U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report:	50-286/96-01
License:	DPR-64
Licensee:	New York Power Authority
Facility:	Indian Point 3 Nuclear Power Plant
Location:	Buchanan, New York
Inspection Period:	January 14, 1996 to March 2, 1996
Inspectors:	R. Rasmussen, Acting Senior Resident Inspector T. Frye, Resident Inspector R. Temps, Senior Resident Inspector, IP2 B. Welling, Reactor Engineer R. Barkley, Project Engineer P. Wilson, Senior Reactor Analyst R. Bhatia, Reactor Engineer G. Morris, Reactor Engineer D. Silk, Reactor Engineer

Approved by: C. J. Cowgill III 3/21/96

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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. 50-286/96-01

Plant Operations: Operators demonstrated good performance in response to a loss of all 138 kv power. However, some weakness in operator performance occurred while responding to the failure of the 32 EDG ventilation system and during the loss of buses 2A and 3A during a protective tagout.

The investigation of high oxygen levels in the waste gas system identified two vent header isolation valves not being operated as described in the FSAR. Additionally, the operations department use of an alarm response procedure for routine operation of these valves resulted in a loss of configuration control.

A discrepancy between the emergency operating procedures and the FSAR involving the operation of the containment spray system was identified by the NRC.

Maintenance: Leaking lube oil check valves resulted in two emergency diesel generators becoming inoperable. The identification of the failures were a result of the management observation program.

NRC identified two instances of scaffolding installed near safety related equipment that did not meet NYPA procedural requirements. This was a violation of NRC requirements. **(VIO 96-01-01)**

The NRC considered NYPA's actions to address fan cooler unit leakage to be good.

The NRC performed a detailed walkdown of the auxiliary feed water system and concluded that the AFW system was generally being maintained in a condition to support its intended safety function. However, a number of system deficiencies were identified or revealed through NRC inspection. These items indicated weaknesses in operations and system engineering walkdown/inspection processes.

Engineering: NRC identified three examples of incomplete engineering by the design engineering group. Additionally, in two of the cases maintenance and system engineers did not identify the requirement for design engineering input and developed temporary solutions without formally addressing the problem or implementing the corrective action process. The incomplete engineering was a violation of NRC requirements. **(VIO 96-01-02)**

A system engineering evaluation report prepared to evaluate the aggregate impact of service water system leakage revealed a heightened level of awareness and sensitivity to service water system leaks.

NRC reviewed the failure analysis and corrective actions taken for a failed lightning arrester in the 138 kv electrical system and the failure of 31 EDG output breaker to function and determined these actions were appropriate.

NYPA failed to effect appropriate immediate corrective actions for the identified inoperability of backup power to containment lighting. This was identified when containment lighting was lost due to the loss of the 138 kv power. After the 138 kv power event, appropriate corrective actions were taken.

NRC review of the engineering analysis performed related to four previously-identified performance-related electrical issues determined the corrective actions were appropriate, and the root cause determination evaluations were comprehensive with adequate basis.

NYPA review and evaluation of the effects of low temperature in the electrical tunnel were good. The review of all equipment in the tunnel for low temperature effects was a good initiative by Design Engineering.

Workers installing temporary heaters deviated from the PORC approved method for routing the cables and the system engineer permitted this without changing the temporary modification. Awareness of cable separation concerns was lacking in this, as well as other electrical temporary modifications.

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

The security response to the loss of 138 kv power was timely and appropriate.

NRC inspection of areas not routinely accessed indicated declining material condition, unidentified deficiencies and poor implementation of housekeeping standards. Examples include the boron injection tank room, the waste holdup pit area and the spent fuel pool cooling pump pit.

The continuing trend of oxygen intrusion into the PRT and CVCS HUTs was well evaluated by NYPA chemistry and operations personnel. Thorough reviews of system operating parameters and procedure instructions identified several necessary equipment repairs and procedure enhancements to prevent future oxygen intrusion.

The operating experience program was effectively implemented by NYPA staff in response to the low service water temperature issue identified at another utility.

Safety Assessment/Quality Verification: Several LER's were reviewed, found acceptable and are considered closed. URI 92-28-11, Adequacy of Flow Measurement for Radioactive Machine Shop Building Vent Calibration Check, was reviewed and closed.

NYPA identified three cases involving procedure changes made with inadequate 10 CFR 50.59 reviews. This issue is left unresolved pending further NYPA evaluation and NRC review to ensure that plant procedures are reflective of the licensing basis of the plant. **(URI 96-01-03)**

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DETAILS

1.0 SUMMARY OF PLANT ACTIVITIES

1.1 New York Power Authority (NYPA) Activities

At the start of the inspection period, the plant was in cold shutdown after having cooled down from hot shutdown to repair a leaking steam generator hand hole on December 25, 1995. On January 20, 1996, all 138 kv power was lost due to a failed lighting arrester on the station auxiliary transformer. Heatup to above 200°F was accomplished on February 25, 1996. Heatup to greater than 350°F occurred February 28, 1996, and the plant achieved normal operating temperature and pressure at the end of the period on March 2, 1996.

1.2 NRC Activities

From January 25 through March 1, 1996, the NRC conducted a special operational readiness review inspection. The findings and conclusions from this inspection are contained in NRC inspection report 50-286/96-02.

Regional inspectors the Division of Reactor Safety, Electrical Branch, conducted inspections on-site between January 22 and February 1, with follow-up inspection activities performed in the Region I office during the weeks of January 22, February 5 and 12, 1996. The results of these inspections are contained in this report.

On February 5-6, 1996, the Deputy Regional Administrator toured the facility and interviewed licensee personnel. On February 29 and March 1, 1996, the NRR project directorate visited the site in preparation for the systematic assessment of licensee performance (SALP) review. March 2, 1996, was the end of the SALP period which began January 9, 1995. The Project Branch Chief toured the facility periodically. During the tours he met with utility personnel, toured the plant and provided oversight to the inspectors.

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: Containment spray system, 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
- Screenwell Structure
- Auxiliary Feed Pump Building

- Diesel Generator Building
- Primary Auxiliary Building
- Containment Building
- Containment Penetration Areas
- Spent Fuel Building
- Waste Holdup Tank Area
- Turbine Building
- Intake Structure
- Access Control Points
- Yard Areas
- BIT Room

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 Loss of Offsite Power

On January 20, 1996, IP3 experienced a complete loss of offsite power (LOOP) from the 138 kv power system. In response to this event, the 32 and 33 emergency diesel generators (EDGs) automatically started and restored power to 480 v buses 5A and 6A. The 31 EDG also started and loaded onto its 480 v bus, however the 31 EDG output breaker tripped opened approximately 12 seconds later. Therefore 480 buses 2A and 3A, and their associated safety related equipment, remained deenergized. NYPA declared the 31 EDG inoperable due to the failure to automatically energize the safety related loads on its 480 v bus.

Residual heat removal (RHR) cooling for the core was restored approximately 5 minutes after 138 kv power was lost. No appreciable heatup of the reactor coolant system was observed while RHR cooling was secured. Spent fuel pool cooling was restored approximately one hour after the loss of offsite power. There was also no appreciable heatup of the spent fuel pool water observed. After stabilizing plant conditions and verifying the extent of faults on the electrical distribution system, offsite power to the 480 v buses was restored using an alternate 13.8 kv power source about 3 hours after the loss of 138 kv power. The loss of 138 kv power and actuation of the EDGs was reported to the NRC as required by 10 CFR 50.72 and licensee event report (LER) 96-02.

Operator response to the LOOP event was very good. Strong command and control of the event was evident with a good approach taken to stabilizing plant conditions and restoring an alternate source of offsite power when appropriate. The operators reviewed the emergency plan and appropriately concluded that the declaration of an Unusual Event was not required. Other departments provided good support to the plant during the event. I&C personnel provided significant assistance to operations in analyzing the cause of the loss of power and evaluating the electrical distribution system protective relays which actuated in response to the electrical disturbance. Security personnel took appropriate and timely compensatory measures in response to the event. All vital areas were searched to verify no unauthorized entry and extra watches were established for the protected area fence and the 13.8 kv power supply.

While securing the 32 EDG after restoring 13.8 kv offsite power, a nuclear plant operator (NPO) noted that the 32 EDG cubicle ventilation system was not operating as required. The NPO promptly reported this information to the control room operators. However, operations management was not informed of this situation by the control room operators until about 3 hours later. Operator performance in identifying and responding to the failure of the 32 EDG ventilation system is discussed in NRC inspection report 50-286/96-02.

NYPA declared the 32 EDG inoperable due to the failure of the ventilation system. The 32 EDG ventilation system failed due to a loss of control air to the ventilation dampers. NYPA repaired the ventilation control air system and restored the 32 EDG to operation approximately 24 hours after the failure.

NYPA investigation revealed that the cause of the loss of 138 kv power was attributed to the failure of a surge arrestor on the A phase of the station auxiliary transformer (SAT). The failure of the 31 EDG output breaker was attributed to a loose wire on the undervoltage sensing circuit for the breaker. The causes of these failures and corrective actions taken by NYPA are discussed in section 4 of this report.

Another instance of weak operator performance occurred while applying a protective tagout to the SAT on January 21. While removing fuses for the SAT to facilitate troubleshooting, an NPO inadvertently removed fuses for the 6.9 kv power supply to 480 v bus 2A. This resulted in the loss of power to 480 v buses 2A and 3A and the initiation of a start signal to the 31 EDG due to the undervoltage condition on bus 2A. The 31 EDG did not start since it was still out of service and the loss of electrical equipment on buses 2A and 3A did not adversely affect plant operation. NYPA reported the EDG start signal actuation to the NRC per LER 96-03. The licensee concluded that the loss of power to buses 2A and 3A was due to inattention on the part of the NPO while applying the protective tagout.

The plant remained on the 13.8 kv source of offsite power pending the restoration of the EDGs and the availability of 138 kv power via the SAT. While operating on 13.8 kv offsite power, NYPA elected not to restart a reactor coolant pump so as not to challenge the power source. NYPA developed temporary operating procedure (TOP)-117 to provide instructions for operating the plant with a pressurizer bubble without a reactor coolant pump operating. The NRC reviewed the final safety analysis report (FSAR) and determined that this mode of operation was consistent with the FSAR. However, the NRC concluded that the evaluation of TOP-117 as required by 10 CFR 50.59 was weak in that the nuclear safety screen prepared for TOP-117 did not provide the detailed bases required by MCM-4 to determine that a NSE was not required. The NRC discussed this observation with plant management and was still being evaluated by NYPA as of the end of this inspection period.

When the SAT repairs were completed, the licensee intended to use procedure SOP-EL-5, Operation of Onsite Power Sources, to restore the 138 kv power supply. With the 31 EDG still inoperable, this would have resulted in the cross-connection of 480 v buses. Prior to performance, the licensee questioned whether this operation had been properly evaluated. Operations appropriately elected to wait for the repairs to the 31 EDG to complete so that the 480 v buses were not cross-connected during the power restoration.

Operations management issued a shift order directing that 480 v buses would not be cross-connected pending further evaluation. The NRC was concerned that an operating procedure may have contained instructions which may not have been evaluated as required by 10 CFR 50.59. This issue and other examples of plant procedures which may have not been properly evaluated in accordance with 10 CFR 50.59 is discussed further in section 7.2 of this report.

The NRC concluded that the NYPA staff overall responded well to the LOOP event. The operations staff responded well to the event and restored the plant to a stable condition in a safe and timely manner. Other departments, particularly I&C and security, provided valuable assistance to operations in responding to the event. The review and identification of concerns with cross-connecting 480 v buses demonstrated a good questioning attitude by NYPA. Some weakness in operator performance occurred while responding to the failure of the 32 EDG ventilation system and during the loss of buses 2A and 3A during a protective tagout.

2.3 Valves AOV-1786/1787 Out of Position

While investigating the cause of the January 12, 1996, oxygen intrusion into the waste gas system, NYPA identified that Vent Header Containment Isolation Valves AOV-1786/1787 were in positions contrary to those required by the FSAR. These valves were found in the open position, contrary to FSAR table 5.2-3 which requires the valves to be closed with the plant shutdown. These valves had been aligned to the open position in accordance with checkoff list (COL)-WDS-2, Gaseous Waste Disposal System. NYPA initiated DER 96-0073 to evaluate this discrepancy and closed the valves.

NYPA prepared an action plan to review and revise operating procedures as necessary to ensure that valves AOV-1786/1787 were maintained in the correct positions. NYPA review of COL-WDS-2 noted that term procedure change (TPC) 95-1029 was written on July 7, 1995, to change the positions of these valves from the closed to the open positions. This was the correct position for these valves at the time since the plant was operating and FSAR table 5.2-3 requires the valves to be open. However, this change to COL-WDS-2 did not recognize that the FSAR stated that the normal position for these valves is open while operating, but closed when shutdown.

NYPA reviewed the preparation of TPC 95-1029 and concluded that it was inadequate and not properly performed. NYPA concluded that a nuclear safety and environmental impact screen should have been performed on this TPC since it modified a system flow path and alignment as described in the FSAR. The NRC noted that this failure to evaluate a TPC as required by 10 CFR 50.59 was similar to an inadequately prepared TPC to SOP-RHR-1 which was characterized as a severity level IV violation in NRC inspection report 50-286/95-17. As of the end of this inspection period, NYPA was still implementing corrective actions for the violation and evaluating the similar problem with TPC 95-1029. This issue and other examples of plant procedures which may have not been properly evaluated in accordance with 10 CFR 50.59 is discussed further in section 7.2 of this report.

On February 3, 1996, an operator identified that valves AOV-1786/1787 were again open, contrary to closed as required when the plant is shutdown. NYPA initiated DER 96-285 to document this discrepancy. The NRC was concerned with this occurrence since it indicated that NYPA had apparently lost configuration control of valves AOV-1786/1787 again. DER 96-285 was closed by the operational review group (ORG) to DER 96-0063, which chemistry was evaluating in response to oxygen intrusion in the waste gas system. Subsequently the inspectors asked NYPA about DER 96-285. Chemistry had not evaluated this DER and stated that it was being captured by operations in the response to DER 96-0073. The NRC reviewed the response to DER 96-0073 and concluded that DER 96-285 was not addressed. The NRC concluded that DER 96-285 was not evaluated. ORG initiated DER 96-639 to document the inadequate process of closing DERs to previous ones and reopened DER 96-285.

NRC review of the events surrounding valves AOV-1786/1787 being found open again on February 3 concluded that these valves had been operated approximately twice a day from January 16 to February 23, 1996, using alarm response procedure (ARP)-22. ARP-22 was being used by operations during this time period to vent the reactor coolant drain tank (RCDT) due to pressure buildup caused by maintaining valves AOV-1786/1787 shut. The NRC concluded that these valves were found open on February 3 because ARP-22 did not contain adequate instructions to ensure the valves were shut after RCDT venting was completed. Further, the action plan developed by operations to revise procedures was inadequate because it did not include ARP-22. The NRC noted that ARP-22 was revised on February 5, 1996, to provide adequate configuration control for the valves.

The NRC questioned the appropriateness of using an ARP to perform routine plant evolutions such as venting the RCDT twice a day for approximately 5 weeks. The NRC also noted that on several occasions, the actions of the ARP were entered prior to receiving the high pressure alarm on the RCDT. Both of these operating practices were similar to the operation at reduced pressure event in July 1995 which was precipitated by the use of ARP-3 to reseat a leaking pressurizer safety valve. A review of operator logs also indicated that configuration control of valves AOV-1786/1787 was also lost from the period from January 22 to February 3, 1996, in that the valves were open during that period. The inspector noted that on January 28 it was logged that the valves were found open and that the CRS was notified. However, the CRS did not direct that the valves be shut and no apparent action such as initiating a DER was taken.

A TPC to SOP-WDS-2 was written on February 23, 1996, to provide instructions for the routine venting of the RCDT. At this time, the routine use of ARP-22 was discontinued. Additional corrective actions taken by NYPA included briefing operators on management expectations for ARP usage. Similar instances of a weak questioning attitude in identifying deficient conditions were noted in NRC inspection report 50-286/96-02. These observations were discussed with NYPA plant management and corrective actions were being implemented as of the end of this inspection period.

2.4 FSAR Issue with the Containment Spray System Operation

While reviewing Section 6.3.2 of the FSAR for design information on the sodium hydroxide (NaOH) tank associated with the containment spray system, the inspector noted a discrepancy between the FSAR and the plant's emergency operating procedures (EOPs). Specifically, in discussing the two-minute time delay feature associated with the injection of NaOH following receipt of a containment spray signal, the FSAR states that emergency procedures will provide guidelines to the operators on terminating NaOH addition during the two-minute time delay in the event of a false containment spray signal. Section 6.3.2 also contained a paragraph that implied that instructions would be provided to operators for distinguishing between valid containment spray actuations and invalid ones in determining whether to cancel NaOH addition.

A review of the EOPs by the inspector indicated no such instructions exist. In accordance with the "symptom-based" EOPs implemented in the late 1980s, operators are not required to distinguish between valid and invalid actuation of the containment spray system while initially responding to an event. Natural progression through the EOPs will eventually require the operators to evaluate whether containment spray is required, and if it isn't, will direct the operators to secure the system at that time. Inadvertent admission of NaOH into containment from a false containment spray signal actuation is not a consideration in the "symptom-based" EOPs. The methodology of these EOPs was created by the Westinghouse Owners Group (WOG) and approved by the NRC as part of the post-TMI action plan requirements.

In discussing this issue with the operations group procedure writers, it appears that the information in the FSAR was previously contained in the old-style "event-based" emergency procedures which were supplanted by the "symptom-based" EOPs. The inspector concluded that when the new EOPs were issued, Section 6.3.2 of the FSAR should have been revised to reflect the changed methodology with respect to containment spray actuation. NYPA issued a DER to track this issue to ensure future revision of the FSAR.

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.

WR 96-00202-00	Replace BFD-FCV-406A Positioner
TM 96-00266-00	RC-513 Packing Gland
WR 96-00278-01	Repair 33 FCU Flanges
WR 96-06388-00	Replace 32 EDG Air Regulator
WR 96-00418-06	Repair 32 EDG Ventilation System
WR 95-06006-08	Maintenance of the Electrical Tunnel Fire Dampers
WR 96-00172-00	Replace Piping to the 32 Service Water Zurn Strainer
WR 96-00421-00	Rebuild Valve RC-AOV-519
WR 96-00252-00	Replace Valve BFD-65-15

3.2 Emergency Diesel Generator Lube Oil Check Valves

On March 1, 1996, NYPA made a one hour non-emergency notification to the NRC regarding the inoperability of two of their three emergency diesel generators (EDG's). The EDG's were declared inoperable due to leakage through lift check valves in the engine lube system. The leakage through the valves had the potential of over oiling the valve train components on the cylinder head area of the EDG's which could lead to hydraulic locking of the EDG's during a start. Hydraulic lock could be caused by excess oil leaking through the valve guides and collecting in the cylinder with the EDG in standby.

On February 29, 1996, a maintenance supervisor detected a leaking check valve on the 33 EDG by feeling the piping downstream of the valve during a management observation tour. The EDG was declared inoperable and repair activities were initiated. At that time the other EDG's were checked for similar conditions and found satisfactory. As required by technical specifications, the operability of the remaining EDG's was subsequently verified. After operation of the 32 EDG, the shift manager found a second leaking check valve on the 32 EDG which placed the plant outside of the design basis. A one hour non-emergency notification was made as required by 10 CFR 50.72. The 33 EDG was repaired and declared operable at 0650 hrs on March 1, 1996. The 32 EDG was declared operable at 0502 hrs on March 2, 1996.

The lift check valves were designed to seat during operation of the pre-lube pump and pass flow during operation of the main motor driven pump. Three of the valves are used per EDG to prevent over lubrication of both banks of cylinder head assemblies and the turbocharger. The valves were designed to operate in any orientation, however NYPA reports industry experience with failures of the valves mounted in the horizontal position.

NYPA had an annual maintenance requirement to replace the check valves. Prior to the failure of the 33 EDG check valve, this procedure was successful in preventing failures of the valves. The 33 EDG check valve was not overdue, however it was near the one year point. The 32 EDG check valve had only been in service for a few months prior to the failure. Additionally, the 32 EDG check valve was a new style that was installed by a design modification because the original valves were no longer available.

NYPA sent the failed 32 EDG check valve to a laboratory for failure analysis. Conclusions as to the acceptability of the valves will be based on this analysis. As immediate corrective actions, the maintenance engineer was periodically walking down the check valves, and a change was being written to the engine operating procedure to verify the check valves reseat after a run. The NRC considered these actions adequate, however it was noted that the operations department personnel were generally unaware of the details of the valve failure, how it was detected, how the valve made the EDG inoperable or what the immediate corrective actions were.

The NRC concluded that the identification of the leaking check valves was a significant finding resulting from the management observation program. The immediate corrective actions were adequate, and the maintenance engineering analysis of long term corrective actions was a positive initiative.

3.3 Unevaluated Scaffolding Erected Adjacent to Safety-Related Equipment

On February 6, 1996, the inspectors identified two scaffolds that NYPA erected without the required minimum clearance from safety-related equipment. NYPA had erected one scaffold in the 33 emergency diesel generator (EDG) room that had less than the minimum allowed clearance (less than 1 inch) at three locations from the 33 EDG safety-related support components. In addition, NYPA had erected another scaffold in the control building switch gear room that had less than minimum allowed clearance (less than 1 inch) from safety-related conduit. IP3 design engineers had not evaluated these scaffolds for its potential impact on the adjacent safety-related equipment.

Prior to December 1995, NYPA performed engineering evaluations of all scaffolds erected in safety-related areas to evaluate the scaffold's potential impact on adjacent safety-related equipment during a postulated seismic event. In order to minimize engineer work load, NYPA engineering developed construction standards regarding the building of scaffolding adjacent to safety-related equipment. NYPA engineering concluded that scaffolding built to these standards would not adversely affect safety-related equipment during any postulated seismic event and would not require scaffold specific engineering evaluation.

On December 18, NYPA revised CON-AD-006, "Scaffolding Control" to incorporate the construction standards. NYPA revised CON-AD-006 step 3.1.1 to require, "Scaffolds that are located in safety-related areas and cannot be installed to conform with the technical requirements of this procedure shall be reviewed and approved by a designated Civil/Structural Engineer from Design Engineering-IP3 prior to use." In addition, NYPA revised Step 3.1.9 to state, in part, that a "Scaffold shall maintain at least a 1" clearance with safety-related equipment." NYPA conducted training with all applicable construction department personnel regarding the changes to CON-AD-006.

NYPA took several corrective actions following the identification of the nonconforming scaffolding. Construction department personnel modified the scaffolding in the 33 EDG room to provide sufficient clearance from adjacent safety-related equipment. IP3 design engineers performed an engineering evaluation of the second nonconforming scaffold and concluded it was acceptable without modification. NYPA also performed a review of all scaffolding erected in safety-related areas and found that all other scaffolds had been evaluated by design engineering prior to use. NYPA documented the above problem in Deviation/Event Report (DER) 96-0311.

The inspector concluded that there were no safety consequences as a result of the nonconforming scaffolding. However, 10 CFR 50, Appendix B, Criterion V requires activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedures. Therefore, the failure to erect the above scaffolding without the minimum required clearance and without the prerequisite engineering evaluation was a violation of NRC requirements. **(VIO 96-01-01)**

3.4 Fan Cooler Unit Service Water Flange Repairs

On January 22, 1996, NYPA discovered water on the containment floor near containment fan cooler unit (FCU) 34. Upon further investigation, the source of the water was determined to be service water leakage from the service water supply piping to the FCU cooling coils. NYPA conducted additional walkdowns of the four other FCUs and found similar, minor flange leakage on the 33 and 35 FCUs. A total of twelve leaking flanges were identified by system engineering and quality assurance representatives.

Initially, NYPA began torquing the flange bolting in an attempt to stop the leakage. However, while torquing, maintenance workers discovered inconsistent material fasteners on some of the flanges. Although the service water piping specification, 9321-05-248-35, called for alloy steel bolting, the workers found that some bolting was stainless steel. Further, since the stainless steel bolting was torqued using the higher torque values for alloy steel in the site's bolting specification, NYPA determined that the stainless steel bolting had been overtorqued. NYPA's corrective action was to replace all stainless steel bolting with the alloy steel bolting called out by the piping specification. Inspection of the removed fasteners revealed no signs of overstress. NYPA also found that a material substitution evaluation (MSE) performed in 1987, MSE 87-03-153-FCU, permitted the use of stainless steel fasteners in this application. However, the piping specification was not revised to reflect this evaluation.

The inspector reviewed NYPA's actions to resolve the bolting problems and considered them appropriate. The causes of this occurrence were due to historical problems with bolting practices and failure to incorporate or otherwise track a material substitution evaluation as part of a piping specification. NYPA considered the failure to incorporate the MSE to be an isolated occurrence. The identification of the bolting deficiencies revealed an appropriate questioning attitude by maintenance.

The initial torquing of the flanges was not successful in stopping the leakage. NYPA identified the leaking flanges as 3", 150#, flat-faced, stainless steel flanges on the supply piping to the FCUs. Inspection of the flange faces revealed minor corrosion degradation and localized pitting which affected the sealing ability of the gasket material. Following the discovery of pitting on the leaking flanges, NYPA elected to disassemble the remaining

[redacted]

exhibited no signs of leakage, and was considered outside the scope of work. NYPA inspected all flanges by visual inspection and, if questionable, also performed ultrasonic testing of the flange wall thickness.

NYPA determined that the repair of the pitted flange faces was to be performed by applying an epoxy-like material, Belzona. The system engineering department reviewed applicable ANSI and ASME Code documentation and concluded that the treatment of the flat-faced flange did not involve a pressure boundary and was not a Code repair. However, they did note that ASME Boiler and Pressure Vessel Code Section VIII invokes "good engineering practice" with respect to design changes for flat-faced flanges.

The NRC reviewed design change DC 95-3-028 FCU which addressed the Belzona repairs of the flange faces. The inspector noted that the original design change, contrary to good engineering practice, did not set a limit or scope on the extent of the Belzona repair. This issue is discussed further in section 4.1 of this report.

In the process of inspecting the FCU flanges, NYPA discovered three isolated pits or worm hole defects in the service water piping in the vicinity of the flanges. These defects, in the 31, 33, and 34 FCU piping, all required code 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-CS15	31 and 33 ABFP Check Valve Test
3PT-M20A	31 and 33 ABFP (Motor Driven) Surveillance and Inservice Test
3PT-W13	Station Battery Visual Inspection
3PT-M16	Safety Injection Pumps Functional Test
3PT-M13A	Reactor Protection Logic Train A
3PT-V01	Source Range Analog Channel Functional Test
ENG-585	31 EDG Loss of Offsite Power Test
ENG-588	WCCPPS Backup Nitrogen Supply Functional Test

3.6 Auxiliary Feedwater System Walkdown

The NRC performed a walkdown and review of the auxiliary feedwater (AFW) system to verify that system material condition and configuration was controlled in a manner to ensure operability. The AFW system was selected, in part, based on its significant contribution to accident risk reduction, as described in the Indian Point 3 Individual Plant Examination dated June 30, 1994.

The inspector verified that the AFW system was properly aligned and available to perform its intended safety function. The system was aligned in accordance with the most recently completed system checkoff list (COL)-FW-1. The inspector also verified that the AFW system as-built condition in the plant was accurately reflected on system flow diagrams 9321-F-20183 and 9321-F-20193.

The inspector observed a number of minor equipment deficiencies that had been previously identified and evaluated by the licensee. The deficiencies included some leaking valves, a few discrepant instrument air gages, and other minor problems.

The NRC also observed that a previously identified leaking drain valve, BFD-65-15, presented a challenge to housekeeping in the auxiliary feedwater system room. The water leaking from this valve formed large puddles in one area of the room. Although the valve deficiency had been previously identified by NYPA, a system engineering AFW system walkdown report dated January 16, 1996, noted that the source of the water on the floor was not known. The inspector brought the valve deficiency to the attention of the system engineer. The valve was subsequently replaced by maintenance work request WR 96-00252-00.

Additionally, the inspector discovered one AFW system equipment deficiency that was of more than minor significance. Initially, the inspector questioned an audible instrument air leak or leak-by problem at the positioner for valve BFD-FCV-406A (#31 AFW pump to #31 steam generator auxiliary feedwater regulating valve). NYPA initiated a Plant Identified Deficiency tag and investigated the problem. Subsequently, NYPA determined that an improper type positioner was installed at BFD-FCV-406A. Specifically, a positioner of model number 5320255A3 was found, and a model 5321030A10 was specified for this application. The improper model positioner had a pressure rating of 50 psig, which is lower than normal instrument air pressure and the specified positioner's rating of 100 psig.

NYPA was unable to determine how or when the improper positioner was installed. The original valve was installed with the correct positioner, therefore the positioner was apparently changed without a proper modification. No other improper positioners were found in the system.

NYPA determined that BFD-FCV-406A had operated satisfactorily in the past, and no deficiencies had been revealed during previous surveillance testing. NYPA concluded that past operability was not a concern. NYPA replaced the positioner on January 25, 1996, under work request WR 96-00202-00.

NRC observed an additional inconsistency associated with an auxiliary feedwater regulating valve. The inspector found that the pressure gage for BFD-FCV-405D was indicating a notably lower pressure than that for BFD-FCV-405A, B, and C. NYPA initiated DER 96-0337 to investigate and performed troubleshooting under work request 96-00580-00. The valve positioner outlet gage was replaced and pressure regulator IA-PCV-1554 was adjusted. Subsequently BFD-FCV-405D was satisfactorily retested. Although no significant concerns were identified by NYPA during the troubleshooting efforts, the pressure reading inconsistency had not been previously identified or recognized by system engineering or operations.

During the walkdown, the inspector also discovered an uncertified placard that provided procedural guidance. The inspector noted that an indicator panel at the 32 auxiliary boiler feed pump (ABFP) included a placard describing the procedure for resetting an overspeed trip of 32 ABFP. The inspector compared this procedure with the corresponding approved System Operating Procedure (SOP) ESP-1, "Local Operation of Safe Shutdown Equipment." The placard was missing a procedural step in the SOP, specifically to "ensure control switch for PCV-1139 is in trip." The inspector brought this discrepancy to the attention of licensee. Upon review, the licensee determined that the placard was not part of the operator aid program, and thus was not controlled or updated. The licensee subsequently removed the placard.

The inspector reviewed the AFW backup nitrogen supply system, which is required to operate selected AFW valves in the event of a loss of instrument air. Upon questioning by the inspector regarding the basis for the nitrogen bottle pressure requirements specified in operator logs, NYPA determined that the log requirements were in error. Specifically, the log listed the minimum pressure limit for the bottles to be 4800 psig combined for all three bottles. However, the basis for this limit was calculation IP3-CALC-MULT-382, which specified that the bottles add to 5448 psig. The pressure requirement was based on 1/2 hour backup supply after a postulated loss of instrument air. The 1/2 hour time period was identified in a design basis letter, IPP-2033, dated July 18, 1968. Thus, the log minimum pressure requirement error was non-conservative and inconsistent with the design basis. At the conclusion, of the inspection, the logs had been corrected and NYPA was investigating the cause of this log basis error and past data for potential operability or reportability issues.

The inspector reviewed copies of the most recent performance of the following surveillance tests: 3PT-M44, Auxiliary Feedwater System; 3PT-M20A, 31 and 33 Auxiliary Boiler Feed Pump (Motor Driven); 3PT-CS15, 31 and 33 ABFP Check Valve Test; 3PT-5Y4, 32 ABFP Turbine Overspeed Test; 3PT-Q20, Auxiliary Boiler Feed Valves; and 3PC-R60, Auxiliary Feedwater Flow Rate Check and Calibration. The inspector also observed portions of the performance of two surveillance tests, 3PT-M20A and 3PT-CS15. The inspector verified that the tests met technical specification (TS) requirements and were adequate to verify system operability. Test results were promptly reviewed by plant management and discrepancies were properly annotated in the test for resolution.

The NRC concluded that the AFW system was generally being maintained in a condition to support its intended safety function. The system was properly aligned in accordance with station procedures, and there were no major deficiencies that compromised the operability of the system. The inspector reviewed the applicable section of the Final Safety Analysis Report (FSAR) and noted no inconsistencies in the FSAR wording related to this area. However, the NRC noted that a number of system deficiencies were identified or revealed through NRC inspection. These items, which included two problems with auxiliary feedwater regulating valves, an incorrect operator log specification, an unrecognized valve leak creating a housekeeping challenge, and an uncertified procedural guidance placard, indicated weaknesses in operations and system engineering walkdown/inspection processes.

4.0 ENGINEERING (37551, 92903)

4.1 Review of Design Engineering Issues

During the inspection period the NRC reviewed engineering documents associated with several activities. Issues identified during this review are described in the following paragraphs.

- DC 95-03-028, was a type 1 design change to allow the use of Belzona Metals, an epoxy like substance, to repair the pitted flange faces of the containment fan cooler units and associated piping. NRC review identified the design change was incomplete in that it failed to define the scope of allowable repairs. The design change did not include a structural analysis to limit the size pit that could be repaired using the Belzona Metals product.

  System engineering was actively involved with the repairs to the fan cooler units and did not identify the deficiency in the design change. The system engineer evaluated the pitting using engineering judgement. Subsequently, an engineering evaluation was made and the flanges were evaluated based on that criteria. No deficiencies were identified based on this evaluation.

- DC 96-03-053, was a type 1 design change required to allow replacement of the emergency diesel generator lube oil check valves. The original part was no longer available and a replacement was required. During a walkdown the inspector observed a mounting bracket that did not support the replacement valve. This was due to the smaller diameter of the replacement valve. The inspector noted that the condition had been independently identified by a maintenance supervisor.

When notified of the bracket problem by the mechanical supervisor, the system engineer and a maintenance engineer failed to identify that the bracket required engineering evaluation. After NRC identification, NYPA design engineering performed calculations and revised the design change to eliminate the bracket. Additionally the inspector noted that a deficiency event report (DER) was not issued to document the discrepancy.

- TM 96-00488-03, was a temporary modification for installing a backup instrument air system. NYPA management requested the backup system due to problems associated with the normal instrument air compressors. NRC review of the temporary modification identified that cleanliness requirements of the temporary system were not addressed. Further, the temporary modification did not provide requirements for a cleanliness verification prior to connecting the temporary system to the normal system. The temporary system hoses were shipped on a flat bed trailer and were not sealed or cleaned by the vendor. The temporary modification was PORC approved.

  Subsequently the licensee incorporated instructions into the work procedure to blow out the lines prior to connecting the temporary system. However, a DER was not issued and the temporary modification was not revised to address system cleanliness.

The above examples indicate incomplete engineering by the design engineering group. Additionally, in two of the cases maintenance and system engineers did not identify the requirement for design engineering input and developed work arounds without formally addressing the problem or implementing the corrective action process. The inspector notes that the above events were entered into the DER system after discussions with site management.

10 CFR 50, Appendix B, section III, Design Control, requires in part that design changes shall be subject to design control measures commensurate with those applied to the original design. Contrary to the above, the design change for the Belzona Metals application to the service water flanges, DC 95-03-028, failed to address the structural design requirements of the original flange design, the design change for the emergency diesel generator lube oil check valves, DC 96-03-053, failed to address the structural requirements of the supports used by the original design, and the temporary modification of the instrument air system, TM 96-00488-03, failed to address the cleanliness requirements of the original design. This is a violation of NRC requirements. **(VIO 96-01-02)**

4.2 Service Water System Leakage Evaluation

In January 1996, NYPA system engineering performed an evaluation of the cumulative effects of multiple pinhole leaks in the service water system. NYPA drafted Deviation/Event Report 96-0109 to review the impact of six simultaneous leaks in various sections of the service water system. The purpose of the evaluation was to consider the aggregate impact of the leaks on overall system operation.

The evaluation included a review of leaks in the Zurn strainer pit identified by the NRC on January 11, 1996, and discussed in NRC inspection report 50-286/95-17. The NRC expressed concerns over the lack of thoroughness of the initial leak evaluation. Additionally, the NRC questioned the apparent lack of ownership of the operations and maintenance personnel involved in the work.

NYPA's report addressed three leaks in the Zurn strainer pit and three others: at the outlet of the turbine hall closed cooling heat exchanger, at the 33 fan cooler unit motor cooler discharge line, and on a supply line to the control room air conditioning units. Each individual leak was discussed in regards to its significance and impact on the system operation. In all cases, NYPA concluded that the leaks did not impact the function of the system. NYPA determined that the nature of the leaks would not lead to catastrophic failure of the piping and that the structural integrity of the piping was maintained. The overall impact of the leakage was generally discussed and evaluated.

NYPA completed repairs to the identified leaks and inspected similar system piping to determine the extent of condition. Additional service water system leaks and defects were discovered at the fan cooler units, as discussed in section 3.4 of this report.

The inspector noted that the system engineering evaluation report, as well as follow-up system engineering field inspection activities, revealed a heightened level of awareness and sensitivity to service water system leaks. The reviews of the individual leaks indicated that degraded service water system conditions are being more thoroughly evaluated.

4.3 Failure of Lightning Arrester of Station Auxiliary Transformer

The inspectors assessed the licensee's actions to confirm the cause of the January 20, 1996, station auxiliary transformer (SAT), A-phase lightning arrester (LA) failure. The inspectors witnessed the disassembly of the faulted arrester by the Consolidated Edison (Con-Ed) line crew and the New York Power Authority (NYPA) Power Generation Department representative. It appeared that the fault-to-ground, through the arrester, was precipitated by an accumulation of moisture internal to the arrester. The arresters were the original equipment and were of a design that utilized a gap that was designed to breakdown when the spark-over voltage was reached. The licensee postulated that a moisture cloud formed in the gap area and reduced the voltage required to spark-over. The inspectors observed that the caulking material in the LA seal area appeared to have deteriorated and allowed moisture to penetrate the internals of the LA.

As a result of the fault, the licensee replaced the LAs on all three phases with a newer surge arrester design that does not rely on a gap for over voltage protection. The inspectors reviewed the applicable portion of the FSAR electrical system design, Section 8.2, and found that existing LAs being replaced were consistent with the 120 kV duty cycle rating on the SAT.

The inspectors discussed the suitability of the new surge arresters (SAs) for the application with IP3 design engineering and NYPA's power generation engineering staff. The NYPA power generation staff confirmed the new SAs were coordinated with other arresters installed at IP3 and Buchanan substations, and provided adequate protection for the 138 kV IP3 SAT transformer. The inspectors agreed that the capability of the SAT and the new SAs ratings were adequate to protect the SAT.

The inspectors reviewed the previous off-site power failure, documented in LER 50-286/84-11, which was blamed on a dirty insulator. The inspectors confirmed there was no relation to the January 20, 1996, event.

The inspectors witnessed the following licensee corrective actions:

- The replacement SAs were tested prior to installation using a Doble Capacitance Test to assure the acceptable condition of the replacement SAs.

- The original lightning arresters were replaced with equivalent surge arresters. The replacement SA installation consisted of two sections, one rated for 52 kV and the other rated for 46 kV for an overall total of 98 kV. Surge arresters carry two ratings. The 98 kV maximum continuous operating voltage (MCOV) was consistent with the original duty cycle 120 kV (maximum discharge voltage) rating of the LA.

- The inspectors reviewed the SA manufacturer's recommended maintenance. The manufacturer's literature indicated only periodic cleaning to remove any external water soluble contamination. The guidance also indicated there was no ordinary field testing capable of revealing the essential characteristics of a surge arrester. Therefore, the inspectors concluded the existing preventive maintenance procedure is adequate.

- During the installation of the SAs, the licensee noticed some of the standoff insulators supporting the 138 kV lines to the SAT had cracks. The damaged insulator sections were also replaced with equivalent units at this time.

The station service transformer was re-energized on January 28, 1996, following restoration of the EDG 31.

Based on the above observations and review, the inspectors concluded that appropriate corrective actions were taken to address this issue.

4.4 Failure of Emergency Diesel Generator (EDG) Breaker 31

During the January 20, 1996, loss of 138 kv power event, the 31 EDG started as required but failed to supply power to the bus. NYPA determined that the EDG output breaker initially closed and then opened several seconds later. The inspectors assessed the licensee's actions to confirm the cause of the EDG breaker's failure to remain closed (Licensee's Action Plan Number IDSE-APL-96-007).

The inspectors observed the troubleshooting and testing performed on the circuit breaker. The licensee's investigation, along with the circuit breaker manufacturer's representatives, indicated that the circuit breaker was operating normally and no out-of-adjustment components were found. The licensee reviewed the control circuits, both internal and external to the circuit breaker. NYPA confirmed that the circuit breaker maintained a close command, and concluded that the reason the circuit breaker did not close was because the anti-pump relay was energized. No indication was immediately found which would explain why the circuit breaker would hold in for 12 to 14 seconds and then trip. Possibilities included overcurrent or undervoltage trips, but no flags were found at the relays. All relays were checked out satisfactorily by the licensee. The licensee found no problems either with the service water pump or motor, which was sequenced onto the diesel feed bus at 12 seconds.

The licensee performed a special test of the EDG on January 27, 1996, in various unloaded and loaded conditions, with the EDG governor and excitor manufacturer's representatives present. Procedures were developed to perform various tests, including a test that synchronized the EDG to the bus, separated the bus from off site and loaded a service water (SW) pump. The test on January 27, 1996, identified a loose wire termination in the EDG voltage interlock seal-in circuit. The missing seal-in would initiate a trip of the EDG output breaker whenever any loaded component resulted in the bus voltage dropping below the relay setpoint of 432 volts. When the voltage returned above the reset value before the circuit breaker trip spring had been recharged, the anti-pump relay locked out the breaker.

The licensee concluded that the contributing causes of the loose wire was a combination of: 1) the location of the EDG control panel in the EDG cell being subjected to higher vibration than most plant areas; and 2) the loose wire being on a compression-type terminal block, not typical of the type used at IP3. As a short-term corrective action, the licensee completed a visual inspection of all similar wire terminations and performed a tightness check of the terminal wires on EDG No. 31. Similar checks on the remaining EDGs were later completed to ensure such generic concerns did not exist. As a long-term corrective action, the licensee was considering a preventive maintenance program to check panel wire tightness. In addition, the licensee was also planning to develop a surveillance test or calibration procedure to test protective trip circuits for the output breakers of the EDGs to ensure reliable EDG breaker performance.

On February 4, 1996, the licensee performed one additional blackout test (ENG 585). This test successfully demonstrated that the EDG 31 was capable of accepting design loads under loss of off-site power conditions. The inspectors reviewed the test results and had no concern.

During troubleshooting of the control circuit wiring, the licensee also discovered that the emergency exciter (field) voltage shutdown switch slave relay contact (KIX/EG1), shown as connected on the design drawing (93-21-LL-31173, Sheet 14, Revision 12) in the EDG local panel, was not connected. The purpose of relay contact in the EDG control circuit was to short circuit the output of the static exciter and prevent excessive current flow to the generator field while operating the engine at reduced speed. The failure to connect the relay contact in the circuit rendered the field emergency exciter voltage shutdown push-button non functional.

The licensee evaluated this issue and, after consulting with the vendor concluded that the voltage shutdown option was not necessary because the EDGs were not expected to run at low speed and, therefore, rotor protection via the voltage shutdown push-button was not required. The licensee reviewed the applicable portion of the Final Safety Analysis Report (FSAR) and Technical Specification and found no impact on the existing EDG system requirements. The licensee performed a 10 CFR 50.59 safety evaluation of the as-found condition and determined that it did not constitute an unreviewed safety question. They corrected the drawing discrepancy by issuing a temporary modification for each EDG. The licensee's investigation determined that the circuit revision may have occurred during the replacement of various relays per Modification 76-3-08. The inspectors reviewed the related documents and had no further questions.

Based on the above observations and review of applicable documentation, the inspectors concluded that appropriate corrective actions were taken by the licensee to address this issue. The inspectors determined that the licensee's engineering department staff was aggressive and used a good technical and methodical approach to determine the cause of this concern.

4.5 Loss of Containment DC Emergency Lighting During Loss of Off-Site Power, DER 96-128

The inspectors assessed the licensee's actions to evaluate the cause of a loss of DC emergency lighting within the containment building during the January 20, 1996, loss of off-site power event.

The emergency lighting inside containment consists of incandescent light fixtures powered from lighting panel LP-318. The normal power to LP-318 is supplied via an AC source through an automatic transfer switch. If AC power is lost, this panel is supplied from the DC power source. On January 20, 1996, when AC power was lost, and the lighting panel was transferred via the transfer switch, no emergency lighting illuminated within the containment building.

The licensee's investigation determined that the containment lighting panel LP-318 incoming feeder from the transfer switch, feeder breaker, and fuses were under-designed for the total lighting load connected on the panel. The licensee determined that the root cause of the problem was a design error made in 1980, involving the installation of additional emergency lighting units inside the containment. This modification increased the load on the existing feeder circuit without upgrading the feeder capacity. The licensee concluded that an inadequate post modification test was performed, and that poor maintenance practices resulted in some lighting fixtures having incorrect light bulbs installed. NYPA concluded that these contributing causes resulted in the blown fuse on the feeder circuit.

The licensee issued PID 30936 on January 18, 1996, and DER 96-110 on January 19, 1996, to document this issue and resolve this condition. However, its priority was low, and as a result, no corrective actions were taken prior to the January 20, 1996, loss of off-site event. After the loss of offsite power event, immediate corrective actions were initiated to brief personnel entering containment and to require they had a flashlight.

The licensee performed a review of the design basis licensing data base and determined that there were no specific commitments related to the emergency lights installed inside the containment. The inspectors confirmed that Technical Specification 3.7.E required that, whenever the reactor was critical, the circuit breaker in the electrical feeder to emergency lighting panel LP-318 inside containment shall be locked open except when containment access is required. The feeder breaker is typically closed during an outage to provide emergency lighting inside containment for personnel safety. Based on the above findings, the licensee concluded there was no nuclear safety concern to the blown fuse or loss of emergency power.

The licensee planned to use portable lights in containment until the full extent of the problem is determined and appropriate corrective modifications are performed. The resolution of this issue was being tracked by the licensee at a higher priority level under the new DER 96-128 and ACTS 15295 issued on February 1, 1996.

Based on the review of the above licensee evaluation and design basis requirements, the inspectors concluded that the licensee had taken reasonable long term corrective actions to temporarily address this issue, however the review of DER 96-110 failed to effect appropriate immediate corrective actions in a timely manner. Since the remaining issues were being tracked adequately by the licensee, the inspectors had no other concerns.

4.6 Previously-Identified Electrical Issues

The inspectors assessed the licensee's corrective actions taken for the following performance-related issues during this inspection.

- **Spurious Central Control Room (CCR) Alarms**

The inspectors reviewed the licensee's actions to confirm the cause of the spurious CCR alarms received since December 21, 1995 (licensee's action plan No. IDSE-APL-95-004, Revision 1). The licensee had not been able to confirm the cause of the spurious alarms. They have identified some possibilities including water damage from an overflowing commode, and electromagnetic interference from solenoid-operated valves. NYPA indicated they had attempted to duplicate EMI in the associated circuits with no success. They indicated they also took samples of the cables and subjected them to a water bath to test their wet insulation resistance. The inspectors verified the licensee had modified the piping in the commode by adding local isolation valves. The licensee had also instrumented the alarm circuits in the CCR, hoping to record any transients on the circuits that may lead them to the cause of the spurious alarms. The licensee planned to be especially sensitive as the plant heat-up approaches the point where the spurious alarms started in the past. (DER 95-2926, ACTS 14866)

The inspectors concluded that the licensee was adequately monitoring the various plant components and was proceeding in a methodical manner to eliminate potential causes for spurious alarms.

- **Residual Heat Removal (RHR) Motor Current Imbalance**

The inspectors assessed the licensee's actions to evaluate the cause and effect of the 460-volt RHR pump 32 motor current imbalance, as documented in the licensee's ACTS No. 6096 IPE-RPT-ED-01512, dated 5/16/95.

The inspectors reviewed the results of the factory tests and the manufacturer's evaluation of the effect of the imbalance. The factory tests were performed at no load. The test results were all normal and balanced except for the phase currents. One phase was recorded approximately 11 amperes lower than the others.

The licensee tested the motor at 71% load on January 7, 1995, in response to Quality Assurance's original concern and submitted the results to the motor manufacturer for review. Those results showed substantially less imbalance (3.4%). The licensee retested the motor at 78% load on November 29, 1995, at which time no significant imbalance (less than 1%) was noted.

The inspectors reviewed the data from all three tests, and the manufacturer's and licensee's evaluations and concluded that no concern existed with the operation of the RHR pump motor operating with such small imbalance. This was consistent with the recommendations found in the industry standard for motors, National Electrical Manufacturer's Association, NEMA-MG-1, "Motors and Generators." The inspectors also noted that the 460 volt-rated motors were being correctly operated on the 480-volt system also in accordance with industry practice and standards.

- **Battery 31 Low Temperature**

During the course of this inspection, the licensee declared Battery 31 inoperable because it was below its design temperature during a surveillance test performed the previous week (Procedure 3PT-Q1, "Station Battery Surveillance and Charging"). The test data on January 14, 1996, indicated an electrolyte temperature of 56°F. The minimum design temperature is 60°F, as indicated in Calculation IP3-CALC-EL-0184. The licensee's review discovered the electrolyte temperature was recorded in the front of the procedure, while the acceptance criteria was given in the back of the procedure. The licensee determined that this was probably the reason why a limit that had been exceeded was not recognized by the licensee until their review of the surveillance results on January 24, 1996. The battery was only then declared inoperable on January 25, 1996 (DER 96-0179). Details of the failure to identify the out of specification data is discussed in NRC inspection report 50-286\96-02.

The inspector's review of the battery sizing calculation revealed that the derated capacity of the battery, at an average electrolyte cell temperature of 55°F, would have been greater than the required station load requirements during this period. The design of Battery 31 had excess capacity in addition to the standard industry temperature and aging capacity margins.

The inspectors concluded that there was no safety problem at these lower temperature conditions during the above short period.

- **RHR Valve, SI-MOV-899B, Tripped on Thermal Overload, DER 96-115**

The inspectors assessed the licensee's actions to evaluate the cause of the RHR motor-operated valve (SI-MOV-899B) failure and loss of dual indication during the restoration of 31 RHR heat exchanger.

The licensee's initial investigation determined the valve tripped on thermal overload in the full open position, as documented in DER 96-0115, on January 19, 1996. Further investigation found that the root cause of this valve failure was a blown A-phase power fuse. This fuse also supplied control power to the valve control circuits and dual indication lights.

During troubleshooting, the licensee indicated that the condition of the blown A-phase fuse was poor and they had difficulty reading the identification on fuse. The other two fuses were identified as being correct and all three fuses were replaced in accordance with the master fuse list. However, when engineering requested the blown fuse for further evaluation, the fuse had been disposed of. Since the blown fuse was no longer available, the licensee could not definitely determine the cause of its failure. At the conclusion of this inspection, the licensee was in the process of revising their fuse control procedure, DEE-SD-01, to require that all blown fuses be retained, tagged, and turned over to design engineering for possible fuse failure analysis in the future.

To verify the operability of the valve, the licensee reviewed past corrective maintenance actions, performed additional testing on the valve, and checked the electrical coordination of the circuit. There were no other electrical problems with the motor, control circuit or electrical protection components. Upon replacement of the fuse, the RHR motor-operated valve functioned satisfactorily. Based on this, the licensee concluded that the cause of this valve failure and loss of dual indication was a blown fuse on A-phase, and concluded the valve was capable of performing its design functions.

NRC review of the troubleshooting documentation (WR No. 96-251-00) indicated that the licensee checked all appropriate components of the valve control circuits, including the thermal overloads and indicating lights. The results of the NRC review of the valve control drawings was consistent with the licensee's conclusion. The inspectors had no other concerns.

- **OVERALL CONCLUSIONS**

The inspectors concluded that the actions to address the previously-identified performance-related electrical issues were acceptable. The licensee's corrective actions were appropriate, and the root cause determination evaluations were comprehensive with adequate basis. In the case of the CCR spurious alarms event, which was still under investigation, the licensee staff was adequately monitoring various plant components to further evaluate and address this concern.

The inspectors noted that the licensee management and technical staff members put forth significant effort to resolve these issues. Management was actively involved in overseeing and resolving these technical issues.

4.7 Low Electrical Tunnel Temperature

NYPA identified a concern with low temperatures in the electrical tunnel. It was noted that when tunnel temperature dropped below 50 °F, that the reactor vessel level indication system (RVLIS) core exit thermocouples (CETs) would be marked as bad. This would result in the affected trains of RVLIS being declared inoperable. This was due to the RVLIS CET reference junction boxes, located in the electrical tunnel, only being calibrated down to 50 °F. When the junction box temperature dropped below 50 °F, the CETs are marked as bad since they are no longer reliable.

The cause of the low temperature in the electrical tunnel was due to two louvers which were not fully shutting. NYPA repaired both and tunnel temperature has since stayed above 50 °F. NYPA engineering reviewed all equipment in the tunnel and evaluated the effects of low temperature on them. In addition to the RVLIS CET junction box, NYPA identified low temperature concerns with the Hydrogen Recombiner power supply, Appendix R Safe Shutdown Panels, and Appendix R lights. NYPA identified additional long term corrective actions for all of this equipment.

NYPA review and evaluation of the effects of low temperature in the electrical tunnel were good. The review of all equipment in the tunnel for low temperature effects was a good initiative by Design Engineering. Corrective actions taken to repair the ventilation louvers were adequate to prevent the tunnel temperature from lowering below 50 °F. Long term corrective actions to address equipment reliability at lower temperatures and to improve tunnel temperature monitoring were appropriate.

4.8 Cable Separation Issues with Temporary Tank Heaters

During a tour of the PAB, the inspector noted that temporary heaters were installed around the containment spray system sodium hydroxide (NaOH) spray additive tank. The inspector observed that the power leads to the four heaters appeared to be routed alongside safety-related cable trays in the overhead in such a manner as to violate cable separation criteria. A review of Temporary Modification (TM) 95-2993-05, that installed the heaters, indicated that the cables were to be placed along the floor, not in the overhead.

The mechanics who installed the TM ran the cables in the overhead along the cable trays contrary to the instructions in the TM. They did this for safety concerns, because placing the cables on the floor presented a potential tripping hazard. This was brought to the attention of the system engineer while he was present at the worksite for a different concern, and he gave approval for this; however, the system engineer admitted that when he gave this approval, he had not considered cable separation criteria. Also, cable separation criteria had not been considered in the preparation of the TM package. Lastly, the TM package was not changed to reflect the change in the cable routing.

NYPA's initial review of the concern indicated that cable separation criteria, as specified in the FSAR, may have been violated. Only after further engineering review did NYPA conclude that the actual installation was acceptable because double fault protection existed for the heater circuits, which met the FSAR criteria. This methodology was reviewed by the NRC and deemed acceptable.

NYPA initiated DER 96-0263 to capture the cable separation concern. NYPA also performed walkdowns of all electrical TMs for similar concerns and did not identify any other possible violations of the separation criteria. Results of the walkdowns were summarized in NYPA Memorandum IP-DEE-96-39, dated February 8, 1996. The inspectors reviewed this memorandum and concluded that NYPA had performed a through walkdown and adequately resolved the inspector's concerns.

In conclusion, workers in the field deviated from the PORC approved method for routing the cables and the system engineer permitted this without changing the TM. Awareness of cable separation concerns was lacking in this TM as well as other electrical TMs; NYPA's corrective actions included the walkdown of existing electrical TMs and a planned revision to the TM procedure, AP-13, to ensure that cable separation criteria are considered in the development of TMs in the future.

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. The security response to the loss of 138 kv power event was timely and appropriate.

5.3 Housekeeping

The NRC assessed the control of plant housekeeping in safety-related areas. General plant housekeeping in easily accessible safety-related areas and containment during the period was adequate. However, NRC inspection of areas not routinely accessed indicated declining material condition, unidentified deficiencies and poor implementation of housekeeping standards. Examples include the boron injection tank room, the waste holdup pit area and the spent fuel pool cooling pump pit. Efforts to improve the housekeeping standards in these areas were being implemented at the close of the inspection period. Issues relating to housekeeping standards and the functions of the nuclear plant operators during rounds are discussed in NRC inspection report 50-286/96-02.

5.4 High Oxygen Concentration in Waste Gas System

Since plant shutdown on September 14, 1995, there have been several instances of high oxygen concentrations and explosive gas mixtures in the waste gas system. These occurrences were last documented in NRC inspection report 50-286/95-15. Since that time, there have been several more unexpected occurrences of high oxygen concentrations detected in the waste gas system. Although these subsequent occurrences did not result in explosive gas mixtures, the NRC was concerned that the source of oxygen intrusion be resolved prior to plant operation.

The IP3 waste gas system provides a means of controlling and processing noncondensible gases from various tanks throughout the primary plant. Some of the tanks connected to the waste gas system are, for example, the pressurizer relief tank (PRT), reactor coolant drain tank (RCDT) and the chemical and volume control system (CVCS) holdup tanks (HUTs). These tanks are connected to the large gas decay tanks (LGDTs) through a vent header. Waste gas compressors are then used to transfer gas from tanks such as the PRT to the LGDTs. The waste gas system is normally a hydrogen rich system during plant operation due to the hydrogen chemistry maintained in the reactor coolant system (RCS). Therefore the oxygen concentration must be maintained low to prevent the possibility of an explosion in the system.

On November 28, 1995, a high oxygen concentration was detected in the PRT after extended operation of the eductor during RCS fill and vent operations. Chemistry department investigation concluded that this extensive eductor operation drew a small vacuum in the PRT and allowed oxygen to enter the PRT through a small system leak. On January 12, 1996, during another RCS fill and vent evolution, a high oxygen concentration was again detected in the PRT and also the CVCS HUTs. While oxygen was expected in the PRT due to RCS fill and vent, it was determined by NYPA that the oxygen had unexpectedly migrated to the CVCS HUTs through Vent Header Containment Isolation Valves AOV-1786/1787. These valves were found in the open position as required by checkoff list (COL)-WDS-2, Gaseous Waste Disposal System. However, FSAR table 5.2-3 requires that these valves be shut while the plant is shutdown. The operation of valves AOV 1786/1787 is discussed in more detail in section 2.3 of this report. High oxygen concentration was again detected in the CVCS HUTs on February 2, 1996. Subsequent sampling determined that a high oxygen concentration was not present and the indication was due to water condensation in the waste gas analyzer sample lines.

Chemistry personnel reviewed all of these instances of oxygen intrusion into the waste gas system and evaluated necessary corrective actions per DER 96-0063. A small leak was found on the PRT level transmitter which chemistry determined was the likely source of oxygen above that normally expected during RCS fill and vent operations in November 1995 and January 1996. The leaking joint was repaired and the PRT was leak tested by ENG-581, which demonstrated no significant leakage on the PRT. The accumulation of water in the waste gas analyzer has been a problem which has caused several false indications of oxygen in the waste gas system. Sampling procedures were revised to blowdown the sample lines with nitrogen to correct the water accumulation. NYPA engineering is further evaluating a modification to relocate a water trap in the sample header.

NYPA evaluation of the operation of the waste gas system identified several problems which could contribute to oxygen intrusion. The waste gas system was manually operated in accordance with SOP-WDS-2, Gaseous Waste Disposal System Operation. Chemistry concluded that this procedure did not provide adequate guidance for the operation of waste gas compressor (WGC) pressure control valves and for controlling the moisture separator tank levels. Chemistry also determined that some vent header pressure gauges were out of tolerance and that the moisture separators level indicators were not properly calibrated. The combination of inadequate procedure guidance and instrumentation not reading correctly resulted in operating the vent header with pressure near vacuum which could allow the intrusion of oxygen through small system leaks. NYPA revised SOP-WDS-2 to include appropriate operating instructions and completed the component repairs necessary to prevent future oxygen intrusion.

The NRC reviewed the operation of the waste gas system as described in the FSAR. FSAR section 11.1.2.1 described a system where the WGCs operated automatically based on vent header pressure. Manual operation of the system per SOP-WDS-2 was contrary to this FSAR description. This was a condition which had been previously identified by NYPA operations personnel and documented in DER 96-133. A nuclear safety evaluation (NSE) was approved on

February 21, 1996, which concluded that manual operation of one WGC was acceptable and did not present an unreviewed safety question.

Although the NSE adequately addressed the acceptability of manual operation of the waste gas system, the NRC was concerned as to why SOP-WDS-2 contained operating instructions which had not been previously evaluated as required by 10 CFR 50.59. The waste gas system had originally been designed for automatic operation. Since the retirement of the waste evaporator system in the mid-1980s, reduced system loading required the manual operation of the waste gas system. The NRC reviewed previous revisions of SOP-WDS-2 and noted that the operating instructions for the system were also changed from automatic to manual in the mid-1980s. As of the end of this inspection report, NYPA was still evaluating why the change in the operation of the waste gas system as described in procedure SOP-WDS-2 had not previously been evaluated as required by 10 CFR 50.59. This issue and other examples of plant procedures which may have not been properly evaluated in accordance with 10 CFR 50.59 is discussed further in section 7.2 of this report.

The continuing trend of oxygen intrusion into the PRT and CVCS HUTs was well evaluated by NYPA chemistry and operations personnel. Thorough reviews of system operating parameters and procedure instructions identified several necessary equipment repairs and procedure enhancements to prevent future oxygen intrusion.

5.5 Operating Experience Review

On February 8, 1996, the operations review group (ORG) received information that another utility operated with their ultimate heat sink temperature below that specified in their design basis. The ORG obtained and distributed this information in accordance with administrative procedure (AP)-37.3, Feedback of Operating Experience to Plant Staff. In accordance with AP-37.3, an action and commitment tracking system (ACTS) item was initiated on February 9 for NYPA reactor engineering to evaluate the IP3 licensing basis for a similar limit. Review by NYPA reactor engineering concluded that a temperature limit of 35 °F was assumed in the plant accident analysis basis document. A records review determined that IP3 had previously operated with service water temperature water below 35 °F. DER 96-0384 was initiated by NYPA on February 11 to further evaluate this issue.

A nuclear safety evaluation (NSE) was prepared by NYPA reactor engineering which concluded that IP3 could be safely operated with a service water temperature down to 28 °F. The NSE for plant operation above the cold shutdown condition was prepared and reviewed by the plant operating review committee (PORC) on February 13, 1996. A subsequent revision to the NSE which covered power operation was reviewed by PORC on March 5, 1996. The NRC reviewed these NSEs and concluded that they appropriately evaluated plant operation with low service water temperature.

The operating experience program was effectively implemented by NYPA staff in response to the low service water temperature issue identified at another utility. The industry information was obtained by ORG and promptly disseminated to NYPA reactor engineering for evaluation. NYPA reactor engineering reviewed the licensing basis and evaluated continued operation of the plant in a timely manner.

5.6 Review of Emergency Plan (the Plan) and Implementing Procedures (IPs)

A Regional in-office review of revisions to the Plan, IPs, and associated forms was completed. A list of the reviewed documents is included in Attachment 1 of this inspection report. The inspector concluded that changes made were acceptable and did not decrease the effectiveness of the Plan.

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (40500, 92901)

6.1 Licensing Event Report (LER) Review

The following LERs were reviewed and found satisfactory and are considered closed:

LER 96-02, Automatic Actuation of Emergency Diesel Generators Following a Loss of Offsite Power Due to a Failed Surge Arrester on the A Phase of the Feed to the Station Auxiliary Transformer, dated February 19, 1996.

LER 96-03, Automatic Actuation of Undervoltage Logic Designed to Strip Safety Buses and Initiate a Signal for Emergency Diesel Generator Start, dated February 20, 1996.

LER 95-003-01, Potential Single Failure of the Carbon Dioxide Fire Protection System Could Result in the Loss of Ventilation to the 480 VAC Switchgear Room, dated May 5, 1995. LER 95-003-00 was previously reviewed during NRC inspection 95-03 and the corrective actions planned were considered acceptable, but not yet implemented. Since that time, modification MMP 95-03-056 CBHV was installed and satisfactorily tested to correct this potential single failure vulnerability. The inspector confirmed the administrative completion of this modification and inspected the condition of the motor operated louvers to the switchgear room (L-319) and the 33' elevation of the control building (L-320). Both louvers were recently included in NYPA's procedure for the inspection of Category I and Category M plant ventilation fans (FAN-002-VSS, Rev. 4) and are exercised and cleaned annually.

The inspector noted that a previous single failure design vulnerability with the 480 VAC Switchgear Room ventilation system was included in a Notice of Violation issued on April 26, 1994, which also detailed a multiple number of NRC violations identified in 1993. Enforcement discretion was exercised for that ventilation system design vulnerability (as well as the other violations listed in the NOV) in accordance with the NRC enforcement policy. The design upgrade implemented to correct that single failure vulnerability did not include the carbon dioxide fire protection system interface because it was outside the system boundary. Nevertheless, the Electrical Engineering Group later identified in mid-1994 that the carbon dioxide system interface relay was also a potential single failure vulnerability. NYPA determined that the cause of the delay in addressing and reporting this design vulnerability until early 1995 was due to an error in decision-making and miscommunication between two engineering groups, each of which assumed the other group had resolved the issue. The implementation of the Action Commitment Tracking System (ACTS) and the restructuring of the engineering organization should preclude a recurrence of the failure to promptly address and report future such identified design problems. Based on NYPA's review of this matter, this event was considered an isolated occurrence. The NRC independently reviewed the evaluation and agreed with NYPAs conclusion. Therefore, no further action in this matter is warranted.

6.2 (Closed) URI 92-28-11, Adequacy of Flow Measurement for Radioactive Machine Shop Building Vent Calibration Check

The inspector reviewed NYPA's actions in response to NRC questions regarding the adequacy of test procedure 3PC-R53, "Radioactive Machine Shop (RAMS) Building Vent Calibration Check." This test procedure was performed to verify that the RAMS building ventilation instrumentation is functioning properly in accordance with Indian Point 3 Technical Specifications, Appendix B. NYPA's resolution of the questions resulted in substantial improvements to the test procedure and changes to System Operating Procedure (SOP)-V-11, "Operation of Radioactive Machine Shop Ventilation System." The inspector found that NYPA's initial resolution of the issue was weak. However, based on NYPA's additional investigation and actions, the inspector concluded that the test procedure was adequate, and that NYPA had addressed the issues related to the test. This item is considered closed.

In inspection report 50-286/92-28, the inspector questioned the adequacy of test procedure 3PC-R53, "Radioactive Machine Shop (RAMS) Building Vent Calibration Check." Specifically, the inspector noted that flow readings fluctuated significantly for measurements obtained using both the installed and test instrumentation. This item was considered unresolved pending licensee evaluation of the test methodology.

The inspector conducted further review of this issue as discussed in inspection report 50-286/94-31. The inspector noted that 3PC-R53 was revised to reduce the maximum allowable difference between the installed and test instrumentation from 50% to 10%. The surveillance was performed using the revised procedure and a more accurate digital flow meter as the test instrument. The licensee noted that flow oscillations were significantly lower than had been previously observed. Although improvement was noted in the substantially revised 3PC-R53 procedure, the inspector considered that four issues remained unresolved. Specifically: (1) the licensee provided no technical basis for the revised acceptance specification for the test; (2) the issue of flow stability as it related to accuracy of the flow meter was not resolved; (3) the flow measurement averaging process could lead to masking of a defective flow probe; and (4) the test procedure contained no quantitative acceptance criteria for the zero flow check of installed detector flow probes.

The inspector reviewed NYPA's actions taken in response to the four issues discussed in inspection report 50-286/94-31. These actions revealed weaknesses in the original resolution, in that both further changes to the test procedure and a significant revision to SOP-V-11 were required. The actions are summarized as follows:

1. In order to address question on the technical basis for the revised acceptance criteria, NYPA consulted with the flow instrument vendor and reviewed related industry publications. NYPA determined that a 10% maximum allowable difference between the installed and test instrumentation to be consistent with vendor recommendations and the reviewed publications.

2. In response to NRC questions about the role of various RAMS building ventilation fan combinations in the magnitude of the flow oscillations observed during the test, NYPA conducted further testing and review of the system operation. NYPA's investigation revealed deficiencies in RAMS building ventilation flow indication whenever the doors between the Primary Auxiliaries Building (PAB) and the RAMS building were open. The review found that interaction between the two building's ventilation systems led to inaccurate flow readings. NYPA issued Deviation/Event Report 95-364 to document and resolve the deficiency. The resolution included a substantial change to SOP-V-11, "Operation of Radioactive Machine Shop Ventilation System," which specified that the doors between the RAMS building and the PAB remain shut to prevent interaction between the ventilation systems. NYPA also reviewed the potential impact of the inaccurate ventilation flow measurements on effluent release data. NYPA determined that the past deficiencies associated with the flow readings did not result in inaccurate radiation release information. In addition, further revisions were made to test procedure 3PC-R53 to provide tighter controls over the test conditions.

3. NYPA determined that the flow measurement averaging process would not lead to masking of a defective flow probe. Licensee calculations indicated that a postulated defective probe would result in a flow deviation that would exceed the 10% acceptance criteria. The test failure would then result in further investigation and troubleshooting.

4. In regards to the question on the acceptance criteria for the zero flow check of installed detector flow probes, NYPA consulted with the flow instrumentation vendor and determined that zero flow checks were not required. Further, the revised test procedure and system operating procedure resulted in substantially reduced indicated flow at the zero flow condition.

The inspector reviewed NYPA's RAMS building ventilation flow investigation information, supporting data, and the revised test and operating procedures. The inspector concluded that the test procedure was adequate, and that NYPA's actions had resolved the issues related to the test. Previous RAMS building ventilation flow inaccuracies did not result in erroneous radiation release information. Additionally, the inspector reviewed the applicable section of the Final Safety Analysis Report (FSAR) and noted no inconsistencies in the FSAR wording related to this area. This item is considered closed.

7.0 ADMINISTRATIVE

7.1 INPO Inspection Review

NRC reviewed the INPO evaluation report dated January 25, 1996. The report covered the weeks of August 21 and 28, 1995. The findings were consistent with NRC findings during that period. No issues requiring additional NRC follow-up were identified.

7.2 Review of FSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the Final Safety Analysis Report (FSAR) description highlighted the need for a special focused review that compares plant practices, procedures and/or parameters to the FSAR description. While performing the inspections discussed in this report, the inspectors reviewed the applicable portions of the FSAR that related to the areas inspected. The following inconsistencies were noted between the wording of the FSAR and the plant practices, procedures and/or parameters observed by the inspectors.

While reviewing Section 6.3.2 of the FSAR for design information on the sodium hydroxide (NaOH) tank associated with the containment spray system, the inspector noted a discrepancy between the FSAR and the plant's emergency operating procedures (EOPs) regarding the termination of NaOH addition. This discrepancy is described in detail in section 2.4 of this report.

Section 2.2 of this report described how procedure SOP-EL-5 provided instructions for cross-connecting 480 v buses while transferring offsite power sources. Performing this procedure would have resulted in both RHR pumps being powered from the same, cross-connected buses. NYPA determined that further evaluation was required prior to performing this procedure. NRC review noted that section 1.3 of the FSAR described the RHR system as having redundant components, and further stated that equipment was arranged electrically so that multiple items received their power from different sources.

Section 2.3 of this report described how the NYPA investigation of oxygen intrusion into the waste gas system identified that valves AOV-1786/1787 were in the open position, contrary to the FSAR. TPC 95-1029 was written to COL-WDS-2 to establish the required position of valves AOV-1786/1787 as open. FSAR table 5.2-3 stated that these valves were normally open, but closed when the plant is shutdown. Operations used COL-WDS-2 to align valves AOV-1786/1787 to the open position during the current forced outage, contrary to the FSAR. With these valves open, oxygen was allowed to migrate to the CVCS HUTs during several oxygen intrusion events. NYPA review of TPC 95-1029 determined that this change to the COL was not properly evaluated as required by 10 CFR 50.59. Similar inadequate procedure change reviews contributed to plant operation at reduced pressure as documented in NRC inspection report 50-286/95-12 and the inadvertent lift of a CCW relief valve as documented in NRC inspection report 50-286/95-17.

Section 5.4 of this report reviewed the NYPA evaluation in response to several instances of oxygen intrusion into the waste gas system. NRC review of FSAR section 11.1.2.1 noted that the waste gas system was described as an automatic system. The waste gas system currently is operated manually per SOP-WDS-2. NYPA had previously identified this discrepancy and a NSE was approved on February 21, 1996, which determined that manual operation was acceptable. Operation of the waste gas system went from automatic to manual operation after the retirement of the waste evaporator in the mid-1980s and procedure SOP-WDS-2 was changed accordingly. The inspector noted that a 50.59 evaluation was not completed at that time.

NYPA identified these last three issues and is taking corrective actions to address each of them. Some of these examples are indicative of previous weaknesses in the procedure revision and modification review processes which may have resulted in inadequate 10 CFR 50.59 evaluations. These three examples and other recent occurrences such as the lifting of the CCW relief valve indicate that plant procedures may not have been consistently and adequately evaluated against the FSAR as required by 10 CFR 50.59. This issue is left unresolved pending further NYPA evaluation and NRC review to ensure that plant procedures are reflective of the licensing basis of the plant. **(URI 96-01-03)**

7.3 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 March 8, 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

Emergency Plan and EPIPs Reviewed

Document	Procedure Title / Section	Revision
Plan	Section 1.0	24
	Section 2.0	24
	Section 3.0	24
	Section 4.0	25,26
	Section 5.0	26
	Section 6.0	25,26
	Section 7.0	26
	Section 8.0	26,27
IP-1003	Obtaining Meteorological Data	15,16
IP-1004	Midas Computer System - Dose Assessment Model	13,14
IP-1011	Offsite Monitoring/Site Perimeter Surveys	18,19
IP-1019	Emergency Use of Potassium Iodine (KI)	7,8
IP-1021	Radiological Medical Emergency	23
IP-1027	Emergency Personnel Exposure	11
IP-1038	Offsite Emergency Notification	19
IP-1040	Habitability of the Emergency Response Facilities and Assembly Areas	15
IP-1050	Accountability	22
IP-1052	Hazardous Waste Emergency	6
IP-1054	Search and Rescue Teams	9
IP-1063	Vehicle/Equipment Radiological Check and Decontamination	10
IP-1076	Roster Notification Methods	19,20
IP-2001	Emergency Director (ED), Plant Operations Manager (POM), Shift Manager (SM) Procedure	5
IP-2100	Emergency Activation of the Technical Support Center (TSC)	1
IP-2101	Technical Support Center Manager	1
IP-2105	TSC Accountability Officer	Deleted
IP-2106	TSC Clerks	1,2
IP-2200	Emergency Activation of the Operations Support Center	1
IP-2201	Operations Support Center Manager	1
IP-2203	OSC Dispatch	1
IP-2204	OSC Team Leaders	1
IP-2205	OSC HP Team Leaders	2
IP-2206	OSC Accountability Officer	Deleted
IP-2207	OSC Clerk	1
IP-2301	Emergency Director Procedure	2
IP-2302	EOF Technical Advisor	1
IP-2311	EOF Offsite Radiological Communicator	2
IP-2500	Security Emergency Activation Responsibilities	4