UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406-1415

April 13, 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 05000247/2000-001

Dear Mr. Blind:

This letter transmits the results of safety inspections conducted by NRC inspectors at your Indian Point 2 reactor facility from January 25 through February 28, 2000. This report does not include NRC reviews of the February 15-16, 2000 event involving a steam generator tube failure. The operators declared an Alert emergency and responded to a leak from the reactor coolant system into the No. 24 steam generator on February 15, 2000. The operators implemented procedures to isolate the faulted generator and minimize the internal steam generator leak . The plant staff terminated from the Alert on February 16 after placing the plant in cold shutdown, and began inspections of the tubes in all steam generators. The NRC detailed reviews of this event include an Augmented Inspection Team (AIT) and an ongoing technical review of steam generator issues; these reviews will be the subject of future correspondence.

Weaknesses in planning and work control were evident in several outage activities. Past efforts to reduce the outstanding backlog of post-maintenance tests have not been effective. The inspectors reviewed the adequacy and implementation of the radiological control program including the access control program, external and internal exposure controls, radioactive material and contamination controls, and the as low as reasonably achievable (ALARA) program. Your program performance in these areas was found to be effective. Also reviewed were your self-identification and corrective action processes in the area of radiation protection. Your program performance in the problem identification area was effective. However, there was a backlog of identified issues requiring corrective action and resolution.

Based on the results of this inspection, the NRC has determined that a violation of NRC requirements occurred. The violation is being treated as a Non-Cited Violation (NCV), consistent with Section VII.B.1.a of the Enforcement Policy (November 9, 1999, 64 FR 61142). The NCV involved the failure to complete post-maintenance tests on safety related equipment prior to returning the components to service. If you contest this violation or the severity level of the NCV, 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. 05000247/2000-001

cc w/encl:
J. Groth, Senior Vice President - Nuclear Operations
J. Baumstark, Vice President, Nuclear Power Engineering
J. McCann, Manager, Nuclear Safety and Licensing
B. Brandenburg, Assistant General Counsel
C. Faison, Director, Nuclear Licensing, NYPA
J. Ferrick, Operations Manager
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
P. Eddy, Electric Division, Department of Public Service, State of New York
T. Rose, NFSC Secretary
F. William Valentino, President, New York State Energy Research
and Development Authority
J. Spath, Program Director, New York State Energy Research
and Development Authority
County Clerk, West Chester County Legislature
Westchester County Executive
Putnam County Executive
Rockland County Executive
Orange County Executive
T. Judson, Central NY Citizens Awareness Network
M. Elie, Citizens Awareness Network

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	05000247
License No.	DPR-26
Report No.	05000247/2000-001
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York
Dates:	January 25 through February 28, 2000
Inspectors:	William Raymond, Senior Resident Inspector Peter Habighorst, Resident Inspector Jennifer England, Resident Inspection, Indian Point 3 Peter Drysdale, Senior Resident Inspector, Indian Point 3 Raymond Lorsen, Senior Resident Inspector, Seabrook Stephen Pindale, Senior Project Engineer John McFadden, Senior Radiation Specialist Lois James, Project Engineer Gregory Smith, Senior Physical Security Inspector
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. 05000247/2000-001

This integrated inspection reviewed operations, engineering, maintenance, and plant support. The report covers a four week period of inspection by resident and regional inspectors.

Operations

The conduct of routine operations was acceptable to monitor plant status. Examples were noted where log keeping, communications and procedure use could be improved. (O1.1)

Con Edison increased monitoring of steam generator conditions when radiochemistry data and N16 radiation monitors showed an active tube leak in steam generator #24. Con Ed performance in this area will be the subject of the aforementioned AIT report. (O1.1)

Plant operators declared an Alert emergency and responded to a leak from the reactor into the #24 steam generator on February 15, 2000. The operators implemented procedures to isolate the faulted generator and minimize the internal steam generator leakage. Con Edison terminated from the Alert on February 16 after placing the plant in cold shutdown. Con Ed performance in this area will be the subject of the aforementioned AIT report. (O1.2)

The lack of clear licensee guidance resulted in an incorrect calculation of the NRC performance indicator for the availability of emergency power sources. The consequence of the error was to increase the unavailability of the 23 emergency diesel generator; however, the indicator remained within the licensee response band. (O1.3)

Con Edison properly controlled the transfer of radioactive liquid waste from the #24 steam generator to the Unit 1 waste collection tank. After the transfer, two separate and unexpected alarms occurred during planned releases as a result of incomplete flushing and removal of the 24 steam generator water from a portion of the release pathway. No regulatory limits were exceeded. This problem showed weaknesses in the control of the process. (O2.1)

Con Edison adequately controlled the risks associated with reactor operation in reduced inventory and at mid-loop in the reactor coolant system. Configuration controls and management oversight of the evolution were acceptable. (O2.2)

Indian Point 2 reviewed an RCS draindown event at another facility as described in Generic Letter (GL) 98-02, and adequate administrative controls were in place to preclude a similar drain down event. (O7.1)

The Nuclear Facilities Safety Committee fulfilled its responsibilities in accordance with technical specifications. (7.2)

Executive Summary (cont'd)

Maintenance

Maintenance and surveillance observed during the period were acceptable. Insufficient planning for degraded pool clarity delayed fuel handling activities. (M1.1)

The work control process was revised to address lessons learned operating experience and to improve efficiency. The training on the revised process changes was adequate. (M1.2)

A large backlog of post-maintenance tests (PMTs) for safety-related equipment existed. The inspector did not identify any outstanding operability concerns for the untested equipment. Con Edison's proposed short-term actions were appropriate to reduce the backlog. Previous Con Edison actions on the PMT process had been ineffective to preclude a large backlog. A NCV was identified for this issue. (M2.1)

Engineering

Con Edison's actions to address boraflex degradation in the spent fuel racks are unresolved pending the completion of an assessment of the boraflex and the impact on spent fuel rack licensing and design basis, and subsequent review by the NRC. (E2.1)

The LERs reviewed met the reporting requirements of 10 CFR 50.73. For some reports, the inspector noted problems in meeting the administrative requirements to process the LERs and track commitments to issue supplemental reports. (E8.1)

Plant Support

ConEd implemented effective radiological controls. The radiation work permit program was adequately implemented. Personnel occupational exposure was maintained within applicable regulatory limits and as low as reasonably achievable (ALARA). Access controls to radiologically controlled areas were effective, and appropriate occupational exposure monitoring devices were provided and used. No violations were identified. (R1.1)

ConEd implemented overall effective surveys, monitoring, and control of radioactive materials and contamination. Health Physics technicians properly conducted and documented survey results. In general, radiological housekeeping conditions were noted to be good with some exceptions on the Unit 1 side. The number of personnel contaminations was being tracked and trended. The radiological surveys, monitoring, and controls were implemented with calibrated and properly used devices. No violations were identified. (R1.2)

The ALARA program was effective. The program was being implemented in accordance with established procedures. Person-rem goals were established, tracked, and used to gauge the overall effectiveness of the station's program in this area. Planning and preparation for the 2000 outage was in progress. (R1.3)

Con Edison anticipated changing radiological conditions in the facility when inducing a reactor coolant system crud burst; however, the magnitude of the radiation doses impacted access to

vital plant areas. Industry guidance predicted this phenomenon due to plant configuration; however, this guidance was not fully considered in the pre-planning for the crud burst. (R1.4)

Con Ed's self-identification processes in the area of radiation protection were effective; however, there was a backlog of identified issues requiring corrective action and resolution. Audit and surveillance reports, radiological assessor reports, self-assessments, and the corrective action program were effective in identifying improvement opportunities. An action plan was initiated to improve the timeliness in processing and resolving identified problems and issues. No violations were identified. (R7)

An in-office review of changes to the Physical Security Plan, submitted to the NRC in accordance with 10 CFR 50.54(p) was conducted. Based on the licensee's determination that the changes did not decrease the overall effectiveness of the security plan and after limited review, no NRC approval was determined to be required. Implementation of these changes will be subject to future inspection. (S3)

TABLE OF CONTENTS

EXECUTIVE SUMMARY ii

TABLE OF CONTENTS v

I. OPERATIONS 1
- O1 Conduct of Operations 1
  - O1.1 Operational Safety Verification 1
  - O1.2 Response to Reactor Trip - Steam Generator Tube Leak 2
  - O1.3 NRC Performance Indicator 3
- O2 Operational Status of Facilities and Equipment 4
  - O2.1 Transfer of 24 Steam Generator Water 4
  - O2.2 Reduced Inventory and Mid-Loop Operation 6
- O7 Quality Assurance in Operations 8
  - O7.1 Response to Generic Letter (GL) 98-02 8
  - O7.2 Nuclear Facilities Safety Committee 9

II. MAINTENANCE 10
- M1 Conduct of Maintenance and Surveillance 10
  - M1.1 Maintenance and Surveillance Observations 10
  - M1.2 Work Control Process Revisions 11
- M2 Maintenance and Material Condition of Facilities and Equipment 12
  - M2.1 Post-Maintenance Test Program (NCV 05000247/2000001-01) 12
- M8 Miscellaneous Maintenance Issues 14
  - M8.1 Review of Previous Inspection Items (92902) 14

III. ENGINEERING 14
- E2 Engineering Support of Facilities and Equipment 14
  - E2.1 Spent Fuel Rack Boraflex (UNR 05000247/2000-001-02) 14
- E8 Miscellaneous Engineering Issues 15
  - E8.1 Review of Licensee Event Reports (92903) 15

IV. PLANT SUPPORT 17
- R1 Radiological Protection and Chemistry (RP&C) Controls 17
  - R1.1 Radiological Controls-External and Internal Exposure 17
  - R1.2 Radiological Controls-Radioactive Materials, Contamination, Surveys, and Monitoring 18
  - R1.3 Radiological Controls-As Low As Reasonably Achievable (ALARA) . . . 19
  - R1.4 Reactor Coolant System Crud Burst 20
- R7 Quality Assurance in RP&C Activities 21
- S3 Security Program Plans 22
- X1 Exit Meeting Summary 23
- X2 Regional Management Visit 23

ATTACHMENTS

Attachment 1 - Inspection Procedures Used
- Items Opened and Closed
- List of Acronyms Used

Report Details

I. OPERATIONS

O1 Conduct of Operations

The plant operated at full power from the start of the inspection period until February 15 when the operators tripped the reactor and declared an Alert in response to a tube failure in steam generator #24. The operators placed the plant in cold shutdown at 6:50 PM on February 16. The plant entered reduced inventory to install SG nozzle dams. The reactor remained partially drained with the vessel head installed and level at about the 69 foot 6 inch elevation while Con Edison inspected tubes in all four steam generators.

O1.1 Operational Safety Verification

a. Inspection Scope (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. The safety injection system was selected for a detailed review and system walk down. Specific observations are described below.

b. Observations and Findings

Plant Status Observations

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

The inspector observed various shift turnovers. The shift turnovers provided the crew with information about equipment status, recent problems, and scheduled maintenance activities. The shift turnovers were performed consistent with station expectations.

The conduct of routine operations was acceptable in the routine shift briefings, the monitoring of plant status, and meeting technical specification limiting conditions for operations. Examples were noted where the conduct of operations could be improved, such as in the details provided in the shift logs, communications with operations personnel and use of the current procedure to fill the reactor coolant system (RCS). These specific concerns are considered to be a minor violation and are not subject to formal enforcement action.

Trending of Primary to Secondary Leakage

Prior to the February 15 event, Con Ed monitored primary to secondary leakage as described below. Further detail on associated performance will be provided in the AIT report, 05000247/2000-002. Con Edison uses nitrogen-16 (N16) radiation monitors mounted on the main steam lines to monitor for reactor coolant leakage into the secondary side of the steam generators at levels above the N16 monitors threshold of

1 gallon per day (gpd). The leakage of primary system (reactor) water into the #24 SG increased on February 3 as indicated by the N16 monitor coming on scale for the first time this operating cycle. The leak rate into the #24 SG increased to 1.23 gpd for a few hours and then went below 1 gpd. The N16 monitor periodically came on-scale and gradually increased after February 3. Once on-scale, Con Edison used the N16 readings to trend the #24 SG leak rate. The inspectors requested on February 9 to be informed by Con Ed if the leak rate reached 5 gpd (half the first alarm limit on the N16 monitor).

The steam generator status was reviewed by the inspectors with NRC management and engineering personnel. Based on industry experience and the EPRI guidance regarding SG leak progression rate history (EPRI TR-10478-R1), the existing leak conditions were not considered to be indicative of imminent tube failure.

On February 15, the SG leak rate was reported at the daily Con Ed plant status meeting as 3 gpd coming from the #24 SG. The chemist recorded the N16 monitor readings at 7:17 p.m. on February 15 to be 3.2 gpd. At 7:30 p.m., the operators tripped the reactor and declared an Alert after the leakage into the #24 steam generator increased to about 100 gallons per minute. The NRC response to the transient is described in Section O1.2 below. The NRC detailed review of the February 15 SG tube failure will be covered in the aforementioned AIT Report.

c. Conclusions

The conduct of routine operations was acceptable to monitor plant status. Examples were noted where log keeping, communications and procedure use could be improved. Con Edison increased monitoring of steam generator conditions when radiochemistry data and N16 radiation monitors showed an active tube leak in steam generator #24. Other performance related conclusions will be the subject of the aforementioned AIT report.

O1.2 Response to Reactor Trip - Steam Generator Tube Leak

a. Inspection Scope (92703)

Using Inspection Procedure 93702, the inspector reviewed the licensee response to a reactor trip and plant transient. The aforementioned AIT report provides the NRC detailed review of this event and performance conclusions.

b. Observations and Findings

The inspector responded to the site on February 15, 2000 when Con Edison declared an Alert emergency and shutdown the reactor in response to a tube failure in the #24 steam generator (SG). The resident staff was augmented by other inspectors dispatched by NRC Region I management. The inspectors observed plant status and monitored actions by the operators and the Con Edison emergency response organization to isolate the faulted SG, minimize the release of radioactivity, cooldown the plant to below 200 degrees F, and terminate the event on February 16th.

Con Edison established a recovery organization to restore plant conditions, and initiated independent reviews to evaluate the cause of the event. Actions were completed during this inspection period to lower reactor coolant level to allow the installation of steam generator nozzle dams, reduce radiation levels in the primary side of the steam generators, and begin steam generator tube inspections. 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. Con Edison initiated actions to inspect all in-service tubes in all four steam generators.

Con Edison reviews were in progress at the conclusion of the inspection to identify the root cause of the tube crack, and identify the appropriate corrective actions. One object of the review was to determine why the tube crack deteriorated so rapidly from an operational leak rate that was far below the action levels recommended in the industry guidance for primary-to-secondary leakage, and determine what additional operational limits may be necessary to assure the operator can respond conservatively to future SG leakage.

The NRC initiated an Augmented Inspection Team (AIT) to determine the sequence of events and causal factors for significant occurrences in the sequence. The NRC Office of Nuclear Reactor Regulation initiated a review to determine the cause of the tube leak and evaluate Con Edison's steam generator inspection program.

c. Conclusions

Plant operators declared an Alert emergency and responded to a leak from the reactor into the #24 steam generator on February 15, 2000. The operators implemented procedures to isolate the faulted generator and minimize the internal steam generator leak. Con Edison terminated from the Alert on February 16 after placing the plant in cold shutdown. The aforementioned AIT report provides the NRC detailed review of this event and performance conclusions.

O1.3 NRC Performance Indicator

a. Inspection Scope (71707)

Using Inspection Procedure 71707, the inspector reviewed the performance indicator for the unavailability of the emergency power supplies. Specific observations are described below.

b. Observations and Findings

The inspector identified that Con Edison failed to properly account historical fault exposure unavailability hours for the 23 emergency diesel generator. The unavailability hours are part of NRC's new performance indicator data. Con Edison has recently provided this information to the NRC for inclusion into the revised oversight program. Fault exposure unavailability hours are defined as the estimated hours that a train was in an undetected failed condition. From July 2 through August 31, 1999, the 23 emergency diesel generator output breaker amptector setting was improper. This resulted in an additional 1,444 hours of unavailability. The cause of the improper over

current setting on the 23 emergency diesel generator output breaker is documented in NRC report 05000245/1999013.

Con Edison documented the error in the NRC performance indicator in condition report (CR) 200000867. When the unavailability hours were recalculated for the 23 emergency diesel generator, the train unavailability increased from 1.9 to 3.7. The indicator remained within the regulatory response band (green), and will be noted as a correction during the next quarterly submittal of performance indicators.

The error occurred because the system engineer did not fully understand the industry guidance, and Con Edison had inadequate guidance on the method to gather and calculate the NRC performance indicators. The Con Edison Quality Assurance (QA) department recently verified the performance indicator data and noted that fault exposure unavailability hours were not collected prior to August 31, 1999. The QA observations were not placed into the corrective action system until after the inspector raised this concern. The inspector did not identify any other historical conditions that could impact Con Edison's accounting of fault exposure unavailability hours for the emergency diesel generators.

c. Conclusions

The lack of clear guidance resulted in an incorrect calculation of the NRC performance indicator for the availability of emergency power sources. The consequence of the error was to increase the unavailability of the 23 emergency diesel generator; however, the indicator remained within the licensee response band.

O2 Operational Status of Facilities and Equipment

O2.1 Transfer of 24 Steam Generator Water

a. Inspection Scope (71707)

During a steam generator tube leak on February 15, 2000, reactor coolant system water entered the 24 steam generator and portions of the steam and condensate systems. On February 17, 2000, Con Edison initiated actions to drain the water inventory from the 24 steam generator to the 11 waste collection tank for processing. The inspection scope included a verification of the system alignment, actions taken by Con Edison to monitor for leakage, and the quality of radiological support.

b. Observations and Findings

The inspector observed the station nuclear safety committee (SNSC) review changes to system operating procedures (SOPs) 1.11, “Steam Generator Filling and Sparging Operation,” and SOP 7.2, “Secondary Boiler Blowdown Purification System Units 2 and 3.” The two procedures provided instructions to align the 24 steam generator blowdown line through the secondary boiler blowdown purification system to the waste collection tank. SNSC questions on the procedures indicated that procedure preparations and review were incomplete. Specifically, the procedures did not consider potential lifting of a relief valve on the Unit 1 flash tank, isolation of a valve to ensure nitrogen pressurization, and a pressure limitation that would have precluded secondary pressurization.

The inspector verified the system alignment was appropriate using plant drawings and the SOPs. The inspector interviewed various nuclear plant operators who were responsible for monitoring the alignment and reporting if any system leakages were to occur. The operators were knowledgeable of the system configuration. Con Edison drained the 24 steam generator to the 11 waste collection tank between February 18 and February 20.

After draining the 24 steam generator inventory, Con Edison used portions of the drain path for a planned release of liquid from the unit 1 processing system. An unexpected radiation alarm occurred and the operator terminated the planned release. The unexpected alarm associated with this planned release was the result of incomplete flushing and removal of steam generator water from the secondary boiler blowdown purification system (SBBPS). A second attempt to release Unit 1 water using this pathway was also terminated when unexpected effluent pathway radiation alarms occurred for the same reason. No regulatory limits were exceeded during either occasion when small releases occurred. Numerous unsuccessful attempts were made to flush the system. At the end of the inspection period, Con Edison was developing methods to clear the lines to allow for proper release of radioactive inventory from the Unit 1 processing system.

c. Conclusions

Con Edison properly controlled the transfer of radioactive liquid waste from the #24 steam generator to the Unit 1 waste collection tank. After the transfer, two separate and unexpected alarms occurred during planned releases as a result of incomplete flushing and removal of the 24 steam generator water from a portion of the release pathway. No regulatory limits were exceeded. This problem showed weaknesses in the control of the process.

O2.2 Reduced Inventory and Mid-Loop Operation

a. Inspection Scope (71707)

The inspectors provided continuous observations of operator actions to reduced reactor coolant system inventory, establish conditions at the centerline of the hot leg, install nozzle dams and increase reactor coolant system inventory. The inspection included control room observations, vital support system alignments (i.e., reactor vessel level monitoring instrumentation, containment closure process, and nozzle dam monitoring instrumentation), and observations of steam generator manway removal and nozzle dam installation. The inspectors also verified implementation of Con Edison commitments to NRC Generic Letter 88-17, " Loss of Decay Heat Removal."

b. Observations and Findings

Mid-loop operation is a common evolution in support of steam generator work. Mid-loop operation causes an increase in shutdown risk because of the reduced reactor vessel inventory and resultant shorter time to boil reactor coolant in the event of a loss of decay heat removal. Reduced inventory is a reactor coolant system level three feet below the reactor vessel flange. The evolution to drain inventory from the reactor coolant system is also risk significant. The increased risk is based upon the potential to lose residual heat removal pumps (loss of suction), the short time to lose reactor coolant subcooling, the opening within the reactor coolant system, and the reliance on administrative controls instead of automatic actions to establish containment closure.

Generally, procedure quality was adequate; however, examples of incomplete detail or inconsistencies existed. In one case, the lack of detail in Temporary Operating Instruction (TOI) 265, "Draining the Reactor Coolant System with Tube Leak in 24 Steam Generator," to verify the audible alarm function for a reactor coolant system level instrument resulted in extended time in reduced inventory. The procedures also lacked sufficient guidance to transition from TOI 265 to system operating procedure (SOP) 1.1, "Reactor Coolant System Fill." Specifically, the operators did not have guidance on how frequently to monitor reactor coolant system level and what the level alarm set points would be during refill operations. This information was provided by operation's management when questioned by the operators.

The operators appropriately implemented applicable procedures. The operators controlled residual heat removal flow to prevent vortexing at the mid-loop condition, and monitored variations within the four reactor coolant system level instrument systems. When the levels channels disagreed, the drain down was appropriately suspended by the operators until the transmitters were vented.

The senior reactor operator controlling the reactor coolant system fill evolution (SOP 1.1) had the wrong procedural revision. The inspector notified the operator and the correct procedure was used thereafter. The issue was documented in Con Edison's corrective action program. The inspector confirmed that use of the incorrect procedure version did not impact the actions to fill the reactor coolant system. This specific concern is considered a minor violation and is not subject to formal enforcement action. The crew shift briefings and inter-crew coordination were adequate. An additional

reactor operator assisted the control room staff in support of the drain down evolution. The inspector observed a station stand down prior to the drain down to discuss the risk significance of the evolution, awareness of plant conditions and management reinforcement of the importance of the evolution.

Appropriate management oversight was provided during the evolution, and during entry into reduced inventory. Con Edison's President, Vice President of Operations, and Operations management were present within the control room during the evolution.

Configuration controls were appropriately maintained during the evolution. The inspector observed that the control room annunciator for "MID LOOP" did not initially annunciate when level went below the defined entry point for the reduced inventory operations of 65 feet. The alarm subsequently annunciated prior to level reaching the mid-loop condition at 62 feet. The inspector discussed the alarm with the shift personnel. The inspector concluded that operators could have questioned this sooner. The significance of the oversight was minimal since the operators used the multiple level channels and were closely monitoring reactor coolant system inventory during the drain down.

The inspector verified that the 24 steam generator (tube leak) was properly isolated to support containment closure, performed periodic alignments on the reactor coolant system level monitoring instrumentation, verified support system alignments for residual heat removal system, and confirmed operators had reviewed and were briefed on the abnormal operating instruction (AOI) 4.2.1, "Loss of Residual Heat Removal." The inspector noted that multiple reactor coolant of makeup and emergency power sources remained availability during the evolution.

The inspector reviewed the maintenance activities scheduled during the evolution and verified the work would not impact reactor coolant system inventory or key shutdown safety features. The watch engineers appropriately monitored shutdown risk and adjusted the risk values as plant conditions changed during the evolution. The inspector confirmed that commitments made to NRC Generic Letter 88-17 were verified complete and properly maintained.

c. Conclusions

Con Edison adequately controlled the risks associated with reactor operation in reduced inventory and at mid-loop in the reactor coolant system. Configuration controls and management oversight of the evolution were acceptable.

O7 Quality Assurance in Operations

O7.1 Response to Generic Letter (GL) 98-02

a. Inspection Scope (TI2515/142)

The inspectors reviewed Consolidated Edison's (ConEd) response to Generic Letter (GL) 98-02 which required licensees to determine whether their emergency core cooling systems (ECCS) were susceptible to common-cause failure due to design features such as a common pump suction header. The inspector reviewed the measures employed by ConEd to prevent such occurrences, including:

- Station Operating Procedure (SOP) 1.2, Draining the Reactor Coolant System
- SOP 4.2.1, Residual Heat Removal (RHR) System Operation
- SOP 4.2.2, Operation with Reduced Reactor Coolant System Inventory
- Station Administrative Order (SAO) 136, Management of Outages

b. Observations and Findings

In the response to GL-98-02, ConEd concluded that the Indian Point 2 (IP2) plant was susceptible to a common cause failure that could potentially result in RCS draindown similar to that described in the generic letter. ConEd further determined that existing procedural and administrative controls provided sufficient protection against an unanticipated draindown event. Valve 883 is a motor-operated valve which is normally closed, sealed and electrically deenergized at its motor control center with a caution tag. This valve provides isolation capability for the RHR return line between the discharge of the RHR pumps and the Refueling Water Storage Tank (RWST). The RHR return line is used during refueling operations to provide a draindown path from the refueling cavity to the RWST. The inadvertent opening of Valve 883 would direct the RCS water inventory into the common supply from the RWST to the ECCS pumps.

The inspectors reviewed administrative controls in place to preclude the loss of RCS inventory, including SAO 136, "Management of Outages," and SOP 4.2.1, "Residual Heat Removal System Operation." SAO 136 requires Pre-outage Shutdown Safety Issues Review to evaluate the availability of systems (temporary or installed) to support key shutdown safety functions, and to ensure alternative capability for areas of high risk to key safety functions. SOP 4.2.1 requires the completion of Check Off List 10 which, in part, ensures that Valve 883 is closed and sealed.

The inspectors determined that ConEd's response to GL 98-02 and subsequent determination that additional corrective measures were not required were acceptable. Administrative controls outlined in station procedures provided reasonable assurance that an event similar to that experienced at Wolf Creek would be prevented.

c. Conclusions

Indian Point 2 reviewed an RCS draindown event at another facility as described in Generic Letter (GL) 98-02, and adequate administrative controls were in place to preclude a similar draindown event.

O7.2 Nuclear Facilities Safety Committee

a. Inspection Scope (71707)

The inspector observed activities of the Nuclear Facilities Safety Committee (NFSC) on February 1, 2000.

b. Observations and Findings

The committee fulfilled responsibilities in accordance with technical specifications. NFSC membership changes were noted by the inspector. The changes included the replacement of the NFSC chairman, operations sub-committee chairman and radiation protection, chemistry and environmental subcommittee chairman. The membership qualifications were consistent with technical specification requirements.

The presentations to the full committee included recent quality assurance audit results, quality assurance self-assessment results, refueling outage preparations, status of corrective actions program improvements, and follow-up questions associated with the flow accelerated corrosion program for balance-of-plant piping. The quality assurance self-assessment conclusions were critical. The conclusions noted that QA was ineffective in characterizing issues and facilitating timely corrective actions. The recommended corrective actions were to provide benchmark trips to other facilities, implementation of rotational assignments with the line organization, and increase QA management review of audits. NFSC committee members appropriately raised questions of proposed corrective actions associated with ineffective characterization of performance issues. The NFSC deliberations on the above topics were probing and appropriately challenged the line organization.

c. Conclusions

The Nuclear Facilities Safety Committee fulfilled its responsibilities in accordance with technical specifications.

II. MAINTENANCE

M1 Conduct of Maintenance and Surveillance

M1.1 Maintenance and Surveillance Observations

a Inspection Scope (62707, 61726)

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

b. Observations and Findings

Adjustments of Output Frequency on 24 Static Inverter

An emergent work activity was identified on February 13, 2000 when a reactor operator questioned the voltage on the 24 vital instrument bus. The voltage was reading higher than normal (120 volts) but not out of specification. Subsequent follow-up by a nuclear plant operator noted that the IN-SYNC light was not illuminated for the vital inverter and that the output frequency was out of specification high at 60.7 hertz.

The corrective maintenance activity (work order NP-00-14094) was to adjust the output frequency of the 24 static inverter to within the acceptance criteria. The operators were directed by the control room to transfer the 120 volt bus supply from the static inverter to a backup power supply and then allow instrument and controls technicians to perform frequency adjustments. The inspector observed appropriate operator actions consistent with system operating procedures to remove the vital inverter from service. During the evolution good peer checking and communications with the control room were observed by the inspector. The inverter output frequency was appropriately adjusted to within acceptable limits.

Spent Fuel Moves and Boraflex Testing

The inspector reviewed activities under work order NP-2000-12730 to relocate spent fuel in the spent fuel pool to allow blackness testing of the spent fuel racks to verify the amount of boraflex degradation. The technical issue is discussed further in Section E2.1 of this report. Con Edison relocated 152 spent fuel bundles within the pool to support the boraflex evaluation. The inspector verified that spent fuel pool operations were completed per the controls established in the work package, which included appropriate staffing, prerequisite plant conditions, fuel handling equipment checkout and operation, foreign material exclusion control, radiological controls, and controls for special nuclear material. No inadequacies were identified. Con Edison appropriately suspended fuel movement activities when pool clarity deteriorated on February 7, and used temporary filters to remove impurities from the pool. There were delays in providing pool filtration due to insufficient planning for degraded pool clarity. Fuel handling activities were completed on February 14.

PT-Q13, Inservice Valve Testing

This surveillance test was performed on February 2, 2000, to confirm the proper operation of containment isolation valves in accordance with Technical Specification 4.2. Steam generator blow down isolation valve PCV-1216A did not meet the acceptance criteria when moved to the open position. The valve opened in 20.8 seconds instead of the required time of less than 20 seconds. The stroke time in the closed direction was acceptable, which assured that the safety function was maintained. Con Edison took appropriate actions to isolate the penetration per Technical Specification 3.6.A.3.a, evaluate the cause of the failure as a defective air regulator, and evaluate valve operability in the degraded condition. Test anomalies were properly documented and entered into the corrective action system for resolution (reference CR 20000653). The surveillance was completed satisfactorily.

PT-R-8B, Spent Fuel Pit Bridge Refueling Crane

The objective of the surveillance was to perform a dead-load test of the spent fuel pit bridge crane in accordance with technical specification 3.8.B.5. The pre-job briefing adequately discussed personnel responsibilities, communication points, and emphasis of foreign material controls when working near the spent fuel pool. The test was performed satisfactory.

c. Conclusions

Maintenance and surveillance observed during the period were acceptable. Insufficient planning for degraded pool clarity delayed fuel handling activities.

M1.2 Work Control Process Revisions

a. Inspection Scope (62703)

The inspection included confirmation of training and evaluation of changes to the work control process as documented in station administrative order (SAO)-204, “Work Control.”

b. Observations and Findings

Revision to the work control process occurred during the inspection period. The changes were actions to address lessons learned from prior implementation of the process, operating experience input, and efficiency improvements for the work control process. Major changes to the process included schedule freeze dates from four weeks to two weeks prior to implementation, actions to identify work orders that are safety significant as defined by NRC Generic Letter 91-18, “Information to Licensee’s Regarding NRC Inspection Manual Section of Degraded and Nonconforming Conditions,” expansion of responsibilities for the “Fix-It-Now” maintenance group, and improved definition of who develops either post-maintenance tests for safety related equipment, or functional check outs for non-safety related components. The inspector confirmed required training was performed prior to full implementation of the process change.

c. Conclusions

The work control process was revised to address lessons learned operating experience and to improve efficiency. The training on the revised process changes was adequate.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Post-Maintenance Test Program (NCV 05000247/2000001-01)

a. Inspection Scope (62703)

The inspector reviewed the post-maintenance testing program and backlog status.

b. Observations and Findings

There was a large total backlog of uncompleted post maintenance tests (PMTs) for equipment returned back to service. Approximately 115 PMTs were outstanding that could be accomplished during power operations and 13 PMTs required a refueling outage to accomplish. A majority of the outstanding PMTs were about 2 years old, with the oldest PMT being October, 1995.

Approximately 37% of the PMT backlog was related either to safety-significant, important-to-safety, or risk significant components. Examples of outstanding tests included the need to confirm no external leakage on a containment spray isolation valve, no leakage in a residual heat removal purification check valve (829A), and proper function on two pressurizer backup heaters.

Prior to this inspection, Quality Assurance audit 99-0-8-C, “Equipment Control and Inspections and Test,” identified that an ineffective PMT process jeopardized equipment control. The conclusion was based upon delays in completion of PMTs, and no process requirement for on-shift operations to verify scope and results of test performed.

The inspector reviewed the outstanding PMT for the safety-significant, important-to-safety, or risk significant components to determine if any outstanding operability issues exist for the untested components. The inspector review did not identify any outstanding concerns. For example, a safety injection line flow element was not leak tested, however this line is neither in-service to support routine surveillances nor aligned during a design basis event. The failure to test for external leakage on the containment spray isolation valve did not result in inoperability since the body-to-bonnet bolts were torqued as part of the maintenance.

Technical specification 6.8.1 requires, in part, that written procedures shall be implemented and maintained covering the requirements of section 5.1 and 5.3 of American National Standards Institute (ANSI) N18.7-1976 and Appendix A of Regulatory guide 1.33, revision 2. ANSI N18.7-1976 section 5.1.6.1, “Quality of Equipment,” states in part, a suitable level of confidence in such structures, systems, or components shall be attained by appropriate inspection and performance testing in accordance with applicable requirements. ANSI N 18.7-1976 section 5.2.6, in part, states that until suitable documentary evidence is available to show the equipment or material is in conformance, affected systems shall be considered to be inoperable and reliance shall not be placed on such systems to perform their intended safety function.

Con Edison station administrative order (SAO)-204, Work Control, revision 18 requires that the Test and Performance Manager is responsible for determining whether a PMT is required, development of the PMT, and review of the completed PMT to verify that testing is satisfactory. Further, the SAO-204 requires that when acceptance criteria are not met, but the equipment is determined to be operable by the shift manager, the shift manager may permit equipment operation. The shift manager shall document the basis for this decision on the PMT. Inspector review of the PMT backlog noted numerous examples where documentation did not exist to conclude that the safety related component could perform its intended safety function. The failure to complete post-maintenance testing is considered a non-cited violation of NRC requirements. This issue is in the Indian Point Unit 2 corrective action program as condition report (CR) 199903055. **(NCV 05000247/2000001-01)**

At the end of the inspection period, Con Edison initiated actions to review the outstanding PMT, develop a schedule for testing and complete the outstanding PMTs by the end of the inspection period. Outstanding PMTs were placed on the station schedule and received station focus. Due to the steam generator tube failure on February 15, the goal was not achieved at the end of the period. For newly developed PMTs, station management continued to reinforce that all testing for components identified in the technical specifications are to be fully tested prior to restoration to service.

The inspector noted that an upgrade to the PMT program was identified in the Con Edison Business Plan within both the Work Control Improvement Plan and the Test and Performance Improvement Plan. The initially planned upgraded PMT program was scheduled for October 2000. The basis for the program upgrade was previous Con Edison identification of program deficiencies. The history included a 1997 Maintenance self-assessment, independent industry evaluations of IP2 in 1998 and the Indian Point Plan for Excellence. None of the above actions precluded a large backlog of PMTs for safety-related equipment.

c. Conclusions

A large backlog of post maintenance tests (PMTs) for safety-related equipment existed. The inspector did not identify any outstanding operability concerns for the untested equipment. Con Edison's proposed short-term actions were appropriate to reduce the backlog. Previous Con Edison actions on the PMT process were ineffective to preclude a large backlog. A NCV was identified for this issue.

M8 Miscellaneous Maintenance Issues

M8.1 Review of Previous Inspection Items (92902)

(Closed) IFI 98-19-01. Inspection item IFI 98-19-01 was reviewed. The item was addressed in the licensee's corrective action program, and had low safety and risk significance. Based on this review, this items is closed.

(Closed) IFI 98-201-10. Inspection item IFI 98-201-10 was reviewed. The item was addressed in the licensee's corrective action program, and had low safety and risk significance. Based on this review, this items is closed.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 Spent Fuel Rack Boraflex (UNR 05000247/2000-001-02)

a. Inspection Scope (37551)

The purpose of this inspection was to review the Con Edison actions to assure the spent fuel pool met the design basis of Technical Specification 5.4.2.B.

b. Observations and Findings

The spent fuel storage racks at Indian Point 2 use boraflex as a neutron absorber material to control the neutron effective multiplication factor (Keff) for spent fuel stored in the pool. Operating experience has shown that the boraflex material can degrade after long term exposure to radiation. Con Edison developed an analytical model (RACKLIFE computer code) to estimate the rate of boraflex loss for each panel in the spent fuel storage racks as a function of service life. Con Edison obtained results from the RACKLIFE analysis in June 1999 which indicated that the extent of boraflex degradation may have caused the effective multiplication factor (Keff) for the racks to be greater than 0.95 with unborated water in the pool (reference Condition Report 199904943). The licensing and design basis for the spent fuel pool described in Technical Specification TS 5.4.2.B requires Keff to be less than or equal to 0.95 with unborated water in the pool.

Con Edison verified that no immediate safety concern existed because the analytical results were considered conservative and the concentration of soluble boron in the spent fuel pool is normally maintained at or above 1500 ppm per Technical Specification 3.8.D.2. The SFP boron concentrations was typically in the range of 2180 to 2215 ppm. The testing of boraflex coupons in 1997 and 1995 did not show evidence of accelerated degradation. Con Edison initiated a multi-faceted action plan to address the boraflex issue. The action included sensitivity studies to refine the conservatism in the analyses, and in-situ neutron absorption measurements (BADGER testing) of a statistical sample of boraflex panels to quantify the actual boraflex loss. The test results will be used to determine the reactivity effects of the degraded boraflex and an assessment of Keff.

Con Edison plans to complete the spent fuel pool assessments by the end of March 2000, along with operability and reportability evaluations.

c. Conclusions

Con Edison's actions to address boraflex degradation in the spent fuel racks is unresolved pending the completion of an assessment of the boraflex and the impact on spent fuel rack licensing and design basis, and subsequent review by the NRC (**UNR 05000247/2000-001-02)**.

E8 Miscellaneous Engineering Issues

E8.1 Review of Licensee Event Reports (92903)

a. Inspection Scope (92903)

The inspector reviewed licensee actions to make reports per 10CFR 50.73 and to address degraded conditions.

b. Observation and Findings

(Updated) LER 1999-013; Design Anomaly Identified in Emergency Diesel Loading Sequence. Con Edison identified that a postulated single failure of 125 volt DC power could result in an emergency diesel improperly sequencing loads during a postulated loss of offsite power with unit trip and no safety injection signal. The event was reported as a condition outside the design basis of the facility. The condition was part of the original plant design and Con Edison could not identify a root cause for this deficiency. The inspector noted that Con Edison's commitment to update the LER on December 20, 1999 did not occur. Previous NRC inspection report 05000247/1999009 documented that operability of the diesel generators was maintained with this design deficiency.

The purpose of the supplemental report was to document the completion of the corrective actions to preclude repetition of the event. Further, the inspector noted that within Con Edison's corrective action program (condition report 199906411) no assignment existed to ensure the recommended supplemental report was provided to the NRC. The investigation results and proposed corrective actions for this event did not meet the timeliness expectation within station administrative order (SAO)-112, "Corrective Action Program." Specifically, the investigation is expected to be completed within 30 days from the event date. The event date of this issue was August 19, 1999. Corrective actions to address timeliness of causal investigations are currently within the corrective action leadership plan. This LER remains open pending completion of proposed modifications.

(Closed) LER 1999-016, Engineering Review of Field Condition Revealed a Missing Jumper Wire. Con Edison identified during an installation of a modification, variations existed between the design documents and the actual field conditions. Specifically, jumper wires indicated on the design drawings did not physically exist in the installed circuitry for the two motor driven auxiliary feedwater pumps. The consequences were in the event of a loss of 480 volt bus 3A, the 21 auxiliary feedwater pump breaker would

require operator action to start the pump for a low-low steam generator water level signal or an anticipated transient without scram (ATWS) mitigating system actuation circuitry (AMSAC) signal. This event was reported as a condition outside the design basis of the facility. Specifically, as described in the NRC safety evaluation report dated May 16, 1989, “Equipment, Diverse from the Reactor Trip System,” the requirement to initiate the AFW system and a turbine trip for ATWS events was not fully met for the 21 auxiliary boiler feedwater pump.

Immediate corrective actions included reinstallation of the missing jumper wires, completing extent of condition reviews on other circuits that may have similar anti-pump lockup problems or missing jumper wires, and review of the post maintenance test with the installation of AMSAC for 21 AFWP to assure the wiring was correct. Con Edison committed to a supplemental report on December 20, 1999 to document apparent causes and possible additional corrective actions. This commitment was not fulfilled, in part, due to a lack of tracking within Con Edison’s corrective action program on the issuance of a supplemental report. This LER is closed.

(Closed) LER 1999-018; Plant Operation in Condition Prohibited by Technical Specifications. This event was previously documented in NRC inspection report 0500247/1999-009. During the in office review of the LER the inspector noted that a supplemental report was not issued to the NRC by the expected date of December 20, 1999. The purpose of the supplemental report was to document any additional corrective actions to preclude repetition of the event. Further, no assignment existed within Con Edison’s corrective action program (condition report 199907583) to ensure the recommended supplemental report was submitted. The investigation results and proposed corrective actions for this event did not meet the timeliness expectation within station administrative order (SAO)-112, “Corrective Action Program.” Specifically, the investigation is expected to be completed within 30 days from the event date. The event date of this issue was October 6, 1999. Corrective actions to address timeliness of causal investigations are currently within the corrective action leadership plan. This specific concern is considered to be a minor violation and is not subject to formal enforcement action. This LER is closed.

(Closed) LER 1999-007; Failure of 480 Bus 2A DC Control Power Transfer Switch. NRC inspection report 0500247/1999-002 previous documented the event and concluded that Con Edison’s reporting, evaluation of the degraded condition associated with the redundant control power, and implementation of corrective actions were appropriate. This LER is considered closed.

c. Conclusions

The LERs reviewed met the reporting requirements of 10 CFR 50.73. For some reports, the inspector noted problems in meeting the administrative requirements to process the LERs and track commitments to issue supplemental reports.

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiological Controls-External and Internal Exposure

a. Inspection Scope (83750)

The inspector evaluated the effectiveness of selected aspects of the applied radiological control program. The evaluation included a selective review of the adequacy and implementation of the following radiological control program elements and activities:

- implementation of the radiation work permit (RWP) program
- access controls to radiologically controlled areas (RCAs)
- use and adequacy of personnel occupational exposure monitoring devices
- maintenance of personnel occupational radiation exposures (external and internal) within applicable regulatory limits and as low as reasonably achievable (ALARA)
- status of the National Voluntary Laboratory Accreditation Program (NVLAP)-accredited personnel dosimetry program providing whole-body thermoluminescent dosimeters (TLDs)
- dose calculations for internal uptakes

The inspector evaluated performance in the above-selected areas via observation of work activities, tours of the radiologically controlled area (RCA), discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

Consolidated Edison Company of New York, Inc. (ConEd) maintained personnel occupational radiation exposures (external and internal) within applicable regulatory limits and as low as reasonably achievable (ALARA). A review of personnel exposure data for 1999 identified that the individual exposure results for total effective dose equivalent (TEDE), lens of the eye dose equivalent (LDE), shallow-dose equivalent (SDE), and extremity dose equivalent were well below regulatory requirements. Further, the maximum individual committed effective dose equivalent (CEDE) for any one individual was well within applicable NRC limits in Title 10 Part 20.1201 of the Code of Federal Regulations (10 CFR 20.1201).

ConEd used their own TLDs and maintained a NVLAP-accredited personnel dosimetry program. ConEd maintained a copy of the current accreditation certificates for this dosimetry program. ConEd also maintained and operated several whole-body counters.

ConEd implemented effective access controls to the radiologically controlled areas of the station including use of RWPs, bar code readers, and computerized log-in stations. No access control deficiencies were identified. Appropriate personnel monitoring devices for access to the RCA were supplied and used. TLDs and personnel alarming electronic dosimeters were observed to be properly worn to measure external dose. Access controls for high radiation areas (HRAs) were effective. Radiological postings and labels throughout the toured areas provided additional administrative controls and information to the worker. Survey maps with radiological data were kept by the health physics (HP) technicians at the HP control points and were used for RWP briefings.

c. Conclusions

ConEd implemented effective applied radiological controls. The radiation work permit program was adequately implemented. Personnel occupational exposure was maintained within applicable regulatory limits and as low as reasonably achievable (ALARA). Access controls to radiologically controlled areas were effective, and appropriate occupational exposure monitoring devices were provided and used. No violations were identified.

R1.2 Radiological Controls-Radioactive Materials, Contamination, Surveys, and Monitoring

a. Inspection Scope (83750)

The inspector evaluated the effectiveness of surveys, monitoring, and control of radioactive materials and contamination. The evaluation included a selective review of the adequacy and effectiveness of the following radioactive material and contamination control program elements:

- surveys and monitoring of radioactive material and contamination
- the calibration status of survey and monitoring equipment
- the proper uses of personal contamination monitors and friskers
- the tracking of personnel contamination events and goals

The inspector evaluated performance in the above selected areas via observation of work activities, tours of the RCA, discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

Survey records contained appropriate radiological information. A HP technician was observed while performing a routine survey in section 1 of the piping penetration area on the 51-foot level of the primary auxiliary building. During the counting of the smears taken during this survey, a smear count indicated that there was contamination in an area which was not posted as contaminated. The HP technician took appropriate and immediate actions including a whole-body frisk to re-verify that there was no resultant personnel contamination involved and an immediate re-posting of the surveyed area as a contaminated area. In general, radiological housekeeping conditions in the unlocked areas of the radiologically controlled area were good. However, housekeeping conditions in the Unit 1 areas were not up to the same standard as that in Unit 2 in that several examples of cluttered pathways and one roof leak were noted. Radioactive material and radioactive waste were clearly labeled, segregated, and stored in an orderly manner.

ConEd implemented an effective radioactive material and contamination control program. Continuous air monitors were in use in the RCA. Hand-held contamination monitors (friskers) and radiation survey meters exhibited current calibration stickers and were appropriately used by personnel. A review of instrument calibration records indicated that the calibration program was being implemented in accordance with

procedures. Personnel were properly frisking at the RCA exit using whole body contamination monitors.

Goals to assist in monitoring and tracking personnel contamination rates and percent clean area were maintained and used to gauge the overall effectiveness of the station's programs in this area.

c. Conclusions

Overall, ConEd implemented effective surveys, monitoring, and control of radioactive materials and contamination. Health Physics technicians properly conducted and documented survey results. In general, radiological housekeeping conditions were noted to be good with some exceptions on the Unit 1 side. The number of personnel contaminations was being tracked and trended. The radiological surveys, monitoring, and controls were implemented with calibrated and properly used devices. No violations were identified.

R1.3 Radiological Controls-As Low As Reasonably Achievable (ALARA)

a. Inspection Scope (83750)

The inspector evaluated the effectiveness of the program to maintain occupational radiation exposure as low as is reasonably achievable. The evaluation included a selective review of the adequacy and effectiveness of the following ALARA program elements/documents:

- Radiological Support 1997 Outage ALARA Summary Report
- Source Term Reduction Program
- Station ALARA Committee Meeting Minutes, Meeting No. 99-05, December 2, 1999
- Con Edison Indian Point 1999ALARA Summary Report
- Person-Rem Exposure Goals for Year 2000, December 16, 1999

The inspector evaluated performance in the above selected areas via observation of work activities, tours of the RCA, discussions with cognizant personnel, review of historical documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The ALARA program was being implemented in accordance with procedural controls. Procedures were in place to address ALARA pre-job reviews, post-job reviews, and RWP person-rem estimating. For RWPs requiring an ALARA review, the reviews considered a wide range of considerations for exposure control. Annual and outage person-rem goals for the site were established, and person-rem goals for each work group were developed. There was an active source term reduction program with a list of identified tasks for source term reduction which were being actively addressed on a priority basis as resources became available. At the 0800 Daily Management Team Meetings, the agenda included a Radiation Protection/Chemistry Daily Report by the RP Manager which addressed actual person-rem accumulated for the site on a month-to-date and year-to-date basis compared to projections based on established goals.

The person-rem goal for 1999 for the site was 42 (i.e., 12 and 30 for Unit 1 and Unit 2, respectively), and the actual result was 40.4 (i.e., 18.1 and 22.3 for Unit 1 and Unit 2, respectively). The actual annual person-rem for Unit 2 was a historical low value for that Unit.

The person-rem goal for 2000 for the site was set at 197.7 (i.e., 4 and 18 for nonoutage periods for Unit 1 and Unit 2, respectively, and 175.7 for the Unit 2 outage). The actual person-rem for the last refueling outage in 1997 was 291.8 which was the lowest historical outage dose.

c. Conclusions

The ALARA program was effective. The program was being implemented in accordance with established procedures. Person-rem goals were established, tracked, and used to gauge the overall effectiveness of the station's program in this area. Planning and preparation for the 2000 outage was in progress.

R1.4 Reactor Coolant System Crud Burst

a. Inspection Scope (71750)

On February 19, 2000, Con Edison initiated a station "stop work" in response to unexpected in-plant radiological condition changes due to a planned reactor coolant system crud burst.

b. Observations and Findings

A crud burst is a routine activity performed during major outages to reduce personnel exposures. A crud burst is the release of elemental nickel-58 and cobalt-58 deposited on the reactor coolant system by the injection of hydrogen peroxide. The purpose is to allow the purification system to clean up the activated metals and reduce personnel exposure when opening the reactor coolant system.

The inspector confirmed the Con Edison had controlled the induced reactor coolant system crud burst consistent with procedural expectations in IPC-A-311, "Induced

Reactor Coolant Crud Burst." Specifically, the health physics technicians performed radiological surveys in areas impacted by the crud burst, and appropriately posted the areas based upon expected radiation level increases.

The crud burst in February 2000 was different than those conducted in the past, since no reactor coolant pump was in operation, but rather only the residual heat removal loop. Due to this configuration, activated products concentrated in the residual heat removal system. Specifically, the concentration of cobalt-58 was approximately two times higher than the crud burst performed in 1995. Consequentially, this resulted in various areas within the primary auxiliary building changing from radiation areas to locked high radiation areas. The inspector confirmed that appropriate controls were being implemented to limit access to the areas during the purification of the residual heat removal system.

Industry guidance does indicate that without reactor coolant pump in operation the radiation fields are expected to concentrate within the residual heat removal system. The stop work was initiated to control personnel access and maintenance work due to a higher than expected change in radiological conditions primarily within selected areas of the primary auxiliary building and vapor containment. Radiological conditions within the primary auxiliary building and vapor containment returned to pre-crud burst values on February 20, 2000.

c. Conclusions

Con Edison anticipated changing radiological conditions in the facility when inducing a reactor coolant system crud burst; however, the magnitude of the radiation doses impacted access to certain plant areas. Industry guidance predicted this phenomenon due to plant configuration; however, this guidance was not fully considered in the pre-planning for the crud burst.

R7 Quality Assurance in RP&C Activities

a. Inspection Scope (83750)

The inspector evaluated the effectiveness of the self-identification and corrective action processes. The evaluation included a selective review of the adequacy and effectiveness of the following program elements and documents:

- Audit Report No. 99-03-C, Radiation Protection Program, June 3, 1999
- Surveillance Report 99-SR-010, High Radiation Area (HRA) Controls, April 16, 1999
- Surveillance Report 99-SR-039, Radiation Protection Training and Qualification Matrixes, December 3, 1999
- First Half 1999 Radiological Assessor Report
- RP Instrument Self-Assessment Report, June 7, 1999
- IP2 Radiation Protection - A Self-Assessment, November 1999
- Condition Reports (various)
- Radiation Protection Improvement Plan (Vision and Mission of the Indian Point 2000-2004 Business Plan)

The inspector evaluated the performance in the above area via observation of work activities, tours of the RCA, discussions with cognizant personnel, review of applicable documentation, and review and evaluation of applicable station procedures.

b. Observations and Findings

The Audit Report No. 99-03-C of the Radiation Protection Program was appropriate in scope, detailed, and resulted in the generation of three Condition Reports. The two surveillance reports (99-SR-010 and 99-SR-039) were detailed, well documented, and resulted in five Condition Reports. The Radiological Assessor Report addressed emergent issues and RP performance from a management oversight perspective. The two Self-Assessments were in-depth reviews and produced seven Condition Reports and numerous improvement action items. A Radiation Protection Improvement Plan, which was currently being implemented, addressed the resolution of systematic weaknesses in programs and processes identified in various recent audits, assessments, condition reports, and industry inspections. The number of Condition Reports since the beginning of 1999 to the time of this inspection showed that problems and areas for improvement in the area of radiation protection were being effectively identified and tracked. However, ConEd self-identified that there was a backlog of Condition Reports which were not being addressed for corrective actions and resolution on a timely basis. RP management recently initiated an action plan to correct this situation and to eliminate the backlog.

c. Conclusions

Con Ed's self-identification processes in the area of radiation protection were effective; however, there was a backlog of identified issues requiring corrective action and resolution. Audit and surveillance reports, radiological assessor reports, self-assessments, and the corrective action program were effective in identifying and improvement opportunities. An action plan was initiated to improve the timeliness in processing and resolving identified problems and issues. No violations were identified.

S3 Security Program Plans

a. Inspection Scope (81700)

Area inspected was: Security Program Plans

b. Observations and Findings

Security Program Plans. An in-office review was conducted of changes to the Physical Security Plan, identified as Revision 19, submitted to the NRC on June 15, 1999 and September 14, 1999, in accordance with the provisions of 10 CFR 50.54(p).

c. Conclusion

Based on a limited review of the changes, as described in the plan revisions, no NRC approval of these changes is required, in accordance with 50.54(p). These changes will

be subject to future inspection to confirm that the changes, as implemented, have not decreased the overall effectiveness of the security plan.

X1 Exit Meeting Summary

The inspector presented the health physics inspection findings to members of management at the end of the inspection on February 11, 2000. Management acknowledged the findings. The resident inspector presented the inspection results to Con Edison's management at an exit meeting on March 9, 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.

X2 Regional Management Visit

The Regional Administrator for Region I, toured the facility on February 25, 2000, and met with U. S. Congresswoman Susan Kelly and a delegation of the New York State and local officials. The purpose of the tour was to discuss the February 15, 2000, steam generator tube leak, and the Con Edison and NRC responses to the event.

Attachment 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

+* T. Burns, HP Supervisor-Instruments
+ M. Dampf, RP Special Projects
+* M. Donegan, HP/RW Manager
+* J. Groth, Chief Nuclear Officer
R. Mages, HP Rad Support
+* M. Miele, Radiation Protection Manager
+* V. Nutter, Radiation Support Manager
E. Salisbury, Dosimetry Supervisor
* D. Smith, Radiological Assessor/QA Auditor
+* M. Zeoli, HP Supervisor Unit 2
W. Zolotas, HP Technician

NRC

+* J. McFadden, Regional Health Physics Inspector
* P. Habighorst, Resident Inspector

+ Denotes those present at entrance meeting on February 7, 2000
* Denotes those present at exit meeting on February 11, 2000

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
83750 Occupational Radiation Exposure
92902 Followup-Maintenance
92903 Followup-Engineering
61725 Surveillance Testing and Calibration Control Program
92901 Followup-Operations
92904 Followup-Plant Support
93702 Response to Events
TI 2515/142 Draindown During Shutdown And Common-Mode Failure (NRC GL 98-02)

ITEMS OPENED and CLOSED

Open
NCV 05000247/2000001-01: Post Maintenance Test Program
UNR 0500247/2000-001-02) Spent Fuel Rack Boraflex Degradation (37551)

Updated
LER 1999-013: Design Anomaly Identified in Emergency Diesel Loading Sequence (92903)

Closed
NCV 05000247/2000001-01: Post Maintenance Test Program
LER 1999-018: Plant Operation in Condition Prohibited by Technical Specifications (92902)
LER 1999-007: Failure of 480 Bus 2A DC Control Power Transfer Switch (92902)
LER 1999-016: Engineering Review of Field Condition Revealed a Missing Jumper Wire (92903)

LIST OF ACRONYMS USED

AFW	auxiliary feedwater
AIT	augmented inspection team
ALARA	As Low As is Reasonably Achievable
AMSAC	ATWS (anticipated transient without scram) mitigating system actuation circuitry
AOI	abnormal operating instruction
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
CM	corrective maintenance
COL	check off list
ConEd	Consolidated Edison
CR	condition report
EDG	emergency diesel generator
EPRI	Electrical Power Research Institute
GL	Generic Letter
gpd	gallons per day
gpm	gallons per minute
HP	Health Physics
HRA	High Radiation Area
IP2	Indian Point 2
IP3	Indian Point 3
Keff	effective multiplication factor
KV	kilovolt
LARP	local alarm response procedure
LDE	Lens of the eye Dose Equivalent
LER	licensee event report
MM	minor maintenance
MOD	modification
MSIV	main steam isolation valve
NOUE	notification of unusual event
NPO	nuclear plant operator
NRR	Nuclear Reactor Regulation, Office of
NVLAP	National Voluntary Laboratory Accreditation Program
OD	operability determination
OTR	other
PAP	plug a plug
PMT	post maintenance test
psig	pounds per square inch, gauge
RCA	Radiologically Controlled Area
RCP	reactor coolant pump
RCS	reactor coolant system

RHR	Residual Heat Removal
RP&C	radiological protection and chemistry controls
RWP	radiation work permit
RWST	Refueling Water Storage Tank
SAO	station administrative order
SBBPS	secondary boiler blowdown purification system
SDE	Shallow Dose Equivalent
SE	safety evaluation
SE	system engineer
SER	safety evaluation report
SJAE	steam jet air ejector
SNSC	station nuclear safety committee
SOP	system operating procedure
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
TS	technical specification
UNR	unresolved
WO	work order