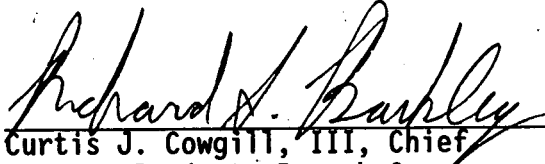


U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report: 50-286/95-17
License: DPR-64
Licensee: New York Power Authority
Facility: Indian Point 3 Nuclear Power Plant
Location: Buchanan, New York
Inspection Period: December 5, 1995 to January 13, 1996
Inspectors: R. Rasmussen, Acting Senior Resident Inspector
T. Frye, Resident Inspector

Approved by:

for 
Curtis J. Cowgill, III, Chief
Reactor Projects Branch 2

2/12/96
Date

Areas Inspected:

Plant operations; maintenance; engineering; plant support; and, safety assessment and quality verification.

Results:

Inspection results are summarized in the attached executive summary.

EXECUTIVE SUMMARY

Indian Point 3 Nuclear Power Plant

NRC Inspection Report No. 50-286/95-17

Plant Operations: The decision to return to cold shutdown to evaluate and repair the leak on the 34 steam generator hand hole was an example of conservative decision making and good safety perspective by management.

Approximately 1300 gallons of CCW drained into containment over a 16 hour period due to a partially stuck open component cooling water (CCW) relief valve. The NRC concluded that operator response to the indications of CCW leakage was weak.

PORC did not fully assess the safety significance of the relief valve lifting event and concurred on two separate occasions that plant heatup could continue without first correcting an RHR system operating procedure deficiency. NRC review of operating procedure revisions and the FSAR identified that the procedure revisions had not been evaluated as required by 10 CFR 50.59 prior to implementation. A similar oversight contributed to plant operation in July 1995 at reduced pressure, a violation which was the subject of an escalated enforcement action in October 1995. The failure to take effective corrective actions to prevent recurrence following the operation at reduced pressure event is considered a violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Actions. (VIO 95-17-01)

The incorrect interpretation of plant technical specifications related to the boric acid heat trace system was an example of weak understanding of the plant licensing basis.

The draining of the 34 steam generator was poorly controlled in that the time the SG was drained was not minimized.

Maintenance: NYPA performed a thorough failure analysis for a leaking 31 RHR pump seal. Expanding the scope to verify the seal on the 32 RHR pump resulted in removing a potential failure due to a manufacturing defect within the seal and represented prudent, conservative decision-making.

Evaluation and repair of the 34 steam generator hand hole leak was performed well by maintenance. The evaluation of the remainder of the steam generator hand holes was a positive initiative.

Engineering: Response and evaluation of the auxiliary component cooling water (ACCW) pump power supply design question was slow. Inadequate communications in the operations department resulted in guidance to the operators not being developed in a timely manner. NYPA did not adequately evaluate the reportability of the ACCW pump power supply design prior to closing DER 95-2863. (URI 95-17-02)

(EXECUTIVE SUMMARY CONTINUED)

The NRC concluded that the evaluation of a service water leak was weak in that lagging was not removed for inspection prior to deferring the work to a future outage. A service water team was established to address the long term system operability concerns.

The initial troubleshooting of the Appendix-R diesel overspeed condition was good. The recommendation to remove the fuel rack booster per a design modification will remove a potential failure mode. However, the bases for the operability determination was weak and the root cause of the AOV failures was not fully considered.

Plant Support: Radiological controls were observed to be adequate during this inspection period.

An inattentive security guard and an inadvertently removed vital area door were promptly identified and NYPA security responded well to both with appropriate compensatory measures.

The NRC noted several plant areas containing safety-related equipment with lighting deficiencies. NYPA is reviewing the process of lighting deficiency identification and repair.

NYPA response to blizzard conditions was good. NYPA maintained an appropriate safety focus and effectively handled the challenges without significant problems.

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ATTACHMENTS

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| Attachment 1 | Predecisional Enforcement Conference on December 13, 1995,
Attendees List |
| Attachment 2 | Slides for the Meeting Between NRC and NYPA on December
13, 1995 |

DETAILS

1.0 SUMMARY OF PLANT ACTIVITIES

1.1 New York Power Authority (NYPA) Activities

At the start of this inspection period, the plant was in hot shutdown with a reactor coolant system (RCS) heat-up in progress. The plant exceeded 350 °F on December 14 and reached normal operating temperature and pressure on December 16, 1995. On December 21, a steam leak was discovered at a 34 steam generator hand hole. The decision to cooldown was made and the plant returned to cold shutdown on December 25, 1995. The plant remained in cold shutdown for the remainder of the period.

1.2 NRC Activities

On December 13, 1995, a predecisional enforcement conference was held at the Region I Office in King of Prussia, PA to discuss the circumstances related to NYPA exceeding cold shutdown with the containment spray pump and recirculation pump switches not in automatic. On January 2, 1996, a severity level III violation with a \$50,000 civil penalty was issued in response to this event. Attachment 1 is a list of meeting attendees and attachment 2 contains slides from the meeting.

On December 19, 1995, senior NRC managers visited the site to conduct tours and interview plant personnel. On January 11, 1996, regional NRC managers visited the site in preparation for the systematic assessment of licensee performance (SALP) review.

2.0 PLANT OPERATIONS (71707, 71715)

2.1 Operations Safety Verification

Using the insights from the Indian Point 3 (IP3) Nuclear Power Plant Individual Plant Examination, and applicable drawings and checkoff lists, the NRC independently verified safety system operability by performing control panel and field walkdowns of the following systems: Boric Acid Heat Trace Systems, Appendix R safe shutdown systems, auxiliary feedwater system, service water system, containment pressure boundary, safety injection system, residual heat removal system, component cooling water system, and the emergency diesel generators. These systems were properly aligned for the existing plant conditions.

The NRC observed plant shutdown operations, and verified that the plant was operated safely and in accordance with licensee procedures and regulatory requirements. Regular tours were conducted of the following plant areas:

- Control Room
- Control Building
- Diesel Generator Building
- Primary Auxiliary Building
- Containment Building
- Containment Penetration Areas
- Spent Fuel Building
- Screenwell Structure
- Auxiliary Feed Pump Building
- Turbine Building
- Intake Structure
- Access Control Points
- Yard Areas

During the course of the inspection, discussions were conducted with operators concerning knowledge of recent changes to procedures, facility configuration, and plant conditions. The NRC verified adherence to approved procedures for ongoing activities observed. Shift turnovers were witnessed and staffing requirements confirmed. The NRC found that proper control room access and a professional atmosphere were maintained. NRC comments or questions resulting from these reviews were satisfactorily resolved by licensee personnel.

Control room instruments and plant computer indications were observed for correlation between channels and for conformance with technical specification (TS) requirements. Operability of engineered safety features, other safety related systems, and onsite and offsite power sources were verified. The NRC observed various alarm conditions and confirmed that operator response was in accordance with plant operating procedures. Compliance with TS and implementation of appropriate action statements for equipment out of service were inspected. Logs and records were reviewed to determine if entries were accurate and identified equipment status or deficiencies. These records included operating logs, turnover sheets, and system safety tags.

2.2 Component Cooling Water Relief Valve AC-819A Failure

The reactor coolant system (RCS) was heated above the cold shutdown condition on December 2, 1995. On December 3, with the plant at approximately 230 °F, operators noted that an excessive amount of water had been added to the component cooling water (CCW) surge tanks over the previous two shifts. Further investigation by the operators determined that CCW relief valve AC-819A was leaking into the containment sump. Relief valve AC-819A provides overpressure protection for the shell side of the 31 residual heat removal (RHR) heat exchanger (HX), which is cooled by CCW. The operators were able to reseal the relief valve and stop most of the CCW leakage into containment. It was determined by NYPA that valve AC-819A had been leaking at approximately 1.3 gallons per minute. Further, this CCW leakage had continued for approximately 16 hours. Although this leak rate resulted in five makeups to the CCW surge tanks to maintain surge tank level, the presence of the CCW leakage was not detected by the operators for two shifts.

NYPA initiated deviation/event report (DER) 95-2784 to document this event. The Director, Quality Assurance (QA), performed a review of the performance of the Operations staff in identifying the leaking relief valve. NYPA supplemented this review by performing a root cause analysis (RCA) for DER 95-2784 to determine the cause of the slow identification by operations. NYPA further assembled a task force to evaluate why valve AC-819A lifted and failed to reseal, and the necessary actions to prevent recurrence.

2.2.1 Cause of Relief Valve Failure

The NYPA task force determined that relief valve AC-819A lifted due to thermal effects on the CCW portion of the 31 RHR HX. To facilitate plant heatup on December 2, the operators shut MOV-822A in accordance with system operating procedure (SOP)-RHR-1, revision 15, Residual Heat Removal System. MOV-822A is the motor operated CCW discharge valve for the 31 RHR HX and shutting this valve secured the flow of CCW through the 31 RHR HX. Further, with valve MOV-

822A shut, CCW was trapped in the shell side of the HX due to the presence of a check valve on the CCW supply line. RCS flow through the tube side of the 31 RHR HX continued as the RCS temperature was raised to approximately 230 °F. As the RCS temperature increased, the pressure of the trapped CCW on the shell side of the 31 RHR HX increased until AC-819A lifted. A review of the plant computer data indicated that steady leakage from the CCW system initiated at about 2:30 pm on December 2. The leakage from the CCW system was not identified by the operators until 8:00 am on December 3.

The task force further concluded that relief valve AC-819A had probably lifted during the three previous plant heatups from cold shutdown which had been performed during 1995. Although the valve may have lifted during each of these previous heatups, it reseated properly and did not result in excessive CCW leakage. The valve vendor concluded that these repeated challenges could have damaged the valve seat or resulted in the accumulation of foreign material on the valve seat. Both of these factors could have resulted in the valve failing to reseat properly on December 2. AC-819A was removed, inspected and replaced. Visual examination of the valve by maintenance noted debris on the valve internals. Maintenance engineering concluded that the cause of the valve leakage was debris under the seat. NYPA planned further valve modifications to correct this leakage problem for future plant outages.

Although the procedural deficiency which led to the valve leakage was not corrected, NYPA concluded the consequences of a repeat failure were not safety significant. This was based on the low leak rate observed and the ability to makeup to the CCW system to compensate for any leakage. These conclusions were presented to the plant operating review committee (PORC) on December 5, 1995. PORC concurred with these conclusions and recommended that plant heatup could continue. The NRC reviewed the NYPA evaluation and PORC review and had several concerns. The NRC questioned the prudence of continued plant heatup and power operation when neither of the relief valves were modified or SOP-RHR-1 revised to prevent recurrence. The NRC was further concerned that the effect of this valve failure during performance of the emergency operating procedures (EOPs) had not been fully evaluated by NYPA. The operations department agreed not to change the plant mode (exceed 350 °F) until the SOP-RHR-1 was reviewed and revised as necessary to prevent recurrence.

On December 6, 1995, PORC again met to discuss continued plant heatup. The operations department stated that the EOPs were reviewed and referred to SOP-RHR-1 for long term cooling. Operations stated that a detailed engineering and licensing review was required to develop another means of operating the RHR system per SOP-RHR-1, besides shutting MOV-822A or 822B. Operations stated that they would continue to work to revise the procedure, but stated that plant heatup could continue. PORC agreed with this recommendation. The NRC remained concerned with this course of action. Specifically, the NRC was concerned that a relief valve lift during a post accident plant cooldown could result in the inability to secure the CCW leakage due to the potential inaccessibility of the containment building. Operations management again stated that the operation of the RHR system would be revised prior to plant heatup above 350 °F.

On December 7, 1995, SOP-RHR-1 was revised to delete the option of shutting MOV-822A or 822B. This revision prevented the possibility of securing CCW flow to either the 31 or 32 RHR HXs with RCS flow through them. The NRC reviewed this change and determined that it was adequate to prevent recurrence of the relief valve lift. Plant heatup above 350 °F continued on December 14, 1995.

2.2.2 Operator Performance in Identifying the Leakage

The Director, QA performed a review of the operator performance in identifying the CCW leakage from relief valve AC-819A. NYPA determined that the operators added water to the CCW surge tanks to maintain level five times over a two shift period between December 2 and December 3. Two makeups were performed on the December 2, 3-11 shift. Two more makeups occurred on the December 3, 11-7 shift. The fifth makeup occurred at 8:00 am on the December 3, 7-3 shift and led to the investigation into the cause of the leakage. The operators on the previous two shifts attributed the need for makeup on their shift to normal contraction of the CCW after securing RHR cooling flow. NYPA concluded that weaknesses in shift turnovers, communications within and between the shifts, and log keeping resulted in the 11-7 shift not realizing that the previous 3-11 shift had already made two makeups to the CCW system. Operator unawareness of telltale indications such as excessive CCW surge tank water addition prevented this abnormal condition from being identified. Further, the shift managers on both of these shifts were only aware of one CCW makeup on each of their shifts.

The NYPA review also identified several instances of failure to adhere to plant procedures. SOP-CC-1B, Component Cooling System Operation, directs Chemistry to sample the CCW system after a makeup to the CCW surge tanks. Contrary to SOP-CC-1B, chemistry was requested to sample after only two out of the five makeups performed. Further, both alarm response procedure (ARP)-10 and SOP-CC-1B direct operations to restore surge tank level to normal (50%) if a low level condition exists. Contrary to these requirements, the December 3, 11-7 shift only filled the CCW surge tanks enough to clear the low level alarms (46%). Finally, SOP-WDS-10, Monitoring Leaks Within Containment Building, required that the containment sump flow totalizer be logged once per shift and graphed daily in the control room. This daily monitoring was based on a NYPA commitment to NRC Bulletin 80-24. However, the containment sump totalizer had been out of service since October 19, 1995, and no measures had been taken to continue the daily monitoring of containment sump flow as required by procedure. All of these procedure violations are significant since they represent missed opportunities to identify the CCW leakage earlier.

Subsequent to the review of operator performance by the Director of QA, a root cause analysis (RCA) was completed by operations and the operational review group (ORG). This RCA concluded that the root cause of the relief valve lift was poor engineering judgement used in generating changes to a system operating procedure. The RCA also determined that the root cause of the delay in detecting the leak was due to a lack of a questioning attitude to changing plant conditions combined with inadequate verbal and written communications. The RCA also compared this event to previous similar events involving

operating at reduced pressure and exceeding cold shutdown with required safety pumps inoperable. Common causes were identified and additional corrective actions were developed to address previous, ineffective corrective actions.

The NRC reviewed the RCA prepared by NYPA and attended the PORC meeting held to review the evaluation. The RCA was a detailed review which identified several inappropriate actions during the event and developed several corrective actions to address these concerns. The RCA also compared this event to previous, similar events and previous corrective actions were identified which needed further emphasis.

NRC review noted that RCS shutdown cooling using the RHR system is described in the final safety analysis report (FSAR), section 9.3. The use of the RHR HX cooling water discharge motor operated valves (MOV-822A and 822B) is not described in the FSAR. The NRC reviewed the revisions to SOP-RHR-1 which added the operation of these valves to the procedure. Term procedure change (TPC) 95-444 was written by operations on March 18, 1995, to allow the use of MOV-822A to control RCS temperature. This TPC was later incorporated into revision 14 to SOP-RHR-1, which was issued on April 21, 1995. Although both TPC 95-444 and revision 14 to SOP-RHR-1 changed the way the RHR system was operated as described in the FSAR, the NRC noted that neither change was evaluated in accordance with 10 CFR 50.59. The NRC also noted that a similar failure to evaluate procedure changes contributed to the operation of the plant at reduced pressure in July 1995. Although NYPA took corrective actions in response to that event to review the adequacy of operations procedure changes, the NRC questioned why TPC 95-444 was not addressed by this review. The NRC raised this concern to NYPA plant management and expressed concerns regarding the completeness of the RCA; NYPA is currently reviewing the RCA.

NYPA reviewed the process used for the procedure change review effort undertaken in response to the July 1995 operation at reduced pressure. NYPA determined that the change in revision 14 to SOP-RHR-1, which added the operation of MOV-822A and 822B, was not reviewed due to personnel error when reviewing the change description. NYPA re-reviewed all other procedure changes previously examined and did not identify any other similar oversights. However, the NRC noted that this procedure change review effort was also not comprehensive since it only looked at procedure revisions. TPCs which were active at the time were not reviewed as part of this reduced pressure event corrective action. The NRC was concerned that this may have led to an oversight of additional procedure changes which have not been properly evaluated per 10 CFR 50.59. As of the end of this inspection, NYPA was reviewing current active TPCs to verify that they have been properly evaluated.

TPC 95-444 was not specifically reviewed since it had been incorporated into revision 14 of SOP-RHR-1. NYPA agreed that this TPC was initially screened incorrectly and performed a Nuclear Safety and Environmental Impact Screen on January 10, 1996. The results of this screen determined that a nuclear safety evaluation (NSE) was not required since shutting MOV-822A did not alter activities as described in the FSAR. The NRC reviewed this screen and concluded that it did not provide adequate justification for these conclusions. The NRC further concluded that shutting MOV-822A did alter the

RHR cooling alignment as described in the FSAR, and that a NSE was required in accordance with 10 CFR 50.59. As of the end of this inspection period, NYPA management was still reviewing this issue.

2.2.3 Conclusion

The NRC reviewed the circumstances involving the cause of CCW relief valve AC-819A lifting event, and the adequacy of the operator identification of the leakage of CCW into containment. The NRC concluded that operator response to the indications of CCW leakage was poor. Many of the causes of this poor response were similar to those related to the July 1995 operation at reduced pressure event and the October 1995 control switch in "trip pull out" event. These include the lack of a questioning attitude, inadequate communications, log keeping and shift turnovers, and the failure to respond to telltale indications such as excessive CCW surge tank water makeups.

Conservative decision making was not demonstrated by PORC while reviewing the cause of the relief valve lift. Although NYPA identified that incorrect instructions in SOP-RHR-1 led to the valve failure, PORC concurred on two separate occasions that plant heatup could continue without first correcting the procedure deficiency.

NYPA identified several procedure adherence violations during the review of this event. These include: 1) the failure to restore CCW surge tank level to normal while filling as directed by alarm response procedure (ARP)-10, 2) the failure to notify chemistry to sample CCW after filling the surge tanks as directed by SOP-CC-1B, and 3) the failure to graph the VC sump totalizer readings on a daily basis as required by SOP-WDS-10. These violations all represented missed opportunities for the operators to identify the CCW leakage earlier. Corrective actions taken and planned by the licensee include revising ARP-10, SOP-CC-1B, and SOP-WDS-10 to clarify the operator actions required. In accordance with the NRC enforcement policy, Section VII.B.1, these violations will not be cited due to the identification by the licensee and the comprehensive corrective actions taken and planned to prevent recurrence.

NRC review of the RCA performed for the relief valve lift noted that the cause of the valve failure was not fully evaluated by NYPA. NYPA concluded in the RCA that poor engineering judgment allowed revisions to be made to SOP-RHR-1 which resulted in the valve failure. NRC review of the procedure revisions and the FSAR identified that the procedure revisions had not been evaluated as required by 10 CFR 50.59 prior to implementation. A similar oversight contributed to plant operation in July 1995 at reduced pressure. The procedure change review performed in response to the reduced pressure event did not identify that the changes to SOP-RHR-1 had not been evaluated as required. Further, the NRC determined that this change review had not been comprehensive since TPCs active at the time were not included in the review effort. The NRC noted that station procedures have been changed to require nuclear safety evaluation (NSE) screens to be performed for all procedure changes. However, the procedure change review was not effective in that it did not identify the improperly evaluated procedure changes to SOP-RHR-1.

The failure to take effective corrective actions for significant conditions adverse to quality following plant operation in July 1995 at reduced pressure is considered a violation of 10 CFR 50 Appendix B, Criterion XVI, Corrective Actions. (VIO 95-17-01)

2.3 Boric Acid Heat Trace Circuit 54

The primary channel of boric acid heat trace (BAHT) circuit 54 was declared inoperable by operations personnel on December 8, 1995, due to a failure to maintain temperature in the required band. BAHT circuit 54 provides heat protection for the normal charging path from the boric acid storage system to the reactor coolant system (RCS) to prevent the boric acid solution from solidifying in the piping. Technical specification (TS) 3.2.C required that the primary channel be restored within seven days, provided that the redundant channel is still operable. About two hours after declaring the primary channel inoperable, the operators noted that the redundant channel of BAHT circuit 54 was also not maintaining temperature within the required band. With both channels inoperable, the operators entered TS 3.2.D and determined that the plant would be required to return to the cold shutdown condition if both channels of BAHT circuit 54 were not restored to service within an additional 48 hours.

The NRC reviewed the application of the TS requirements to BAHT circuit 54 and noted that the correct TS action statements may not have been applied. TS 3.2.D states that the reactor shall be brought to the cold shutdown condition after an additional 48 hours if the time period specified in TS 3.2.C (seven days with one channel inoperable) is exceeded. The NRC considered that TS 3.2.D did not apply with both channels inoperable and that TS did not provide an action statement for this condition. To address situations where TS requirements do not exist, the operations department has established a policy that the plant would be cooled down from the hot shutdown condition within 30 hours of the identification of the problem. The NRC concluded that the 30 hours to the cold shutdown condition should apply in this case if both channels of BAHT circuit 54 were not restored to service.

These questions were raised to operations department management on December 8, 1995. At the time the question was raised by the NRC, both channels had been inoperable for about 8 hours. Operations reviewed the entry into TS 3.2.D and concluded that this TS had been applied when it was not necessary. Although the heat trace circuits for the normal charging flow path (BAHT circuit 54) were inoperable, alternate charging paths to the RCS were still available. Based on this information, operations exited the TS required actions for BAHT circuit 54. However, operations concurred that in addition to entering a TS action unnecessarily, the incorrect action was applied assuming a loss of both BAHT circuits to the primary charging path. Operations initiated DER 95-2832 to document that TS required actions were entered erroneously and that the incorrect TS action statement was applied.

The incorrect interpretation of plant TS was a concern to the NRC since this was an example of weak understanding of the plant licensing basis. Operations management issued shift orders to provide guidance to the operators for the

application of TS 3.2.D and other similarly worded TS limiting conditions of operation. The NRC reviewed this guidance and concluded that it was adequate for plant startup. Operations and licensing will continue to develop additional guidance for the licensed operators in applying TS action statements.

3.0 MAINTENANCE/SURVEILLANCE (61726, 62703)

3.1 Routine Maintenance Review

The NRC reviewed selected maintenance activities to assure that: the activity did not violate Technical Specification Limiting Conditions for Operation and that redundant components were operable; required approvals and releases were obtained prior to commencing work; procedures used for the task were adequate and work was within the skills of the trade; activities were accomplished by qualified personnel; radiological and fire prevention controls were adequate and implemented; quality control hold points were established where required and observed; and equipment was properly tested and returned to service.

The maintenance work requests (MWRs) listed below were observed/reviewed. Unless otherwise indicated, the activities observed and reviewed were properly conducted.

- 95-04659-00 PAB Supply Fan Motor PM
- 94-04321-06 31 IA Compressor Unloader Valve Modification
- 94-00850-16 Repair 32 RHR Pump
- 95-05085-11 Troubleshoot Appendix-R Diesel Generator
- 95-04606-00 33 Emergency Diesel Generator PM

3.2 Residual Heat Removal Pump Seal Failure

On December 29, 1995, NYPA observed leakage from the 31 residual heat removal (RHR) pump seal. This seal had been previously replaced in late November, 1995, as described in NRC inspection report 50-286/95-16. NYPA determined that complications with the November installation led to the December failure.

In November, NYPA replaced the 31 RHR pump motor and seal assembly. The replacement seal was a design change which involved installing a Chesterton seal package. A Chesterton package had previously been installed in the 32 pump. During the initial motor bump of the 31 RHR pump, the seal package sleeve slipped on the pump shaft. After discussions with the seal vendor, NYPA reset the sleeve and the seal appeared to function normally until the failure on December 29. NYPA could not determine why the sleeve slipped on the pump shaft.

As a result of the December failure, NYPA and the vendor performed a detailed review of the seal to determine the failure mode. The review determined that the seal internals were damaged in November during the pump bump and the December failure was a result of that damage. However, during the investigation, NYPA discovered a seal component with incorrect dimensions.

The stationary seal ring was 0.159 inch shorter than required. The seal package was designed to accommodate a maximum of 0.149 inch of cumulative manufacturing tolerances. This had the potential to create another failure mode of the seal, however it did not have any effect in this case.

Further investigation revealed that the discrepant part was actually designed for the next size smaller seal. During manufacture, it was incorrectly installed in the seal package. The seal ring was not accessible once the seal package was assembled by the vendor. NYPA purchased the seal in 1991 from the vendor who performed the commercial grade dedication. The vendor informed NYPA that the RHR pump seals were special order items that were only applicable to NYPA and Con Edison. At the end of the inspection period, NYPA was working with the vendor to determine the reportability requirements. Upon completion, the NRC will review NYPA's reportability determination.

Because the seal which was installed in the 32 RHR pump was procured at the same time as the 31 seal, NYPA elected to replace the 32 RHR pump seal. However, the 32 RHR pump seal, which had been in service for about one year, did not have any indications of pending failure. Nevertheless, on January 10, 1996, NYPA entered a special evolution to replace the seal. Subsequent investigation of the old seal revealed that it also had an improperly sized wear ring.

The NRC concluded that NYPA demonstrated excellent judgement and conservative decision making in performing the investigation of the 31 RHR pump seal failure and in opting to examine and replace the 32 RHR pump seal.

3.3 RHR Pump Seal Replacement

The NRC observed the 32 RHR pump seal repair effort by maintenance and questioned NYPA's approach for the removal of the pump impeller. Difficulty was encountered during the impeller removal, which prompted maintenance personnel to use pry bars and mallet force to attempt to remove the impeller from the shaft. The NRC was concerned that the use of these tools could damage the wear rings on the impeller. The NRC reviewed maintenance procedure PMP-001-RHR, RHR Pump Overhaul and Inspection, and noted that impeller removal using these methods was not addressed in the procedure. The NRC further noted that the maintenance procedure did not require that the impeller wear ring surfaces be inspected for damage prior to reassembly. Maintenance eventually used wooden wedges and hydraulic force to successfully remove the impeller.

NYPA maintenance management later determined that inappropriate work practices were employed in the impeller removal. DER 96-048 was initiated by NYPA maintenance to document this deficiency, and the maintenance crew was briefed to review the work effort. The impeller was inspected by quality control and a pump vendor representative and no damage was found. Maintenance decided to also procure more appropriate equipment as required for future pump work. Although poor work practices were used during the pump impeller removal, the NRC determined that appropriate corrective actions were taken by NYPA maintenance to improve future pump overhauls.

3.4 Steam Generator Hand Hole Repair

While in hot shutdown, NYPA identified a steam leak from one of the seven hand holes at the bottom of the 34 steam generator (SG). The hand holes, which are located on the secondary side of the SG, are used for SG maintenance. On December 22, 1995, NYPA commenced taking the reactor to cold shutdown conditions to perform the repairs. On December 27, 1995, 34 SG was drained for the repairs.

Work to repair the leaking handhold was performed promptly. Based on the condition of the hand hole, it was determined that the remaining hand hole cover bolts should be checked for proper torque. Maintenance determined that the hand holes were not properly installed during a 1992 outage. The hand hole cover bolts on the other SG's were torqued and no other steam generators required draining. Long term corrective actions are being evaluated by NYPA.

Procedure SOP-SG-2B, Steam Generator Draining and Dry Layup, contained requirements that the SG may only be drained for a 96 hour period unless the site executive officer's permission is obtained and the amount of time with the steam generator drained shall be minimized. These precautions are designed to minimize the effects of harmful corrosion to the SG tubes, which can be accelerated in the partial draindown condition.

On December 26, 1995, the operations manager identified the need for a revised procedure for filling the steam generator. The procedure revision was required due to a concern with refilling with cold water from the condensate water storage tank. The concern was that the cold water could cause an excessive cooldown of the reactor coolant system if the evolution was performed too quickly. Although the procedure to refill the SG was not in place, operations proceeded to drain the SG on December 27.

Delays associated with developing a suitable procedure for refilling the SG extended the time in partial draindown to 94 hours. The NRC noted the delays in refilling the SG and brought this to the attention of NYPA management. Of further concern to the NRC was this issue was not identified by NYPA. After identification by the NRC, a deficiency event report (DER) was issued to evaluate the lessons learned.

The NRC considered the decision to return to cold shutdown to evaluate and repair the leaks was an example of conservative decision making and good safety perspective by management. The evaluation of the other SG hand holes was a positive initiative by maintenance. However, the NRC was concerned that the time the SG was drained was not minimized by NYPA, as prescribed in SOP-SG-2B, although the time limitation posed in the procedure was not ultimately violated.

3.5 Routine Surveillance Review

The NRC witnessed/reviewed selected surveillance tests to determine whether properly approved procedures were in use, details were adequate, test instrumentation was properly calibrated and used, technical specifications were satisfied, testing was performed by qualified personnel, and test results

satisfied acceptance criteria or were properly dispositioned. The performance tests (Pts) listed below were observed and reviewed. The activities observed and reviewed were properly conducted without any notable deficiencies.

- 3PT-M1 Power Range Functional Test
- 3PT-M18B RHR Pump Functional Test
- 3PT-Q17 Test of Valve SP-AOV-956D Position Indication
- 3PT-Q92E 35 Service Water Pump Operational Test

3.6 RHR Pump Test Deficiencies

Following the November replacement of the 31 RHR pump seal and recirculation flow throttle valve, NYPA performed a flow test of the pump. During the test, the pump was operated at various flowrates and flow and pressure data was obtained. When this data was compared to the pump curve in surveillance procedure 3PT-M18B, Residual Heat Removal Pump Functional Test, the pump performance fell below the curve at the higher flowrates (above 2500 gpm). Operations declared the pump inoperable and initiated a deviation event report (DER) to address the discrepancy.

NYPA analyzed the problem and determined that the pump performance was satisfactory. The problem was due to pressure losses associated with the location of the tap used to measure discharge pressure. Calculations showed that these losses were only significant at high flow rates. Routine in service testing (IST) is performed on the pump monthly and data is collected only at a single point normally low on the operating curve. The data collected at the high flowrate was obtained as part of the retest for previous maintenance. As a result of this discrepancy, NYPA revised the acceptance curve in the procedure to account for the dynamic effects at high flowrates. NYPA initiated a second DER to identify similar issues elsewhere in the IST program.

The NRC reviewed the bases for altering the pump performance curve and found the issue well documented. The personnel involved with the IST program were knowledgeable of pump operating characteristics and IST program requirements.

4.0 ENGINEERING (37551, 92903)

4.1 Auxiliary Component Cooling Water Pump Power Supplies

While reviewing system status prior to exceeding cooldown shutdown, the NRC questioned the power supply arrangement for the auxiliary component cooling water (ACCW) pumps. The ACCW pumps supply cooling water to the recirculation pumps during accident conditions. There are four ACCW pumps, two for each recirculation pump. The 31 recirculation pump is supported by the 31 and 32 ACCW pumps. The 32 recirculation pump is supported by the 33 and 34 ACCW pumps. The power supplies, however are cross connected for the 32 and 33 ACCW pumps. Specifically, 480 volt bus 5A supplies the 31 recirculation pump and the 31 and 33 ACCW pumps, and 480 volt bus 6A supplies the 32 recirculation pump and the 32 and 34 ACCW pumps. Above the cold shutdown condition, technical specification (TS) 3.3.E.1 requires that only one ACCW pump per

recirculation pump be operable. TS 3.3.E.2 allows both ACCW pumps serving a recirculation pump to be inoperable provided that one of the ACCW pumps is restored to service within 24 hours.

The NRC was concerned that with this electrical configuration and these TS requirements, that the recirculation system was not meeting single failure requirements. For example, the TS would allow the 31 ACCW pump to be out of service indefinitely for maintenance or testing. With the 31 ACCW pump inoperable, assume a single failure of 480 volt bus 6A during a loss of coolant accident (LOCA) coincident with a loss of offsite power (LOOP). The failure of bus 6A to energize would directly result in the loss of recirculation pump 32 due to the loss of its power supply. The 32 ACCW pump would also not have any power, and this coupled with the 31 ACCW being inoperable for maintenance would result in the 31 recirculation pump also being inoperable. Therefore, the single failure of bus 6A could result in the loss of both 31 and 32 recirculation pumps during a LOCA. The NRC raised this concern to the operators in the control room prior to the plant exceeding the cold shutdown condition on December 2, 1995. At that time, all ACCW pumps were operable so there was not an immediate operability or safety concern.

The NRC noted that this question had not been resolved by December 4, 1995, and notified operations and licensing management of this concern. After a review of the issue by licensing personnel, this concern was documented by DER 95-2863 on December 13, 1995. The NYPA plant leadership team (PLT) discussed this issue on December 14, 1995, and decided that a TS interpretation was needed to clarify TS 3.3.E requirements. The PLT also decided that in the interim, guidance would be given to the operators that the 24 hour action statement of TS 3.3.E.2 should be applied to any inoperable ACCW pump.

On December 18, 1995, the NRC noted that the TS interpretation for ACCW pump operability had not yet been issued. The NRC also noted that guidance had not been provided to the operators via the operations shift orders or standing orders, and several operators interviewed by the NRC were not aware of the problem. The NRC again raised this issue and the lack of timely resolution to the PLT on December 18, 1995. The TS interpretation was issued later that day and the interpretation was highlighted in the operations shift orders.

The TS interpretation concluded that credit could not be taken for the 32 and 33 ACCW pumps to meet TS 3.3.E, and that the 24 hour action statement specified in TS 3.3.E.2 would be applied if either 31 or 34 ACCW pumps were out of service. A similar power supply problem involving the emergency diesel generator cubicle ventilation fans was identified by NYPA and documented in licensee event report (LER) 95-15. Based on the repetitive nature of these design deficiencies, NYPA engineering developed an action plan to review all safety related systems to determine other similar configurations which may not have adequate TS requirements to control the allowed outage times. The NRC reviewed the TS interpretation and the engineering action plan and concluded that they adequately addressed the operability of the ACCW pumps and other similar configurations for plant startup.

The evaluation of DER 95-2863 was reviewed and closed by the operations review group (ORG) on January 11, 1996. NRC review noted that the DER evaluation did

not include a review of past operability of the ACCW pumps or evaluate the reportability requirements. The NRC independently reviewed the work history of the ACCW pumps and noted two instances in 1991 and 1992 when the 31 ACCW pump may have been out of service for more than the 24 hours specified by TS 3.3.E.2. The failure to evaluate past operability and reportability was documented by NYPA in DER 96-097.

The NRC concluded that NYPA response and evaluation of the ACCW pump power supply design question was weak. The concern was not promptly documented in a DER after the issue was brought to the attention of the operators by the NRC. Afterward, inadequate communications in the operations department resulted in guidance to the operators not being developed in a timely manner. NYPA did not adequately evaluate the reportability of the ACCW pump power supply design prior to closing DER 95-2863. This item will remain open pending the completion of the NYPA reportability evaluation and the completion NYPA's review of the adequacy of the reportability process. (URI 95-17-02)

4.2 Service Water Leak in Zurn Strainer Pit

On January 11, 1996, the NRC noted a several gallon per minute service water (SW) leak coming from the 33 Zurn strainer blowdown line. The leak was located in a section of three inch carbon steel non-code piping. At the time of discovery, the leak was covered by lagging and the water was leaking out from under the lagging. The Zurn strainer blowdown system normally operates automatically for a period of five minutes per hour to clean the strainers. However, due to reduced heat loads in cold shutdown, all six of the strainer blowdown lines were operating in bypass to reduce total system flow.

On December 19, 1995, the system engineer identified a slight leak of several drops per minute coming from the pipe and initiated a plant identified deficiency (PID) tag. Based on input from the system engineer, the PID tag was dispositioned in the system as "outage to be determined." This means that the discrepancy was not intended to be worked in the current plant outage. On the morning of January 11, the leak rapidly degraded from several drops per minute to several gallons per minute in a period of about 20 minutes. Leakage in the Zurn strainer pit is not desirable because the pit contains all of the SWS Zurn strainers as well as strainer instrumentation, motors and solenoid control valves, albeit the pit does have a sump pump to remove water leakage.

The NRC questioned the adequacy of the evaluation performed for deferral of the PID on December 19. The repair of the leak was deferred without removing the lagging and performing visual or ultrasonic evaluations of the discrepancy. Evaluation of the piping performed after the leak worsened identified several perforations in the area of the leak and small through wall leaks in the piping to the 31 and 32 Zurn blowdown lines. The system engineer stated that his recommendation to defer the repair was based on the fact that the leak was in non-code piping and experience with similar leaks indicated the piping would not fail catastrophically and cause an operational problem. However, operations stated that the isolation required for repairs to the piping would require the plant to return to the cold shutdown condition.

Prior to performing the replacement of the pipe, NYPA removed the lagging and applied a temporary rubber patch to contain the spray. However, on January 12, when the inspector toured the work site, there was significant water leaking from the patch and splashing onto adjacent electrical components. The NRC informed the shift manager who directed that the water spray be contained.

By the end of the inspection period, NYPA had examined all of the similar piping in the Zurn strainer pit and scheduled the replacement of piping segments to three strainers. Prior to this event, several other piping segments had been replaced. Because the condition of small bore service water piping continues to be a problem, NYPA established a team to evaluate the problem and determine long term corrective actions. In response to NRC concerns regarding the thoroughness of the leak evaluation, NYPA established a policy of removing insulation to evaluate leaks whenever possible prior to the determination of the work priority. NYPA also began removing the lagging from previously identified leaks that were being deferred in the work control system.

The NRC concluded that the evaluation of the PID prior to deferral was weak. The sudden failure of the piping could have caused an unnecessary challenge to the operators if the failure had occurred after plant restart. Additionally, the failure to adequately contain the leakage indicated a lack of ownership by the maintenance and operations personnel involved with the work. The evaluation performed after the identification by the NRC was thorough and provided adequate bases for limiting the scope of repairs. The evaluation will also address long term service water system operability concerns.

4.3 Appendix-R Diesel Generator Tripped on Overspeed

On December 16, 1995, the Appendix-R diesel generator tripped during the start sequence. Subsequent troubleshooting determined the engine tripped on overspeed due to the fuel rack booster remaining pressurized too long into the start sequence. The fuel rack booster functions to assure full fuel delivery during starting for diesels that are required to start and load quickly.

The fuel rack booster receives starting air when the air start system air operated valves (AOVs) are open supplying the air start motors. The air start system includes two air start AOVs and two air start motors. Only one AOV and air start motor is used at a time to start the engine, and the failure only occurred with one of the air motors. To correct the problem, both AOVs were replaced. During the restart, one valve failed to function and was again replaced. This failure was attributed to be a faulty replacement valve. Both AOVs were previously replaced in October 1994 after one valve failed to close properly.

As a result of the recent failures, NYPA performed inspections of the starting air system. Although some moisture and rust was found in the accumulator, the 40-micron screen and downstream piping was noted in excellent condition. NYPA also noted the starting air pressure regulator allowed pressure to become too high. Pressure downstream of the regulator was observed to be 198 psi immediately after a start. The normal pressure is 120 psi and the design

pressure is 150 psi. Approximately 20 minutes after the start, the pressure returned to the normal band with no flow through the regulator.

The system engineer made an assessment that the high pressure was the probable cause of the valve failures. After testing, the diesel was returned to operable status on January 13, 1996, based on the system engineer's determination that the regulator caused the AOVs to fail over a long period of time. However, through documentation reviews and interviews, the NRC was concerned that the mechanism of the failure was not adequately understood to support the operability determination. Additionally, as part of the justification for restoration of the diesel, a priority work item was issued to replace the pressure regulator prior to the next monthly engine test. Additionally, a modification to remove the fuel rack booster was scheduled to be accomplished as a priority. The fuel rack booster is not required for the Appendix-R application because the diesel is not intended for fast loading.

The inspector noted that the operability determination did not specify the assumption as to how long the regulator has been spiking or any trend on the spikes. Further, the expected number of cycles to failure was not adequately described. From October 1994, the diesel was started only 17 times. This means that each AOV would have only operated eight or nine times. However, the system engineer characterized the failures as long term. The failure analysis from the 1994 AOV failure was not reviewed to determine if the cause was similar. Based on these questions, the NRC expressed concern whether the effect of the spiking regulator was adequately considered and the AOVs would continue to fully close within the time required to prevent engine overspeed during starting.

The NRC raised concerns to the general manager of operations following an event involving the loss of 138 KV power to the station on January 20, 1996. In response to NRC concerns, operations tested the Appendix-R diesel. Although the diesel started properly, the NRC remained concerned that the root cause of the failure had not been corrected and could degrade the AOVs. Operations again evaluated the concerns and declared the Appendix-R diesel inoperable on January 22, 1996. Engineering evaluation of this issue is continuing.

The NRC concluded that the initial troubleshooting of the overspeed condition was good. The recommendation to remove the fuel rack booster per a design modification will remove a potential failure mode. However, the bases for the operability determination was weak. Although the operability of the diesel was questioned with regard to the spiking pressure regulator, the evaluation was informal and the root cause of the AOV failures was not fully considered.

5.0 PLANT SUPPORT (71750, 902904)

5.1 Radiological Controls

Radiological protection activities were reviewed on a periodic basis. Posting and control of radiation and high radiation areas were inspected; radiation work permit compliance and use of personnel monitoring devices were checked. Conditions of step-off pads, disposal of protecting clothing, radiation control job coverage, area monitor operability and calibration (portable and permanent), and personnel frisking was observed on a sampling basis. Licensee personnel were observed to be properly implementing the radiological protection program.

5.2 Security

Implementation of the physical security plan was observed in various plant areas with regard to the following: protected area and vital area barriers were well maintained and not compromised; isolation zones were clear; personnel and vehicles entering and packages being delivered to the protected area were properly searched and access control was in accordance with approved licensee procedures; persons granted access to the site were badged to indicate whether they had unescorted access or escorted authorization; security access controls to vital areas were maintained; and security personnel were alert and knowledgeable regarding position requirements. Licensee personnel were observed to be properly implementing the Physical Security Plan.

On December 8, 1995, a vital area security door was inadvertently removed by maintenance personnel. Prior to performing this work, a pre-job brief was held by maintenance with security personnel present. It was discussed and agreed upon that prior to removing the door, a security officer would have to be present to provide compensatory measures for the inoperable security door. However, the maintenance workers then proceeded to remove the door prior to security being present. Maintenance determined that the pre-job brief was not effective in that the maintenance workers did not understand the importance of having security present prior to removing the door. Maintenance is working to improve and formalize the pre-job brief process. Security immediately responded to the alarm generated by the removed door and provided the required compensatory measures. This included posting a security officer at the door and performing a search of the building for unauthorized entry.

On December 24, 1995, a NYPA security guard was found inattentive to duty while assigned for a brief time at a compensatory post for degraded detection capability at the site perimeter. When discovered, compensatory measures were taken as required and the guard was relieved of duty. Assessment of the affected area was promptly verified.

The NRC reviewed both of these events with security management and determined that NYPA security responded well to both. NYPA security took prompt and appropriate compensatory measures for both of these events. The reportability criteria was also appropriately reviewed and applied by NYPA security.

5.3 Housekeeping

The NRC assessed the control of plant housekeeping in safety-related areas. General plant housekeeping in normally accessible safety-related areas and containment during the period was adequate. NYPA generally kept these areas free of dirt and debris. Transient equipment was properly secured. However, the NRC noted several plant areas which contained safety-related equipment had lighting deficiencies. NYPA is reviewing the process of lighting deficiency identification and repair.

5.4 Station Response to Blizzard

On January 7 and 8, 1996, the area was hit with a blizzard which resulted in considerable snow accumulation. NYPA invoked procedure OD-8, Guidelines for Severe Weather, in preparation for the storm. The NRC observed the preparations, and monitored the situation at the station throughout the storm. Prior to the storm, NYPA performed the required walkdowns and verified the readiness of key equipment. During the storm, the availability of essential personnel was assessed and effectively managed by plant management. Security maintained watches and promptly cleared fouled perimeter alarms. Support personnel kept required passage ways clear of snow. After the storm, NYPA evaluated lessons learned and revised OD-8 where warranted. The NRC concluded that NYPA maintained and appropriate safety focus and effectively managed the challenges posed by the storm.

6.0 ADMINISTRATIVE

6.1 Management Changes

Effective January 8, 1996, NYPA announced further restructuring of their plant organization. Mr. Robert Barrett was appointed as the General Manager of Operations, reestablishing the position which was previously deleted. Mr. Barrett was formerly the General Manager of Operations for NYPA's FitzPatrick plant. Under the reorganization, operations, site planning and outage services, radiological and environmental services, training, and performance and reliability will report to the General Manager of Operations.

The position of General Manager of Support Services was also eliminated. This resulted in other restructuring: Configuration management now reports to engineering; the Operations Review Group now reports to the Tactical Assessment Coordinator; and, Security and emergency preparedness now report directly to the Site Executive Officer.

Additionally, procurement engineering was moved from maintenance to engineering. Other procurement and warehouse functions will come under a new materials management department that will report to the Vice President of Procurement and Real Estate in the White Plains Office.

6.2 NRC Inspection Exit Meeting

At periodic intervals during the inspection, meetings were held with senior facility management to discuss the inspection scope and findings. The issues in this inspection were discussed with site management throughout this inspection, and an exit meeting was held on January 23, 1996, to discuss the findings and conclusions of this report period. During the discussion, the licensee did not identify any 10 CFR 2.790 material and did not take exception to any of the findings of this inspection.

ATTACHMENT 1

PREDECISIONAL ENFORCEMENT CONFERENCE

DECEMBER 13, 1995

LIST OF ATTENDEES

NEW YORK POWER AUTHORITY - INDIAN POINT 3

W. Zizzo, Co-Chairman, Local 1-2
J. Kelly, General Manager, Supply Services
S. Zulla, Director, ISEG
J. Lafferty, System Engineer Manager
M. Caskey, Operations Senior Reactor Operator
C. Faison, Director, Nuclear Licensing
R. Patch, Director, Quality Assurance
R. Deasy, Vice President, Appraisals and Compliance
T. Williams, Shift Mentor
R. Bernard, Shift Manager
M. Pearson, Operations Manager
L. Hill, Site Executive Officer
W. Cahill, Chief Nuclear Officer
K. Peters, Licensing Manager
D. Spoerry, General Manager, Training
J. Steets, Public Affairs Manager
F. Cherry, Nuclear Plant Operator
C. Yun, Nuclear Plant Operator
R. Barrett, Assistant to Site Executive Officer
B. Vangor, Field Support Supervisor
D. Vinchkoski, Field Support Supervisor
J. Comiotes, Tactical Assessment Coordinator
A. Levine, Senior Attorney

U.S. NUCLEAR REGULATORY COMMISSION

K. Smith, Regional Counsel, RI
T. Martin, Regional Administrator, RI
R. Cooper, Director, Division of Reactor Projects, RI
C. Cowgill, Chief, Projects Branch 2, RI
J. Beall, Enforcement Specialist, NRR (via phone)
C. Goodman, Human Factors Representative, NRR (via phone)
G. Meyer, Chief, Operator Licensing and Human Performance Branch, RI
B. Welling, Reactor Engineer, RI
D. Lew, Senior Resident Inspector, RI
J. Harold, Acting Project Manager, NRR
J. Zwolinski, Deputy Director, DRPE, NRR
R. Reyes, Reactor Engineer, RI
R. Barkley, Project Engineer, RI
D. Holody, Technical Program Manager, RI

ATTACHMENT 2
NRC/NYPA MEETING
DECEMBER 13, 1996
SLIDES

Meeting Between NRC

and

Indian Point 3

Docket No. 50-286

December 13, 1995

Agenda

1. *Opening Remarks*
2. *Brief Sequence of Events*
3. *Summary of Causes*
4. *Initial Corrective Actions*
5. *Ongoing Actions and Results Seen*
6. *Analysis of Event*
7. *Closing Remarks*

Brief Sequence of Events

Date (and Approx. Time)	Event
9/17/95	Plant achieved cold shutdown using Plant Operating Procedure 3.3.
10/14/95 (2000-2300)	Control Room Supervisor (CRS) initialed steps in pre-200°F checklist for containment spray and recirculation pumps as "operable."
10/15/95 (0630)	Majority of pre-200°F checklist completed by the Shift Manager.
10/15/95	Oncoming shift manager completed last pages of checklist. At about 0820, he received the Site Executive Officer's permission to exit cold shutdown.
10/15/95 (1100)	CRS turnover completed with heatup in progress (temp. approx. 190°F).
10/15/95 (1125)	Exceeded cold shutdown.
10/15/95 (1523)	Quality Assurance auditor noted switches in trip pullout and informed the CRS.
10/15/95 (1533)	Switches returned to the "automatic" position.

Summary of Causes

- *Control Room Supervisor Failed to Follow Procedure*
- *Lack of Awareness of Equipment Status by Reactor Operators*
- *Shift Manager Failed to Verify Switches in "Automatic"*
- *Procedures*

Initial Corrective Actions

- *Switches Returned to "Automatic" within Approximately 10 Minutes of Discovery*
- *Board Walkdown*
- *Suspended Plant Heatup*
- *Disciplinary Action*
- *Procedure Change (TPC)*
- *Re-organized Operations*
- *Tactical Assessment Group*

Why?

What's Different Now?

What I Saw

- *Ownership and Accountability*
- *Planning and Preparation*
- *Training Effectiveness*
- *Self Assessment*

Ownership and Accountability

- *Candor and Attention to Detail*
- *Learn from Mistakes*
- *Procedural Compliance*

What We're Seeing

OWNERSHIP AND ACCOUNTABILITY

- *Positive Feedback*
- *Conservative Decisions*
- *Questioning Attitude*

CCW Relief Valve

EVENT DESCRIPTION

- *CCW Isolated to RHR Heat Exchangers*
- *Relief Valve Lifted*
- *Several Water Additions to CCW Surge Tank*

PROBLEMS

- *Inadequate Procedure Revision*
- *Operator Performance*

CCW Relief Valve

- **CORRECTIVE ACTIONS**

- *Accountability*

- *Shift Routine Changes*

- *Log Keeping Expectations*

- *Improve On-Shift Communications*

- *Post Turnover Briefings*

- *Reinforcement by Shift Mentors*

Planning and Preparation

- *Procedure Upgrades*

- *Trailblazing*

- *Staff Development*

What We're Seeing

PLANNING AND PREPARATION

- *Procedure use*
- *NPO Rounds Analysis*
- *Improved Communications*

Training

- *Management Involvement*

- *Instructor Involvement*

What We're Seeing

TRAINING

- *Independent Evaluation*
- *Shift Manager Ownership*
- *Improved Communications*

Monitoring and Assessment

- *Operations Mentoring Program*
- *Tactical Assessment Group*
- *Rapid Response*

What We're Seeing

MONITORING AND ASSESSMENT

- *Tactical Assessment Group Observations*
- *Procedural Adherence*
- *Rapid Investigations to Keep Raising the Bar*

Analysis of Event

- *Self-Identified*
- *Promptly Returned Switches to "Automatic" Within Approximately 10 Minutes*
- *Strong Corrective Action*
- *Minimal Safety Significance*