U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.	50-247
License No.	DPR-26
Report No.	98-01
Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
Location:	Buchanan, New York
Dates:	January 1 through February 9, 1998
Inspectors:	Robert Temps, Senior Resident Inspector Peter Habighorst, Resident Inspector Scott Schwind, Resident Inspector Jason Jang, Radiation Specialist, Region I Frank Arner, Reactor Engineer, Region I
Approved by:	John F. Rogge, Chief Projects Branch 2 Division of Reactor Projects

9804030255 980325
PDR ADOCK 05000247
Q PDR

EXECUTIVE SUMMARY

Indian Point 2 Nuclear Power Plant
NRC Inspection Report No. 50-247/98-01

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

Operations

The inspector reviewed Con Edison's implementation of freeze protection procedures and concluded that they were adequately implemented; however, the inspector noted a lack of operable heaters on the service water traveling screens. This is a long-standing deficiency that has not been appropriately evaluated. This issue is another example of a concern expressed in NRC Inspection Report 50-247/97-15 regarding Con Edison's failure to adequately review degraded conditions on equipment for their impact on system operation. (O2.2)

The inspectors reviewed an event involving a small steam release that occurred in the auxiliary feedwater pump room during testing of a feedwater flow instrument on July 19, 1997. During this incident, a perturbation of the feedwater system was observed by operators in the control room, and the event causing it was investigated by control room operators. However, no record of this event or its investigation was entered in the operators' log books, nor was the start of the test evolution. This was contrary to the specified requirements for logkeeping practices in the Operations Administrative Directives. (O3.1)

The inspectors reviewed the results of a peer review of Indian Point 2 performed by the World Association of Nuclear Operators (WANO) that was conducted in February of 1997. The peer review was performed by WANO in lieu of the Institute of Nuclear Power Operations (INPO). The inspector reviewed the final report, issued October 21, 1997, and concluded that the results documented in it were generally consistent with the NRC's assessment of Con Edison's performance at the time of the review. (O6.1)

The inspectors updated an inspector followup item dealing with the venting of non-condensibles gases from the reactor vessel. In reviewing corrective actions take to date, the inspector concluded that communication between engineering and operations on this issue could be improved and that actions to address adequate venting of the reactor head have been slow and are still not fully resolved. (O8.3)

Maintenance and Surveillance

The inspectors reviewed a number of maintenance and surveillance activities and concluded that overall, the activities observed were performed in accordance with procedures. However, the inspectors identified several instances in which procedures did not provide sufficient guidance to the test or maintenance personnel. In one instance, this resulted in the failure of the post-maintenance test following corrective maintenance on an

oiler for a component cooling water pump. In a separate instance, test conditions for the 22 auxiliary boiler feedpump could not be obtained as the result of failing to incorporate lessons learned from past performance of the test using compressed air. (M1.1)

Con Edison identified three instances of long-standing failures to implement Technical Specification (TS) surveillance requirements. The causes for these implementation failures were under investigation by Con Edison at the end of the inspection period. The inspectors concluded that these examples, coupled with those identified in past NRC reports, could indicate a programmatic weakness in the TS surveillance program at Indian Point Unit 2. The inspectors note that a comprehensive TS surveillance review was initiated by Con Edison at the end of the inspection period and that the review was added as a restart criterion. (M3.1) These three issues are considered an unresolved item (URI) which will be updated in the future to reflect additional TS deficiencies identified by Con Edison's review, at which time the overall result will be reviewed against the NRC's enforcement policy for appropriate action. **URI (50-247/98-001-06)**

The inspectors reviewed several open item reports (OIRs) generated during the current outage that documented examples of poor work control and lack of attention to detail with the potential for adverse impact on safety related equipment and on personnel safety. In the case of nitrogen relief valve 4105, associated with power operated relief valve (PORV) PCV-456, inattention to the work environment rendered one train of the overpressure protection system (OPS) inoperable on two separate occasions and unnecessarily challenged the operators. With respect to the improper release of a tagout to support work on the 22 residual heat removal (RHR) pump, operators performed well in recognizing that plant conditions could not support removing the 22 RHR pump from service as this would result in a violation of TSs. However, poor operator performance was noted in the improper removal of a protective tagout with an active work order still open. A modification to the EDG air start motor supports was performed out of sequence, raising questions regarding work control and contractor oversight. Collectively, the issues described in the OIRs represent continued problems in the work control process and unnecessary challenges to operators, plant equipment, and personnel safety. (M6.1)

A licensee event report (LER) was reviewed dealing with the inadvertent start of the three EDGs during a maintenance activity in a breaker cubicle. As stated in the LER, the principal root cause of this event was worker error on the part of the mechanic working in the breaker cubicle compounded by not considering the consequences of potential worker error at the work location during the pre-job briefing. (M8.1)

Engineering

During post maintenance testing, the K3 relay and resistor R65 in the 23 EDG control panel failed. This 480 volt relay had been replaced with an incorrect 120 volt model, as part of a seismic upgrade modification, due to an erroneous determination of equivalency and errors in the controlled drawing for the system. This is an additional example of configuration control and drawing discrepancies resulting in challenges to operators and equipment and is similar to concerns expressed in NRC Inspection Report 50-247/97-15, Section O4.2, regarding drawing configuration errors. A violation was issued for this event. (E2.1)

The inspectors reviewed two modifications that were being implemented at the plant. The first involved modification to the drain lines for the main steam safety valves. The inspector concluded this modification was installed as designed. In reviewing the other modification involving protection to the 480 volt switchgear from a postulated high energy line break, the inspector concluded that the reviews for this modification were incomplete in that they did not consider the consequences of a potential worst-case single active failure vulnerability for the 480 volt room intake louvers used to mitigate the impact of a postulated line break of the auxiliary steam piping located outside the room. While subsequent Con Edison analysis concluded that the modification was still acceptable assuming the new postulated failure mode, the modification was subsequently canceled when Con Edison decided to reroute the auxiliary steam piping. (E2.2)

Plant Support

The corrective actions for a violation and a deviation, regarding the technical support center ventilation system, were inspected by a Region I Radiation Specialist during this inspection period. This inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector of the licensee's corrective actions. Based on the licensee's corrective actions and technical reviews, the violation and deviation were closed. (R8.1)

The inspectors concluded that on January 8, 1998, two individuals violated Con Edison's procedures for hot work activity when a worker continued grinding while his fire watch had left his assigned post. The inspectors also concluded that: 1) personnel performing fire watch duties had not received training on the specific Indian Point 2 administrative requirements governing hot work and fire watch requirements; 2) the fire watch administrative procedure was deficient in that it required personnel to perform actions from an inactive procedure; 3) supervisory personnel who initiated and signed hot work permits had not been rigorous in ensuring that fire watch personnel were instructed and posted in accordance with the administrative procedure governing fire watch activities; and 4) plant management was not proactive in identifying and ensuring that expectations regarding the control of hot work and fire watches were met. These conclusions are in line with

concerns expressed by the NRC in previous inspection reports regarding problems at Indian Point 2 with lack of procedure adherence, poor procedures, poor supervisory oversight, and plant management expectations not being met and or enforced. A violation was issued for this event. (F1.1)

TABLE OF CONTENTS

EXECUTIVE SUMMARY ii

TABLE OF CONTENTS vi

Summary of Plant Status 1

I. OPERATIONS 1
- O1 Conduct of Operations 1
  - O1.1 General Comments 1
- O2 Operational Status of Facilities and Equipment 1
  - O2.1 Verification of System Alignment and Safety-Related Tagouts 1
  - O2.2 Status of Cold Weather Preparations; URI 50-247/98-001-01 2
  - O2.3 Verification of Sealed Valves and Components 3
  - O2.4 Observations from Vapor Containment Tours 4
- O3 Operations Procedures and Documents 5
  - O3.1 Inadequate Log Keeping Practices; NCV 50-247/98-001-02 5
- O6 Operations Organization and Administration 6
  - O6.1 Inspector Review of Third Party Review (71707) 6
- O8 Miscellaneous Operations Issues 7
  - O8.1 Closed: Licensee Event Report 50-247/97-009-00;ESF of Automatic Containment Isolation Valves 7
  - O8.2 Closed: VIO 50-247/96-001-01;Activities Performed Without Approved Procedures 7
  - O8.3 Update: IFI 50-247/97-015-02;Reactor Vessel Head Non-Condensible Gas Removal 8
  - O8.4 Closed: URI 50-247/96-080-01;Actions on Pressurizer Level During Reactor Trip 8

II. MAINTENANCE 10
- M1 Conduct of Maintenance 10
  - M1.1 Maintenance and Surveillance Observations; IFI 50-247/98-001-03; NCV 50-247/98-001-04and 05 10
- M3 Maintenance Procedures and Documentation 13
  - M3.1 Missed Technical Specification Surveillances; URI 50-247/98-001-06; Closed LER 50-247/97-025 13
  - M6.1 Work Control and Quality of Maintenance; NCV 50-247/98-001-07 and 08 15
- M8 Miscellaneous Maintenance Issues 17
  - M8.1 Closed: Licensee Event Report 50-247/97-016-00;Auto-Start of Emergency Diesel Generators 17

Table of Contents (cont'd)

III. ENGINEERING 18
E2 Engineering Support of Facilities and Equipment 18
E2.1 Failure of 23 Emergency Diesel Generator K3 Relay; VIO 50-247/98-001-09 18
E2.2 Plant Modification Review 19
E8 Miscellaneous Engineering Issues 21
E8.1 (Closed): Licensee Event Report 50-247/97-020, Technical Specification 3.0.1 Required Plant Shutdown 21

IV. PLANT SUPPORT 21
R8 Miscellaneous Radiological Protection and Chemistry Issues Miscellaneous 21
R8.1 Review of Corrective Actions to Apparent Violation Identified in IR 50-247/92-22 (Reference EA 92-245; item 01014) 21
R8.2 Review of TSC Filtration System 22
R8.3 Closed: URI 50-247/96-004-03 22
F1 Control of Fire Protection Activities 23
F1.1 Failure to Observe Fire Watch Requirements; VIO 50-247/98-001-10 23

V. MANAGEMENT MEETINGS 25
X1 Exit Meeting Summary 25

ATTACHMENT

Attachment 1 - INSPECTION PROCEDURES USED
ITEMS OPENED AND CLOSED
LIST OF ACRONYMS USED

Report Details

Summary of Plant Status

At the start of the inspection period on January 1, 1998, the unit was in the cold shutdown condition with the reactor coolant system below 200° F and depressurized. The plant remained in this condition for the remainder of the inspection period.

I. OPERATIONS

O1 Conduct of Operations[1]

O1.1 General Comments

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. Specific events, noteworthy activities, and observations are detailed in the sections below.

O2 Operational Status of Facilities and Equipment

O2.1 Verification of System Alignment and Safety-Related Tagouts

a. Inspection Scope (71707)

Using Inspection Procedure 71707, the inspectors performed a walkdown of accessible portions of the residual heat removal system and performed a review of a safety related tagout.

b. Observations and Findings

Verification of System Alignment

The inspectors performed a walkdown of accessible portions of the residual heat removal (RHR) system. System alignment was verified using system prints, system operating procedures, and the Updated Final Safety Analysis Report (UFSAR). No previously unidentified discrepancies were noted.

Verification of Safety Related Tagout

The inspectors reviewed safety related tagout 97-8393 issued for work on electrical bus 2A. The inspectors noted that the positions of the 22 service water pump breaker and the 21 MG set 480V bus breaker were listed on the tagout as racked out; however, the breakers were actually removed. The inspectors verified that this was allowed per Operations Administrative Directive (OAD) 19, "Tagout Program," since the actual positions were more conservative than the tagout required position

---

Topical headings such as 01, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

and minimum required protection was not reduced. Portions of the tagout restoration were also observed and no deficiencies were noted.

c. Conclusions

The inspectors noted no deficiencies during review of the bus 2A tagout. No discrepancies were noted in system alignment during the walkdown of the residual heat removal system.

O2.2 Status of Cold Weather Preparations; URI 50-247/98-001-01

a. Inspection Scope (71714)

Using inspection procedure 71714, the inspector reviewed Con Edison's measures to protect safety related systems against cold weather. The inspection included a review and independent verification of Con Edison's freeze protection procedures as well as a comparison of present system configurations against the Updated Final Safety Analysis Report (UFSAR) as they relate to freeze protection.

b. Observation and Findings

Operations Administrative Directive (OAD) 22, "Freeze Protection," implements Con Edison's cold weather protection actions. This directive references System Operating Procedures (SOP) 11.5, "Space Heating and Winterization," and SOP 30.1, "Electric Heat Trace System," and the associated check-off-lists (COL). The inspector reviewed the completed procedures and walked down portions of the procedures applicable to the service water pump and strainer pits, the condensate storage tank, areas inside the primary auxiliary building (PAB) and the vapor containment (VC).

During the walkdown of SOP 30.1, the inspector noted that all six circulating water traveling screens and the two service water traveling screens had deficiency tags written against the associated electric heaters. These deficiency tags were dated as early as 1994 and stated that the heaters would not energize or that the heaters cause fuses to blow. The heaters for the traveling screens in both the circulating water (CW) and service water (SW) systems are described in the applicable portions of the UFSAR; however, the inspector determined that no evaluation had been performed, to date, to determine if operability of the screens is maintained during freezing conditions with their heaters inoperable. After this issue was discussed with the Operation Department Con Edison wrote an open item report (OIR) for this condition. In addition to the heaters, the OIR also documented that the warm water recirculation portion of the CW system, which is also described in the UFSAR, was inoperable and that no evaluation was performed for this feature being inoperable. This item remains unresolved pending further NRC review of the design basis history of the service water traveling screen heaters and the impact of the degraded heaters on operability of the service water system. **URI (50-247/98-001-01)**

The inspector did not observe any other significant or previously unidentified discrepancies with space heating or heat tracing requirements.

c. Conclusions

The inspector concluded that overall, Con Edison adequately implemented their freeze protection procedures; however, the lack of operable heaters on the service water traveling screens is a long-standing deficiency that has not been appropriately evaluated. This issue is another example of a concern expressed in NRC Inspection Report 50-247/97-15 regarding Con Edison's failure to adequately review degraded conditions on equipment for their impact on system operation.

O2.3 Verification of Sealed Valves and Components

a. Inspection Scope (71707)

From January 19 to January 23, 1998, the inspectors independently verified the adequacy of controls for locked and sealed components pursuant to COL-10, "Locked Safeguards Valves."

b. Observations and Findings

The purpose of COL-10 is to provide administrative controls in the positioning and sealing of safeguards valves to establish proper configuration. During the current outage, Con Edison intended to maintain COL-10 up-to-date in reflecting real-time status of sealed components to preclude having to perform the COL at the end of the outage. Normally, deviations from the lineup are allowed when the valve is controlled under another process, such as the caution tagging process.

The inspector verified the status of 30 valves and identified overall agreement between the documented status in COL-10, maintained in the central control room, and the status in the plant. However, on January 19, 1998, the inspector observed that valves BFD-62-2 and BFD 62-3 (stop valves for the 23 auxiliary feedwater pump flow control valves FCV-406C and FCV-406D) were caution tagged closed with the seals removed. This conflicted with their status in COL-10 which indicated the valves had the seals removed and were positioned open on December 14, 1997. The inspector verified that the controls in the caution tag program would have restored the valves to the correct alignment with seals installed per COL-10.

This observation was discussed with operations management, who acknowledged the deficiency and stated they would reinforce their expectations to operators in maintaining COL-10 current during the outage.

c. Conclusions

Overall, good agreement existed between the position of valves in COL-10 and the position in the plant. However, one example was identified where two auxiliary feedwater isolation valve positions and status were not in agreement with the COL.

This was contrary to Con Edison management's expectations for maintaining the COL up-to-date during the outage.

O2.4 Observations from Vapor Containment Tours

a. Inspection Scope (71707)

The inspectors performed various tours within the vapor containment. Selected observations regarding plant housekeeping and material conditions were used to independently assess licensee performance.

b. Observations and Findings

On January 6, 1998, the inspectors observed extensive cracking and flaking paint on the supply and return headers for component cooling water (CCW). This condition existed from the containment penetration along the length of piping all the way to the residual heat removal (RHR) cell. The inspector determined that Con Edison had identified this condition in 1995 and that the CCW piping was being evaluated for repainting.

On January 29, 1998, the inspectors observed that various cable trays on the 46 foot elevation of the vapor containment had marinite fire barrier boards installed that appeared to be inadequately supported and that in some areas there were multiple layers of fire boards. The inspectors also noted a general accumulation of dust and debris on top of the fire boards. The inspector discussed the condition of the fire boards with Con Edison with respect to potential adverse seismic interaction with the cable trays. The inspectors also discussed with Con Edison the observed dust and debris with respect to their potential impact on containment sump performance in the post-accident environment. Con Edison documented these concerns in OIR 98-E00735. The inspectors also observed poorly attached or missing insulation on the letdown line located between the reactor coolant system and the regenerative heat exchanger. The inspectors confirmed that Con Edison had previously included this line in an on-going project to replace and upgrade containment insulation. The upgrade or repair of containment insulation is considered to be a criteria for restart by Con Edison.

During the tour, the inspectors questioned the controls that provide reasonable assurance that the cavity drain line remains free of debris. This drain path is safety significant as it ensures that post-accident flow from the reactor cavity to the containment floor. Con Edison provided reasonable assurance that the line is free of debris as the line is used to drain the cavity post-refueling.

A deficiency observed by the inspectors was a tear in the flexible boot on the 21 iodine fan discharge duct. The inspectors determined that Con Edison had earlier identified this deficiency on January 11, 1998, and documented it in an open item report (OIR).

c. Conclusions

The plant housekeeping and material condition deficiencies inside the vapor containment, noted during the inspector's tours, were appropriately being resolved or tracked in the corrective action system.

O3 Operations Procedures and Documents

O3.1 Inadequate Log Keeping Practices; NCV 50-247/98-001-02

a. Inspection Scope

The inspectors investigated an incident that occurred on July 19, 1997, in which a steam release occurred in the auxiliary feedwater pump room while performing a surveillance test (ST) on a feedwater flow instrument. The ST was procedure PC-R32-1, "Main Feedwater Flow Transmitters," that was performed between July 19 and 21, 1997. Eight feedwater flow (FW) transmitters are checked for calibration, one at a time, by the procedure.

b. Observations and Findings

The inspectors interviewed one of the two Instrument and Control (I&C) technicians who performed the ST on July 19, 1997, and was informed that during restoration from testing the first flow transmitter, the individual went to the level above the auxiliary feedwater (AFW) pump room to open the transmitter root stop valve. As he opened the valve, he said he heard noise (the steam release) from the room below and his partner yelling, so he shut the valve. He stated that the steam/water that was released through a drain line came from a transmitter blowdown valve, manipulated earlier in the test, that apparently had not been fully closed. He also stated that these valves are sometimes difficult to operate and this may have caused the technicians to think it was closed when in fact it was not. The actual steam release was apparently not significant as the steam was mostly dissipated by the time he came back to the AFW pump room.

The technician recalled that the central control room (CCR) was contacted and that the event was discussed with the on-shift senior reactor operator (SRO). He recalled that the SRO was interested in what happened as the steam release apparently activated a fire alarm and because the steam flow through the blowdown line had caused a "blip" in feedwater indications in the CCR. The technician stated that he went to the CCR and talked with the SRO, and that the SRO later went out to the AFW pump room with him to discuss and validate what had happened with respect to the indications observed in the CCR.

The inspector reviewed the senior reactor operator (SRO) and senior watch supervisor (SWS) logs for July 19, 1997, and did not identify any reference to a steam release in the AFW pump room, nor logging of any fire alarm in the building as a result of the steam release, nor any record of a feedwater flow anomaly. The

SWS log made no reference to the event or that the SRO was sent in the field to investigate the anomaly. The SRO log made no reference to the event; it makes no reference to the test being in progress nor any reference to the flow instruments bistables being tripped, as was done by the other SROs in log entries during subsequent testing. There was also no reference in the SRO log to the SRO being relieved by the SWS so the SRO could go to the AFW building, and there was no reference in the SWS log that the SRO was relieved. Lastly, there was no reference in the SRO log to the fire alarm coming in nor of any perturbation in the feedwater system indications.

The inspectors discussed the log keeping issues with the Operations Manager (OM) who agreed that the feed water perturbation and its investigation probably should have been logged. The OM also stated that it would have been more appropriate for someone other than the on-watch SRO to leave the control room to investigate the anomaly with the I&C technician. The OM discussed this event with the SWS and SRO involved and confirmed that there had been a feedwater perturbation, but no fire alarm, during the testing on July 19, 1997.

The inspector reviewed Operations Administrative Directive (OAD) 2, "Watch Relief," and noted that it does not specify an exact protocol for logging turnovers when the SWS temporarily relieves the SRO. The inspector also reviewed OAD 3, "Plant Surveillance and Log Keeping," and noted that Section 3.2.1.1 specifies, in part, that log entries be made for: 1) abnormal indications or conditions which could affect plant status, and 2) tests of equipment, components, or systems. Contrary to this requirement, when PC-R32-1 was initiated on July 19, 1997, initiation of the test was not recorded in the SWS or SRO logbooks, nor was the feedwater perturbation and the subsequent investigation by the SRO logged. This failure constitutes a violation of minor significance and is being treated as a Non-Cited Violation consistent with Section IV of the NRC Enforcement Policy. **NCV (50-247/98-001-02)**

c. Conclusions

The inspectors concluded that a small steam release occurred in the auxiliary feedwater pump room during testing of a feedwater flow instrument on July 19, 1997. During this incident, a perturbation of the feedwater system was observed by operators in the control room and the event causing it was investigated by control room operators. However, no record of this event or its investigation was entered in the operators' log books, nor was the start of the test evolution. This is contrary to the specified requirements for logkeeping practices.

O6 Operations Organization and Administration

O6.1 Inspector Review of Third Party Review (71707)

The inspectors reviewed the results of a peer review of Indian Point 2 performed by the World Association of Nuclear Operators (WANO) that was conducted in February of 1997. The peer review was performed by WANO in lieu of the Institute

of Nuclear Power Operations (INPO). The inspector reviewed the final report, issued October 21, 1997, and concluded that the results documented in it were generally consistent with the NRC's assessment of Con Edison's performance at the time of the review.

O8 Miscellaneous Operations Issues

O8.1 Closed: Licensee Event Report 50-247/97-009-00;ESF of Automatic Containment Isolation Valves

This licensee event report (LER) was submitted in response to an event which occurred on May 2, 1997, when an alarm on R-44 (station ventilation radiation monitor) caused the isolation of the vapor containment (VC) purge supply and exhaust valves and the activation of the weld channel pressurization system (WCPS) to the VC purge valves. This occurred when radiogases leaked by the primary relief tank (PRT) rupture disk during a safety injection (SI) actuation (reference NRC Special Inspection Report 50-247/97-05). The inspector performed an in-office review of Con Edison's root cause determination and corrective actions stated in the LER and assessed that they were appropriate. Corrective actions included: revision of the maintenance procedure used to replace the PRT rupture disk and replacement of the disk; revision of the radioactive release permit procedure and development of a computerized release permit; and a review of this event and corrective actions with all operations crews. The inspector verified that while all corrective action assignments were completed, two corrective actions were still listed as open in the problem identification and tracking system. This is another example of a weakness (lack of timeliness) in the tracking and closing of corrective action assignments. Similar weaknesses were documented in NRC Inspection Report 50-247/97-12, Section M8.1. No violation of NRC requirements was identified during this review and the LER is closed.

O8.2 Closed: VIO 50-247/96-001-01;Activities Performed Without Approved Procedures

This violation was issued because the NRC identified that safety-related equipment was manipulated without safety committee approved procedures as required by the Technical Specifications (TSs) and Regulatory Guide (RG) 1.33. Specifically, it was identified that Operating Instructions (OIs) were being used beyond their original intent as described in Operations Administrative Directive (OAD) 33, "Procedure Adherence and Use."

Con Edison responded to the violation in a letter dated April 29, 1996. In their response, they agreed with the violation as stated. Corrective actions described in the response letter included a temporary change to OAD 33 restricting the use of OIs to evolutions other than those specified in RG 1.33 and a review with the operators as to the proper use of OIs. The inspectors verified that the procedure change was instituted and that operators were instructed as to the appropriate use of OIs. The inspectors note that since this issue was initially raised by the NRC, no

OIs have been issued by the operators and that the process is essentially not used although it remains proceduralized in OAD 33. This violation is closed.

O8.3 Update: IFI 50-247/97-015-02;Reactor Vessel Head Non-Condensible Gas Removal

This inspector followup item involved an assessment of Con Edison's actions concerning the inspector's questions on the accuracy of the reactor vessel level indication system (RVLIS) in a depressurized reactor coolant system condition, and the unexpected relationship between pressurizer level and RVLIS. These issues were previously documented by Con Edison OIR 97-E04896.

The inspector observed that Con Edison had modified their mode of operation during the current shutdown from a periodic venting of the reactor vessel head to a continuous vent of the head to remove non-condensible gases. During this continuous venting, operators have observed that pressurizer level slowly (daily) increases and then drops rapidly (maximum variations in the order of 10 percent level). The use of continuous venting of the reactor head was not a recommendation from engineering personnel involved with responding to OIR 97-E04896. Engineering had recommended not to continuously vent the vessel head due to the formation of loop seals in the vent line and possible loss of positive control of reactor coolant system level; however, the Operations department implemented actions anyway to provide a continuous vent for the reactor head. The inspectors note that operators do continuously trend and monitor level changes within the reactor coolant system and that the level perturbations to date have not adversely impacted on reactor coolant system inventory nor the ability to provide adequate core cooling. At the end of the inspection period, Con Edison was performing actions to confirm the formation of a loop seal in the reactor vessel head vent and reviewing longer term actions to improve the vent capability for the vessel head system.

The inspector concluded that communication between engineering and operations on this issue could be improved and that actions to address adequate venting of the reactor head have been slow and are still not fully resolved. This item remains open pending completion of Con Edison's final actions on this matter.

O8.4 Closed: URI 50-247/96-080-01;Actions on Pressurizer Level During Reactor Trip

This item was originally opened during the 1996 Integrated Performance Assessment Process (IPAP) inspection regarding operators' actions described in the post-trip report for a reactor trip that occurred on January 17, 1995. Specifically, the report stated that an operator observed that level on one of the three pressurizer level channels went below 9%, but that the average of the three channels remained above 9%. Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection," requires that safety injection be manually initiated if level goes below 9%. A computer generated trace of pressurizer level showed that average pressurizer level went below 9% for approximately one minute, reaching a minimum

of 7% before trending upwards. This item was opened pending further NRC review of the operator actions described in the post-trip report.

In followup, the inspector discussed the post-trip report and the operator actions discussed in it with the licensed operator training manager. The inspector obtained and reviewed the computerized printout for pressurizer level during the reactor trip transient period and for the steady-state period prior to the transient. The inspector also reviewed log readings of the three pressurizer level channels that operators use in the control room when implementing E-0 for the pre-trip condition and reviewed EOP E-0 procedure actions.

From this review, the inspector made the following observations and conclusions:

1) A comparison of the three control room pressurizer level channel readings to the computer generated data in the pre-trip period showed that the control room channels read slightly higher (about .3%) than the computer generated indications.

2) The computerized level information is not available in "real time" and is not used by the operators when implementing the EOPs. While the computer printout shows that average level did decrease below 9%, this information was not available to the operators at that time. Further, in steady state conditions, the control room channels read slightly higher than the computer generated indication, so that the indicated decrease in the control room channel was probably less than indicated by the printout. Also, the control room channels are marked in 1% increments so the visual difference between a reading of between slightly greater than 7% to a reading of 9% is not that great.

3) The statement in E-0 to initiate safety injection if pressurizer level cannot be maintained greater than 9% does not specify the number of channels that must be less than 9% to take the action. The operators are trained to use the average of the three level channels. Based on the inspector's previous experience as a licensed operator examiner, this is an acceptable practice as it ensures that a faulty reading on any of the three level channels will not cause the operators to take an unnecessary or adverse action.

4) When the operators enter E-0 on a reactor trip or safety injection, they have a number of actions that must be carried out in a limited time frame. Pressurizer level is just one of many parameters that are checked. While the post-trip computer generated data shows that average level decreased to around 7%, the operators' attention while implementing E-0 is not on pressurizer level alone, and therefore it is quite possible that the average indicated level in the control room did decrease below 9% for a short period without this being noticed. As pressurizer level then increased from the minimum level indicated by the computer generated data, the average level indicated in the control room was probably back above 9% during subsequent checks by the operators.

From this review, and given the factors discussed above, the inspector concluded that the actions taken by the operators during the January 1995 trip with respect to pressurizer level were appropriate. No violation of NRC requirements was identified. This unresolved item is closed.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 Maintenance and Surveillance Observations; IFI 50-247/98-001-03;NCV 50-247/98-001-04and 05

a. Inspection Scope (61726, 62707)

The inspectors observed the performance of various maintenance activities including the review of completed work packages and surveillance tests.

b. Observations and Findings

The following surveillance activities were observed or reviewed by the inspectors:

PC-EM26	Condensate Storage Tank Level Calibration
WO 97-96928	Preventative Maintenance on Bus 2A
PT-Q35	Containment Spray Pump Test
PT-Q24B	22 Emergency Diesel Generator Fuel Oil Transfer Pump
WO 97-95037	Troubleshooting Main Feedwater Regulating Valve FCV-437
WO 98-97018/019	Freeze Seal to Inspect and Repair Valves 839H and 839E
WO 97-96757	21 Component Cooling Water Pump Inboard and Outboard Bearing Oiler Leakage
PT-V8A	Test of 22 Auxiliary Boiler Feedwater Pump

Comments on specific maintenance activities observed are as follows:

PT-Q35

The inspectors reviewed the results of PT-Q35, "Containment Spray Pump Test," performed on December 30 and 31, 1997, for the 21 and 22 containment spray pumps (CSP) respectively. Vibration measurements taken on both pumps were above the alert level. Both CSPs have experienced vibration levels in the alert range intermittently since 1994. The inspector reviewed Con Edison's evaluation and corrective actions for both pumps and concluded that while the system engineer and engineering analysis section have performed evaluations on both pumps, and recommended corrective actions, their overall response to the issues did not appear focused at a comprehensive resolution of the problem. In addition, corrective actions already completed have had limited success in reducing pump vibration. During the 1997 refueling outage, the impeller for 21 CSP was balanced, which

reduced the vibration levels significantly; however, vibration levels continued to trend upward into the alert range during the December 1997 test. The impeller for 22 CSP will be balanced during the current outage. The inspector will follow-up on the ongoing evaluation and corrective actions to reduce vibration levels on both pumps. **IFI (50-247/98-001-03)**

PT-Q24B

On January 7, 1998, the inspectors observed Con Edison personnel perform a quarterly surveillance to demonstrate the operability of the 22 emergency diesel generator fuel oil transfer pump and various check valves in accordance with TS 4.2. The inspector observed that personnel adhered to the surveillance as written, and test instrumentation was within calibration frequency. The inspector observed that one person involved with the test was unfamiliar with how to verify the valve position for the fuel oil level control valves (LCV-1207A and B, 1208A and B, 1209A and B), and that no guidance existed within the surveillance procedure. Also, PT-Q24B requires a jumper to be installed across the level switch for the 22 emergency diesel generator day tank to initiate starting of the fuel transfer pump. The inspector observed that in this instance, the "jumper" was a worker's credit card which was used to ensure that the float switch was positioned to simulate a low level condition for the fuel transfer pump. Both observations were discussed with and acknowledged by the Performance and Test Manager. The inspector concluded that the two observations did not adversely impact the ability to perform the test satisfactorily; however, they did indicate a weakness in procedural detail and guidance in the surveillance test in ensuring consistency in performance.

PT-V8A

On January 8, 1998, the inspectors observed the performance of PT-V8A, which was used to test the overspeed function of the 22 turbine driven auxiliary boiler feedwater pump (ABFP). As the plant was in cold shutdown, compressed air was used to provide the motive force for the ABFP turbine, vice steam. At one point in the test, the operators were directed to locally increase turbine speed using the governor speed changer adjustment wheel. The inspector observed that there appeared to be some confusion over this step as to which handwheel on the turbine this applied to. The operators initially operated the handwheel for HCV-1118 but discovered that this did not result in a change in turbine speed. The operators concluded this was not the correct handwheel and it was placed back in its original position. The test proceeded using the correct adjustment wheel, but the operators were unable to increase the turbine speed to the trip point required by the procedure. The test was stopped and two OIRs were written; one to document the operator performance problem with the incorrect operation of HCV-1118, and a second to document the inability to raise turbine speed to the test required value. The inspectors were subsequently informed by the test director that when the test was performed several years ago using compressed air, the same problem had occurred with respect to not obtaining the desired speed on the turbine. Apparently the actions taken at that time to address the issue were not incorporated into the

procedure. The failure to have an adequate test procedure constitutes a violation of minor significance and is being treated as a Non-Cited Violation consistent with Section IV of the NRC Enforcement Policy. **NCV (50-247/98-001-04)**

WO 98-97018/019

On January 14, 1998, the inspector observed maintenance and contractor personnel establish a freeze seal to isolate test lines from the 22 and 23 safety injection accumulators. The inspector compared Con Edison controls to inspection guidance contained in NRC inspection manual chapter 9900. Overall, the maintenance controls were consistent with this guidance in that: an adequate supply of nitrogen existed, the freeze plug was supplied from a single source of nitrogen, adequate contingency actions had been developed and discussed if the freeze seal failed, temperature and environmental monitoring was established, and criteria were defined for establishing that the freeze seal plug had formed.

WO 97-96757

On January 22, 1998, during corrective maintenance to address an oil leak for the inboard and outboard bearing oilers on the 21 component cooling water pump, the post maintenance test failed and was documented in OIR 98-E00634. The inspector learned that mechanics replaced pipe fittings from both bearing covers to the constant level oiler. The replacement fittings and configuration did not have adequate work order guidance, resulting in the improper setting of the constant level oiler and overfilling of the bearing pumps. This resulted in excessive oil leakage during pump start and failure of the post-maintenance test. Corrective actions were to improve the guidance to mechanics on adjustments and replacement of connecting piping within the work order. The failure to have adequate work order guidance constitutes a violation of minor significance and is being treated as a Non-Cited Violation consistent with Section IV of the NRC Enforcement Policy. **NCV (50-247/98-001-05)**

c. Conclusions

Overall, the maintenance and surveillance activities observed by the inspectors were performed in accordance with procedures. However, the inspectors identified several instances in which procedures did not provide sufficient guidance to the test or maintenance personnel; in one instance, this resulted in the failure of the post-maintenance test following corrective maintenance on an oiler for a CCW pump. In a separate instance, test conditions for the 22 ABFP could not be obtained as the result of failing to incorporate lessons learned from past performance of the test using compressed air.

M3 Maintenance Procedures and Documentation

M3.1 Missed Technical Specification Surveillances; URI 50-247/98-001-06;Closed LER 50-247/97-025

a. Inspection Scope (61726, 90712)

The inspection scope involved follow-up and assessment of Con Edison's identification of three issues involving deficiencies in the technical specification (TS) surveillance program.

b. Observations and Findings

Con Edison identified three problems with the TS surveillance program in early December, 1997, and documented these issues in OIRs 97-E04756, 97-E04812, and 97-E04849. The problems included: 1) failure to perform a surveillance to ensure service water in-leakage into the vapor containment is less that 0.36 gallons per minute (gpm) per fan cooler unit, 2) failure to incorporate a TS change involving a coefficient in the reactor protection system, and 3) failure to perform a channel check and functional check for the steam generator effluent blowdown flowrate measurement. The failure to perform an adequate channel check and functional check for the blowdown effluent flowrates was documented in licensee event report (LER) 50-247/97-025.

Missed Service Water Inleakage Test

TS surveillance requirement 4.4.D.2.b requires that "the leakage rate into containment for the isolation valves sealed with the service water system shall not exceed 0.36 gpm per fan cooler." The TS basis of this inleakage limit is to preclude flooding of the internal recirculation pumps for a full 12-month post-accident period in the event of a fan cooler leak.

The service water system alignment for each of the fan cooler units has two containment isolation valves, an inboard and outboard, for both the supply and return lines. These are referred to in the TS as the A and B valves. TS Table 4.4-1 and Updated Final Safety Analysis Report (UFSAR) section 5.2 state that either the A or B valves for the SWN-41, SWN-44, and SWN-71 series valves may serve as a required containment isolation valve. Con Edison's procedure PT-27B and PT-27B-1, "Service Water Containment Isolation Valve Leakrate Determination," test penetration leakage out of containment. The inspector learned through discussions with Con Edison engineering personnel that the inboard and outboard containment isolation valves are motor-operated tricentric butterfly valves that have a preferred seating direction. The outboard isolation valves are designed to seat with service water fluid and the inboard isolation valves are designed to seat with containment pressure. Given that the A and B valves are designed to seat tighter in different directions, the inspector questioned Con Edison as to the appropriateness of TS Table 4.4-1 allowing either the inboard or outboard isolation valve to be credited in

fulfilling the containment leakage criteria. During past surveillances, Con Edison has credited and tested the inboard isolation valves for satisfying the TS containment leakage criteria.

A Con Edison letter to the NRC, dated December 22, 1980, documented that service water in-leakage was tested between 1976 and 1979; however, indications are that inleakage testing has not been performed since 1980 as required by TS 4.4.D.2.b on a refueling outage frequency.

During this outage, Con Edison performed inleakage testing for the outboard containment isolation valves in both the supply and return lines to the fan cooler units. Six of the total ten outboard isolation valves failed the leakage criteria of 0.36 gpm, with some of the failures between 10 to 15 gpm in-leakage. Further testing using P-MT-155, "Fan Cooler Unit A-Valve Inleakage Determination," to test the inboard isolation valves determined that if all (A and B) isolation valves were closed, then each fan cooler unit's "as found" condition was less than 0.36 gpm, which would satisfy the TS requirement. Con Edison subsequently took corrective actions to improve the leak tightness of the valves with elevated in-leakage and is changing their testing methodology to be more in accordance with the TSs.

During review of Con Edison's specific corrective actions to this missed surveillance, the inspector noted that no proposed actions were planned to provide specific operator guidance within applicable procedures on how to isolate a fan cooler unit should a service water leak occur. Procedures reviewed by the inspector included:

- Emergency Operating Procedure FR-Z.2, Response to Containment Flooding
- Abnormal Operating Instruction (AOI) 28.0.5, Containment Building Flooding
- AOI 24.1, Service Water Malfunction

Con Edison stated that no procedural changes were necessary since operators would be expected to initiate a tagout to isolate any affected service water line to protect other safety-related equipment. The inspector concluded that this was an inadequate response in that procedures should reflect design considerations. Given that testing shows that in some cases both the A and B valves are needed to satisfy the inleakage criteria, then procedures that call for isolation of fan cooler units should reflect this fact. Subsequently, Con Edison issued a procedure change providing specific guidance on fan cooler unit isolation.

LER 50-247/97-025 and Incomplete Reactor Protection System Surveillance

Two other discrepancies with TS surveillances involved: 1) failure to change the acceptance criteria for the "as-found" condition for a coefficient in a reactor protection system setpoint for overtemperature differential temperature value, and 2) a Quality Assurance (QA) surveillance that identified that the daily channel check by operators for the steam generator effluent flowrate occurred without flow in the line; however, the TS requires flow and Con Edison incorrectly concluded that the

daily channel check satisfied the functional test. The latter issue was documented in LER 50-247/97-025. Based on an in-office review, the LER is considered closed; however, the issue raised in this LER with respect to not performing a required TS surveillance, is a broader issue which is being tracked as an unresolved item (URI) as described below. The reactor protection system setpoint surveillance issue is still under evaluation by Con Edison for apparent causes and reportability determination.

c. Conclusions

Con Edison identified three instances of long-standing failures to implement TS surveillance requirements. The causes of these implementation failures were under investigation by Con Edison at the end of the inspection period. The inspectors concluded that these examples, coupled with those identified in past NRC reports, could indicate a programmatic weakness in the TS surveillance program at Indian Point Unit 2. The inspectors note that a comprehensive TS surveillance review was initiated by Con Edison at the end of the inspection period and that the review was added as a restart criterion. The three TS issues discussed in this section are considered an unresolved item (URI). The URI will be updated to reflect additional TS deficiencies identified and reported as part of Con Edison's comprehensive review, at which time the overall result will be reviewed against the NRC's enforcement policy for appropriate action. **URI (50-247/98-001-06)**

M6.1 Work Control and Quality of Maintenance; NCV 50-247/98-001-07and 08

a. Inspection Scope

The inspector reviewed several significant open item reports (OIR) concerning work control during the current outage.

b. Observations and Findings

The inspector reviewed the following Con Edison identified or self-revealing events which were documented in open item reports (OIRs) by Con Edison:

1) On January 28, 1998, a scaffold plank fell and struck the nitrogen supply line to PCV-456 (pressurizer power operated relief valve, train A), that caused relief valve 4105 (relief valve to PCV-456 nitrogen supply line) to lift and stick open. The Technical Specification action statement for one train of the over pressure protection system (OPS) being inoperable was entered and actions were taken to reseat the relief valve. The January 28 event was a repeat occurrence in that in December 1997, personnel inside the VC accidentally struck relief valve 4105 with a ladder causing the relief valve to lift.

2) On November 20, 1997, work was scheduled to be performed on the 22 residual heat removal (RHR) pump and a tagout had been issued which would have rendered the pump inoperable. At the time, 21 RHR pump was

inservice, 22 RHR pump was in standby and the steam generators were unavailable for decay heat removal. Removing the 22 RHR pump from service, therefore, would have caused an inadvertent entry into the technical specification action statement of Table 3.1.A-1 for decay heat removal. No provisions were made in the tagout or work package for an alternate means of decay heat removal. Fortunately, this condition was identified by the control room operators prior to tagging the 22 RHR pump out of service. Con Edison initiated a root cause evaluation for this "near miss" event.

3) On January 26, 1998, an active work permit and tagout for the primary auxiliary building (PAB) ventilation supply fan was cleared in error. Specifically, a tagout was issued to support Instrument and Control (I&C) work permit 36731; however, the tagout was erroneously closed and a tagout restoration permit was issued to clear the associated tags. During the process of restoration from the tagout, an additional job was identified by a system engineer and the tagout was re-opened against a new work permit, 36737. After completion of this new work, work permit 36737 was closed and the tagout was cleared. Two days later, I&C technicians were preparing to start work under permit 36731 and discovered that the associated tagout had been removed. This failure to properly control the tagout for the I&C work permit is a violation of Con Edison's tagout process identified in SAO-105, "Work Permits." This issue was identified as the result of a pre-job tagout verification by Con Edison I&C personnel and was not reasonably preventable by corrective action for a previous violation or licensee finding. Therefore this non-repetitive, licensee-identified and corrected violation is being treated as a **Non-Cited Violation**, consistent with Section VII.B.1 of the NRC Enforcement Policy. **NCV (50-247/98-001-07)**

4) An OIR was submitted by field engineering for improper controls on a generic piping and support modification on all three emergency diesel generators (EDG). At the time, no diesel generators were required to be operable. The modification was initiated to eliminate leaks from the six relief valves on the diesel starting air lines. These leaks were believed to be caused by excessive vibration during EDG operation. The modification was to install flex hoses between the air start motors and the relief valves. After the installation of the flex hoses, additional piping supports were to be installed on both sides of the relief valves. When the field engineer inspected the work in progress, he noticed that some of the piping supports had been installed prior to the flex hoses, which was not per the modification package. Only one EDG at a time was supposed to have been modified. This issue raised concerns regarding the control of this work and oversight of the contractors performing the work which will be addressed by Con Ed as part of the resolution of the OIR in this matter. Therefore this non-repetitive, licensee-identified and corrected violation is being treated as a **Non-Cited Violation**, consistent with Section VII.B.1 of the NRC Enforcement Policy. **NCV (50-247/98-001-08)**

5) OIR 97-E02707 was reviewed which detailed Foreign Material Exclusion (FME) incident, in this case where a ratchet was accidentally dropped into the Service Water Bay in July 1997 during installation of an insulating jacket. The Watch Engineer was notified of this event and this OIR drafted to track the recovery (or disposition) of the ratchet. The licensee evaluated that the ratchet would not impact the service water pumps and could remain until retrieved. The worker was counseled regarding the need to preclude this type of event. The inspectors noted that during this outage, the ratchet was planned to be recovered along with another object found in the SW bay.

c. Conclusions

During the current outage, Con Edison OIRs have documented examples of poor work control and lack of attention to detail with the potential for adverse impact on safety related equipment and on personnel safety. In the case of nitrogen relief valve 4105 for PORV PCV-456, inattention to the work environment rendered one train of the OPS inoperable on two separate occasions and unnecessarily challenged the operators. With respect to the improper release of a tagout to support work on the 22 RHR pump, operators performed well in recognizing that plant conditions could not support removing the 22 RHR pump from service as this would result in a violation of the plant's technical specifications. However, poor operator performance was noted in the improper removal of a protective tagout with an active work order still open. A modification to the EDG air start motor supports was performed out of sequence, raising questions regarding work control and contractor oversight. Collectively, the issues described in these OIRs represent continued problems in the work control process and unnecessary challenges to operators, plant equipment and personnel safety.

M8 Miscellaneous Maintenance Issues

M8.1 Closed: Licensee Event Report 50-247/97-016-00;Auto-Start of Emergency Diesel Generators

This LER was submitted in response to an auto-start of the emergency diesel generators. The event occurred when a mechanic inadvertently tripped breaker 52/2AT3A (bus 2A - 3A tie breaker) which was supplying bus 3A from bus 2A. The mechanic was working on breaker 52/2AT5A which is located directly above 52/2AT3A when his knee hit the manual trip button on the front of the breaker. As a result, bus 3A was de-energized and the three emergency diesel generators started. The inspector performed an in-office review of Con Edison's root cause and corrective actions stated in the LER and assessed that they were appropriate and timely. As stated in the LER, the principal root cause of this event was worker error on the part of the mechanic working in the breaker cubicle compounded by not considering the consequences of potential worker error at the work location during the pre-job briefing. No violation of NRC requirements was identified during this review and the LER is closed.

III. ENGINEERING

E2 Engineering Support of Facilities and Equipment

E2.1 Failure of 23 Emergency Diesel Generator K3 Relay; VIO 50-247/98-001-09

a. Inspection Scope

The inspector reviewed the modification and subsequent failure of the K3 relay and resistor R65 in the 23 emergency diesel generator (EDG) control panel.

b. Observations and Findings

On January 29, 1998, Con Edison was running the 23 EDG for post maintenance testing following work to upgrade the seismic qualification of the EDG. During this upgrade, the K3 relay associated with the field flashing circuit, which was not seismically qualified, was replaced with an equivalent seismically qualified relay. After approximately eight minutes of run time, the operator noticed a burning smell around the control panel. He opened the panel to investigate and saw white smoke inside the panel, at which point he secured the EDG. Later investigation by Con Edison revealed that resistor R65 in series with the K3 relay had burned and that the new K3 relay coil was rated for 120 volts whereas the old relay had been rated for 480 volts.

A determination of equivalency (DOE) and a material substitution authorization (MSAP) were performed in order to replace the K3 relay with a different model. Engineering stated that while preparing the DOE, specifications for the original relay were obtained from name plate data as well as the vendor drawings which had been entered into the Con Edison drawing control system. In addition, Revision 3 to drawing number 59M196-Z3, "EDG exciter voltage regulator," which is generic to all three EDGs, shows a 480/120 volt step down transformer supplying the K3 relay coil; however, this transformer is not installed in the 23 EDG control panel. The inspector noted that revision 2 of this drawing did not show the transformer. The DOE/MSAP incorrectly identified the relay coil as 120 volt vice 480 volt and the replacement relay was specified as 120 volt. Therefore, when the diesel was run, resistor R65 in series with the K3 relay coils overheated and failed.

Con Edison documented this event in Significant Occurrence Report (SOR) 98-E00747. Immediate corrective actions included replacement of the new K3 relay with the original relay and replacement of the R65 resistor. Similar modification packages for the 21 and 22 EDGs were canceled. Con Edison initiated a root cause investigation into this event.

The inspectors noted that Con Edison has a plant restart criterion for the resolution and prioritization of the outstanding backlog of drawing discrepancies and changes. This restart criterion will be inspected by the NRC in the near future.

10 CFR Part 50, Appendix B, Criterion V, states in part, that activities affecting quality shall be prescribed by drawings of a type appropriate to the circumstance and shall be accomplished in accordance with these drawings. Drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to this requirement, an inaccurate drawing contributed to the installation of an improperly sized relay in the control circuitry for the 23 emergency diesel generator. During post-modification testing on January 29, 1998, the relay was damaged and required the diesel to be secured from testing. This is a **VIOLATION** of NRC requirements. **VIO (50-247/98-001-09)**

c. Conclusions

During post maintenance testing, the K3 relay and resistor R65 in the 23 emergency diesel generator control panel failed. This 480 volt relay had been replaced with an incorrect 120 volt model, as part of a seismic upgrade modification, due to an erroneous determination of equivalency and errors in the controlled drawing for the system.

E2.2 Plant Modification Review

a. Inspection Scope (37701)

The inspection scope was to evaluate selected plant modifications to assess the thoroughness of design inputs and, in the case of one modification, to evaluate the quality of its implementation.

b. Observations and Findings

The inspector reviewed two modifications that were in various phases of implementation. The modifications were: FPX-97-12642, 480 Volt Switchgear Room Steam Pipe Break Mitigation, and FPX-97-12794, Main Steam Safety Valves (MS-49A through D) Drain Line Modifications.

Modification FPX-97-12794 involved modification to the main steam safety valve drain lines. Drain piping was installed to meet seismic qualification of the main steam system to preclude adverse room conditions if a failure of the drain line occurred. The inspector concluded that the safety evaluation supporting the 10 CFR 50.59 determination appropriately focused on the impacts of the main steam line break analysis for the steam and feedwater penetration area in the auxiliary feedwater building, and impact during a steam generator tube rupture in relationship to offsite release calculations.

On January 27, 1998, the inspector accompanied the design engineer during a walkdown of the modification using design modification drawing (DMD) 31109-AA. The inspector observed that supports were in the correct location and orientation as compared to the design drawing.

The inspector observed a station nuclear safety committee (SNSC) meeting on January 15, 1998, held to approve modification FPX-97-12642. The purpose of the modification was to mitigate a postulated high energy line break from a steam line (auxiliary steam) located outside the 480 volt switchgear room. This postulated high energy line break was previously mitigated by limiting steam flow from the house service boilers that supply the auxiliary steam line; this modification was intended to permit lifting these steam flow restrictions. The need for restricting steam flow was identified following an earlier review in April of 1997 as part of the corrective actions regarding a problem with fire dampers. (Reference NRC Inspection Report 50-247/97-03 and Enforcement Action 97-191) This would be accomplished by installing shields in front of the intake louvers to the switchgear room to provide pipe whip and radiant heat protection, and modification to actuation circuitry so that generation of a high temperature signal in the switchgear room would cause the intake louvers to close and the exhaust fans to trip off. The SNSC raised questions on the adequacy of the safety review as it relates to controls when Indian Point Unit 3 provides heating steam to Indian Point Unit 2. The conclusion was that administrative controls would exist to prevent the consequences of a main steam line break outside the switchgear room.

The inspector walked down the plant modification plans on January 15, 1998, with the design engineers. The inspector noted in the modification package that the engineers had selected as the most limiting postulated failure mode the mechanical binding of one of the eight intake dampers. The inspector questioned whether mechanical binding on one of the eight intake dampers was truly the most-limiting failure mode, since all of the intake louvers receive motive electrical power for dropping from a common power supply breaker in the vital 120 volt system, and failure of this circuit would preclude automatic actuation of all eight dampers. Con Edison prepared open item report (OIR) 98-E00713 to document this potential vulnerability in the design of the modification and placed a hold on the modification while calculations were reperformed assuming the new postulated failure mode. While revised calculations demonstrated acceptability, Con Edison decided to cancel the modification and instead physically relocated the auxiliary steam line away from the 480 volt switchgear room.

c. Conclusions

Inspector walkdowns of the completed modification for the drain lines to the main steam safety valves concluded installation was as designed. The inspectors concluded that engineering and SNSC reviews of a modification to address a steam line break outside the 480 volt switchgear room were incomplete in that they did not consider the consequences of a potential worst-case single active failure vulnerability involving switchgear room intake louvers used to mitigate the impact of a postulated high energy line break. While subsequent Con Edison analysis concluded that the modification was acceptable, the modification was subsequently canceled after Con Edison decided to reroute the auxiliary steam piping away from the 480 volt switchgear room, thus obviating the need for the originally planned modification.

E8 Miscellaneous Engineering Issues

E8.1 (Closed): Licensee Event Report 50-247/97-020, Technical Specification 3.0.1 Required Plant Shutdown

The details of this issue were previously discussed in NRC inspection report 50-247/97-08 and 50-247/97-09. NRC report 50-247/97-08 issued a notice of violation on the testing controls for pressurizer code safety valves, the response to which will be reviewed and documented in a future NRC inspection. Based on in-office review of the LER, no new issues requiring NRC followup inspection were identified and the LER is closed.

IV. PLANT SUPPORT

R8 Miscellaneous Radiological Protection and Chemistry Issues Miscellaneous

R8.1 Review of Corrective Actions to Apparent Violation Identified in IR 50-247/92-22 (Reference EA 92-245; item 01014)

a. Inspection Scope (IP 92702)

An enforcement conference was held on January 13, 1993, regarding apparent violations identified in NRC IR 50-247/92-22 concerning the Technical Support Center (TSC) ventilation system. By letter dated January 29, 1993, the NRC issued a severity level IV violation and a Notice of Deviation for these issues. The nature of the violation was that Con Edison had an unacceptable program to maintain the TSC habitability, as required by: (1) 10 CFR 50.47(b)(8); and, (2) NUREG-0737, Supplement 1. The nature of the deviation was that the actual design of the TSC ventilation system was different from the design referenced in Section 12.7 of the Updated Final Safety Assessment Report (UFSAR) .

The scope of this inspection was to evaluate the TSC habitability, including the technical evaluations to support closure of the violation and deviation.

b. Observations and Findings

The inspector noted that Section 12.7 of the UFSAR was deleted as per Con Edison's 1993 response letter to the deviation. The inspector reviewed the following documents to determine the adequacy of the TSC habitability requirements: (1) safety evaluation for removing Section 12.7 of the UFSAR; (2) dose assessment results; (3) monitoring capability; and (4) TSC pressure test results. With respect to each of these items, the inspector noted that the safety evaluation was proper, including its evaluation for unreviewed safety questions. Dose assessment results were well below the regulatory requirements of 0.506 Rem whole body dose (limit 5 Rem) and 6.84 Rem thyroid (limit 30 Rem) within 30 days. Con Edison has the capability to measure (monitor) radioiodine activity to lower than

1E-7micro curies per cubic centimeter (μCi/cc). In addition, surveillance test results indicated that the TSC would reach positive pressure within 15 minutes.

The inspector also reviewed the following procedures and determined that Con Edison has an adequate program to maintain and control TSC habitability:

- Section 7.1.7 of the Emergency Plan;
- IP-1035, Technical Support Center;
- IP-1020, Airborne Activity Determination;
- IP-1041, Use of the TRITON for Monitoring Radiogas; and
- PT-EM1, TSC Filtration.

c. Conclusion

Based on the above reviews, the inspector determined that the violation and deviation were closed. **CLOSED (EA 92-245; item 01014)**

R8.2 Review of TSC Filtration System

During the review of surveillance test results for the TSC filtration system, performed under procedure PT-EMI1, "TSC Filtration," the inspector noted that positive pressure would be obtained within 15 minutes per the 1994 test result. However, the inspector noted that the 1996 test was missing the data sheet supporting the test result. Without the data sheet, the inspector could not verify the test result was correctly evaluated, although the procedure indicated that the system met the test acceptance criteria.

R8.3 Closed: URI 50-247/96-004-03

This URI was opened because the containment purge air flow rate defined in the UFSAR is not consistent with actual containment purge air flow rates. Currently, Table 9.8-1 of UFSAR Section 9.8, denotes that the containment air purge system is capable of an air flow rate of 55,500 cubic feet per minute (cfm) at a standard condition. Procedure PT-R28, "Containment Building Air Filtration System," requires a system air flow rate from 27,000-33,000 cfm.

Con Edison evaluated the containment purge air flow rate listed in Table 9.8-1 of the UFSAR. (Note: Table 5.3-1 of UFSAR also defines the containment purge exhaust rate as 55,500 cfm) Con Edison concluded this flow rate was the design value from the original system specification. They also concluded that the flow rates for the roughing filters and high efficiency particulate (HEPA) filters listed in Table 9.8-1, which were 55,500 cfm, were also incorrect because the fan capacity at a standard condition (i.e. just the fan without system additions/resistance) was 55,500 cfm, as stated in the Table 9.8-1.

Adding ducts, vents, and an air cleaning system (roughing and HEPA filters) to the containment purge system reduces the flow rate. Therefore, the containment purge air flow rate should be lower than the capacity based on the standard conditions.

Con Edison determined the potential containment purge flow rate as 40,000 cfm based on previous test results. Con Edison is tracking this item through OIR 96-E02001, which has been identified as having to be resolved prior to restart of the unit from the current outage. As a result of this review the FSAR inconsistency does not constitute a violation of regulatory requirements and this item is closed.

F1 Control of Fire Protection Activities

F1.1 Failure to Observe Fire Watch Requirements; VIO 50-247/98-001-10

a. Inspection Scope (37551)

The inspectors observed "hot work" (i.e. work that generates sparks capable of being an ignition source) activity in the auxiliary feedwater building (AFWB) and noted that grinding was being performed without a fire watch present. The inspectors discussed this observation with Con Edison management; reviewed the results of Con Edison's interviews with the individuals; and reviewed the administrative procedures that govern hot work activities and fire watches at Indian Point 2 as well as the training that personnel receive on them.

b. Findings and Observations

On January 6, 1998, the inspectors toured the AFWB and noted a grinding operation being performed on piping on an elevated platform outboard of a main steam isolation valve. The grinding operation produced a trail of sparks that were observed to be cascading to the floor approximately fifteen feet below where the maintenance worker was performing the grinding. The inspectors noted that a fire extinguisher was stationed at the floor level but did not observe the presence of a fire watch. The individual performing the grinding was the only individual in the area.

The inspectors asked the maintenance worker if he had a fire watch. The individual responded that he did, but when questioned as to where the fire watch was, he appeared surprised that his fire watch was not present. On further questioning by the inspectors, he stated that the fire watch must have just left a few minutes ago, perhaps to get something. He also stated that he was aware of the hot work requirements and that a fire watch should have been present while he was grinding. He assured the inspectors that he would not resume grinding until the fire watch returned.

The inspectors exited the AFWB and informed the Work Week Manager of the incident so that the individual's supervisor could be informed and appropriate actions taken. The issue was also discussed with the Plant Manager with a request that the inspectors be informed of the results of Con Edison's followup interviews with the two individuals involved, the mechanic performing the grinding (grinder) and the fire watch who left the area. The inspectors also obtained the security log printout for the AFWB and noted that the fire watch had left the area approximately fifteen minutes prior to the inspectors questioning the mechanic performing the

grinding activity, which indicated that the fire watch had left more than "just a few minutes ago" as stated by the grinder.

In a subsequent discussion, Con Edison provided the inspectors with a summary of their interviews with the grinder and fire watch. In that summary, the grinder acknowledged that he was aware of the fact that the fire watch had left the area to obtain a special grinding attachment from the tool room.

Technical Specification (TS) Section 6.8.1 requires that written procedures be implemented covering activities referenced in Regulatory (Safety) Guide 1.33, November 1972. Appendix A of Regulatory Guide 1.33 recommends written procedures that govern procedure adherence. Station Administrative Order (SAO)-133, "Procedure, Technical Specification and License Adherence and Use Policy," section 5.1.1, states that procedures shall be followed. SAO-702,"Control of Ignition Sources," Section 4.3.3, states that a fire watch shall be present during all work activities (i.e. work involving an ignition source) and that fire watch responsibilities are delineated in SAO-705, "Fire Watch Tour." Contrary to these requirements, on January 6, 1998, a maintenance worker performed grinding (an ignition source activity) absent the required fire watch, and the fire watch left his assigned post to retrieve a tool from the tool room. This is a VIOLATION of NRC requirements. **VIO (50-247/98-001-10)**

As part of their review of this incident, the inspectors requested information on what training fire watches receive on Indian Point 2 specific fire watch responsibilities as delineated in Station Administrative Order (SAO)-702, "Control of Ignition Sources," and SAO-705, "Fire Watch/Fire Tour," as the assigned fire watches usually work in the non-nuclear side of the Con Edison system and are not permanently assigned to the site. In a subsequent discussion with the Maintenance Manager, the inspectors were informed that fire watches assigned to Indian Point 2 do not receive specific training on SAO-702 or 705 prior to assuming duties as fire watches. They do, however, receive annual training in New York City to obtain/maintain a Certificate of Fitness as a "fire guard." While the training material for this certification encompasses many of the concepts in SAO-702 and 705, there are significant differences between them with the SAO's giving more specific and, in some cases, more restrictive requirements.

The inspectors also questioned whether the supervisors who issue hot work permits per Attachment 1 to SAO-702 are ensuring that the assigned fire watches are familiar with the SAO requirements. Attachment 1 requires the supervisors to initial on a daily basis that the job location has been inspected, the precautions delineated in SAO-702 taken, and a fire watch instructed and posted in accordance with SAO-705.

The inspectors had previously asked questions about step 4.6.6 of SAO-705, Section 4.6, "Continuous Fire Watch Responsibilities." This step applies to fire watches and states that "The Fire Watch personnel shall become familiar with the area requiring fire watch. The pre-fire plans in procedure SE-SQ-12.401 shall be consulted." The inspectors had confirmed through questioning in the field that fire

watches were unfamiliar with the meaning of this step and the inspectors had informed Con Edison management of this observation. The Maintenance Manager subsequently informed the inspectors that SE-SQ-12.401 was an inactive procedure and that a temporary procedure change was issued on February 2, 1998, removing reference to this document from step 4.6.6 of SAO-705. The inspectors concluded that if fire watches or their supervisors were truly following the requirements of SAO-705, then this reference to an inactive procedure would have been identified and questioned earlier and a procedure change implemented. The inspectors also concluded that supervisors were not rigorous in ensuring that fire watches were adequately instructed and posted in accordance with SAO-705 when signing the hot work permit in Attachment 1 to SAO-702.

c. Conclusions

The inspectors concluded that on January 8, 1998, two individuals violated Con Edison's procedures for hot work activity when a worker continued grinding while his fire watch had left his assigned post. The inspectors also concluded that: personnel performing fire watch duties had not received training on the specific Indian Point 2 administrative requirements governing hot work and fire watch requirements; the fire watch administrative procedure was deficient in that it required personnel to perform actions from an inactive procedure; supervisory personnel who initiated and signed hot work permits had not been rigorous in ensuring that fire watch personnel were instructed and posted in accordance with the administrative procedure governing fire watch activities; and plant management was not proactive in identifying and ensuring that expectations regarding the control of hot work and fire watches were met. These conclusions are in line with concerns expressed by the NRC in previous inspection reports regarding problems at Indian Point 2 with lack of procedure adherence, poor procedures, poor supervisory oversight, and plant management expectations not being met and or enforced.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of Con Edison management at an exit meeting held on March 6, 1998. Con Edison acknowledged the findings presented. The inspectors confirmed that none of the materials examined during the inspection period should be considered proprietary.

Attachment 1

INSPECTION PROCEDURES USED

37551	Onsite Engineering
40500	Effectiveness of Licensee Controls in Identifying, Resolving, and Preventing Problems
61726	Surveillance Observation
62707	Maintenance Observation
71707	Plant Operations
71750	Plant Support
90712	In-Office Review of Written Reports of Nonroutine Events at Power Reactor Facilities
92702	Followup on Corrective Actions for Violations and Deviations
92904	Followup - Plant Support

ITEMS OPENED and CLOSED

Opened

URI 50-247/98-001-01	Operability of Service Water Screens without Operable Heaters
NCV 50-247/98-001-02	Logkeeping Practices not in Accordance with Procedures
IFI 50-247/98-001-03	Resolution to Containment Spray Pump Vibration Issue
NCV 50-247/98-001-04	Inadequate Procedure for Turbine Driven Pump Test
NCV 50-247/98-001-05	Inadequate Work Order Guidance for Maintenance Activity
URI 50-247/98-001-06	Missed Technical Specification Test Requirements
NCV 50-247/98-001-07	Improperly Cleared Tagout
NCV 50-247/98-001-08	Inadequate Work Control during Diesel Modification Work
VIO 50-24798-001-009	Drawing Discrepancy Contributing to Error in Installing Improper Relay in Emergency Diesel Generator Control Panel
VIO 50-247/98-001-10	Failure to Follow Fire Watch Requirements

Closed

E 92-245 01014	Unacceptable Program for Technical Support Center Habitability
VIO 50-247/96-001-01	Actions Performed without Approved Procedures
URI 50-247/96-004-03	Containment Purge Rate Different than Design Purge Rate

LER 50-247/97-009	Actuation of Automatic Containment Isolation Valves
LER 50-247/97-016	Automatic Start of the Emergency Diesel Generators
LER 50-247/97-020	Technical Specification 3.0.1 Required Plant Shutdown
LER 50-247/97-025	Inadequate Surveillance Requirements for Radioactive Liquid Effluent Monitoring Instrumentation

Discussed

URI 50-247/96-080-01	Actions on Pressurizer Level During Reactor Trip
IFI-50-247/97-015-02	Reactor Vessel Head Non-condensible Gas Removal

In accordance with recent guidance on the entry of items into the NRC's inspection followup system (IFS), and in accordance with guidance in NRC inspection manual chapter (IMC) 0610, the following items originally designated as unresolved items (URIs) have been redesignated as inspector followup items (IFIs).

URI 50-247/95-013-01
URI 50-247/ 96-080-03, 06, 08, 10 & 13

LIST OF ACRONYMS USED

ABFP	Auxiliary boiler feedwater pump
AFWB	Auxiliary feedwater building
AOI	Abnormal Operating Instructions
CCW	Component cooling water
COL	Check-off list
CSP	Containment Spray Pumps
CW	Circulating water
DMD	Design modification drawing
DOE	Determination of equivalency
EDG	Emergency diesel generators
FCV	Flow control valve
gpm	gallons per minute
HEPA	High efficiency particulate air
INPO	Institute of Nuclear Power Operations
LCV	Level control valve
LER	Licensee event report
MSAP	Material substitution authorization procedure
NCV	Non-cited violation
OAD	Operations Administration Directive
OI	Operating Instructions
OIR	Open item report
OPS	Overpressure Protection System
PAB	Primary auxiliary building
PDR	Public Document Room
PRT	Primary relief tank
QA	Quality Assurance
RG	Regulatory Guide
RHR	Residual heat removal
RVLIS	Reactor vessel level indication system
SAO	Station Administrative Order
SI	Safety injection
SNSC	Station Nuclear Safety Committee
SW	Service water
TS	Technical Specifications
TSC	Technical Support Center
UFSAR	Updated Final Safety Assessment Report
URI	Unresolved item
VC	Vapor containment
VIO	Violation
WANO	World Association of Nuclear Operators
WCPS	Weld Channel Pressurization System