U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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EXECUTIVE SUMMARY

Indian Point 3 Nuclear Power Plant
NRC Inspection Report No. 50-286/97-07

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covered a six-week period of resident inspection.

Operations

The test procedure for the testing of the turbine generator thrust bearing trip was inadequate in that a required step to direct operators to reset the generator lockout relays was omitted. This resulted in an inadvertent trip of the reactor and the turbine generator, and was identified as a violation of NRC requirements. **(VIO 97007-01)** Operator performance during the test of the turbine generator thrust bearing trip was weak and failed to provide an effective barrier to preventing an inadvertent reactor and main turbine generator trip. Operators did not identify the procedural inadequacy when the 86P and 86BU lock relays were directed to be unblocked, although indications were available in the control room that the relays were in the trip condition. Operators did not detect the change in states of the P-10 and P-7 permissives and bistables, which occurred about 12 minutes before the trip. Operators did not effectively control reactor power below the P-10 setpoint for power range nuclear instrument N-41 during the test. (Section O1.2)

The plant operating procedure POP 1.1, "Plant Heatup from the Cold Shutdown Condition," did not provide specific requirements to verify that the overpressure protection system (OPS) was operable, or refer to an existing procedure which provided specific requirements to verify OPS operability. This inadequate procedure, which resulted in the failure of OPS to be operable when required for a period of 13 hours, is the second example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." **(VIO 97001-01)** Operator identification of this problem during the fill and vent of the reactor coolant system was good; however, weak performance by the operations staff relative to the development of protective tagouts failed to identify the impact on OPS during two separate occasions. Also, the operators did not recognize this impact when reviewing the reactor coolant system valve deviation list. (Section O1.3)

The start-up activities from the refueling outage were generally performed well. Good operator control and communications were observed; several examples of good questioning attitude were exhibited by operators; and, good management involvement was maintained. However, inconsistencies with good performance were also identified by the NRC throughout the start up activities. These inconsistencies include: weak operator performance by not detecting and terminating a main feed pump slow ramp up in speed for over two minutes which resulted in a feedwater pump trip, weak corrective actions in response to a degraded ventilation boundary, operator informality with respect to rod movement during the reactor start up, and a weak protective tagout during a maintenance activity. (Section O1.4) The licensee's efforts in reducing the amount of equipment being stored in the containment building was excellent. The closeout inspection procedure was good and was implemented in a thorough manner. (Section O2.1)

Executive Summary (cont'd)

The system operating procedure SOP-TG-4, "Turbine Generator Operation," did not specify that local control was the preferred method for raising turbine speed on the load limiter during the adjustment of the governor impeller orifice. This inadequate procedure, which resulted in the inadvertent trip of the turbine generator on overspeed, is the third example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings." **(VIO 97007-01)** (Section O3.1)

The licensee's modification/procedure update was weak in that checkoff list COL-RW-2, "Service Water System," was not appropriately revised before being used to verify system operability. Checkoff list COL-RW-2 did not reflect the addition of a new valve and the revision of valve positions from an outage modification and an outage test. This inadequate procedure is the fourth example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings." **(VIO 97007-01)** (Section O3.2)

Maintenance

The maintenance personnel's questioning attitude was effective in assuring an appropriate drain path for working on the charging system. However, not including valve HCV-142 in the protective tagout (PTO) initially or adding it as a valve controlled within the maintenance procedure, is deemed a weakness. The maintenance workers were unnecessarily challenged at the work site due to a poor PTO and weak pre-job planning walkdown. (Section M1.2)

The initial performance of several equipment indicated that the outage related activities on those equipment were successful. Notable examples included the excess letdown line repair, charging pump seal modification, pressurizer safety valve upgrade, service water piping replacement, turbine generator vibration reduction and control room upgrade. However, equipment failures were encountered during startup activities, such as the generator bearing oil leak and the relay contact failure, which caused plant transients. (Section M1.3)

Procedure 3PT-5Y4, "32 Auxiliary Boiler Feed Pump (ABFP) Overspeed Test," was not revised to reflect valve position changes implemented by a nuclear safety evaluation before it was used. This inadequate procedure is the fifth example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings." **(VIO 97007-01)** In addition, checkoff list COL-MS-1, "Main and Reheat Steam System," was not revised to reflect the similar changes in valve positions implemented by the same nuclear safety evaluation. (Section M1.5)

The surveillance tests on the weld channel system were performed well. Good communications and questioning attitudes were exhibited throughout the evolutions. The resolution and replacement of the failed air regulator was timely and well coordinated; however, the licensee missed an opportunity to preclude this failure by not blowing down the weld channel air lines after finishing the air receiver maintenance. (Section M1.6)

The performance of the safety injection test reflected excellent preparation and was well implemented. The command and control demonstrated by the control room supervisor and the test coordinator was excellent. (Section M1.7) The new hardware used to perform the safety valve testing provided efficient and technically sound surveillance results while maintaining a safer work environment for those performing the test. The planning and implementation of the surveillance test was good. (Section M1.8)

Engineering

The operator response and equipment response to the plant trip were good. However, the failed contact which caused the trip was another example of equipment impact on the plant operations. The engineering support and actions to address aging relays and contacts were slow as evidenced by limited progress in completing action plan and action commitment tracking system items. (Section E1.1)

The operations and reactor engineer's decision to terminate the startup due to a questionable estimated critical position (ECP) was good, and was consistent with management expectations to stop critical activities when the plant response is not clearly understood or readily explainable. However, engineering preparation for the activity was weak in that the computer estimated critical position was not available and the limitation of the manual ECP on how the inverse count rate ratio plot could vary due to the recent power history was not fully accounted for. (Section E1.2)

System engineering expended a lot of time and effort to verify the service water system hydraulic performance. Overall, the test was well supervised and the technical dispositions for the components which failed to meet the minimum acceptance criteria defined in the procedure, were good. (Section E1.3)

The system engineer's identification of an original construction deficiency was good and exhibited a questioning attitude. A non cited violation was issued because the design deficiency rendered the system incapable of meeting its Technical Specification requirement. (Section E1.4)

Plant Support

The health physics organization performed well by proactively returning much of the plant to a clean and workable status in an efficient manner. (Section R1.1) Control room habitability during off normal conditions is addressed appropriately by plant procedures and operator training. (Section F8.1)

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Summary of Plant Status

At the beginning of this inspection period, the plant was in the cold shutdown condition. A plant startup was performed and the reactor was brought critical on September 8. On September 10, the reactor inadvertently tripped from about 7% power during performance of turbine generator testing. On September 11, the plant was returned to criticality. However, on September 15, the reactor inadvertently tripped from 30 percent power due to a failed relay contact while performing reactor protection system testing. On September 17, the licensee commenced a reactor startup; however, the startup was terminated due to uncertainties associated with the estimated critical position. Another reactor startup was commenced and reactor criticality was achieved on September 19. At the end of the inspection period, the plant was operating at 74% power.

I. OPERATIONS

O1 Conduct of Operations

O1.1 General Comments (71707)

During this inspection period, the inspectors conducted frequent reviews of operational activities with specific focus on plant startup activities, including containment closeout, plant heatup, reactor criticality, main turbine startup and generator synchronization. The plant systems, which were required to be restored and operable for changing modes, were walked down and the associated equipment testing was observed. The systems that were walked down included the emergency diesel generators, the control room ventilation system, the service water system, weld channel penetration pressurization system, isolation valve seal water system, containment spray system and component cooling water system. The licensee conducted the above activities in a satisfactory manner.

O1.2 Reactor Trip During Turbine Generator Mechanical Testing (VIO 97007-01)

a. Inspection Scope (61726, 71707, 93702)

On September 10, at about 3:00 a.m., the main turbine generator and reactor inadvertently tripped during the performance of the testing of the turbine generator thrust bearing trip. The inspectors reviewed operator performance regarding the trip and assessed the licensee's investigation and corrective actions in response to the event.

b. Observation and Findings

At the time of the event, the licensee was performing test procedure 3PT-V06, "Turbine Generator Mechanical Trip Test." Procedure 3PT-V06 includes testing of the thrust bearing oil pressure trip, during which the signals from the generator lockout relays 86P and 86BU are blocked by installing test stabs to prevent an actual trip of the turbine generator. However, during the restoration from this portion of the test, the procedure failed to direct the operators to reset the relays before unblocking 86P and 86BU relay signals. Further, operators did not detect this inadequacy in the procedure, although indications that the relays were still tripped were available in the control room.

When the operators unblocked the relay signals, the main turbine generator tripped, which in turn caused the reactor to trip. The reactor protection system is designed to trip the reactor on a turbine trip. The design feature is anticipatory and protects the reactor on a loss of heat sink. This anticipatory trip is in effect when the at-power permissive (P-7) logic is met. The P-7 logic is met when either one of two turbine first stage pressures are above about 10% or when the nuclear at-power permissive (P-10) logic is met. The P-10 logic is met when the two of four power range nuclear instruments are at 8.5 to 9% power.

During the event, the operators were unaware that the P-7 logic was satisfied and that the reactor will trip upon a turbine trip. About 1:30 a.m., the bistable input to P-10 from power range nuclear instrument (PRNI) N-41 tripped and provided a one of four signal to the P-10 logic. The inspector reviewed the computer data, which indicated that power on PRNI N-41 had peaked at about 8.8%. Since P-10 was set between 8.5% and 9%, the inspector concluded that the trip of the bistable from PRNI N-41 was consistent with the power history as recorded in the computer data. At the time the P-10 input from PRNI N-41 tripped, PRNIs N-42, N-43 and N-44 were indicating about 7.7%, 7.7% and 6.7%, respectively. The operators were cognizant that another bistable trip would satisfy the logic for P-10, which would in-turn satisfy the P-7 logic, and that a reactor trip would occur upon a turbine trip. Therefore, the operators reduced power slightly and maintained power below 7% on the PRNI channels. Based on computer data, about twelve minutes prior to the trip of the turbine, the bistable from PRNI N-42 tripped, which provided the second signal to satisfy the P-10 logic. The indication that P-10 was satisfied was not noticed by the operators. No audible alarms are associated with P-10 and P-7 changing states. As a result, when the main turbine tripped, the reactor also tripped inadvertently.

Procedure 3PT-V06 was revised in April 1997 to include testing the thrust bearing oil pressure trip with the 86P and 86BU relays in test. The procedure change was as a result of an event in June 1995 during which the procedure inadequately reflected a modification to the turbine control oil system to prevent disabling the turbine trip features while operating the overspeed trip mechanism test handle. In response to that event, the licensee had revised the procedure (which actually trips the main turbine generator), and the procedure was successfully performed as recently as February 1997. However, an action commitment tracking system (ACTS) item remained opened for engineering to review a methodology to test the main thrust bearing pressure trip with the relays in test. The procedure change to satisfy the ACTS item was made in April 1997, but the change was inadequate in that it failed to direct the operators to reset the generator lockout relays before unblocking the signal. This inadequate procedure, which resulted in a reactor and turbine generator trip, is a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." **(VIO 97007-01)**

The licensee conducted a post transient review of the event prior to proceeding with plant restart, and was performing a root cause review of the event. Immediate corrective actions taken included correcting the procedural deficiency, configuring the plant computer to alarm when reactor permissives change state, instituting an "at the controls operator" during the subsequent plant startup to maintain an overview of activities in progress, and emphasizing the expectation for the shift technical advisor to maintain a broad perspective of plant operations. Also, the settings and annunciator function associated with P-10 for PRNIs N-41 and N-42 were performed and found to be within specifications. Although the licensee did not determine the specific cause for why PRNI N-42 input to P-10 permissive picked up early, the licensee considered that the subsequent testing and monitoring of the channel provided assurance that the PRNI was operable. The licensee's root cause analysis and long term corrective actions continue to be developed, and an independent transient review group effort had not been finalized.

The inspectors reviewed the circumstances surrounding the event and concluded that the operator performance was weak. The operators did not recognize that the unblocking of the lockout relays would result in a turbine generator trip, although there were indications in the control room that a trip signal was still in for the lockout relays. Also, the operators did not recognize that the P-10 logic was satisfied for twelve minutes before the trip of the turbine generator, although indications for the P-7 and P-10 permissive and bistable lights were available. The operators control of power was weak in that it resulted in the pickup of the input to P-10 from channel N-41 about an hour and a half before the event.

c. Conclusions

The test procedure for the testing of the turbine generator thrust bearing trip was inadequate in that a required step to direct operators to reset the generator lockout relays was omitted. This resulted in an inadvertent trip of the reactor and the turbine generator, and was identified as a violation of NRC requirements. **(VIO 97007-01)**

Operator performance during the test of the turbine generator thrust bearing trip was weak and failed to provide an effective barrier to preventing an inadvertent reactor and main turbine generator trip. Operators did not identify the procedural inadequacy when the 86P and 86BU lock relays were directed to be unblocked, although indications were available in the control room that the relays were in the trip condition. Operators did not detect the change in states of the P-10 and P-7 permissives and bistables, which occurred about 12 minutes before the trip. Operators did not effectively control reactor power below the P-10 setpoint for power range nuclear instrument N-41 during the test.

O1.3 Overpressure Protection System Inoperable (VIO 97007-01)

a. Inspection Scope (71707, 93702)

On August 13, 1997, the licensee identified that two of three pressure inputs to the overpressurization protection system (OPS) were isolated, which rendered the system inoperable. The inspector reviewed the circumstances leading to the inoperability of the system and the licensee's response to the event.

b. Observations and Findings

The licensee was performing fill and vent of the reactor coolant system in accordance with procedure SOP-RCS-9, "Reactor Coolant System Fill and Vent." The procedure required that the reactor coolant system (RCS) be pressurized to support operation of a reactor coolant pump. Control room operators closed the power operated relief valves (PORVs) and commenced pressurizing the RCS by controlling charging and letdown. As RCS pressure increased, the operators noted that the pressure indicators associated with the OPS were not tracking. The operators stopped the pressure increase to investigate the anomalous condition, and identified that two of the three OPS pressure transmitters, PI-413 and PI-433, were not responding. A nuclear plant operator was dispatched to verify the lineup and discovered that the pressure instruments were isolated by a protective tagging order. The operators took immediate corrective action to begin RCS depressurization in order to reopen the PORVs and establish a minimum two square inches vent path. Compliance was achieved about 53 minutes after discovery of the condition.

The OPS system is required to prevent brittle fracture of the reactor coolant system when the vessel is cold. It operates on a two of three logic which has inputs from the reactor coolant system pressures and temperatures. An increased pressure condition would cause a signal to the PORVs to open. The pressure setpoint for the plant temperature at the time of the event corresponded to 410 psig.

The licensee reviewed the event and concluded that the cause was inadequate procedures, in that procedural requirements lacked specifics for verifying OPS operable. The plant operating procedure POP 1.1, "Plant Heatup from the Cold Shutdown Condition," specified the requirement for OPS to be operable or a minimum vent opening be maintained, but did not provide specific requirements to verify that OPS was operable, or reference any existing procedure which provided specific requirements that OPS was operable. This inadequate procedure, which resulted in the failure of OPS to be operable when required, is the second example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." **(VIO 97007-01)**

The licensee identified other contributing causes to this event, including the failure to recognize that OPS was affected when the sample system was being tagged for maintenance. Operations personnel failed to recognize the impact on OPS during the development of protective tagouts on two separate occasions. The impact was not identified by the operations impact review and, as a result, the administrative tracking of a potential limiting condition for operation (PLCO) for OPS was not performed. Also, the operators did not recognize this impact when reviewing the reactor coolant system valve deviation list.

The licensee formed a post transient review group to review the event. Corrective actions included developing and verifying specific checkoff lists which verify the operability of required functions which cross multiple system boundaries. The system operability checklist was enhanced for closing all limiting conditions for operations and potential limiting conditions for operations. A shift order was issued to clarify expectations for verifying operability, and the licensee planned to include the event into licensed operator requalification training.

The inspector considered the safety consequence of the event to be minimal due to the short time in which OPS was inoperable, and the maximum RCS pressure during the event of 90 psig. The OPS system was inoperable for about 13 hours. During this event, one of the pressure instruments was still operable, and would have provided an alarm function to notify the operators of an impending RCS overpressurization condition. The residual heat removal system was in service with a relief capability for the capacity of three charging pumps. The safety injection pumps were in trip pullout, and the RCS pressure was below the minimum required for starting a reactor coolant pump.

c. <u>Conclusions</u>

The plant operating procedure POP 1.1, "Plant Heatup from the Cold Shutdown Condition," did not provide specific requirements to verify that the overpressure protection system (OPS) was operable, or refer to an existing procedure which provided specific requirements to verify OPS operability. This inadequate procedure, which resulted in the failure of OPS to be operable when required for a period of 13 hours, is the second example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." **(VIO 97001-01)**

Operator identification of this problem during the fill and vent of the reactor coolant system was good; however, weak performance by the operations staff relative to the development of protective tagouts failed to identify the impact on OPS during two separate occasions. Also, the operators did not recognize this impact when reviewing the reactor coolant system valve deviation list.

O1.4 Plant Startup Activities

a. <u>Inspection Scope (71707, 93702)</u>

The inspectors observed several activities associated with the completion of the refueling outage. These include portions of four reactor startups, heatup activities, low power physics testing, mode changes and secondary plant equipment start up.

b. Observations and Findings

Several examples of effective management oversight, good communications, good questioning attitudes and good operator performance were observed as the plant was returned to service after the refueling outage.

- Activities affecting plant start up had a high level of management interaction as evidenced by thorough special evolution briefings and assessment hold point meetings.

- Three part communication was used throughout the start up and plant heat up evolutions. Additional reactor operators were brought on shift in order to have personnel dedicated to observing reactivity changes, maintaining steam generator level with the feedwater controller in manual and bringing secondary plant components on line.

- The operators exhibited good questioning attitudes throughout the startup process. For example, the radiation monitor (R12) was noted to be trending up while the reactor operator was performing a final board walkdown to commence rod withdrawal. All control room start up activities were suspended until clear resolution of the issue was made. Also, prior to pulling rods, the control room personnel discovered that the RCS leak rate calculation far exceeded its previous value and was near its administrative limit. Again, all startup preparations were suspended until the operations personnel could investigate this matter.

  Operator performance during startup activities; however, was not consistently good, as evidenced by NRC identified performance weaknesses and informality.

- On September 9, the main boiler feed pumps (MBFPs) tripped while the plant was at 3 percent power. The 31 and 33 auxiliary boiler feed pumps (ABFP) started appropriately in response to the trip. Instrument and controls department (I&C) performed troubleshooting on the 32 MBFP controller and identified that there was a slight rub between the toggle switch and the controller face plate. I&C believed that the rubbing could cause the toggle switch to stick in the slow raise position. The licensee replaced the controller.

  The inspector reviewed the computer data for the speed curves for the MBFPs and noted that the 32 MBFP had ramped up in speed for about two minutes and fifteen seconds, from 3700 rpm to 4700 rpm, before causing both pumps to trip on high discharge pressure. The toggle switch is normally bumped to raise the speed slowly. The inspector considered it likely that the speed increase was initiated by an operator bumping the toggle switch to speed up the pump. Consequently, the inspector considered that operator performance was weak in not detecting and terminating the slow ramp up in speed over two minutes.

- On August 30, 1997, the inspector observed that a panel to the 32 control room ventilation system was held in place with six untightened screws. A one-quarter inch gap was observed between the panel and the heads of the screws. The remaining 18 screws were lying adjacent to the ventilation unit. The inspector informed the operations shift of this discrepancy. However, the operations shift indicated that the deficiency was identified about a week ago, when the plant was in the cold shutdown condition. At the time of the inspector's identification, the plant had transitioned to the hot shutdown condition. The operations shift further indicated that "maintenance" was asked to reinstall the panel properly.

  The inspector considered that the operations shift resolution of the deficiency prior to the NRC's identification was weak. The identification of the deficiency was informal and not effectively tracked. Although subsequent evaluation by the licensee indicated that the panel deficiency did not impact operability, the inspector considered that the ventilation boundary was in a degraded condition, for which the licensee's corrective actions were weak.

- On September 6, 1997, while preparing for reactor criticality, with all rods out, the operators were asked by the reactor engineer to insert control bank "D" three steps from its fully withdrawn position, and then return it to the fully withdrawn position in order to verify the rod movement indicator in the reactivity computer was working. The rod motion was not recorded in the operators log book nor was it included in any of the start up or reactor physics procedures.

  Although the initial start up was dependent on dilution rather than control rods for criticality, the inspector considered the operators informality and lack of documentation of the rod movement a weakness. Additionally, communication between the shifts was not clear as to which rods would be used to verify the operation of the reactor physics computer's rod movement detector.

- A protective tagout for the maintenance on the charging line containment isolation valve (CH 226) was weak as it was unclear to the maintenance workers performing the job if whether the system which is normally at 2000 psig was properly vented and drained. When preparing to disconnect the valve actuator, the workers noticed that although a vent line had been tagged open, there was a closed valve between the vent and the portion of the system to be worked. Section M1.2 of this report documents additional observations of this maintenance activity.

c. <u>Conclusions</u>

The start-up activities from the refueling outage were generally performed well. Good operator control and communications were observed; several examples of good questioning attitude were exhibited by operators; and, good management involvement was maintained. However, inconsistencies with good performance were also identified by the NRC throughout the start up activities. These inconsistencies include: weak operator performance by not detecting and terminating a main feed pump slow ramp up in speed for over two minutes which resulted in a feedwater pump trip, weak corrective actions in response to a degraded ventilation boundary, operator informality with respect to rod movement during the reactor start up, and a weak protective tagout during a maintenance activity.

O2 Operational Status of Facilities and Equipment

O2.1 Containment Building Walkdown

a. Inspection Scope (71707)

The inspectors performed an independent walkdown of the containment prior to closeout to assess containment cleanliness and housekeeping after RO9. The inspectors accompanied the licensee while they performed the final pre-criticality containment closeout, reviewed the resolution of licensee identified deficiencies and reviewed the seismic evaluations performed by engineering for certain components left in the containment building.

b. Observations and Findings

Overall, the cleanliness of the containment building was excellent. The inspector noted that there was considerably less equipment being stored in the containment building at the end of this outage compared to last operating cycle. The initial walkdown conducted by the inspectors looked for loose material, completeness of system restorations (i.e. tagouts removed, valves restored to safe operations position), and the overall cleanliness of those areas where there was a lot of maintenance activity during the outage. The inspectors then accompanied the licensee on their final walkdown and observed good questioning attitudes, thorough searches throughout containment for any loose material as well as detailed inspections of systems to ensure no piping or component deficiencies remained. After the walkdowns were completed, the inspectors reviewed the completed SOP-CB-2, "Containment Closeout Prior to Exiting Cold Shutdown," attachment and determined that the checkoff list for exceeding cold shutdown was thorough and those items which required technical disposition were addressed in a timely manner. The inspector also determined that the seismic evaluations were technically sound.

c. Conclusions

The licensee's efforts in reducing the amount of equipment being stored in the containment building was excellent. The closeout inspection procedure was good and was implemented in a thorough manner.

03 Operations Procedures and Documentation

03.1 Inadvertent Overspeed Trip of the Turbine Generator (VIO 97007-01)

a. Inspection Scope (62707, 71707)

The inspector reviewed the circumstances surrounding the trip of the turbine generator on overspeed on September 12.

b. Observations and Findings

The licensee was performing adjustments of the governor impeller orifice in accordance with system operating procedure SOP-TG-4, "Turbine Generator Operations." The procedure directed the operators to place turbine speed control on the load limiter during the adjustment. This was performed by decreasing the load limiter until it took control of turbine speed and lowering turbine speed below 1800 rpm. The governor control was then raised above the 1800 rpm setting, so that the load limiter could be used to raise speed back up to 1800 rpm. However, when the operators raised turbine speed using the load limiter, the turbine speed exceeded 1800 rpm and inadvertently tripped on overspeed at about 1832 rpm.

Procedure SOP-TG-4 did not provide specific guidance on how to slowly raise turbine speed using the load limiter. The operators used the load limiter control switch in the control room. However, the system engineer indicated that the vendor recommended using the local load limiter control because local control provided finer control of turbine speed. The inspector concluded that the procedure was not appropriate to the circumstances in that the use of the local control was not specified as the preferred method for raising turbine speed during the adjustment. This inadequate procedure, which resulted in the inadvertent overspeed trip of the turbine generator, is the third example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings." **(VIO 97007-01)**

c. Conclusions

The system operating procedure SOP-TG-4, "Turbine Generator Operation," did not specify that local control was the preferred method for raising turbine speed on the load limiter during the adjustment of the governor impeller orifice. This inadequate procedure, which resulted in the inadvertent trip of the turbine generator on overspeed, is the third example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings." **(VIO 97007-01)**

O3.2 Service Water Checkoff List (VIO 97007-01)

a. Inspection Scope (71707)

On August 28, 1997, the service water checkoff list, which the licensee performed to establish the operability of the system, did not reflect a modification and an engineering test performed during the refueling outage. The inspector reviewed the circumstances surrounding this deficiency.

b. Observations and Findings

During a walkdown of the service water system, the inspector identified that the throttle position of the fan cooler motor service water outlet return valves were not revised to reflect ENG-281, "Service Water System Flow Balance Test." Also, the inspector identified that an outage modification which added a cross-connect isolation valve in the control room air conditioning system was not included in the checkoff list. The inspector noted that the new cross-connect valve, if closed, would isolate service water to a train of control room air conditioning. However, the valve was in the required open position.

When the inspector brought these deficiencies to the attention of the licensee, the procedure revision group showed the inspector a revised service water procedure scheduled to be approved by the plant operations review committee. In this event, the procedure change process failed to ensure that required procedure changes were made before the procedure was used to line up the system for operability. The licensee also indicated that other weaknesses in the engineering and operations interface for ensuring that procedures were appropriately revised were encountered during the outage. As a result, the license initiated deviation event report 97-2255 to address this issue.

The inspector determined that the modification/procedure update process was weak, in that checkoff list COL-RW-2 was not revised appropriately before being used to lineup the system for operability. As a result, checkoff list COL-RW-2 was inadequate in that it did not reflect a new valve and new valve positions from an outage modification and an outage test. This inadequate procedure is a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." **(VIO 97007-01)**

c. Conclusions

The licensee's modification/procedure update was weak in that checkoff list COL-RW-2, "Service Water System," was not appropriately revised before being used to verify system operability. Checkoff list COL-RW-2 did not reflect the addition of a new valve and the revision of valve positions from an outage modification and an outage test. This inadequate procedure is the fourth example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings." **(VIO 97007-01)**

O8 Miscellaneous Operations Issues (92700, 92901)

O8.1 (Closed) Inspector Follow Item 50-286/97004-01: Switchyard control issues in response to a loss of offsite power event. On June 16, 1997, while in the refueling condition, the 138 kilovolt off-site power supply to the plant was lost due to improper grounding activity in the Buchanan switchyard. The Buchanan switchyard is owned and operated by Con Edison non-nuclear switchyard operators. NRC inspection report 50-286/97-04 discussed this event, however at the time the corrective actions for switchyard control had not been established. The root cause of the event was determined to be personnel error on the part of the switchyard operators, however the inspector considered that NYPA oversight of switchyard activities was not effective in averting the event. In response to this event, a joint team of NYPA and Con Edison representatives were tasked to evaluate current policies and procedures that control switchyard activities and make recommendations to improve overall control and coordination of switchyard operations to prevent problems in the future. The joint committee developed seven recommendations for improving switchyard activities. Included in the recommendations is a provision to create a Plant Transmission Activities Coordinator (PTAC) who would be responsible for serving as the single point of contact for planning and coordinating activities affecting the station. Additionally, a revision was made to the Memorandum of Understanding (MOU) between Con Edison and NYPA to include notification of the PTAC when scheduling outages on transmission facilities that affect the station and training switchyard personnel on the new MOU. A Con Edison Outage scheduling system would be installed at the NYPA station to facilitate the PTACs review of outage requests. The inspectors reviewed the licensee's corrective actions in response to this event and have concluded that they are adequate in exercising a more proactive role in switchyard evolutions. This item is closed.

O8.2 (Closed) Licensee Event Report (LER) 50-286/97017: overpressurization protection system inoperable due to inadequate procedure. This event is described in section O1.3 of this inspection report and is one of several examples of inadequate procedures.

II. MAINTENANCE

M1 Conduct of Maintenance

M1.1 General Comments (62707)

The inspectors observed all or portions of the following work activities:

WR 95-03858-19, Replace Logic Relay SW6X for 36 Service Water Pump
WR 96-07254-11, Retest 318 Emergency Diesel Generator Room Exhaust Fan
WR 97-03347-00, Clearing the Line Between CH-332 and CH-333
WR 97-04097-03, Reactor Coolant Pump Seal Injection Valve Test
WR 97-04459-00, Weld Channel Zone 2 Nitrogen Supply Header Check Valve
WR 97-04166-07, Blowdown Weld Channel Piping
WR 97-04475-02, Replace Acoustic Monitor Accelerometer Cable
WR 97-04466-00, Charging Return Valve Pressure Seal Replacement

The inspectors observed that the work performed under the above work requests (WR) was conducted satisfactorily and in accordance with applicable maintenance and administrative procedures. The inspector also reviewed significant equipment failures that occurred, such as the pressurizer pressure relay contact failure (Section E1.1), to determine whether the equipment was within the scope of the maintenance rule, whether the licensee's corrective actions were appropriate, and whether the licensee was appropriately monitoring equipment performance. The inspector concluded that the licensee was properly implementing the maintenance rule for these equipment failures.

M1.2 Repair of MOV 226 Pressure Seal Leak

a. Inspection Scope (62707)

The inspectors observed the disassembly of the charging line containment isolation valve (MOV-226) in preparation for repair of the pressure seal which prohibits body to bonnet leakage. Additionally, the inspector reviewed the work package and protective tagout with respect to station procedures.

b. Observations and Findings

While observing the maintenance on valve MOV-226, the inspector noted that the vent path established by protective tagout (PTO) 97-2367 could not be used as such because there was a closed charging hand control valve (HCV-142) between the valve to be worked and the vent line. Knowing that the piping usually contained 2000 psi, the maintenance personnel requested that valve HCV-142 be stroked open to assure that no high pressure liquid was trapped in the isolated piping. A maintenance contractor telephoned the control room to request that the valve be cycled prior to performing the work, operations stroked the valve and the work continued per the maintenance package. The inspector considered the controls for the work to be weak, because valve HCV-142 was not included in the PTO nor noted in the maintenance work package. Further, the pre-job maintenance walkdown did not identify this deficiency. As a result, the maintenance worker was the last barrier for ensuring the system was properly vented and drained, before beginning intrusive work.

c. Conclusion

The maintenance personnel's questioning attitude was effective in assuring an appropriate drain path for work on the charging system. However, not including valve HCV-142 in the protective tagout (PTO) initially or adding it as a valve controlled within the maintenance procedure, is deemed a weakness. The maintenance workers were unnecessarily challenged at the work site due to a poor PTO and weak pre-job planning walkdown.

M1.3 Observations Regarding Equipment Performance

a. Inspection Scope (62707)

The inspectors assessed the equipment performance during startup activities from the RO9 refueling outage.

b. Observations and Findings

The inspector noted several indications that maintenance completed during this outage was effective. Specifically, the licensee repaired and successfully used the excess letdown line. The modified seals for the 31 and 32 charging pumps were performing well; the 33 charging pump seal remained to be modified. Turbine generator and lead box vibrations were reduced. The control room upgrade modification and service water piping replacements were notable improvements. The generator voltage regulator repair and pressurizer safety valve upgrades appeared effective.

During the startup, there were several equipment failures which impacted plant operations. The equipment failures included the excessive leakage from the main generator exciter bearing, which resulted in a manual trip of the turbine generator. Also, the pressurizer pressure relay contact failure caused an inadvertent reactor trip. Other equipment failures noted during startup activities included the failure of the nitrogen supply containment isolation valve, the 33 static inverter, containment vent valves, 35 service water pump and control room air conditioning compressor trips. The licensee took or was taking appropriate actions to address these deficiencies.

c. Conclusions

The initial performance of several equipment indicated that the outage related activities on those equipment were successful. Notable examples included the excess letdown line repair, charging pump seal modification, pressurizer safety valve upgrade, service water piping replacement, turbine generator vibration reduction and control room upgrade. However, equipment failures were encountered during startup activities, such as the generator bearing oil leak and the relay contact failure, which caused plant transients.

M1.4 Surveillance General Comments (61726)

The inspectors observed all or portions of the following surveillances:

- 3PT-5Y4, 32 Auxiliary Boiler Feed Pump Turbine Overspeed Test
- 3PT-M01, Nuclear Power Range Channels Functional Test
- 3PT-M79A, 31 Emergency Diesel Generator Functional Test
- 3PT-R03D, Safety Injection Test
- 3PT-R06A, Main Steam Safety Valves Setting Test
- 3PT-R09, Leakage Test of Weld Channel Zone 2
- 3PT-R136, Weld Channel Backup N_2 Supply Test
- 3PT-V12, Nuclear Power Range Permissive and Low Range Trip Functional Test

The licensee conducted the above surveillances appropriately and in accordance with procedural and administrative requirements. As applicable, good coordination and communication with the operations department was observed during performance of the surveillance. Procedures supported the timely completion of the surveillance. In particular, the operations test group were knowledgeable about the weld channel system and were proactive in identifying potential failures of the system (Section M1.6).

M1.5 Auxiliary Boiler Feed Pump Overspeed Test (VIO 97007-01)

a. Inspection Scope (62707)

On August 27, 1997, the inspector observed the performance of surveillance procedure 3PT-5Y4, "32 Auxiliary Boiler Feed Pump (ABFP) Turbine Overspeed."

b. Observations and Findings

The inspector observed the test being performed appropriately with good adherence to procedures. The procedure adequately verified the operational readiness of the governor, overspeed trip mechanism, throttle valve and stop valve.

During the performance of the test, the inspector noted that the operator positioned drain valve MS-112-1 in the crack open position per procedure 3PT-5Y4. However, nuclear safety evaluation (NSE) 97-3-365-MS, "32 ABFP Turbine Drains," revised the normal position of valve MS 112-1 from cracked open to closed. Similarly, the NSE revised normal position for strainer blowdown isolation valves MS-111 and MS-113 from cracked open to shut.

The inspector discussed this observation with the shift manager, who acknowledged the problem and indicated that operations identified that checkoff list COL-MS-5, "Main Steam Traps," was not revised to reflect the new position of valve MS-112-1 during the previous day. The licensee initiated deviation event report 97-2194 to address this procedural issue. Subsequent to the identification of the inadequate surveillance test, the inspector also identified that procedure COL-MS-1, "Main and Reheat Steam System," also did not reflect the change in position of these valves.

The inspector concluded the revision of affected procedures to reflect valve position changes for the ABFP drain valves was not effectively implemented. Although the licensee identified that COL-MS-5, "Main Steam Traps," was not appropriately revised, the inspector identified that procedure 3PT-5Y4 was not revised as required before the procedure was used. Further, the inspector identified that COL-MS-1, "Main and Reheat Steam System," also was not revised as a result of the NSE. Checkoff list COL-MS-1, however, was not used during this inspection period. Test procedure 3PT-5Y4 was inadequate in that the procedure did not specify the correct valve position of the drain valve MS-112-1. This inadequate procedure is a violation of 10 CFR 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings." **(VIO 97007-01)**

c. Conclusions

Procedure 3PT-5Y4, "32 Auxiliary Boiler Feed Pump (ABFP) Overspeed Test," was not revised to reflect valve position changes implemented by a nuclear safety evaluation before it was used. This inadequate procedure is the fifth example of a violation of 10 CFR 50, Appendix B, Criterion V, "Instruction, Procedures, and Drawings." **(VIO 97007-01)** In addition, checkoff list COL-MS-1. "Main and Reheat Steam System," was not revised to reflect the similar changes in valve positions implemented by the same nuclear safety evaluation.

M1.6 Weld Channel Leakage Verification Testing and Pressure Regulator Failure

a. Inspection Scope (61726, 62707)

The inspectors observed several surveillance procedures performed on both the air and nitrogen portions of the weld channel system. Additionally, the inspectors observed the troubleshooting of a pressure regulator that failed to maintain pressure within the acceptance criteria of the system operating procedure.

b. Observations and Findings

The weld channel containment penetration pressurization system (WCCPPS) consists of four independent zones which provide air at above accident pressure to piping penetration welds to preclude leakage during an accident. Each air zone has a separate nitrogen zone which serves as a backup pressure supply in the event of an air system failure. The inspector observed the implementation of 3PT-R09, "Leakage Test of Weld Channel Zone(2)," and 3PT-R136, "Weld Channel Backup N_2 Supply Test." The inspector noted excellent communications between the nuclear plant operators (NPOs) and the control room throughout the performance of the weld channel surveillance tests. The NPO's were knowledgeable regarding past performance of the system during these tests and sensitive to fluctuations in pressure indications due to the pressure regulators. This knowledge enabled several faults to be identified prior to system restoration.

While attempting to return Zone 3 to service, an air pressure regulator failed to maintain pressure within the acceptable range. Subsequently, the system was declared inoperable. Upon further investigation, the licensee performed a blowdown of each of the air lines upstream of the pressure regulators to determine if the air lines contained debris that could be getting into the regulators. The blowdown revealed that there was in fact particulate in the lines. The debris was likely to be from work that was performed on the air receiver tanks earlier in the outage in which the inside of the tanks was cleaned. The regulator was replaced and the system was restored to an operable status. The inspector noted that the air lines should have been blown down after the maintenance on the air receiver tanks to preclude any further regulator failures. The sensitivity of these regulators to particulate clogging was not a new issue and several corrective action steps to prevent future failures had been developed. However, due to the large amount of work planned for RO9, this worked was delayed until RO10.

c. Conclusions

The surveillance tests on the weld channel system were performed well. Good communications and questioning attitudes were exhibited throughout the evolutions. The resolution and replacement of the failed air regulator was timely and well coordinated; however, the licensee missed an opportunity to preclude this failure by not blowing down the weld channel air lines after finishing the air receiver maintenance.

M1.7 Safety Injection Test

a. Inspection Scope (61726)

The inspectors observed the performance of procedure 3PT-R003C, "Safety Injection Test, Train 1 and Train 2."

b. Observations and Findings

The licensee implemented the surveillance well. The command and control demonstrated by the control room supervisor and the test coordinator was excellent. Responsibilities between the two were clearly delineated, with the test coordinator controlling the performance of the test and the control room supervisor maintaining oversight and control of safe plant operations. In addition, the pre-evolution briefing was conducted well, and the procedural adherence to the test was good. Equipment problems identified through the test were appropriately addressed.

c. Conclusions

The performance of the safety injection test reflected excellent preparation and was well implemented. The command and control demonstrated by the control room supervisor and the test coordinator was excellent.

M1.8 Main Steam Safety Valves Set Point Test

a. Inspection Scope (61726)

The inspectors reviewed the procedure and vendor information for the new testing device for testing the main steam safety valves. Also, the inspectors attended the pre-job brief, observed a portion of the equipment set up and observed the testing of several main steam safety valves.

b. Observations and Findings

The inspectors observed the implementation of 3PT-R06A, "Main Steam Safety Valves Setting Test Using Set Pressure Verification Device." A special evolution brief was conducted in the control room prior to performing the test. The operations manager emphasized personnel safety and retaining a questioning attitude throughout the duration of the procedure. All twenty main steam (MS) valves were stroked such that either the as-left or as-found set point was verified via the set pressure verification device (SPVD). Ten of the twenty valves had maintenance performed on them during RO9. This was this first time the SPVD test device was used at IP3. It lifts the valve off its seat using air pressure and once a load cell verifies the seat has changed position, the air pressure is immediately eliminated and the spring returns the disk to its seat. The quick action of this mechanism precludes a large disk travel distance and consequently reduces the time the disk is off the seat. By reducing the disk travel distance and time, the valve is less likely to fail open during testing. The performance engineer responsible for the test provided good contractor oversight and support to the operations personnel assisting in the test.

c. Conclusions

The new hardware used to perform the safety valve testing provided efficient and technically sound surveillance results while maintaining a safer work environment for those performing the test. The planning and implementation of the surveillance test was good.

III. ENGINEERING

E1 Conduct of Engineering

E1.1 Reactor Trip Due to a Failed Contact Relay

a. Inspection Scope (93702, 71707, 62707)

On September 15, at 4:29 p.m., the reactor tripped from 30% power while performing testing of the pressurizer pressure instrumentation. The inspector reviewed the licensee's investigation and actions in response to this event.

b. Observation and Findings

The licensee determined that the trip was caused by a high resistance contact in a BFD-66 relay in the reactor protection system (RPS). The high resistance in this normally closed contact provided one of two required low pressurizer pressure inputs to trip the B reactor trip breaker. However, no other indications were available to indicate that this condition existed because the other contacts associated with the relay operated properly. When the low pressurizer pressure bistable of another RPS channel was placed in the trip condition for the surveillance test, the second low pressurizer pressure input satisfied the RPS two of four logic and caused the B reactor trip breaker to open. The plant was placed in the hot shutdown condition without complication, and plant equipment performed as required in response to the trip.

The inspector reviewed the licensee's corrective action in response to the event. A post transient review group was convened to review the event. The group thoroughly identified actions required to return the unit to service. The sequence of events printout was reviewed and confirmed the proper operation of equipment. An extent of condition review was performed to verify the continuity of reactor protection matrix relay contacts. The monthly surveillance test was clarified to ensure that the potential for a high resistance contact causing a trip signal can be identified before it causes a trip during quarterly surveillance testing.

The inspector reviewed the past history of relay and contact performance at Indian Point 3. Several relay failures had occurred in the past, in part due to aging of these relays. However, the inspector noted that deviation event reports (DERs) in 1995 documented failures of relays which resulted in the initiation of technical specification shutdowns. These DERs (DERs 95-1735 and 95-2028) and system health reviews resulted in the development of instrument & controls engineering and system engineering action plans, and the initiation of several action commitment tracking items. However, the inspector considered the licensee efforts regarding implementing these actions slow. Many of the actions, although identified in late 1995 and early 1996, have not yet been completed. These actions included evaluating replacement of safety related relays, developing a preventive maintenance program and prioritizing relay replacements.

c. Conclusions

The operator response and equipment response to the plant trip were good. However, the failed contact which caused the trip was another example of equipment impact on the plant operations. The engineering support and actions to address aging relays and contacts were slow as evidenced by limited progress in completing action plan and action commitment tracking system items.

E1.2 Estimated Critical Position

a. Inspection Scope (37551, 71707)

On September 18, 1997, a reactor startup was commenced but was subsequently terminated due to a question concerning the accuracy of the estimated critical position. The inspector reviewed the circumstances surrounding this event.

b. Observations and Findings

During the reactor startup on September 18, the operators and the reactor engineer noted that the inverse count rate ratio (ICRR) plot appeared to be extrapolating to a critical rod position in excess of one hundred steps above the estimated critical position. Although the reactor engineer understood that the ICRR plot was conservative and would begin plotting closer toward the estimated critical position, the operators and reactor engineer raised concern that the plot was significantly different than the plot performed for the reactor startup on September 11. Specifically, the reactor engineer noted that the ICRR plot was outside the plus or minus fifty step band of the estimated critical position, and that a second doubling had not yet occurred with Bank D control rods at 27 steps. In the September 11 curve, the second doubling had been reached before Bank D control rods were withdrawn and the curve was much closer to the plus or minus fifty step band. As a result, the operators reinserted the control rods and terminated the startup.

The licensee's investigation concluded that the estimated critical position (ECP) was performed correctly. However, because the computer ECP was not available due to a software problem, the manual ECP was used. Procedurally, the computer ECP is preferred because it is more accurate. The recent power history of the plant before the shutdown is modelled by the computer more accurately, whereas the manual ECP provided a grosser and typically more conservative number. After the event, the computer ECP was brought back in service and indicated about a 31 step higher ECP. A computer generated ECP by another computer model, which was not yet validated, was used for the September 11 startup for information and indicated an ECP that was about 17 steps lower than the manual. The difference of about 48 steps contributed to the difference in the exaggerated differences between the two manual ECPs for September 11 and 18.

The inspector considered that the reactor engineer and operator decision to terminate the startup was a good decision, and reflected management expectations to stop critical activities when the responses are not clearly understood or immediately explainable. Nevertheless, the inspector considered engineering preparation for the activity was weak, because the computer ECP was not available and the limitation of the manual ECP on how the ICRR could vary due to the recent power history was not fully accounted for.

c. Conclusions

The operations and reactor engineer's decision to terminate the startup due to a questionable estimated critical position (ECP) was good, and was consistent with management expectations to stop critical activities when the plant response is not clearly understood or readily explainable. However, engineering preparation for the activity was weak in that the computer estimated critical position was not available and the limitation of the manual ECP on how the inverse count rate ratio plot could vary due to the recent power history was not fully accounted for.

E1.3 Service Water Flow Balance Test

a. Inspection Scope (37551)

The inspector reviewed the results of the service water system flow balance test (ENG 281B) and the engineering evaluations for the data points that failed to meet the acceptance criteria contained in the procedure.

b. Observations and Findings

During the RO9 outage, over 2000 feet of piping was replaced in the service water system. This modification along with questions raised by a recent NRC audit prompted the licensee to perform a system-wide service water flow balance test.

The 34 and 35 fan cooler units (FCUs) were unable to meet the test acceptance criteria during the injection phase of an accident and the 32, 34, 35 FCUs were in question during the recirculation phase of an accident. Additionally, the 31 emergency diesel generator cooler did not meet the acceptance criteria in the procedure. NYPA contracted with Raytheon engineering to perform flow balance calculations using data obtained from the initial phases of the test. New pump performance curves were developed and in conjunction with the service water system hydraulic model, individual FCU flows were determined. Additionally, the emergency diesel generator cooler unit flows were numerically evaluated and determined to be above the minimum acceptance criteria. The inspector considered the vendors basis for the calculations and assumptions to be technically sound.

c. Conclusions

System engineering expended a lot of time and effort to verify the service water system hydraulic performance. Overall, the test was well supervised and the technical dispositions for the components which failed to meet the minimum acceptance criteria defined in the procedure, were good.

E1.4 Weld Channel Airlock Design Deficiencies (NCV 97007-02)

a. Inspection Scope (37551)

The inspector reviewed the licensee's actions with respect to the identification and resolution of an original construction deficiency associated with the weld channel air supply to the containment airlocks was performed.

b. Observations and Findings

While performing several upgrades to the 80 ft. and 95 ft. airlocks, three deficiencies were discovered which revealed that this portion of the weld channel containment penetration pressurization system (WCCPPS) was incapable of performing its intended function. First, the flat gasket which fits between the airlock bulkhead mounting flanges and the shaft seal housings did not have the appropriate holes to allow the air from the WCCPPS to be supplied to the shaft seals for both the 80 ft. and 95 ft. airlocks. Secondly, the equalizing ball valve flanged joints were identified to not have WCCPPS air supplied to them. These deficiencies were attributed to original plant construction errors. The correct gasket design was installed and tested on all 80 ft. and 95 ft. shaft seal housing flanges and the equalizing ball valve flanged joints. Lastly, a 3/4" pipe penetration associated with the inner door seal pressure gauges did not have WCCPPS supplied to the containment boundary welds. This occurred as a result of a modification which was revised without specific consideration of the weld channel configuration requirements. The licensee connected these containment boundary welds to the WCCPPS system prior to exiting the RO9 outage. Technical Specification (TS), section 3.3D requires that all portions of the weld channel system be operable when the reactor is above the cold shutdown condition. Contrary to the above, the deficiencies identified precluded the system from satisfying the TS requirement.

The inspectors reviewed the licensee's corrective actions and concluded that the current design should preclude these events from reoccurring. This 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 97007-02)**

c. Conclusions

The system engineer's identification of an original construction deficiency was good and exhibited a questioning attitude. A non cited violation was issued because the design deficiency rendered the system incapable of meeting its Technical Specification requirement.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) LER 50-286/97014: Lack of Weld Channel Containment and Pressurization System Air Supply to Airlock Seals. This issue is discussed in section E1.5 of this report and is considered a Non Cited Violation (NCV 97-07-02).

IV. PLANT SUPPORT

R1 Radiological Protection and Chemistry (RP&C) Controls

R1.1 Radiologically Controlled Areas

a. Inspection Scope (71750)

The inspectors walked down several areas of the facility where major modifications and/or equipment overhauls were performed during the outage.

b. Observations and Findings

Those areas that were roped off for contamination control during maintenance activities had been decontaminated and returned to a clean status in a timely manner. Additionally, the inspectors noted that the health physics technicians maintained good communications with plant personnel during start-up turnover meetings by providing updates on the changing radiological status of equipment and areas within the radiologically controlled areas during power ascension. Overall, the inspectors observed excellent performance from the health physics organization in proactively returning much of the plant to a clean and workable status in an efficient manner.

c. Conclusions

The health physics organization performed well by proactively returning much of the plant to a clean and workable status in an efficient manner.

F8 Miscellaneous Fire Protection Issues

F8.1 Control Room Fire Suppression Capability

a. Inspection Scope (71750)

The inspectors reviewed NYPA's operational procedures and equipment in place to mitigate a control room fire. This review was conducted in response to an industry event concerning the inadvertent actuation of a control room fire suppression system.

b. Observations and Findings

The control room at IP3 does not have any automatic fire suppression systems. The Pre-Fire Plan, Chapter 28, outlines the IP3 fire suppression techniques to be employed in case of a control room fire. Manual method of extinguishment shall be used, varying the type of extinguisher depending on the type of fire. In case of a fire in the main control room, off normal operating procedure (ONOP) ONOP-FP-1, "Plant Fires," will direct the operators actions to use emergency breathing apparatus and/or, if necessary, to initiate safe shutdown from outside of the control room. Procedure ONOP-FP-1 directs the operator to determine the need for placing emergency breathing apparatus in the control room. The control room utilizes both self contained breathing apparatus' (SCBA) and air lines for emergency breathing apparatus. There are five SCBAs in the control room and the operators receive training on SCBA usage annually. Additionally, there are two emergency air bottles placed just outside the control room. These bottles have 25 feet of airline attached for use under control room un-inhabitability situations. The control room essential shift personnel receive training on its use annually during re-qualification. There are no technical specification requirements regarding the quantity of SCBA's available to the operators; however, procedure FP-13, "Inspection and Testing of Self Contained Breathing Apparatus," establishes guidelines to verify the operability of the SCBAs and control room breathing air units.

c. Conclusions

Control room habitability during off normal conditions is addressed appropriately by plant procedures and operator training.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on September 25, 1997. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT A

PARTIAL LIST OF PERSONS CONTACTED

Licensee
J. Knubel, Chief Nuclear Officer
H. Salmon, Vice President, Nuclear Operations
R. Barrett, Site Executive Officer
J. Comiotes, General Manager, Operations
J. Russell, General Manager, Maintenance
D. Quinn, General Manager, Plant Support
J. DeRoy, Director, IP3 Engineering

NRC
G. Wunder, Project Manager
D. Dempsey, Reactor Engineer

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support Activities
IP 92700:	Onsite Followup of Written Reports of Nonroutine Events
IP 92901:	Followup - Plant Operations
IP 93702:	Prompt Response to Events at Operating Power Reactors

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

VIO 97007-01:	Five Examples of Inadequate Procedures Resulting in Plant Transients and Equipment Inoperability
NCV 97007-02:	Lack of Weld Channel Containment and Pressurization System Air Supply to Airlock Seals

Closed

IFI 97004-01:	Switchyard Controls
LER 97014:	Lack of Weld Channel Containment and Pressurization System Air Supply to Airlock Seals
LER 97017:	Overpressurization Protection System Inoperable Due to Inadequate Procedure
NCV 97007-02:	Lack of Weld Channel Containment and Pressurization System Air Supply to Airlock Seals

LIST OF ACRONYMS USED

ABFP	Auxiliary Boiler Feed Pump
ACTS	Action Commitment Tracking System
DER	Deviation Event Report
ECP	Estimated Critical Position
ft.	foot
I&C	Instrument and Controls
ICRR	Inverse Count Rate Ratio
LER	Licensee Event Report
MBFP	Main Boiler Feed Pump
MOU	Memorandum of Understanding
MS	Main Steam
NCV	Non-cited Violation
NPO	Nuclear Plant Operator
NRC	Nuclear Regulatory Commission
NSE	Nuclear Safety Evaluation
NYPA	New York Power Authority
OPS	Overpressure Protection System
PLCO	Potential Limiting Condition for Operation
PORV	Power Operated Relief Valve
PRNI	Power Range Nuclear Instrument
psig	pounds per square inch gage
PTAC	Plant Transmission Activities Coordinator
PTO	Protective Tag Out
RCS	Reactor Coolant System
RP&C	Radiological Protection and Chemistry
RPS	Reactor Protection System
SCBA	Self Contained Breathing Apparatus
SPVD	Set Pressure Verification Device
TS	Technical Specification
VIO	Violation
WCCPPS	Weld Channel and Containment Penetration Pressurization System
WR	Work Request