May 16, 2000

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

SUBJECT: NRC INTEGRATED INSPECTION REPORT 05000247/2000-003

Dear Mr. Blind:

This letter transmits the results of safety inspections conducted by NRC inspectors at your Indian Point 2 reactor facility from February 29, 2000 through April 1, 2000. The unit was in cold shutdown throughout the inspection period as steam generator examinations continued and the reactor was disassembled for refueling.

Based on the results of this inspection, an apparent violation was identified that is under evaluation in conjunction with other findings from the Augmented Team Inspection described in Report 05000247/2000002. The apparent violation involves the failure to maintain the isolation valve seal water (IVSW) system operable per Technical Specification 3.3.C and as described in the Updated Final Safety Analysis Report (UFSAR) Sections 6.5 and 14.3.6.1. The isolation valve seal water system became inoperable on February 15, 2000 during the steam generator tube failure event shortly after it actuated in response to a phase A containment isolation signal. There were prior opportunities to have corrected deficiencies in IVSW system operation. We understand that modifications to correct system response to a containment isolation signal will be completed prior to plant restart from the present steam generator outage. Should you have a different understanding regarding plans to correct the IVSW system problem before restart, please contact Mr. Peter Eselgroth at 610-337-5234.

Also, based on the results of this inspection, the NRC has determined that three Severity Level IV violations of NRC requirements occurred. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VII.B.1.a of the Enforcement Policy (November 9, 1999, 64 FR 61142). The NCVs involve the failure to maintain the reactor coolant pump oil collection system per regulatory requirements, the failure to follow limits in residual heat removal operating procedures for reactor differential temperature, and the failure to provide adequate maintenance instructions for fire dampers in the cable spreading room. If you choose to contest these violations or the severity level of these NCVs, 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, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Indian Point 2 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. Should you have any questions regarding this report, please contact Mr. Peter Eselgroth at 610-337-5234.

Sincerely,

/RA/

A. Randolph Blough, Director
Division of Reactor Projects

Docket No. 05000247
License No. DPR-26

Enclosure: Inspection Report No. 05000247/2000-003

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No. License No.	05000247 DPR-26
Report No.	05000247/2000-003
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York
Dates:	February 29, 2000 through April 1, 2000
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector Jennifer England, Resident Inspector Gregory Cranston, Reactor Engineer
Approved by:	Peter W. Eselgroth, Chief Projects Branch 2 Division of Reactor Projects

EXECUTIVE SUMMARY

Indian Point 2 Nuclear Power Plant
NRC Inspection Report No. 05000247/2000-003

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a five-week period of inspection by resident and regional inspectors.

<u>Operations</u>

Con Edison had adequate controls for placing two reactor coolant pumps on the backseat, and adequate plans for containment closure in the event of a loss of reactor cooling. An operating procedure lacked guidance to assure a safe shutdown instrument was properly vented. (O1.1)

The operators failed to control RCS differential temperature within limits during RHR system operation. The failure to follow SOP 4.2.1 was a non-cited violation of NRC requirements. Licensee actions continued at the end of the inspection period to evaluate the impact on the baffle-former and baffle-barrel bolts in the reactor vessel internals, and to resolve this matter prior to plant restart. (O1.2)

The operators promptly responded to the loss of power to the steam generator nozzle dams. The nozzle dam normal air supply was lost; however, no loss of reactor coolant system inventory occurred, and no monitoring existed for the nozzle dams for approximately one hour. Con Edison failed to control and integrate several temporary facility changes for the nozzle dam support systems. Inadequate coordination between operators and workers resulted in a near miss for a significant injury. (O2.1)

Plant management presentations to the Nuclear Facilities Safety Committee were incomplete. However, the committee members appeared well prepared and provided good discussions on the February 15 steam generator tube leak event. (O7.1)

Con Edison completed the investigation of the plant response to the February 15, 2000 steam generator tube leak. Corrective actions to address the causes of weaknesses in the plant response to the event were in progress at the end of the inspection period and NRC review will be the subject of an AIT follow-up team inspection. The results of the root cause investigation for the steam generator tube failure were not reviewed and are being provided by Con Edison to the NRC Office of Nuclear Reactor Regulation for review. (O7.2)

<u>Maintenance</u>

Maintenance activities were satisfactorily completed. The conduct of surveillance tests during the period was acceptable. Maintenance and test activities were not consistently performed in accordance with expectations and administrative controls. The initial evaluations in preparation for a turbine load test did not completely consider shutdown risk. (M1.3)

Executive Summary (cont'd)

The containment liner became corroded due to prolonged contact with borated water in areas where moisture barriers were degraded. Con Edison actions continued to investigate and repair liner degradation, and to assure that margins to design limits were maintained. (M2.1)

Engineering

The failure to collect leakage from the vent pipe and the lower oil reservoir drain connections on three RCP motors is considered a violation of 10 CFR 50 Appendix R, Section III. This Severity Level IV violation is being treated as a Non-Cited Violation. A long-standing deficiency in the oil collection system had gone uncorrected. (E2.1)

Con Edison did not recognize a long-standing difference between the design and licensing basis for the isolation valve seal water system. Despite several past events and a design basis verification program which highlighted IVSWS performance issues, Con Edison failed to correct a basic design deficiency and assure that the licensing basis was met. Operability evaluations were less than adequate and corrective actions were narrow and untimely. The failure to assure regulatory requirements were correctly translated into specifications, drawings and procedures was an apparent violation. (E2.2)

Steam generator eddy current testing and analysis was conducted. The eddy current test results revealed defects which resulted in a Classification of C-3 per Technical Specification 4.13. (E2.3) More detailed review of steam generator inspection results is under the purview of the NRC Office of Nuclear Reactor Regulation.

A lack of maintenance installation instructions contributed to the failure of cable spreading room fire dampers to fully close. The faulty dampers caused the suppression system to be degraded for approximately 3 months. The failure to maintain provisions of the NRC-approved fire protection plan as described in the UFSAR and approved NRC Safety Evaluation Report is a Non-Cited Violation. (E2.4)

Plant Support

Con Edison staff appropriately responded to the discovery of trace amounts of contamination in the Unit 1 storm drains and took proper actions to resolve the condition and to investigate the cause. The material was not associated with the Unit 2 steam generator event or any recent plant activities, and there was no radiological dose consequence due to the contamination. (R1.1)

TABLE OF CONTENTS

ATTACHMENTS

Report Details

I. OPERATIONS

O1 Conduct of Operations

The plant was in cold shutdown during the inspection period to inspect steam generators following a tube failure while operating at power on February 15, 1999. On March 27, Con Edison announced its decision to enter the refueling outage. Reactor vessel disassembly was in progress at the end of the inspection period.

O1.1 Operational Safety Verification

a. Inspection Scope (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. Specific observations are described below.

b. Observations and Findings

The inspector performed regular tours in the control room, switchgear room, auxiliary building, diesel generator building, turbine building, primary auxiliary building, containment building, spent fuel building, and areas within Indian Point Unit 1. Plant parameters important to safety were observed within allowable limits during control board and plant status reviews.

Reactor Coolant Pumps on the Backseat

Con Edison placed the 21 and 23 reactor coolant pumps (RCPs) on the backseat to facilitate maintenance on the seal packages. A reactor coolant pump that is disconnected from the motor with the pump shaft lower journal bearing resting on the upper thermal barrier is on the backseat. The backseat minimizes reactor coolant system leakage during maintenance on the reactor coolant pump seal package.

The pre-evolution brief was consistent with procedural guidance in SAO-202, “Infrequently Performed Tests or Evolutions.” Lessons learned from NRC Information Notice 96-58, “RCP Seal Replacement with Pump on Backseat,” were correctly incorporated into temporary operating procedure (TOI) 235, “RCP Backseat Operations with Fuel in the Reactor or During Fuel Transfer”.

The inspector observed nuclear plant operators adequately monitored leakage from the two reactor coolant pumps and control room personnel were adequately aware of contingency actions if reactor coolant system leakage increased significantly. The leakage from the 23 RCP was greater than 2 gallons per minute (gpm) on March 24, 2000, which was greater than the expected leakage of less than 1 gpm. Operators isolated the leakage by securing the seal injection line vent valve. Procedure TOI 235 only provided guidance to increase leakage monitoring if leakage was greater than 2 gpm. After discussions with the inspector, Con Edison processed temporary procedure change 2000-0071 to add a step to close the seal injection vent valve. No consequence occurred from the leakage. The seals were successfully replaced and the RCPs were taken off the backseat.

Containment Closure

Nuclear plant operators involved with containment closure were knowledgeable of their responsibilities. The contingency plans for closure were established in the event of a loss of residual heat removal. Operators had instructions on which containment penetrations needed to be isolated and a number of connections penetrating through the containment equipment hatch. Containment evacuation alarms were tested daily during this period.

On March 16, 2000, Con Edison increased the number of lines penetrating the equipment hatch in support of steam generator sludge lancing. This complicated the actions needed to close the containment. The inspector noted a lack of consistency between operating crews in contingency planning, such as specific roles and responsibilities on closure activities and timely evacuation. These issues were discussed with Con Edison management who acknowledged the inspector's concern. At the end of the inspection, plant condition changes resulted in less urgency for containment closure.

Alternate Safe Shutdown Pressurizer Level Instrument (LI-3101)

During a containment tour, the inspector observed that the alternate safe shutdown pressurizer level instrument (LI-3101) indicated five percent. Reactor coolant system water level at the time was 6 inches below the pressurizer. The remaining pressurizer level instruments indicated zero percent. LI-3101 was indicating inaccurately. This level instrument would be used during a reactor shutdown if the control room was inaccessible. The pressurizer was pressurized with nitrogen at the time. A nitrogen bubble in the reference leg of the level instrument would cause the indicator to read high. Procedure SOP 1.1 did not provide guidance to vent this transmitter. Condition Report 200001596 was for this deficiency. Con Edison initiated actions to assure the instrument was made operable when required.

c. Conclusions

Con Edison had adequate controls for placing two reactor coolant pumps on the backseat, and adequate plans for containment closure in the event of a loss of reactor cooling. An operating procedure lacked guidance to assure a safe shutdown instrument was properly vented.

O1.2 Excessive Core Differential Temperature (NCV 2000-003-01)

a. Inspection Scope (71707)

The purpose of this inspection was to review the licensee control of the reactor temperatures during shutdown operations.

b. Observation and Findings

NRC inspection 05000247/2000002 noted that the reactor differential temperature (delta-T) reached 91 degrees F during plant cooldown activities following the February

15, 2000 Steam Generator Tube Failure. The reactor delta-T is defined to be the difference between the average reactor coolant system (RCS) wide range hot leg and cold leg temperatures. Procedure SOP 4.2.1 (Limitation 2.7 and step 4.1.2) directs the operator to limit delta-T to 72 degree F. The delta-T exceeded and then returned below the limit as the operators realigned the RHR system by removing one residual heat removal (RHR) heat exchanger from service. The maximum core delta-T occurred at 2:45 a.m. on February 17 when the four cold leg temperatures were 105, 105, 102 and 116 degrees F, respectively, and the four hot leg temperatures were 192, 192, 191 and 216 degrees F, respectively. The failure to follow SOP 4.2.1 during RHR system operation is considered a non-cited violation of NRC requirements. This issue is in the corrective action program as condition report (CR) 200001681. (NCV 05000247/2000-003-01)

The maximum core delta-T was established for the reactor vessel internals (reference 1988 safety evaluation SECL 88-612B) to limit lateral deflection of the baffle-former and baffle-barrel bolts to 0.02 inches. Con Edison contacted the reactor vendor to perform an evaluation of the consequences of exceeding the delta-T limit. Con Edison required that the evaluations be resolved prior to plant operation above 200 degrees F. Licensee action on this item continued at the end of the inspection period.

c. Conclusions

The operators failed to control RCS differential temperature within limits during RHR system operation. The failure to follow SOP 4.2.1 was a non-cited violation of NRC requirements. Licensee actions continued at the end of the inspection period to evaluate the impact on the baffle-former and baffle-barrel bolts in the reactor vessel internals, and to resolve this matter prior to plant restart.

a. Inspection Scope

The inspector reviewed the licensee response to a loss of power and air supply to the steam generator nozzle dams.

b. Observations and Findings

On March 2, 2000 with the unit in cold shutdown and reactor coolant system level six inches above the reactor vessel flange, operators were alerted by control room annunciators that electrical power and compressed air supply was lost to the steam generator nozzle dams. The inspector reviewed the cause of the momentary loss of support systems to the nozzle dams, the impact on reactor coolant system inventory, and the performance issues associated with the recovery of the support systems. The nozzle dam air supply was lost when a worker tripped on and inadvertently disconnected a temporary 480 volt power supply.

The operator appropriately followed annunciator response procedures to confirm there was no loss of reactor coolant system inventory, and initiated actions to troubleshoot the loss of power. The power loss was investigated by nuclear plant operators, the watch engineer, and members of the work control center. Troubleshooting activities originating from the work control center were accomplished without communicating with the control room. The control room personnel were not initially notified of the cause for the loss of power and re-energized the temporary 480 volt power supply. Poor coordination between the containment and the control room resulted in a potential personnel safety issue. Maintenance personnel were repairing the 480 volt cable when the operator energized the temporary 480 volt supply. No personnel injuries occurred. Con Edison documented these discrepancies in condition reports 200001433 and 200001441. Plant workers failed to follow the requirements in SAO-105, “Work Permits”, intended to assure personnel safety while working on plant equipment. This specific problem is considered to be a minor violation and is not subject to formal enforcement action.

No adverse consequences occurred when the air supply and electrical power to the nozzle dam monitoring panels were lost for approximately one hour and ten minutes. Con Edison confirmed no reactor coolant system leakage into the generator plenums and backup nitrogen supplied the nozzle dam seals. The steam generator nozzle dams provide a barrier between the reactor coolant system and the steam generator plenum to allow for installation of equipment to inspect steam generator tubes.

The operations response to the loss of power and air was complicated by inadequate descriptions for several temporary facility changes (TFCs). Six TFCs provided temporary electrical power and air supply to the containment; however, none of the TFCs were evaluated to minimize the consequence on a loss of electrical power. The TFC associated with electrical power were from a Unit 1 13.8 kilovolt supply breaker. Further, electrical protective features (i.e., ground fault) only existed at the one supply breaker.

The inspector observed that operation of the redundant electrical and diesel air compressors was not understood by control room personnel. The building and grounds personnel were responsible for maintenance and operation of the compressors. Con Edison documented this failure in condition report (CR) 200001435. The CR was prioritized as a significance level 2 out of a four tier priority system (with level 1 most significant). The inspector confirmed that corrective actions adequately addressed the above deficiencies. Con Edison established a special team to review the causes for industrial safety events.

c. Conclusions

The operators promptly responded to the loss of power to the steam generator nozzle dams. The nozzle dam normal air supply was lost; however, no loss of reactor coolant system inventory occurred, and no monitoring existed for the nozzle dams for approximately one hour. Con Edison failed to control and integrate several temporary facility changes for the nozzle dam support systems. Inadequate coordination between operators and workers resulted in a near miss for a significant injury.

O7 Quality Assurance in Operations

O7.1 Nuclear Facilities Safety Committee

a. Inspection Scope

The inspector observed activities of the Nuclear Facilities Safety Committee (NFSC) on March 15, 2000.

b. Observations and Findings

The committee fulfilled its responsibilities in accordance with the technical specifications. The primary agenda topic included performance observations during the February 15, 2000 steam generator tube leak. The specific agenda items included event chronology, equipment response, and emergency plan execution. Plant management presentations to the Nuclear Facilities Safety Committee were incomplete. For example, discussions on the use 10 CFR 50.54 (X) implementation of emergency operating procedure changes, and delays on establishment of normal pressurizer spray were not initially presented to the NFSC by operations management.

The NFSC raised appropriate questions and appeared to be adequately briefed on the event and performance insights. The NFSC questioned the adequacy of the operator work around process, the long-standing degradation of the steam supply valve to the air ejectors, and questioned whether recurrent problems occurred in emergency planning given recent performance observations.

c. Conclusions

Plant management presentations to the Nuclear Facilities Safety Committee were incomplete. However, the committee members appeared well prepared and provided good discussions on the February 15 steam generator tube leak event.

O7.2 Steam Generator Tube Leak Root Cause (SL-1) Evaluation

a. Inspection Scope (93703)

The inspector reviewed the licensee actions to evaluate the February 15, 2000, steam generator tube leak event response and develop corrective actions.

b. Observations and Findings

Con Edison completed the plant trip analysis and investigation for the February 15, 2000 steam generator tube leak event response. The Significance Level 1 (SL-1) report was issued during this inspection period and described the corrective actions to address the direct and contributing causes of weaknesses in the plant response to the event. The corrective measures included actions to revise procedures, train operators, perform extent of condition reviews, enhance log keeping, repair equipment, modify plant equipment, and address weaknesses in the emergency plan and emergency plan implementation. Licensee actions continued at the end of the inspection period to complete the short and long term corrective action in the SL-1 report. NRC review of the Con Edison corrective actions continued to verify the corrective measures were timely and appropriate, and to determine whether the weaknesses addressed in NRC Report 05000247/2000002 were adequately addressed. Further NRC review will be the subject of an AIT follow-up team inspection.

c. Conclusions

Con Edison completed the investigation of the plant response to the February 15, 2000 steam generator tube leak. Corrective actions to address the causes of weaknesses in the plant response to the event were in progress at the end of the inspection period and NRC review will be the subject of an AIT follow-up team inspection. The results of the root cause investigation for the steam generator tube failure were not reviewed and are being provided by Con Edison to the NRC Office of Nuclear Reactor Regulation for review.

O8 Miscellaneous Operations Issues

O8.1 Reporting of Events

a. Inspection Scope (92703)

The inspector reviewed licensee actions to submit reports per 10CFR 50.72.

b. Observations and Findings

On March 3, 2000, Con Edison retracted an event previously reported as a design deficiency that resulted in plant operation outside the design basis for the auxiliary feedwater system (reference Event No. 36660). Con Edison reviewed the design basis calculation for the system, FIX-00030-02, and found that it contained excess conservatism for nitrogen usage requirements for the assumed number of valve operations and leakages. After revising the calculation to include more realistic assumptions, Con Edison determined that the auxiliary feedwater system met the design basis requirements with adequate margins. The inspector reviewed the revised calculation and identified no discrepancies. The licensee had an adequate basis to retract the event.

c. Conclusions

The reporting of events per 10 CFR 50.72 was appropriate.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 Maintenance Observations

a. Inspection Scope (62707)

The inspectors reviewed selected maintenance work activities and supporting work documentation. Activities were selected for systems, structures, or components in the scope of the maintenance rule.

b. Observations and Findings

23 Reactor Coolant Pump Motor Lift

On March 7, 2000, the inspector observed the lifting of the 23 Reactor Coolant Pump motor. This motor was removed to perform lower bearing and pump seal inspections. The lift followed System Operating Procedure (SOP) 29.8.1 “Polar Crane Operation.” The heavy lift was performed in accordance with this procedure, however, the procedure was not referenced during the actual lift. Station Administrative Order (SAO) 133 “Procedure, Technical Specifications and Licensee Adherence and Use Policy” requires each step of a continuous use procedure to be read prior to performance. SOP 29.8.1 is a continuous use procedure; however it was not observed to be in visible use. This concern was discussed with an outage manager, who assured the inspector that procedures were used. This specific failure to follow procedures on procedure use had minor significance. This issue is being treated as a non-cited violation consistent with section IV of the NRC enforcement policy.

The administrative controls for heavy loads were followed during the lift. However, the inspector determined through interviews that not all personnel associated with the activity were familiar with the administrative controls for heavy loads. This was discussed with the licensee outage managers. The licensee conducted briefings to assure work crews were familiar with the controls for heavy loads.

NP-00-14381 22 Steam Generator Hillside Port Removal and Restoration

On March 6, 2000, the inspector observed the attempt to remove the 22 steam generator hillside port to allow an inspection of the upper support plate. The removal was not successful because one bolt could not be removed. The bolt was later cut and drilled out to allow for the inspection. On March 8, 2000, the inspector observed the restoration of the hillside port. The appropriate portion of the procedure was present in the field. The procedure called for a specific torque sequence, however, that torque sequence was not followed. The opposite torque sequence was followed. This did not impact the effectiveness of the system restoration.

NP-00-14380 23 Steam Generator Hillside Port Removal and Restoration

On March 6, 2000, the inspector observed the removal of the 23 steam generator hillside port. The hillside port was removed consistent with the procedure. On March 8, 2000, the inspector observed the restoration of the 23 steam generator hillside port. The hillside port was removed and restored properly.

M1.2 Surveillance Observations

a. Inspection Scope (61726)

The inspector reviewed selected surveillance activities and supporting documentation. Activities were selected for systems, structures, or components in the scope of the maintenance rule.

b. Observations and Findings

WO 99-12391. Turbine Building Crane Load Test

The inspector reviewed licensee preparations to load test the turbine building crane by lifting 66,000 gallons of water in water bags. The inspector raised a question on the consequences of a failure and the impact on shutdown risk and of internal flooding of the safety-related 480 volt switchgear room. Con Edison stopped load test preparations and initiated condition report 200001553. The load test was conducted following a major overhaul on the main and auxiliary hoist gearboxes, brakes and hook blocks.

Con Edison's corrective actions included an evaluation to determine whether an unreviewed safety question existed (safety evaluation SE-00-188-PR). The safety evaluation adequately documented a basis that an unreviewed safety question did not exist. Con Edison performed two engineering calculations. The first calculation (FMX-00144) concluded that the use temporary dams would be sufficient to prevent impact on the 480 volt buses, service water pump power cabling, and 6.9 kilovolt load center. The second calculation (FMX-00145) determined the maximum water height at the temporary dams. The inspector verified that the temporary dams installed to protect safety-related equipment in the event that the water bags failed were consistent with the engineering calculations. The turbine load test was performed after contingency actions were taken to preclude impact on safety-related equipment.

Incomplete planning occurred for the load test since the potential failure of the load and impact on safe shutdown equipment was not evaluated. Corrective actions were appropriate and the load test was performed successfully.

M1.3 Conclusions for Maintenance and Surveillance

Maintenance activities were satisfactorily completed. The conduct of surveillance tests during the period was acceptable. Maintenance and test activities were not consistently performed in accordance with expectations and administrative controls. The initial evaluations in preparation for a turbine load test did not completely consider shutdown risk.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Containment Liner Degradation

a. Inspection Scope (62707)

The inspector reviewed licensee actions to investigate and evaluate degradation in the containment liner.

b. Observations and Findings

Con Edison completed inspections of the containment liner during the outage to meet the requirements of ASME Section XI, Subsection IWE. The inspections identified degradation in a caulk seal installed in the joint between the insulated liner wall and the containment floor at the 46 foot elevation. The degraded seal allowed water to seep behind the insulation and contact the carbon steel liner. Further investigations in areas behind the degraded seal identified corrosion and wall loss in the liner at and below the floor level. The zinc-based paint and epoxy was missing in some locations with extensive wall loss. The licensee expanded the inspections and conducted ultrasonic (UT) measurements to identify the scope of the corrosion and the extent of the degradation. The results were summarized on Drawing DMD 322097-AA, Revision 0, dated March 27, 2000.

The corrosion occurred because moisture barrier seals were missing or deteriorated, which allowed extended wetting of the liner with borated water. A possible source was water leakage onto the 46 ft elevation floor from sources such as the accumulator tanks. Another source of liner wetting was from past events in which Zone 10 of the weld channel and penetration pressurization system (WCPPS) became flooded. Zone 10 was inadvertently flooded early in plant life while filling the reactor cavity. Zone 10 was also hydrostatically tested in 1995 in an attempt to identify and repair leakage within the zone. Zone 10 has since been retired in place as allowed by Technical Specification 3.3.D.2.c because of leakage that was deemed not repairable. The most significant liner degradation was found to exist at inspection port #9, located behind accumulators 21 and 22, and just above a weld channel located about 6 inches below the floor.

The licensee evaluated the liner conditions and determined that the liner remained acceptable to function as a barrier to leakage. The UT measurements showed the depth of the defects; the minimum wall thickness in the nominal one-half (½) inch liner was 0.360 inches. Engineering evaluations (Raytheon Report 91450.044-S-001) showed that the minimum wall thickness was 0.34 inches for buckling loads (applicable for initial construction only), and that a wall thicknesses as low as 0.25 inches may be allowable for strain in local areas. Licensee evaluations continued at the end of the inspection period to analyze the minimum allowable wall thickness.

The licensee entered this issue in the corrective actions system (reference CRs 200001209, 200001731, 200001652 and 200002024). Corrective actions continued at the end of the inspection period to restore damaged moisture seals and repair concrete, restore liner protection in the area of greatest corrosion, and plan for future inspections to monitor the rate of corrosion. NRC review of the licensee corrective actions continued at the end of the inspection period.

c. Conclusions

The containment liner became corroded due to prolonged contact with borated water in areas where moisture barriers were degraded. Con Edison actions continued to investigate and repair liner degradation, and to assure that margins to design limits were maintained.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 Reactor Coolant Pump Oil Collection System (NCV 2000-003-02)

a. Inspection Scope (92903)

The inspection reviewed Con Edison’s actions in response to low oil level alarms on the upper and lower oil reservoirs for the 23 reactor coolant pump motor and acceptability of the oil collection system as required by 10 CFR 50 Appendix R section III. O.

b. Observation and Findings

On February 15, 2000 at approximately 10:40 p.m., the operators secured the 23 reactor coolant pump (RCP), as part of the activities in response to the steam generator tube failure. Two and one half hours later low oil level alarms were annunciated in the control room for both the upper and lower oil reservoir. The operations watch engineer inspected the 23 RCP motor and found oil on the entire motor frame, and burnt oil residue on the pump main flange, studs and mirror insulation. Condition report (CR) 200001152 documented these observations. The watch engineer estimated based upon local level indications that approximately 4% (8 gallons) of the oil inventory had leaked out of the motor. Con Edison concluded that the oil alarms occurred when oil leaked from the motor on shutdown and due to oil contraction on motor cool down. No smoke or smoldering was observed due to the leaking oil.

The inspector walked down the RCP motor oil collection system using applicable Con Edison design drawings. The oil collection system was designed as depicted on the drawings. The inspector verified that no oil addition and no change in the oil collection tank inventory was identified during the last four months of pump operation.

During the outage, Con Edison performed preventative maintenance on the 23 RCP motor, initiated work orders to remove and or inspect reactor coolant system insulation on both the 21 and 24 cold legs, and initiated a plant modification to address potential external oil leakage not collected by the oil collection system for all RCPs. Con Edison believes the primary source of oil leakage is from the motor cooling air vents. Specifically, oil mist goes past the flywheel oil seal into the vent pipe. The misted oil travels down the exterior side of the motor and is picked up by the motor cooling air vents and sprayed into surrounding areas. Plant modification FPX-00-12334-F was being developed to capture the oil leaks from under the motor flywheel cover and redirect it into the existing oil collection system. The inspector walked down the proposed modification with cognizant design engineering personnel.

During review of this condition, the inspector learned that another long-standing vulnerability existed on the oil collection system. CR 199801646 documented that three of four RCP motor lower reservoir drain lines with flanged connections were not enclosed by the oil collection system. Further, a Quality Assurance Fire Protection Audit in 1998 (98-07-A) documented that Con Edison inappropriately dispositioned this observation. Con Edison concluded that this flanged connection did not need to be enclosed. Enclosure of the lower oil reservoir drain connections for the RCP motors is now planned as part of modification FPX-00-12334-F.

10 CFR 50 Appendix R, Section III.O., requires, in part, that the oil collection system shall be capable of collecting lube oil from all potential pressurized and unpressurized leakage sites in the reactor coolant pump lube oil systems. Leakage points to be protected shall include, in part, flanged connections on oil lines and lube oil reservoirs where such features exist on the reactor coolant pumps. NRC safety evaluation dated October 16, 1984, restated NRC's approval of the reactor coolant oil collection system and approval of an exemption that the holding tanks would not hold the entire lube oil system inventory from the four RCPs. The failure to collect leakage from the vent pipe and the lower oil reservoir drain connections on three RCP motors is considered a violation of 10 CFR 50 Appendix R, Section III. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC enforcement Policy (64 FR 61142, November 9, 1999), and this violation proposed corrective actions are documented in the licensee corrective action program as CR 200001152 and CR 199801646. **(NCV 05000247/2000-003-02)**.

c. Conclusion

The failure to collect leakage from the vent pipe and the lower oil reservoir drain connections on three RCP motors is considered a violation of 10 CFR 50 Appendix R, Section III. This Severity Level IV violation is being treated as a Non-Cited Violation. A long-standing deficiency in the oil collection system had gone uncorrected.

E2.2 Inoperable Isolation Valve Seal Water System (EEI 2000-003-03)

a. Inspection Scope

The inspection involved a review of the deficiencies documented in NRC report 500247/200002 regarding the isolation valve seal water system (IVSWS).

b. Observations and Findings

Con Edison failed to recognize a long-standing difference between the plant design and IVSWS licensing basis. This failure resulted in less than adequate corrective actions for deficiencies in IVSWS performance.

Licensing Basis

Technical specification (TS) 3.3.c. requires the IVSWS to be operable above cold shutdown with the tank maintained greater than 52 psig and a minimum of 144 gallons of seal water. The IVSWS supplies seal water to the containment isolation valves to minimize containment leakage following loss of coolant accidents. The system injects seal water at a pressure higher than the containment design pressure of 47 psig, so that leakage will be from the seal water system into the containment. The system operates automatically for lines communicating with containment atmosphere. IVSWS is an engineered safeguards system and is initiated on a phase A containment isolation signal.

The Updated Final Safety Analysis Report (UFSAR) Section 6.5.1 states that the design basis for the IVSWS is to ensure the effectiveness of the containment isolation valves during any condition which requires containment isolation. The UFSAR states that no single failure in the IVSW system would prevent it from performing the design function. The IVSW tank inventory is sufficient to ensure a 24 hour supply of seal water to all containment isolation valves, with an assumed worst case isolation valve seat leakage. The IVSWS tank volume is sufficient for approximately 2.5 days of operation at design seal water flow rates before makeup is required.

IVSWS Design and Performance

As designed, the IVSWS would operate successfully to assist containment isolation only after a large break LOCA in which a phase B isolation follows immediately after the phase A isolation. The present design does not assure IVSWS operability under all licensing basis conditions for which it could be called upon to function, specifically for the broader spectrum of postulated breaks in the plant licensing basis, and in particular, for those break sizes where the phase B isolation is delayed after the start of the event.

The IVSWS response during past events involving a phase A isolation was to drain the IVSWS tank into the component cooling water system and decrease system pressure below that needed to seal the isolation valves serviced by the seal water. A Phase A isolation signal does not close all containment penetrations, and results in a loss of seal water to open penetrations and the inability to restore the seal water. Pending the completion of manual actions to refill the IVSWS tank and repressurize the system, IVSWS was not capable of performing its design function.

On February 15, 2000, following a phase A containment isolation signal, operators found that the IVSWS tank was empty and could not refill the tank. The IVSWS tank drained through open containment isolation valves in penetrations that remain open by design until a phase B signal occurs (e.g., reactor coolant pump thermal barrier return line). The operators declared the system inoperable and entered TS 3.0.1.

On November 24, 1998, Con Edison identified discrepancies between the IVSWS design as described in UFSAR Section 6.5.2.1 and design documents (CR 199810169). Calculation PGI-00333-00 was prepared to demonstrate per UFSAR 6.5.2.1 that the IVSWS water inventory was sufficient to supply seal water for 24 hours following a design basis accident. CR 199810169 documented that several UFSAR assumptions used in the calculation were unverified, and that differences existed between the calculation and a surveillance procedure used to assure system leakage was within limits. An operability determination on November 24, 1998 concluded that IVSWS was operable. On January 26, 2000, design engineering concluded that the IVSWS tank would empty in about 85 minutes through valves that remain open following a phase A isolation signal, and questioned whether the operator could add water to the IVSWS tank prior to it being emptied. However, no further operability review was performed after this engineering issue was raised.

On June 2, 1997, the IVSWS did not perform as described in the UFSAR following a phase A containment isolation signal at power (LER 05000247/1997-010). The IVSWS tank lost both level and pressure because of leakage through a containment isolation valve. The valve leakage was not identified previously due to insufficient testing. The corrective actions were to revise the IVSWS test and for the operators to verify system performance within the emergency operating procedures (E-0, step 44). These actions were overly narrow and failed to address the basic design problem that allows the seal water to be depleted prior to the 24 hours assumed in the design basis calculation.

Safety and Regulatory Significance

10 CFR 50, Appendix B, Criterion III, “Design Control,” in part, requires that measures shall be established to assure that applicable regulatory requirements as specified in the license application (UFSAR) are correctly translated into specifications, drawings, procedures, and instructions (i.e., the plant design). The IVSWS design basis as documented in UFSAR sections 6.5, 14.3.6.1, and Table 14.3-48 is to minimize leakage following any event requiring containment isolation. However, the plant design as specified in drawings and procedures did not incorporate the regulatory requirements in the UFSAR to assure IVSWS functions following both a phase A and phase B isolation.

Following the safety injection and phase A isolation on February 15, 2000, the operator declared the IVSWS inoperable when the tank level and pressure could not be restored per emergency operating procedure (EOP) E-0, Reactor Trip/Safety Injection, step 44, to the TS limits. The operator action was required much sooner than the 24 hours assumed in UFSAR Section 6.5. On February 15 and during previous occasions, the IVSWS lost functional capability after a phase A isolation under conditions it was assumed to remain operable as described in the licensing basis. The IVSWS has not been operable as described in UFSAR Section 6.5 or TS 3.3.C since initial plant startup. The failure to ensure the licensing basis of the IVSW system was maintained during conditions for which it was required to perform its intended safety function is a violation **(EEI 05000247/2000-03-03)**.

The technical specification bases and the UFSAR state that no credit is taken for IVSWS operation in the calculation to show post-accident offsite doses are well below the limits in 10 CFR Part 100. The original NRC safety evaluation (November 20, 1970) concluded that IVSWS provides an additional means of reducing leakage following a loss-of-coolant accident. UFSAR Table 14.3-48 provides a summary of offsite exposure calculations for loss of coolant accidents with leakage terminated in one minute by isolation valve seal water. The UFSAR shows a reduction in public dose with successful IVSWS operation.

c. Conclusions

Con Edison did not recognize a long-standing difference between the design and licensing basis for the isolation valve seal water system. Despite several past events and a design basis verification program which highlighted IVSWS performance issues, Con Edison failed to correct a basic design deficiency and assure that the licensing basis was met. Operability evaluations were less than adequate and corrective actions were narrow and untimely. The failure to assure regulatory requirements were correctly translated into specifications, drawings and procedures was an apparent violation.

E2.3 Steam Generator Examinations

a. Inspection Scope (61725)

The purpose of the inspection was to review the Indian Point Unit 2 Steam Generator eddy current testing program and the test results, in support of an ongoing review by the NRC Office of Nuclear Reactor Regulation of steam generator tube integrity at Unit 2. The summary provided herein reflects the program status as of March 27, 2000.

b. Observations and Findings

Summary of Steam Generator Activities

Con Edison initiated examinations of the structural integrity of the in-service tubes in all four steam generators and conducted additional activities to evaluate the steam generators. The evaluations on the secondary side included pressure tests, flow slot examinations, hillside port inspections, and sludge lancing. The evaluations and repairs on the primary side included eddy current examinations using bobbin and rotating pancake (RPC) probes, special interest plus point examinations, high frequency plus point examinations, tube re-roll, tube plug replacement and repair, measurement of denting, ultrasonic testing, and in-situ testing. Con Edison modified steam generators #21 and #24 per FMX-00-12323-M to add a 2 inch inspection port in the transition cone and a 1-1/2 inch observation port in the tube bundle wrapper. The modification was supported by safety evaluation SECL-89-1131A, Revision 1, with the determination that no unreviewed safety question was created. The modification allowed inspection of the tubes and supports above the #6 tube support plate. Con Edison planned to replace the Inconel 600 mechanical tube plugs on the cold leg side of the #21 and #24 steam generators.

Con Edison identified an axial crack in the U-bend region of the tube at Row 2 Column 5 in the #24 steam generator, which was the direct cause of the February 15 event. The root cause of the tube failure is being determined by Con Edison and will be submitted to the NRC Office of Nuclear Reactor Regulation. Three additional Row 2 tubes in the #24 Steam Generator (SG) were also found to have indications: row 2, column 4; row 2, column 71; and row 2, column 74. Con Edison planned to conduct in-situ hydrostatic testing of these tube in accordance with industry guidelines. The results of the eddy current program as of March 23, 2000 indicated there were defects in greater than 1 percent of the tubes inspected in the #21 and #24 steam generators. The majority of the defects were at the tube support plate intersections and in Row 2 u-bends. In accordance with Technical Specification 4.13, the defects resulted in a Classification of C-3 for two steam generators. Con Edison reported this to the NRC per 10 CFR 50.72(b)(2)(i) as the plant being in a degraded condition. Con Edison also reported this event as licensee event report LER 2000-003. In accordance with Technical Specification Table 4.13-1, Con Edison must obtain NRC approval prior to plant restart. Con Edison actions were in progress to complete a root cause evaluation of the tube failure, and to complete a Condition Monitoring and Operational Assessment. The review of the steam generator examination program and results by the NRC Office of Nuclear Reactor Regulation continued at the end of the inspection period.

Steam Generator Inspection Results

The inspector observed SG tube testing and data analysis from the eddy current test program. Con Edison identified three tube leaks during a hydrostatic test of SG 22. The leaks were discovered in Row 44, Column 42; Row 45, Column 39; and Row 45, Column 44. The leaks were very slow: about 1 drop per 30 minutes for two tubes and 1 drop per 60 minutes for the third. The potential significance of the findings was that the eddy current testing (ECT) examinations during this current outage had not identified defects in these tubes and followup inspections were planned. The ECT data for one of the tubes had one finding categorized as a non qualified indication (NQI) which had not yet been profiled with a RPC probe. Further licensee review of the ECT data for these tubes, as well as an engineering evaluation of the hydro results, continued at the end of the inspection. The location of the leak was the same on all three tubes and was at the tube to tube sheet filet weld. This is the same issue described in NRC Information Notice 98-27. The leaks were located using the mid-range, +Point probe at 300 KHz. This was identified using the RPC probe and was not seen with the Cecco probe.

The licensee planned to re-examine all Row 2 and 3 tubes with high frequency probes. Additional areas planned for investigation included two indications in the u-bend area of the #24 SG that were scheduled for insitu testing. Con Edison developed a Pluggable Tube Summary to identify the minimum number of additional tubes that will be plugged for each SG. The plugging list included all Row 2 tubes and all tubes that had indications at the support plates. Not identified are those tubes that had indications in the u-bend area, sludge area and tube sheet area that will be retested using the high frequency probes. For tubes with indications in the tube sheet area, including the three tubes in the #22 SG that leaked (see C.1 above), a decision will be made as to whether to plug those tubes or re-roll the tubes.

Con Edison obtained and qualified additional high frequency probes to augment the Cecco and plus point examinations. The advantage of the high frequency probe is that the signal does not penetrate as far. Consequently, if there is crud on the outside of the tube, it will not appear as a ridge on the data analysis presentation, which makes it easier to see crack indications on the inside of the tube wall. Twelve (12) hot leg tubes in the #24 SG will be re-rolled due to indications observed in the tube to tube sheet area. The length of the re-roll varies for each tube. The original roll length is about 2.25". The tube sheet is about 22" thick. The re-rolls will vary from about 2" to 7" in length, done in about 2" increments with some overlap. The inspector reviewed the tubes being re-rolled, along with the number of tests to be conducted and the test status as of March 27, 2000.

This is an interim report on the status of the steam generator tube inspections and test results. The tubes to be plugged were not yet completely established.

c. Conclusions

The steam generator eddy current testing and analysis was in progress. The eddy current test results revealed defects which resulted in a Classification of C-3 per Technical Specification 4.13. More detailed review of steam generator inspection results is under the purview of the NRC Office of Nuclear Reactor Regulation.

E2.4 Cable Spreading Room Fire Dampers (NCV 2000003-04)

a. Inspection Scope

The inspection involved verification of Con Edison’s analysis and corrective actions following a surveillance in which some cable spreading room fire dampers failed to perform their intended safety function.

b. Observations and Findings

During the performance of PT-R36A, “Main Transformer #21 Water Deluge System,” operators were not anticipating fire damper actuation because a test switch used during the surveillance should have blocked the actuation signal to the dampers. The dampers operated during the test due to an intermittent test switch failure. Con Edison could not duplicate the switch failure, however there has been industry experience with this type of switch failure. At the end of the period, the switch was being sent out for failure analysis.

When actuated during the test, four of ten fire dampers did not function properly. Three dampers failed to operate due to mechanical interference between the fusible links and the conduit fittings. The mechanical interference was present since December, 1999. Another damper failed to fully close because the closing latch was bent.

The fire dampers failed to close because of inadequate maintenance instructions and vendor information on proper installation of conduit fittings and methods to properly unlatch closed fire dampers. Maintenance instructions in a work step list did not provide instructions on methods to preclude mechanical interference between conduit fittings and the fusible links. Con Edison had previously considered the method to unlatch a closed fire damper as “skill of the craft”, however this method resulted in mechanical deformation of the latching mechanism causing the damper not to fully close. Con Edison’s corrective actions were to remove the mechanical interference with the fusible links and to correct the latching mechanism deformation. Extent of condition reviews were adequate to identify any other fire damper vulnerabilities. None were identified.

The fire dampers provide a fire barrier to protect equipment in the cable spreading room and battery rooms in the event of a transformer fire. The dampers also serve as a barrier to ensure that the halon fire suppression system is not compromised. The failure of four of the ten dampers constituted a degradation in the fire suppression system between December 1999 through March 2000. Con Edison appropriately implemented fire watch tours per station administrative order (SAO)-703, “Fire Protection Impairment Criteria and Surveillance,” during inoperability of the fire dampers and the associated halon system for the cable spreading room.

License Condition 2.K of License DPR-26 requires Con Edison to implement and maintain all provisions of the NRC-approved fire protection program as described in the UFSAR for the facility and as approved in SERs. The SER dated October 31, 1980 documents operability requirements for the cable spreading room halon system. Those requirements are in SAO-703 addendum II. Poor maintenance instructions resulted in the faulty installation of fire dampers in December, 1999, and caused the halon suppression system to be inoperable. The failure to provide adequate maintenance instructions that resulted in an inoperable cable spreading room halon system was a violation of DPR-26 License Condition 2.K. This Severity Level IV violation is being treated as a Non-Cited Violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy (64 FR 61142, November 9, 1999). The violation was properly corrected and documented in the licensee’s corrective action program as CR 200001825. **(NCV 05000247/2000003-04)**.

NRC inspection report 05000247/1999011 described other deficiencies with the design, testing and operation of the cable spreading room halon fire suppression system. There was no causal relationship between the two events.

c. Conclusions

A lack of maintenance installation instructions contributed to the failure of cable spreading room fire dampers to fully close. The faulty dampers caused the suppression system to be degraded for approximately 3 months. The failure to maintain provisions of the NRC-approved fire protection plan as described in the UFSAR and approved NRC Safety Evaluation Report is a Non-Cited Violation.

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Contamination in Storm Drains

a. Inspection Scope (71750)

The inspector reviewed licensee actions to investigate the discovery on March 30 of low level contamination in the storm drains (reference Condition Report 20002197). This matter was reviewed with the assistance of NRC Region I Health Physics Personnel.

b. Observations and Findings

Trace concentrations of Cs-137 and Co-60 (in the range of E-5 to E-6 uCi/gram) were found in the storm drain silt and debris in the vicinity of the Unit I Chemical Service Building. All water samples were clean; only silt and debris had radioactivity above background. No short-lived activation or fission products were identified, such as Cs-134, which indicated that the radioactivity was not recently produced within the reactor, and thus was not associated with the Unit 2 steam generator tube leak event.

The storm drains are normally sampled annually by taking small sample volumes. While cleaning out all dirt and debris to facilitate drain flow, Con Edison took much larger samples and counted longer (about 8 hours). Consequently, the increased sensitivity resulted in the positive identification of activity. Con Edison confirmed that about half of the 13 drains had detectable activity when sampled and analyzed in this manner. The storm drains connect to the Unit 3 system before discharge to the river. No activity was identified in any drain that flowed into Unit 3, indicating that no activity left the Unit 1 site through this pathway.

Licensee actions continued at the end of the inspection period to clean debris from the storm drain system. Contaminated debris will be handled as radioactive material and disposed of accordingly. Further, Con Edison will investigate if there is an on-site source, or if this material originated from previous spills or leakage events on-site. There was no radiological dose consequence associated with this event. Con Edison took proper actions to resolve the immediate concerns and investigate the cause. The issue was entered into the corrective action system as Condition Report 20002197.

c. Conclusions

Con Edison appropriately responded to the discovery of trace amounts of contamination in the Unit 1 storm drains and took proper actions to resolve the condition and to investigate the cause. The material was not associated with the Unit 2 steam generator event or any recent plant activities, and there was no radiological dose consequence from the contamination.

X1 Exit Meeting Summary

The resident inspector presented the inspection results to Con Edison's management at an exit meeting on April 6, 2000. The inspectors were not informed by Con Edison that any of the issues discussed at the exit or materials examined during the inspection should be considered proprietary

ATTACHMENT 1

INSPECTION PROCEDURES USED

37551	Onsite Engineering
40500	Effectiveness of Licensee Process to Identify, Resolve, and Prevent Problems
61726	Surveillance Observation
62707	Maintenance Observation
71707	Plant Operations
71750	Plant Support
92902	Followup-Maintenance
92903	Followup-Engineering
61725	Surveillance Testing and Calibration Control Program
92901	Followup-Operations
92904	Followup-Plant Support

ITEMS OPENED and CLOSED

Open

2000-03-01	NCV	Failure to Follow Procedures on Core Differential Temperature
2000-03-02	NCV	Failure to Meet Appendix R for Oil Collection System
2000-03-03	EEI	Failure to Meet IVSWS Licensing Basis
2000-03-04	NCV	Inadequate Maintenance Instructions for Fire Dampers

Closed

2000-03-01	NCV	Failure to Follow Procedures on Core Differential Temperature
2000-03-02	NCV	Failure to Meet Appendix R for Oil Collection System
2000-03-04	NCV	Inadequate Maintenance Instructions for Fire Dampers

LIST OF ACRONYMS USED

cc	cubic centimeters
ConEd	Con Edison
CR	condition report
ECT	eddy current testing
EOP	emergency operating procedure
gpm	gallons per minute
IVSWS	isolation valve steam water system
LLD	low level dose
NCV	Non cited violation
NFSC	Nuclear Facilities Safety Committee
NQI	non-qualified indication
OTSG	once-thru steam generator
RCP	reactor coolant pump
RCS	reactor coolant system
REMP	radiological environmental monitoring program
RP&C	radiological protection and chemistry controls
SAO	station administrative order
SE	safety evaluation
SG	steam generator
SOP	system operating procedure
SPI	support plates
TFC	temporary facility change
TOI	temporary operating procedure
TS	technical specification
UFSAR	Updated Final Safety Analysis Report
UT	ultrasonic
WCPPS	weld channel and penetration pressurization system