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REGION I

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Licensee:	Consolidated Edison Company of New York, Inc.
Facility:	Indian Point 2 Nuclear Power Plant
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Inspectors:	Peter J. Habighorst, Resident Inspector Robert R. Temps, Senior Resident Inspector
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ATTACHMENTS

EXECUTIVE SUMMARY

Indian Point 2 Nuclear Power Plant
NRC Inspection Report No. 50-247/97-08

The NRC completed a special inspection regarding three separate issues that were identified during the 1997 refueling outage (RFO). The issues involved: operation of the plant outside of technical specification (TS) requirements with the overpressure protection system (OPS) inoperable, all three reactor coolant system (RCS) pressurizer code safety valves found with their lift setpoints above the TS allowable value, and identification of a rubber hose wrapped around the impeller of the # 21 recirculation pump and apparent degraded pump performance that preceded the identification of the ingested hose.

Operations

An apparent violation involved operation of the plant, for approximately two and one half days, outside the TS pressure and temperature curves with the OPS inoperable. Contributing to the apparent violation was an inadequate procedure to fill and vent the RCS. Identification of this apparent violation was prompted through NRC questioning of operations personnel regarding the pressure temperature curves. Contributing factors to this event included less than adequate configuration controls, deficiencies in operator awareness and training related to OPS operation, and poor procedural quality. The inspectors identified that there were numerous opportunities for Con Edison to have identified the apparent violation earlier, including log turnovers, and watch supervisory reviews. The potential safety consequence of this condition was to reduce the necessary operator response time (less than ten minutes) to mitigate a reactor coolant system overpressure condition.

Maintenance

A second apparent violation was identified concerning inadequacies by Con Edison staff in the consideration of ambient temperature conditions on RCS pressurizer code safety valve setpoint testing. The apparent violation specifically focuses on untimely and ineffective corrective actions in response to a 1996 open item report (OIR), less than adequate implementation of the approved ASME Section XI code, and less than adequate 10 CFR 50.59 safety evaluation for a plant modification in 1995 to remove the pressurizer block house roof. As a result, the plant was apparently operated for an entire operating cycle with the valve setpoints non-conservatively high and above the TS allowable range.

Numerous opportunities existed for Con Edison to identify the setpoint issue prior to the 1997 RFO. The staff had recorded actual RCS pressurizer code safety valve ambient temperatures in early 1996 for an unrelated reason; however, they did not evaluate the information at that time for the impact on the valve setpoints, and the ambient temperatures were not incorporated into testing procedures until May 5, 1997. In October of 1996, Con Edison identified (ref. Open Item Report 96-E02411) that ambient conditions may not be established for ASME Section XI relief valves; however, they failed to adequately evaluate this for the impact of ambient test conditions for the pressurizer safety valves. In February, 1995 Con Edison failed to fully consider the impact of a plant modification to remove the pressurizer block house. The 10 CFR 50.59 evaluation considered the reduction in ambient temperature; however, it failed to evaluate the affect on pressurizer code safety valve lift setpoints. Further, in July, 1994, Con Edison failed to implement provisions within the in service test program for safety valve testing that required simulation of the operating environment.

Engineering

A third apparent violation was identified concerning Con Edison's response to the identification of degraded performance in the 21 reactor recirculation pump (RRP) identified in 1995. Specifically, at the end of the 1995 refueling outage (RFO), the pump just met the minimum engineering acceptance value during surveillance testing. Following restart of the unit from the 1995 RFO, a system engineer reviewed past performance data on the pump and identified to site management that a significant decline in the pump performance had occurred, especially since 1989, and that the he believed the pump would not pass the next surveillance test (in the 1997 RFO). The engineer's concern was not entered into the site corrective action system and therefore did not receive a formal evaluation. The 21 RRP remained in service without further testing until the present RFO, at which time it failed to meet the minimum acceptance criteria and was replaced. Subsequent inspection of the removed pump identified an ingested hose which was the likely cause of the observed degradation in pump performance. Con Edison engineering analysis concluded that in certain circumstances the 21 RRP would not have been able to perform its safety function and that system redundancy and plant procedures would have been required to compensate for this.

Clear evidence of pump degradation existed. The unexplained decrease in pump performance that occurred in 1989 (identified in 1995), that was not recovered during subsequent testing, should have been cause to investigate the condition of the pump further. Instead, the informal resolution in 1995 focused on the acceptability of lowering the test acceptance value and preparing a spare RRP for use in 1997. The decision to keep the # 21 RRP in service throughout an entire operating cycle without a formal operability review, despite the marginal test result obtained in 1995 and the system engineer's memorandum, represents poor engineering resolution to an equipment problem. Based upon the identification of a hose within the pump, engineering initial evaluation in 1995 of pump's ability to achieve the acceptance criteria in the surveillance program, and the failure of the surveillance in 1997 indicates that the #21 recirculation pump may have been inoperable since at least 1995.

The manner in which the pump performance was resolved in 1995 and 1997 also raises concerns regarding the threshold used for entering problems in the site problem identification and corrective action system. When the pump failed the test criteria in 1997, this fact was documented in the surveillance test results section and later entered into the problem identification system. However, in 1995, the marginal test result, together with the system engineer's identification of significantly degraded pump performance, was not entered into the problem identification system and instead received an informal evaluation for acceptability.

Report Details

I. OPERATIONS

O4 Operator Performance and Knowledge

O4.1 Operability of the Overpressure Protection System; EEI 50-247/97-008-01

a. Inspection Scope (71707)

The inspection evaluated operability controls for the overpressure protection system (OPS) between June 14 through June 17, 1997. The inspection also evaluated operator performance in recognizing and maintaining plant conditions within technical specification (TS) requirements.

b. Observations and Findings

On June 17, 1997, during an observation of control room activities, the inspector questioned operators on the status of the OPS since the power operated relief valve (PORV) control switches were in trip-pull out and their associated block valves were closed. The OPS configuration when operable is: PORV block valves open and the PORV control switches in automatic, capable of responding to an OPS actuation signal. The operators referred the inspector to TS Figure 3.1.A-3, which applied when OPS is inoperable, and stated that reactor coolant system (RCS) temperature and pressure were in the acceptable region of the TS figure. In looking at the TS figure, the inspector raised a question with the Operations Manager about the controls of TS figure 3.1.A-3, as they appeared less restrictive for RCS pressure and temperature limits with more injection sources available (3 charging pumps and 1 safety injection pump energized) than the limits in TS Figure 3.1.A-2 which assumes only one injection source available (one charging pump energized).

After researching the inspector's question, the Operations Manager subsequently informed the inspector that Con Edison had been in violation of the requirements of TS figure 3.1.A-3. TS figure 3.1.A-3 was less restrictive on the pressure and temperature limits than figure 3.1.A-2 because figure 3.1.A-3 assumes that pressurizer level is less than 30% and takes credit for the gas space in the pressurizer allowing more injection sources to be energized and available during OPS inoperability. At the time of the inspectors questioning, pressurizer level was at 80%, which violated the maximum allowed level of 30% upon which use of Figure 3.1.A-3 is predicated upon.

Con Edison prepared significant occurrence report (SOR) 97-E02399, documenting the concern, and commenced draining the pressurizer from approximately 80% to 30%. The draining of the pressurizer was completed in six and one half hours, placing the unit in compliance with the conditions of TS figure 3.1.A-3. Con Edison had been in apparent violation of the TS figure 3.1.A-3 requirements since June 15, 1997 at 2:30 a.m., when the last of three pressurizer code safety valves were installed, which eliminated the alternate pressurizer vent path allowed by TSs. This is considered an apparent violation of NRC requirements. **EEI (50-247/97-008-01, part 1)**

The inspectors identified that numerous opportunities existed for Con Edison to have identified the apparent violation of TS figure 3.1.A-3 prior to the inspector's questioning on June 17, 1997. This included twice daily reactor operator turnover log, DSR-17A, which documents whether OPS is required or not. The inspector learned that DSR-17A does not provide specific guidance on what actions are necessary if OPS is not required; however, it is Con Edison management's expectations that if OPS is not required, then the reactor operator confirms administrative controls and plant conditions as required in SOP 1.4.1, "Overpressure Protection System Operation," revision 7. The inspector confirmed that SOP 1.4.1 provided adequate guidance consistent with TS when OPS is inoperable. Another opportunity to reasonably identify this apparent violation was the senior watch supervisor review of the reactor operator logs during each shift. In addition, control room board indications provide indication of OPS operability, and conditions within the RCS. Licensed operators are expected to understand and operate the plant in accordance with rules and regulations as part of the condition for the license as required in 10 CFR 55.53.

On June 14, 1997, the plant evolution that removed the installed RCS vent pathway, and raised RCS inventory was the vacuum fill and venting of the RCS. SOP 1.1.1, "Vacuum Filling and Venting the Reactor Coolant System," revision 35, did not refer to or verify the operability of OPS prior to or during the RCS fill evolution. This is considered an apparent violation of TS 6.8.1 for less than adequate procedures **(EEI 50-247/97-008-01, part 2)**.

The OPS is designed to relieve RCS pressure for certain transients and to prevent those transients from causing pressure to exceed 10 CFR 50 Appendix G limits. In addition to OPS, the RCS overpressure transient assumptions restrict the number of charging and safety injection pumps that can be energized, and temperature differences between steam generators and the RCS for a reactor coolant pump start.

The potential safety consequence of the apparent violation was that if one or more charging pumps started with an isolation of the letdown system, or a safety injection pump inadvertently started, operators would have less than the assumed ten (10) minutes to terminate the overpressure condition and prevent exceeding 10 CFR 50 Appendix G limits on the reactor vessel. The analysis credits pressurizer gas space when OPS is inoperable, which allows operator action to terminate the pressure transient in ten minutes.

On July 14, 1997, the inspector observed Con Edison's post event critique. Various members of Con Edison management were in attendance at the critique including the Plant Manager and the Operations Manager. The inspector observed adequate interaction and discussions on the apparent causes and proposed corrective actions. The depth and quality of the root cause analysis was good. The inspector's basis was that good supporting information was developed for each apparent cause and thorough analysis of failed barriers.

Con Edison's apparent causes for the failure of operators to adhere to TS Figure 3.1.A-3 included:

- A 1995 safety evaluation for RCS fill evolution did not address plant operational requirements;
- the RCS fill procedure should have been controlled under Station Administrative Order (SAO)-202, Conduct of Infrequently Performed Test or Evolution;
- procedural deficiencies and lack of consistency;
- outage scheduling did not ensure OPS operability prior to RCS fill evolution;
- and, operators did not fully appreciate the relationship between OPS, vent pathway, and required RCS conditions.

Con Edison's proposed corrective actions adequately addressed the above apparent causes. The corrective actions included operator training, future outage schedule controls, procedural changes, revision to the 1995 safety evaluation, and evaluation of SAO-202 criterion.

The inspector reviewed past performance by Con Edison operators in recognizing and maintaining plant configuration in accordance with TS requirements. The inspector noted a poor performance record in this area. The review identified several instances where operators failed to implement TS requirements. These included: controls during the replacement of containment pressure bistables (Licensee Event Report (LER) 96-23), service water pump testing with an emergency diesel generator out of service (NRC inspection report 50-247/96-06 and LER 96-20), exceeding the surveillance interval for control room air filtering system (LER 95-19), inoperability of the electric tunnel exhaust fans (LER 96-06), and incomplete surveillance for control room channel checks (LER 96-17).

There have also been recent events that involved inadequacies in operator identification of systems being outside their expected configuration. This includes the actuation of fire dampers in the 480 volt switchgear room (NRC inspection report 50-247/97-03), the long-term inoperability of the filter/fire deluge control panel (IR 50-247/96-04), and the containment isolation valve for nitrogen supply to the PORVs and other components left open despite procedural requirements that it be shut (IR 50-247/96-04).

c. <u>Conclusions</u>

NRC questioning resulted in Con Edison personnel finding an apparent violation of TS on OPS. A less than adequate procedure existed to fill and vent the RCS and is being considered part of the apparent violation. Numerous opportunities existed for Con Edison to have identified the apparent violation including log turnovers, and watch supervisory reviews. The potential safety consequence was to reduce the necessary operator response time to mitigate a potential RCS overpressure condition. Further, poor past performance was noted in Con Edison's ability to maintain plant configuration in accordance with TS requirements and in identifying significant abnormal system configurations.

O5 Operator Training and Qualification

O5.1 Requalification Training on Overpressurization Protection System

a. Inspection Scope

The inspector reviewed training on OPS during the last cycle of the operator requalification program. The inspection also included observations of the simulator that duplicated plant conditions that existed when the OPS was inoperable. The purpose of the simulator observations was to evaluate plant response and available alarms to operators given a postulated transient leading to an RCS overpressure condition.

b. Observations and Findings

The inspector learned that training on OPS is provided by two principal methods. In both methods, operators are exposed to actions to place OPS in service either in a simulator scenario or in a self-performed practical factor. None of the training focused directly on actions to take during OPS inoperability. The self-performed practical factor which was to be completed within two years of issuance (March 1996) placed OPS in service using SOP 1.4.1 step 3.1. The practical factor was assigned to the senior reactor operators, reactor operators, watch engineers, and the support facility supervisors. Management's expectation was that the operators review the entire SOP for familiarity. SOP 1.4.1 provided instructions during OPS inoperability consistent with TS. The practical factor was to read the procedure prior to a walkthrough. The inspector noted that five of the operators on watch between June 14 through June 17, 1997 had completed the self-performed practical factor on SOP 1.4.1. The second portion of training was a simulator scenario exercise for each crew. One of the critical tasks in simulator exercise guide SS.411, "Loss of Residual Heat Removal with Reactor Coolant Pump Start and Loss of Coolant Accident" was to place OPS in service with SOP 1.4.1.

On July 2, 1997, the inspectors observed two simulator scenarios that were run to provide insight into operator response times and availability of alarms and indications for a postulated overpressure condition. The initial conditions were pressurizer level at 80.5%, RCS temperature at 98°F, and the OPS inoperable. The first scenario was to start all three charging pumps at maximum speed and isolated letdown (a condition evaluated in the design basis analysis). Observation of this sequence indicated that operators would have received various control board annunciators prior to exceeding the OPS setpoint. Maximum RCS pressure was approximately 593 psig, which was limited by actuation of relief valves on the residual heat removal discharge and the letdown line. The RCS pressure did exceed the OPS setpoint within ten minutes, which provided consistency with the engineering analysis supporting TS Figure 3.1.A-3.

The second scenario had the same initial RCS conditions, isolation of the letdown line, and the inadvertent start-up of the 23 safety injection pump through one of four loop isolation valves. Control room annunciators again would have provided indication to the operators of an ongoing transient. The RCS pressure rise also exceeded the OPS setpoint in less than ten minutes given no operator response.

c. Conclusions

The recent requalification training on the OPS system did not emphasize procedure actions or scenarios when OPS was inoperable and compensatory measures taken with TS figures 3.1.A-2 and 3.1.A-3.

The two simulator scenarios confirmed that operator response to terminate the design basis overpressure condition due to mass-addition would have been necessary within ten minutes, thus violating one of the design basis assumptions pertaining to TS Figure 3.1.A-3. The scenarios also indicated that various alarms would annunciate to alert operators to the changing plant conditions.

O7 Quality Assurance in Operations

O7.1 Outage Planning and Scheduling for OPS

a. Inspection Scope

The inspector compared the original refueling outage maintenance schedule for OPS to the actual sequence of maintenance activities.

b. Observations and Findings

The inspectors determined that the original outage schedule replaced the PORVs and cable connections, and completely retested and restored the OPS system prior to RCS fill and vent using SOP 1.1-1. The activities to restore OPS to an operable status were the performance of in service test on the nitrogen supply check valves, and the TS required analog testing.

The following lists the actual sequence of maintenance activities related to the OPS system:

Date	Activity
June 13	Failure of check valve in service test and Significant Occurrence Report (SOR) 97-E02338 prepared
June 14	Reactor coolant system evacuated fill and vent (SOP 1.1-1) completed
June 15	Last pressurizer code safety valve installed
June 15	OPS analog test completed
June 16	Nitrogen to PORVs isolated to address failure of in service test

June 17	Con Edison realizes that the plant was not in compliance with TS Figure 3.1.A-3
June 17	Nitrogen unisolated from PORVs
June 19	Second failure of in service test
June 20	In service test completed satisfactorily

The actual sequence of maintenance activities differed from the initial outage plan in that the decision to perform SOP 1.1-1 preceded completion of in service testing of the nitrogen system check valves and the OPS analog testing. Inspector review concluded that had the original outage sequence been accomplished, the OPS would be operable prior to filling and venting the RCS.

The inspectors determined that Outage Management and Operations personnel evaluated the acceptability of proceeding with the RCS fill and vent without OPS operable. The operations department concluded that it was acceptable, however, an undetected error was made since the decision did not include a complete understanding of TS Figure 3.1.A-3 for minimum RCS inventory. The control of outage maintenance sequence changes for the OPS in regards to SOP 1.1-1 was less than adequate.

c. Conclusions

The control of outage maintenance sequence for OPS was less than adequate. The original outage sequence was appropriate, but as test failures occurred with OPS post-modification testing, Con Edison failed to recognized the impact of OPS inoperability during the reactor coolant system fill evolution.

II. MAINTENANCE

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Pressurizer Safety Valve Testing; EEI 50-247/97-008-02

a. Inspection Scope (61726)

The inspection scope evaluated the apparent cause and prior opportunities to identify setpoint failures of all three pressurizer code safety valve during "as found" testing.

b. Observations and Findings

On June 8, 1997, Con Edison informed the NRC pursuant to 10 CFR 50.72(b)(2)(i) that all three pressurizer code safety valves (PCV-464, 466, and 468) failed their "as-found" set pressure tests. The required lift point for the safety valves is 2,485 psig (+/-1%). The "as found" lift pressures were all greater than +1% (or 2510 psig); two of the three valves exceeded 3% of the nominal set pressure.

The pressurizer code safety valves prevent damage to the reactor coolant system (RCS) pressure boundary and reactor fuel by limiting RCS pressure below design limits for certain transients. TS 5.3.B.1 documents that the design value for the RCS system pressure is 2,485 psig. The safety limit for RCS pressure as documented in TS 2.2 is that RCS pressure shall not exceed 2,735 psig, which is 110% of the RCS pressure design limit. The combined capacity of the three pressurizer code safeties is greater than the maximum surge rate resulting from a complete loss of load transient without credit for power operated relief valves (PORVs) operation and pressurizer spray. Inspector review of Updated Final Safety Analysis Report (UFSAR) figure 14.1-37 indicates that with the above assumptions, maximum RCS pressure would reach approximately 2,564 psig. The inspector noted that with all three pressurizer code safety valves exceeding their allowable lift setpoint, a potential increase in peak RCS pressure is expected during the postulated loss of load analysis event. Con Edison's preliminary analysis indicated that the UFSAR analysis would remain bounding for the worst case safety valve setpoint of 2,581 psig, as documented in licensee event report (LER) 97-013.

The inspector determined that Con Edison removes and sends the safety valves to an outside vendor for testing and for refurbishment of the safety valves. Con Edison's procedure to verify pressurizer code safety setpoints is PT-R5A, "Hot Setting of Pressurizer Safety Valves by Wyle Labs." The testing of pressurizer code safeties is required every refueling interval.

The apparent cause of the setpoint failures for the pressurizer code safeties was untimely inclusion of actual valve ambient temperature conditions into the setpoint testing program. As early as 1989, other utilities indicated that code safety valve setpoint shifts can be caused by changes in temperature of the safety valve body and bonnet. An increase in temperature of the valve results in the expansion of the body and elongation of the bonnet. This relieves spring pressure and reduces the lift setpoint of the valve. This information was documented in NRC Information Notice 89-90, Supplement 1, "Pressurizer Safety Valve Lift Setpoint Shift" in April, 1991. As mentioned in LER 97-013, Con Edison is also exploring the possibility that mechanical damage during the shipment process may have contributed to the unacceptable "as found" setpoints and further testing to confirm this is planned.

Con Edison failed to incorporate the requirements of American Society of Mechanical Engineers (ASME) Section XI guidance (1989 edition) through IWV-1100 and ASME/ANSI OM-1987, Part 1, when the pressurizer code safeties were tested in 1995. Con Edison submitted their third ten-year in-service test (IST) program incorporating ASME Section XI (1989) edition in July, 1994. ASME/ANSI OM-1987 Part 1 paragraph 8.1.1.5 states that the operating environment shall be simulated during set pressure testing of relief valves. No changes to procedure PT-R5A occurred to reflect actual ambient conditions prior to testing in early 1995. This is considered an apparent violation of TS 4.2.1. **(EEI 50-247/97-008-02, part 1)**

Prior Opportunities for Identification

Con Edison recorded pressurizer code safety valve ambient temperatures in early 1996 for an unrelated reason; however, they did not evaluate the ramification on setpoints or incorporate the values into PT-R5A until May 5, 1997. The change incorporated actual ambient temperatures and provided a testing environment range between 130 to 150°F instead of the past valve testing vendor recommended values for lower (200 -225°F) and upper (175 - 185°F) bonnet temperatures.

Prior to the PT-R5A procedural revision in May, 1997, Con Edison personnel had identified in Open Item Report (OIR) 96-E02411 (dated October 1996) that no controls on operating environment for ASME Section XI relief valve testing were being specified for the tests as required by the revision to the ASME Code. However, the corrective actions in response to OIR 96-E02411 were inadequate in that resolution of the OIR failed to identify that actual ambient conditions were not used during pressurizer code safety valve testing in April 1995. This is considered an apparent violation of 10 CFR 50 Appendix B, Criterion XVI **(EEI 50-247/97-008-02, part 2).**

Attachment A to the inspection report provides a time line of activities between 1984 through May 1997 on conditions or events that could have provided opportunity to identify temperature effects prior to the surveillance test failure in May 1997. Con Edison internal actions in response to NRC Information Notice 89-90, Supplement 1 included inputs to Westinghouse with the expected normal operating ambient air temperature range. This information was provided, yet no actions were taken to verify ambient conditions or to consider procedure changes to the tested bonnet temperatures specified in PT-R5A.

The inspector determined that ambient temperatures surrounding the pressurizer code safeties were affected by a plant modification in February 1995 when the pressurizer missile shield roof was removed. The modification's safety evaluation concluded that one effect would be reduced ambient temperature around the pressurizer; however, Con Edison's evaluation of this modification was incomplete in that it did not address the impact on code safety valve setpoints with a reduction in ambient temperatures. This is considered an apparent violation of 10 CFR 50.59 **(EEI 50-247/97-008-02, part 3).**

Con Edison's corrective actions were to successfully adjust the code safety "as-left" setpoints to within the TS limits using the revised PT-R5A which used actual ambient temperature data. As documented in LER 97-013, Con Edison's analysis of the root cause will be documented in the final SAO-132A, Analysis of Station Conditions, report.

c. Conclusions

A contributing cause of the pressurizer safety valve setpoint failures was failure of Con Edison to timely incorporate actual valve ambient temperature conditions into the setpoint testing program. Con Edison had recorded pressurizer code safety valve ambient temperatures in early 1996 for an unrelated reason; however, they did not evaluate the ramification on setpoints or incorporate those values into the testing procedure until May 5, 1997. Con Edison's corrective actions in response to Open Item Report (OIR) 96-E02411 were inadequate in accounting for actual ambient conditions during pressurizer code safety valves testing in April 1995, and related valve test failures in May 1997. Con Edison also failed to implement the ASME Section XI program requirements for relief valves in 1995, and failed to perform a thorough 10 CFR 50.59 evaluation for a 1995 plant modification that effected ambient temperature conditions around the code safety valves.

III. ENGINEERING

E4 Engineering Performance and Knowledge

E4.1 Closed (UNR 50-247/97-007-03): Loss of Foreign Material Exclusion (FME) on the # 21 Reactor Recirculation Pump (RRP); EEI 50-247/97-008-03

a. Inspection Scope (37551)

Inspection report 50-247/97-07 reviewed issues related to the testing of the # 21 RRP during the 1997 refueling outage (RFO) and the subsequent identification of a rubber hose found ingested in the pump internals. The issue was left unresolved pending further review by the inspectors and completion of Con Edison's root cause analysis.

The inspectors reviewed the past performance history of the pump and held discussions with engineering personnel about Con Edison's response to the identification of degraded pump performance in 1995. The inspectors reviewed the results of Con Edison's justification for past operation (JPO) that considered the potential impact of the ingested hose on the operability of the #21 RRP.

b. Findings and Observations

On May 3, 1997, surveillance test PT-R16, Recirculation Pumps, was performed on the #21 RRP. The pump failed the surveillance test because the minimum required differential head of 475 feet (ft) was not obtained; the pump's differential head during the test was 470.25 ft. Following identification of the failure, and anticipating that the pump might have to be replaced during the current outage, the #21 RRP was replaced with a refurbished spare RRP. During decontamination and inspection of the removed pump, a length of red rubber hose about twenty feet long was found ingested in the pump impeller. Following identification, Con Edison initiated a root-cause investigation team to review this issue. Results of the root-cause investigation were unavailable at the end of the inspection period as the final report had not been issued yet.

The inspector reviewed the past operating history of the #21 RRP and discussed it with engineering personnel. The inspector determined that when the pump was tested at the end of the 1995 RFO, the pump had just met the minimal acceptance criteria with a differential head of 475.1 feet. Following restart of the unit from the outage, a system engineer reviewed past performance data on the pump and, following a correction to pre-1989 data, identified to management, via an E-mail message dated September 29, 1995, that a significant decline in the pump's performance had occurred, especially since 1989, and that the he believed the pump would not pass its next surveillance test in the 1997 RFO. The message was generated in accordance with instructions in engineering procedure SE-Q-12.105, System Engineer/Specialist Reviews and Trending. Section 5.1.3 of the procedure requires the notification of various department managers of significant changes or developing trends in system performance.

Since the system engineer's concern over degraded pump performance and the likelihood of failure when tested next was not entered into the site problem identification and corrective action system, the inspector questioned how the system engineer's notification was dispositioned in 1995. The inspector was informed that discussions were held at that time concerning the message. In these discussions, it was concluded that the pump performance might just as easily pass the next test, referring to an "increase" in pump performance observed in the 1993 test data, and that the minimum acceptance criteria might be able to be lowered to 460 ft with further analysis; however, for the present time, the pump had passed the last surveillance test. Plans were developed to send a spare RRP out for refurbishment in the event pump replacement was required during the 1997 RFO, and testing of the pump was scheduled to occur at the start of the outage, rather than at the end as had been the previous practice. The 21 RRP was left in service with no further testing or evaluation until the start of the 1997 RFO.

The inspector reviewed the corrected pump performance data and noted that it did show a sharp decline in the pump's performance between 1987 and 1989. Prior to 1989, the pump's differential head was trending around 520 ft; however, in 1989, the differential head decreased by 40 feet to around 480 feet. This drop was not recovered during subsequent refueling outage tests. In 1991, differential head tested at just above 475 ft, it increased slightly to around 490 ft in 1993, and in 1995 was basically at the minimum acceptance value of 475 ft. In contrast, the #22 RRP differential pressure the entire time was averaging around 520 ft.

The inspectors reviewed a Justification of Past Operability (JPO) that was prepared to determine the impact that the ingested hose had on past operability of the # 21 RRP. The JPO discussed the recent test results of the RRP as well as past test results. The JPO described previous issues concerning the introduction of foreign material into the RRP system and how they were addressed. The JPO concluded that the major contributing factor to the failure of the 21 RRP to attain its differential head was the pressure drop resulting from the hose ingested in the pump suction end bell.

The JPO also included analysis of five hypothetical (see Attachment B) scenarios in an attempt to bound the effects of the hose ingestion on pump performance. Four of the scenarios concluded that the pump would have been degraded to the point that it would not have been operable; however, system redundancy in coordination with the plant emergency operating procedures would have satisfied the overall safety function of the recirculation system.

A fifth scenario was performed to bound the 1997 test result of 470.25 feet. This scenario assumed that pump differential head was 460 ft and the degradation was assumed to be uniform across all points of the pump head/flow curve. The evaluation determined that the pump would have met minimum flow requirements and that operability would have been maintained. The inspectors determined that this scenario was initially performed when the pump failed the surveillance test at the beginning of the 1997 RFO, and was performed to bound the unacceptable test result of 470.25 ft in order to determine if the RRP could be left in place with the differential head at 470.25 ft. However, on further review, and apparently at the insistence of the system engineer, the 21 RRP was replaced during the outage.

Con Edison's resolution to the degraded pump performance is an apparent violation **(EEI 50-247/97-008-03, part 1)** of NRC requirements, specifically Criteria XVI to Appendix B of 10 CFR 50, which states in part that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. Contrary to this, inadequate corrective actions were taken by Con Edison in response to the identification of degraded pump performance in 1995. Subsequently, during the 1997 RFO, the 21 RRP failed its surveillance test and was removed. During decontamination of the removed pump, a rubber hose was found wrapped around the pump impeller. Analysis of this situation concluded that under certain conditions, the 21 RRP would not have been to perform its post-accident safety function and would have been inoperable. This is considered an apparent violation of TS 3.3.A.1.f regarding operability of the RRP **(EEI 50-247/97-008-03, part 2)**.

c. <u>Conclusions</u>

The inspectors are concerned with Con Edison's response to the identification of degraded performance in the 21 reactor recirculation pump (RRP) identified in 1995. Specifically, at the end of the 1995 refueling outage (RFO), the pump just met the minimum engineering acceptance value during surveillance testing. Following restart of the unit from the outage, a system engineer reviewed past performance data on the pump and identified to site management that a significant decline in the pump's performance had occurred, especially since 1989, and that the he believed the pump would not pass its next surveillance test (in the 1997 RFO). The engineer's concern was not entered into the site corrective action system and therefore did not receive a formal evaluation. The 21 RRP remained in service without further testing until the present RFO, at which time it failed to meet the minimum acceptance criteria and was replaced. Subsequent inspection of the removed pump identified the ingested hose which was the likely cause of the pump's observed degradation in performance. An engineering analysis concluded that in certain circumstances the 21 RRP would not have been able to perform its accident safety function, and that system redundancy and plant procedures would have been required to compensate for this.

Clear evidence of pump degradation existed. The unexplained decrease in pump performance that occurred in 1989 (identified in 1995), that was not recovered during subsequent testing, should have been cause to investigate the condition of the pump further. Instead, the informal resolution in 1995 focused on the acceptability of lowering the test acceptance value and preparing a spare RRP for use in 1997. The decision to keep the # 21 RRP in service throughout an entire operating cycle without a formal operability review, despite the marginal test result obtained in 1995 and the system engineer's memorandum, represents poor engineering resolution to an equipment problem.

The manner in which the pump performance was resolved in 1995 and 1997 also raises concerns regarding the threshold used for entering problems in the site problem identification and corrective action system. When the pump failed the test criteria in 1997, this fact was documented in the surveillance test results section and later entered into the problem identification system. However, in 1995, the marginal test result together with the system engineer's identification of significantly degraded pump performance was not entered into the problem identification system and instead received an informal evaluation for acceptability.

V. MANAGEMENT MEETINGS

X1 Exit Meeting Summary

The inspection team presented the inspection results to members of Con Edison management at an exit meeting held on July 29, 1997. Con Edison acknowledged the findings presented. The inspectors asked Con Edison whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

Attachment 1

INSPECTION PROCEDURES USED

71707:	Plant Operations
61726:	Surveillance Observation
37551:	Onsite Engineering

ITEMS OPENED, CLOSED, AND UPDATED

Opened

EEI 50-247/97-008-01	Apparent Violation of NRC Requirements for OPS (2 parts)
EEI 50-247/97-008-02	Apparent Violation of NRC Requirements for Testing of Pressurizer Code Safety Valves (3 parts)
EEI 50-247/97-008-03	Apparent Violation of NRC Requirement for Timely Resolution of Degraded Pump Performance (2 parts)

Closed

UNR 50-247/97-007-03	Ingested Hose in the # 21 RRP

Updated

None.

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
EOP	Emergency Operating Procedures
FME	Foreign Material Exclusion
GPM	gallons per minute
IST	Inservice Testing
JPO	Justification of Past Operation
LER	Licensee Event Report
MOV	Motor Operated Valve
NPSH	Net Positive Suction Head
OIR	Open Item Report
OPS	Overpressure Protection System
PORV	Power Operated Relief Valve
RCS	Reactor Coolant System
RFO	Refueling Outage
RHR	Residual Heat Removal
RRP	Reactor Recirculation Pump
SOR	Significant Occurrence Report
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UNR	Unresolved Item

Attachment A
Pressurizer Code Safety Valve Timeline

Date	Activity
1984-1995	PT-R5A Code safety surveillance test records upper (175 to 185°F) and lower bonnet (200 to 225°F) temperatures and valve body (465 to 485°F) temperatures. Tests performed at Wyle Laboratories. Acceptance of temperature limits based upon contractor input on typical industry values.
October, 1990	NRC AEOD Report E90-09 documents root causes for relief valve setpoint shifts. NRC Information Notice 89-90 reference AEOD Report E90-09
Nov-Dec, 1990	Westinghouse Owners group performed testing on safety valves. Tests examine loop seals and other factors regarding set point drift. Results published. Indian Point Unit 2 removed loop seal arrangement on safety valves in 1976.
April, 1991	NRC Information Notice 89-90 Supplement 1, alerts addresses to an additional factor (ambient temperature) that may affect safety valve performance. Indian Point Unit 2 relies on controls of upper and lower bonnet temperatures established in 1984. Indian Point Unit 2 did not acquire actual ambient temperature data.
July, 1994	IST third year interval begins, with adoption of OM-1 provisions for code safety valves. OM-1 requires use of actual ambient temperatures. Indian Point Unit 2 does not change procedure PT-R5A
February, 1995	Mod FCX-94-10649-C Removal of Pressurizer Missile Shield; (i.e., roof removal; results in lower ambient temperatures). Indian Point Unit 2 does not assess impact of code safety valve performance
April, 1995	PT-R5A performed on three code safeties with all three lifting within "as-found" acceptance criteria (+/-1%). Valves 466 and 468 initial lifts were acceptable, however adjustments were made to make 3 final "as-left" lifts satisfactory. Lower Bonnet Temperature were between 223°F-212°F and upper bonnet temperatures were 181°-177°F.
January, 1996	Indian Point Unit 2 records ambient temperature of all 3 code safety valves. PORV leakage and raising tailpipe temperatures. Ambient temperature recorded in the vicinity of the code safety valves was 133°F.

October, 1996	OIR 96-E02411, ASME Section XI testing of pressure relieving devices states "OM-1 paragraph 8.1.1.5 states that the operating environment shall be simulated during set pressure testing of relief valves. The component data base used by planning to obtain setpoints for relief valves does not supply the operating environment, medium or temperature used to develop setpoints." CORRECTIVE ACTION: Review of valves tested since July, 1994. Based on preliminary review all valves except for valve 263 (Nonregenerative Heat Exchanger Relief Valve) have been tested very close to their operating environment, medium and temperature." However, no actual verification of pressurizer ambient test values against actual ambient valves was performed.
December, 1996	Performance Test personnel sent to system engineers the requirements of OM-1 with a hard copy of all relief valves in the program.
May, 1997	PT-R5A revised to add OM-1 (1987) as a reference, and added ambient temperature range of 140 +/- 10°F. Actual temperatures ranges between 143°F and 136°F. This revision removed upper and lower bonnet temperature criteria.
May 19-21,1997	Code Safeties tested a Wyle Laboratories. For information purposes bonnet temperatures recorded: Valve 464 - Bonnet (129 - 131°F); Ambient (132 - 137°F) Valve 466 - Bonnet (128°F); Ambient (134 - 137°F) Valve 468 - Bonnet (134 - 136°F); Ambient (130 - 138°F) Valve body ranges (450 - 470°F)
May 28, 1997	Con Edison informed of a Notice of Anomaly for Code Safety Valve Tests
June 8, 1997	ENS Report 32447 under 50.72(b)(2)(i) Expected Values: 2460 - 2510 psig (+/-1% tolerance) As Found: Valve 464 - 2,560 psig (exceed 3%) Valve 466 - 2,581 psig (exceed 3%) Valve 468 - 2,533 psig (exceed 1%)

Attachment B
 # 21 RRP JPO Scenarios

CASE 1:

Westinghouse was contracted to perform an analysis of past pump performance assuming an additional pump head loss of 10 ft to the 1997 test result of 470 ft. The degradation was assumed to be uniform across all points of the pump head/flow curve. This analysis showed that with a uniform degradation, the 21 RRP would still have met minimum flow requirements, and therefore its operability was maintained.

CASE 2:

This case assumed that the ingested hose acted as a fixed orifice in the pump suction. The analysis indicated that pump flow would be reduced to 480 gallons per minute (GPM) and that pump net positive suction head (NPSH) may not be sufficient, resulting in a further decrease in differential head. For this case, the calculations indicated that insufficient flow would be established from the pump and that alternate actions, via the emergency operating procedures (EOPs), would be required to align the recirculation system to the high head recirculation mode.

CASE 3:

This case assumed that part of the hose is chopped up and passes through the pump internals, resulting in blocking of the downstream residual heat removal (RHR) heat exchanger, and that the remainder of the hose remains in the pump impeller. Although calculations showed the pump flow would be reduced to 670 gpm, the outcome was the same as case 2 in that insufficient flow would be established from the pump and alternate EOP actions would be required.

CASE 4:

This case assumed small pieces of hose pass through the pump and block the RHR heat exchangers. The result of this scenario was the same as case 2 and 3.

CASE 5:

This case assumed that the hose prevented the 21 RRP from starting. For this case, the EOPs would direct the operators to manually start the 22 RRP.