generator replacement project. The plant data for the remainder of 2000 and the first month of 2001 show continued operation within the green band.

4OA2 Cross Cutting Issues

.1 Human Performance Issues

a. Inspection Scope

The inspector reviewed condition reports and licensee actions in response to events that involved human performance errors.

b. Findings

No significant findings were identified.

On February 7, 2001, the inspector observed that two cells in the 22 station battery had electrolyte level below the low level mark. Con Edison initiated condition report 200101436 to document this observation and to further evaluate the electrolyte level on all cells for the 22 station battery. The recent vendor guidance associated with minimizing addition of water to the cells for this recently replaced battery was not properly implemented.

In addition to the above, the NRC noted other human performance events this period, some of which impacted plant equipment or affected plant operation (see Sections 1R04, 1R05, 1R14 and 2OS1 above). Several issues were determined to involve violations of NRC requirements in the areas of radiological controls, procedure use and adherence, and event reporting. Other human performance issues were noted in the corrective action process documentation (water hammer in the secondary systems on plant startup and in the utility tunnel on a house heating steam line, and inadvertent operator actuation of the wrong 480 volt transfer switch). The inspector met with the Plant Manager on February 16, 2001, to discuss concerns with the number and significance of recent human performance events. The Plant Manager described Con Edison’s action to consider performing risk significant events on day shifts and for accelerating the schedule for the planned station-wide stand down to review human performance topics.

4OA3 Meetings

Exit Meeting Summary

On February 23, 2001, the resident inspector presented the inspection results to Mr. A. Blind and other members of the Con Edison staff who acknowledged the findings. The inspectors asked whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF INDIVIDUALS CONTACTED

A. Blind	Vice President - Nuclear Power
R. Masse	Plant Manager
W. Smith	Operations Manager
T. Poirier	Work Control Manager
T. Waddell	Maintenance Manager
C. Tippin	Reactor Engineer
W. Osmin	Reactor Engineer
E. Woody	I&C Manager
G. Schwartz	Chief Engineer
T. Burns	Supervisor, Radiation Support
M. Donegan	Health Physics/Radioactive Waste Manager
R. Majes	Radiation Support Health Physicist
L. Menoscal	Radiation Support Health Physicist
L. Mettey	NEM Technician
M. Miele	Radiation Protection Manager
V. Nutter	Radiation Support Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

05000247/2000-15-01	NCV	Failure to follow operating procedures
05000247/2000-15-02	NCV	Failure to make timely notifications
05000247/2000-15-03	NCV	Failure to follow log keeping procedures
05000247/2000-15-04	NCV	Failure to follow surveillance procedures

LIST OF DOCUMENTS REVIEWED

AOI 27.1.13, Loss of a 480V Bus
Condition Report 200101621, Loss of Bus 3A
PT-M48, 480 Volt Undervoltage Alarm
Drawing A225100-05, Sheet 7
Drawing 9321-LL-3117-19, Sheet 8
Drawing 9321-LL-3118-22, Sheet 2
Drawing 9321-LL-3118-19, Sheet 8
Drawing 9321-LL-3117-18, Sheet 3A
Drawing 9321-LL-3118-23, Sheet 3
Drawing 9321-LL-3117-10, Sheet 22A

LIST OF ACRONYMS USED

ALARA	As Low As reasonably Achievable
AOI	abnormal operating instructions
ARP	annunciator response procedure
CFR	code of federal regulations
COL	check-off list
CR	condition report
EDG	emergency diesel generator
FC	flow controller
FCV	flow control valve
FMBP	fire main booster pump
FSAR	final safety analysis report
FWST	fire water storage tank
gpm	gallons per minute
GT	gas turbine
HP	health physics
LCO	limiting condition for operation
NEI	Nuclear Energy Institute
NEM	nuclear environmental monitoring
OAD	operation administrative directive
OWA	operator work around
POP	plant operating procedure
PRT	pressurizer relief tank
QA	quality assurance
RCS	reactor coolant system
RWP	radiation work permit
SAO	station administrative order
SCBA	self-contained breathing apparatus
SCM	significant control manipulation
SDP	significant determination process
SE	safety evaluation
SG	steam generator
SI	safety injection
SOP	system operating procedure
SSC	selected structure, system or component
Tave	reactor coolant system average temperature
TFC	temporary facility change
TS	technical specifications

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.