U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No.: 95-02

Docket No.: 50-333

License No.: DPR-59

Licensee: New York Power Authority

Facility: James A. FitzPatrick Nuclear Power Plant

Location: Scriba, New York

Dates: January 1, 1995 through February 11, 1995

Inspectors: W. Cook, Senior Resident Inspector
R. Fernandes, Resident Inspector

Approved by: C. J. Cowgill III — Mar 10, 1995
Curtis J. Cowgill III, Acting Chief — Date
Reactor Projects Section 1B, DRP

INSPECTION SUMMARY: Routine NRC resident inspection of plant operations, maintenance, engineering, plant support, and quality assurance/safety verification.

RESULTS: See Executive Summary

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NOTE: The NRC inspection manual procedure or temporary instruction that was used as inspection guidance is listed for each applicable report section.

EXECUTIVE SUMMARY

James A. FitzPatrick Nuclear Power Plant

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Plant Operations: The inspector concluded that the operations staff took timely and conservative actions to restore the fuel pool cooling assist mode of RHR following the discovery of a leaking safety valve. A questioning attitude by the operations staff led to the identification of a procedure weakness when preparing to do maintenance on the reactor building radiation monitors. Inspector review of ST-2AJ determined that the operations staff initiated procedural changes and component labeling to clearly identify each component being tested during the surveillance test to be a good procedure enhancement. Evaluation and corrective actions associated with some licensed operator radiation protection practices were not yet completed by the utility and this area remains unresolved (URI 95-02-04).

Maintenance: NRC staff review of an engineering evaluation of a crack discovered in the body of a main steam isolation valve identified that it lacked sufficient technical detail and quantitative analysis to support leaving it in the "as found" condition. At the end of the inspection period, the NRC staff was provided with a copy of a structural analysis report prepared by a contractor that addressed these shortcomings. The NRC resident staff reviewed the analysis and had no further questions.

The inspector determined that the corrective actions taken following NYPA's discovery of several cracked HCU isolation valve disks were satisfactory. Work done on the core spray system was in accordance with station procedures and documentation. Corrective actions and revised control rod drive leak rate data were reviewed by the inspector and found to be appropriate.

Engineering: The inspectors noted that several engineering aspects of the containment spray surveillance test issue were not resolved at the close of the inspection period. An unresolved item (URI 95-02-01) remains concerning NYPA's root cause analysis for an apparent drawing and design basis documentation discrepancy. The licensee, as well as other utilities (see NRC Information Notice 94-84), have experienced problems with the reactor core isolation cooling system turbine lubricating oil system. The inspectors concluded that NYPA was taking positive corrective action in addressing this long-term issue. The inspector witnessed the performance of special test procedure, STP-76AV, Relay Room Enclosure Integrity Test, as an update to a previous unresolved issue.

Plant Support: The inspectors reviewed the information given by the licensee on the improper use of dosimetry by a contractor visitor and the issue will remain unresolved, pending further review by the NRC staff (URI 95-02-02). Additionally, the licensee informed the resident staff of an issue involving inappropriate signatures on combustion control permits. The inspectors were informed by station management that a broad investigation/critique and a technical/ administrative review of all active and expired combustible control permits was underway. Pending completion of NYPA's review of this issue and detailed follow-up by the NRC staff, this issue remains unresolved (URI 95-02-03). The inspectors found NYPA's response to several identified industrial safety concerns to have been appropriate. Inspector review of the corrective actions following a previous event, in which the fire protection system for the standby gas treatment system was clogged, could have been more aggressive.

DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

1.1 NYPA Activities

During this inspection period, the unit remained de-fueled while conducting vessel shroud inspections and repairs as part of the 1994-1995 refuel outage. Outage activities conducted in parallel with these efforts included: hydraulic control unit refurbishment; torus desludging; emergency diesel generator preventive maintenance; high pressure coolant injection turbine overhaul; reactor core isolation cooling system modifications; core spray piping replacements; motor-operated valve testing and modifications; and electrical bus and breaker preventive maintenance.

Effective January 12, 1995, Mr. A. McKeen replaced Mr. J. Sipp as the interim Radiological and Environmental Services (RES) manager following Mr. Sipp's resignation until a permanent RES manager is selected.

1.2 NRC Activities

A region based specialist inspector conducted a review in the area of inservice inspection the weeks of January 2 and 9, 1995.

During the weeks of January 2 and 9, 1995, the NRC's Non-Destructive Examination (NDE) team was on site with their van to conduct independent examinations of safety related piping repairs and modifications.

The project manager from NRR visited the facility during the week of January 2, 1995.

A region based specialist inspector conducted a review in the area of radiation protection during the week of January 23, 1995.

Staff from the Office of Nuclear Regulatory Research were on site observing core shroud repair activities on January 23, 1995.

The inspection activities during this report period included inspection during normal, backshift and weekend hours by the resident staff. There were 58 hours of backshift (evening shift) and 17 hours of deep backshift (weekend, holiday and midnight shift) inspections during this period.

2.0 PLANT OPERATIONS (71707,93702,92901,62703)

2.1 Followup of Events Occurring During the Inspection Period

2.1.1 Spent Fuel Pool Cooling Event

On January 9, a leaking safety valve (10-SV-35B) delayed the operations staff from placing the B side of the residual heat removal (RHR) system in the fuel pool cooling assist mode. The heat-up rate of the spent fuel pool (SFP), at that time, was greater than the capacity of the fuel pool cooling systems and thus required assistance from the RHR system. After several attempts to fill and vent the system piping and re-seat the leaking safety valve, operators elected to gag the safety valve to stop the leakage.

Initial evaluation by the operations staff was that the momentary rise in system pressure when the RHR pump was started increased pressure high enough to lift the safety valve (lift setpoint was 300 psig). The pressure rise was the result of an abnormal RHR system lineup when in the fuel pool assist mode. Following gagging of the safety valve, the system was again filled and vented and the B fuel pool cooling assist mode was properly initiated. The licensee concluded that the preliminary evaluation was correct.

In case problems were encountered with gagging the safety valve, NYPA's contingency plan was to use the A side of the RHR system in the fuel pool cooling assist mode. However, a timely restoration of the A side would have been hindered by the fact that the A side was tagged out of service in preparation for outage work. Therefore, NYPA determined that actions to begin restoring the A side at 100°F in the SFP would give the plant staff sufficient time to supplement cooling prior to exceeding 135°F. The existing SFP heat-up rate of approximately 1-2°F per hour was not high enough to challenge the maximum allowed temperature by the time the A side could be in operation.

The inspector observed control room operations, reviewed the temporary modification control form, and discussed the safety valve gagging evolution with the operations staff. The inspector concluded that the operations staff took timely and conservative actions to restore the fuel pool cooling assist mode and to minimize the spread of potentially contaminated water from the leaking safety valve.

2.1.2 Reactor Building Ventilation Radiation Monitor

While preparing a protective tagout, the operations staff determined that maintenance performed in accordance with IMP-17.12 on the B reactor building ventilation radiation monitor was adversely impacting the operability of the in-service radiation monitor. The ventilation system radiation monitors provide a safety function of isolating the reactor building ventilation and starting the standby gas treatment system when radiation levels measured in the ventilation system reach a pre-determined setpoint.

The inspector discussed this event with the instrumentation and controls (I&C) staff and learned that the radiation monitor's safety function was only partially impacted. The radiation monitors initiate an isolation signal when a high radiation level is detected or when both the A and B monitors fail downscale. The maintenance procedure (IMP-17.12) directed disconnecting a lead in the monitor circuitry to prevent inadvertent isolation signals to the reactor building ventilation and SBGT systems while performing this maintenance. The operations staff determined that lifting this lead would prevent the in service radiation monitor from sending an isolation signal, should it fail downscale. The licensee determined that the ability of the monitor to trip on a high radiation condition was still available. The inspector concluded that, despite defeating the downscale isolation function, the operable monitor still would have performed its intended safety function. A questioning attitude by the operations staff led to the identification of this procedural weakness. The licensee will correct the procedure and develop an alternate method for performing the maintenance.

2.1.3 Containment Spray Header and Nozzle Air Test

During the performance of surveillance test (ST)-2AJ, RHR Loop A Containment Spray Header and Nozzle Air Test, NYPA discovered that one of the spray nozzles had no air flow. ST-2AJ utilizes air instead of water as the test medium. The containment spray system is part of the RHR system and is used to assist drywell pressure reduction following a loss of coolant accident (LOCA). There are two spray headers in the drywell and one header in the suppression chamber that supply water to numerous nozzle headers. Each nozzle header has either 11 or 13 nozzles that form a spray pattern of water designed to quench the steam and thus reduce pressure in the containment post-LOCA. It was one of these nozzles that was discovered during testing that did not have air flow.

The inspectors reviewed the licensee's action plan for resolution of this issue and to determine whether the Technical Specification requirements were being met. The inspectors determined that subsequent visual inspection of the nozzle identified that it had been blocked by an internally installed plug. A review of this condition by the engineering staff determined, based on previous engineering analysis, that this condition was acceptable and the system was operable.

The inspector learned that the operations staff initiated procedural changes to clearly identify each nozzle header and took action to label each nozzle header in the drywell. The inspectors concluded that these actions were appropriate. However, the inspectors noted that several engineering aspects of the issue were not resolved at the close of the inspection period. Specifically, the origin of the plugged nozzle had not been determined nor the reason why the drawing and design basis documentation did not reflect the as found condition. This is an unresolved item pending completion of NYPA engineerings root cause evaluation and corrective actions, and NRC inspector review. (URI 95-02-01)

2.1.4 Inadvertent Primary Containment Isolation System Actuation

On January 14, at 11:38 p.m., an inadvertent primary containment isolation system (PCIS) actuation occurred while conducting a B side reactor water level transmitter calibration. The PCIS group 2 isolation (reactor water cleanup system, reactor building ventilation, primary containment and reactor water sample valves, containment floor and equipment drains, and standby gas treatment system automatic start) occurred due a concurrent A side reactor protection system bus de-energization. The inspectors verified that the licensee had properly restored the affected systems following this event and had made an appropriate notification per 10 CFR 50.72. At the conclusion of the inspection period, NYPA had not completed their evaluation or submitted the 10 CFR 50.73 report (LER No. 95-003) for this event. The inspectors will review this LER in a subsequent inspection.

2.1.5 Inspector Followup of DER 95-0231

DER 95-0231 addressed some licensed operator radiation protection practices concerns identified by a quality assurance (QA) inspector. The licensee had not yet completed their evaluation of these concerns or developed a comprehensive list of corrective actions by the end of the inspection period. Consequently, this issue is unresolved, pending completion of NYPA's evaluation and NRC inspector review of the corrective actions **(URI 95-02-04)**.

3.0 MAINTENANCE (62703,61726,92902)

3.1 Maintenance Observation

The inspector observed and reviewed selected portions of preventive and corrective maintenance to verify compliance with codes, standards and Technical Specifications, proper use of administrative and maintenance procedures, proper QA/QC involvement, and appropriate equipment alignment and retest. The following inspection activities were conducted:

- Work Request (WR) 94-00292-00, preventive maintenance to replace safety related 125 volt DC breakers in accordance with maintenance procedure, MP 200.16 and Installation Specification IS-E-07, observed January 4, 1995.
- WR 94-05903-01 preventive maintenance to change out control rod drive units (CRD) with new BWR-6 CRDs, observed January 4, 1995.
- WR 94-0452-00, maintenance to replace ASCO scram pilot valve assemblies, observed January 5,1995.
- WR 93-03770-00, maintenance per Maintenance Procedure MP-072.01 to replace the bearings and align the shaft of the B standby gas treatment system centrifugal exhaust fan, reviewed January 25, 1995.
- WR 94-07761-00, field work per modification D1-91-150 that increased the RCIC turbine governor end pedestal drain line size and installed an oil sample valve, reviewed January 11,1995.
- WR 94-06778-01, field work per modification M1-93-059 that replaced the A core spray test line, 10"-W23-152-9A, reviewed and observed January 23, 1995.
- WR 94-06776-00, torus inspection, desludge, and inspection of ECCS strainers for debris. Reviewed and observed during various times during the refueling outage.

No significant concerns were identified during inspector review of the above activities.

3.1.1 Outboard Main Steam Isolation Valve Crack

While performing maintenance on 29 AOV-86C, outboard main steam isolation valve (MSIV), the licensee discovered a crack in the south wear guide. A one inch long by one quarter inch deep crack was found where the top of the wear guide joins the valve body. The wear guide is internal to the valve and is one of three wear guides that help to guide the valve disk into the seat during valve closure. The crack was found in weld material and did not appear to extend to the base material. The wear guide is part of the carbon steel casted valve body. Based on information received from the vendor, NYPA postulated that the weld material was from a previous repair of the casting at the valve manufacturer's facility, Rockwell-Edwards. The weld material is stellite, as is the cladding on the wear surface of the wear guide. The inspector observed the defect, reviewed memorandum JMD-95-020, "Disposition VT3 Examination on 29 AOV-86C from ISI Inspection," and discussed the issue with the maintenance engineering staff. NYPA determined that the presence of the defect did not interfere with the safe and reliable operation of the valve and left the condition "as found."

The NRC staff review of the evaluation in memorandum JMD-95-020 and expressed concern that it lacked sufficient technical detail and quantitative analysis to support leaving it in the "as found" condition. The evaluation did not include a consequence analysis or address postulated scenarios. At the end of the inspection period, the NRC staff was provided with a copy of a structural analysis report prepared by a contractor that was more fully developed and addressed these concerns.

The analysis, in part, reviewed: the effects of flow and flow induced vibration; stress effects of internal pressure (hoop stress); evaluation of crack propagation through the valve body; and fatigue crack growth. The report concluded that the existing crack did not present a structural concern for the valve and that the predicted growth rate for the crack would not be a problem for the remaining plant life. The report also concluded that complete through rib failure would not jeopardize the proper operation of the valve.

The inspectors concluded that the structural analysis provided additional technical basis to substantiate the original determination of accept-as-is made by the NYPA engineering staff. The NRC staff reviewed the analysis and had no further questions.

3.1.2 Control Rod Drive Hydraulic Control Unit (HCU) Isolation Valve Concerns

In response to General Electric Service Information Letter (GESIL) 419, dated March 15, 1985, NYPA implemented a maintenance inspection plan in 1992 to perform dye penetrant inspections on the valve disk of the 03HCU-112 valves. During the 1994-95 refueling outage, the maintenance staff discovered two valve disks with cracks in the area where the disk mounts to the valve stems. The scram discharge volume one-inch manual isolation valves (03HCU-112s) are normally open, and closed only for maintenance. The concern, as identified in GESIL 419, is that the valve wedge (disk) could potentially separate from the stem and block water flow from the HCU, and thus prevent scramming of the control rod. GESIL 419 referenced a plant where this event occurred, but the control rod was subsequently driven in via the reactor manual control system.

NYPA's previous inspections in 1992 did not identify any defects within the 10% sample size (13 valves). However, inspection during the 1994-95 outage identified two of thirteen wedges examined to have a crack in the "L" shaped stem to disk interface area. Additionally, ten of the wedges were found to be pitted severely enough that dye penetrant inspection could not be performed. During the work package closeout, the maintenance engineering staff noted the two cracked wedges and implemented actions to inspect a second sample lot. The second sample of 13 valves did not reveal any more defects. However, because the original lot of 13 wedges were discarded before a failure analysis could be performed, one wedge replaced from the second sample set was sent for further analysis. The inspector concluded that communications between maintenance engineers and work planners could have been better so that the work request could have instructed the workers to retain the removed wedges.

The GESIL identified the failures in the Hancock isolation valves to be caused by intergranular stress corrosion cracking (IGSCC). The GESIL also stated that Henry Vogt valves, also used at various plants, utilize AISI 420 stainless steel as do the Hancock valves. To date, NYPA has found only Henry Vogt valve disks. Based on industry experience and plant staff knowledge, NYPA postulated the cracks to have been of non-IGSCC origin and more likely a manufacturing anomaly.

NYPA engineering disposition JMD-95-041 concluded that the operability of the total population of 03HCU-112 valves and the control rod drive system was not challenged as a result of the inspection findings. The disposition recommended that the next refuel outage sample size be adjusted, if required, following the examination of the wedge mentioned above. The inspector found the above actions to be satisfactory.

3.1.3 Core Spray Test Line

The inspector performed a detailed walkdown of plant modification M1-93-059, "A Side Core Spray Test Line Replacement." The inspector utilized system piping drawings and pipe support drawings to verify proper installation of the modification and restoration of hangers and supports. The inspector reviewed: completed pipe support and installation inspection reports; weld map reports; and reviewed non-destructive examination inspection forms.

During the outage, NYPA replaced a portion of both the A and B side core spray test lines with stainless steel pipe. Excessive wall thinning of the original piping had been identified by the Erosion Corrosion Program. The inspector verified that hangers and supports were made-up properly and that equipment in the vicinity of the work area had not been adversely impacted by the modification work. The inspector concluded that the work was done in accordance with station drawings and procedures. The welding was of good quality and pipe supports were reinstalled correctly. The inspector noted that cutting and welding debris remained in the work area and, in particular, on an adjacent safety related valve. The inspector brought this to the attention of the responsible plant representative and was assured that work area clean-up would be conducted prior to final closure of the work package.

The inspector subsequently observed ST-3T, Core Spray Class 1 Piping System Leakage Test for 10-Year Inspection Interval, to verify the adequacy of the post-work test and compliance with technical specifications (TS). The inspector reviewed the test equipment and discussed its operation with the test technicians. The inspector walked down the pressurized sections of core spray piping both inside and outside of the containment. Test boundaries were discussed with the NYPA quality assurance staff. The inspector concluded that the evolution was controlled well, procedures were used properly, and TS requirements were met.

3.1.4 Control Rod Drive Leak Test Critique

During the previous inspection period (Inspection Report 94-29), the inspector identified a potential problem with a test gauge during leak testing of control rod drive mechanisms. NYPA concurred with the inspector's observations, replaced the gauge with a more accurate gauge and completed the testing. Subsequent to the issuance of the last inspection report, NYPA informed the inspector that the original determination that the gauge was reading conservatively, in the high direction, was incorrect. The gauge was in fact reading 10 psi lower than actual pressure. This information and NYPA calculations used to determine the acceptability of the test results were documented in JAF critique memorandum JMD-95-046. The inspector reviewed the critique, the corrected leak rate data, and the corrective actions. The overall impact of the lower test pressure was of minor consequence with respect to the test results. The inspector determined that the above actions were appropriate.

3.2 Surveillance Observations

The inspector observed and reviewed portions of ongoing and completed surveillance tests to assess performance in accordance with approved procedures and Limiting Conditions for Operation, removal and restoration of equipment, and deficiency review and resolution. The following tests were reviewed:

- ST-20M, Scram Discharge Volume Vent and Drain Valves Full Stroke and Timing Test (IST), performed 1/23/95, reviewed 1/24/95.
- ST-03T, Core Spray Class 1 Piping System Leakage Test For 10-Year Inspection Interval (ISI), performed and observed 01/25/95 (see section 3.1.3).
- ST-15B, Suppression Chamber and Drywell Deterioration Inspection, performed 1/19/95, reviewed 1/20/95.
- STP-76AU, Relay Room Enclosure Integrity Test, performed and observed on 01/25/95.

- ST-2AJ, RHR Loop A Containment Spray Headers and Nozzle Air Test, performed 1/31/95, reviewed 2/2/95 (see section 2.1.3).

No significant concerns were identified during inspector review of the above activities.

4.0 ENGINEERING (37551,92903,71707)

4.1 Reactor Core Isolation Cooling Modifications

The licensee, as well as other utilities (see NRC Information Notice 94-84), has experienced problems with the reactor core isolation cooling (RCIC) system turbine lubricating oil system. The RCIC system provides high pressure makeup water to the reactor vessel utilizing a steam turbine driven pump and associated valves and piping. A gear pump is driven by the turbine shaft through a worm gear to supply lubricating oil to the turbine bearings and governor valve. The lubricating oil then gravity drains to a common sump where the gear pump takes suction. The problems have been occurring when the oil fails to drain fast enough from the governor end (G/E) bearing housing and subsequently rises high enough where the over-speed disk whips air into the oil, causing the level to rise further. The erratic lube oil level would then require the turbine to be shut down.

In October 1993, the licensee installed a temporary sight glass in an attempt to resolve the air entrained oil phenomenon. During subsequent surveillance testing the G/E bearing housing oil level rose high enough to spray out the sight glass. NYPA concluded that eventhough the sight glass installation was part of the problem, it also revealed the underlying problem of an undersized drain line. Corrective actions at that time included: removing the temporary sight glass; changing the procedure to more tightly control oil addition during maintenance; and developing a modification to increase the size of the G/E pedestal drain piping.

During the 1994-95 outage, corrective actions included installing modification D1-91-150, RCIC Turbine Lube Oil Piping Design Change, and lowering the relief valve's setpoint. Industry information revealed that lowering the oil system relief valve setpoint was effective in mitigating oil level problems.

The inspector reviewed plant modification, D1-91-150, and discussed the issue with the licensee staff. The inspector noted that the modification did not address any special post-modification testing. Information provided by the industry and NRC indicated that some oil problems take a period of time to surface, which was longer than the duration of most surveillance testing. The inspector was concerned that a normal surveillance test on the RCIC pump may not adequately verify that the lubricating oil problem was resolved. NYPA subsequently reviewed the modification and added appropriate post-work testing to the modification package.

The inspector concluded that NYPA was taking positive corrective action in addressing this long-term issue. The omission of post-work testing on the modification was viewed as a minor oversight, in that, performance testing of the system is part of normal plant start-up preparations.

4.2 Previously Identified Items

4.2.1 (Updated) Unresolved Item (92-14-01): Relay Room CO_2 System Testing

The inspector witnessed the performance of special test procedure, STP-76AV, Relay Room Enclosure Integrity Test. The purpose of STP-76AV was to collect data for a subsequent engineering analysis and confirmation of the relay room as a carbon dioxide protected enclosure. The test methodology consisted of a tracer gas test (measuring changes in gas concentration) and a door fan test (pressurization and depressurization of the enclosed space). The inspector reviewed the procedure, walked down portions of the ventilation system, and discussed the test methods with station personnel. Data obtained during the test will be utilized by NYPA to model the relay room CO_2 system performance. At the end of the inspection period, the inspectors had not received the final report on the analysis. This unresolved issue remains open.

5.0 PLANT SUPPORT (71707,40500,92904)

5.1 Improper Use of Dosimetry by Contractor Visitor

On January 17, 1995, NYPA informed the NRC staff that following an internal review into an observed procedural violation by a site visitor (visitor B) on December 8, involving the wearing of outer garments under anti-contamination clothing, additional improprieties were identified. Specifically, visitor B was also determined to have entered the radiologically controlled area (RCA) on December 8 and 9 using dosimetry (a thermoluminiscent dosimeter) not issued to him by the site dosimetry office. NYPA investigation revealed that visitor B's escort (a contractor supervisor) provided him with a TLD issued to a different visitor (visitor A) whom the supervisor had escorted on an earlier occasion. The supervisor had visitor B log into and out of the RCA under visitor A's name, in lieu of having dosimetry issued specifically to visitor B. NYPA reading of the visitor TLD identified that neither visitor received a measurable amount of radiation exposure.

This issue remains unresolved, pending further review by the NRC staff. (URI 95-02-02)

5.2 Unauthorized Approval of Combustible Control Permit

On February 3, 1995, NYPA informed the NRC staff that preliminary results of an internal investigation of combustible control permit authorization inconsistencies identified that a fire protection supervisor/fire inspector inappropriately used the Fire Protection System Engineer's signature on Permit No. 94120, dated October 18, 1994. The permit was issued to allow temporary storage of three boxes of HEPA filters. The fire protection supervisor involved admitted to inappropriately using the fire protection engineer's signature. The inspectors were informed by station management that a broad investigation/critique of the station staff's involvement with the inconsistent combustion control permit authorization, and a technical/administrative review of all active and expired combustible control permits was underway at the conclusion of the inspection period. Pending completion of NYPA's review of this issue and detailed follow-up by the NRC staff, this issue remains unresolved. (URI 95-02-03)

5.3 Industrial Safety Review

During the 1994-95 refuel outage, the main condenser was re-tubed. As would be expected, the work was man-power intensive and required a significant amount of scaffolding and temporary staging to complete the work. Concerns were raised with the adequacy of the scaffolding erected and the use of personnel fall protection in those areas where scaffolding was impractical. The station safety supervisor was tasked with assessing the adequacy of the contractor's controls in this area in late December, and periodically examined the industrial safety practices through to the completion of this job.

The inspectors reviewed an internal memorandum dated December 28, 1994, in which the safety supervisor summarized his observations and corrective actions for industrial safety practices and scaffolding deficiencies he identified. The safety supervisor noted that the tubular frame type scaffolding staged in front of the condenser lacked proper toeboards and handrails. In addition, no safety belts or harnesses were being used by the workers on this scaffolding. Because of the necessity to change the elevation of the scaffolding to support re-tubing, fall protection (safety belts and harnesses) was subsequently implemented. The safety supervisor also noted the lack of proper fall protection for work inside the condensers. In this case, the installation of proper scaffolding was impractical. Additional safety ropes were strung and workers inside the condenser were required to wear safety belts or harnesses once they were at their work area inside the condenser. The wearing of hard hats inside the condenser was also more strictly enforced after the safety supervisor's review.

The safety supervisor's review of the contractor's industrial safety practices also revealed a higher than normal (station average) number of minor hand injuries (cuts, contusions, and abrasions) requiring first-aid treatment. This observation was discussed with contractor supervision for follow-up and corrective action.

The inspectors periodically witnessed re-tubing activities and observed only minor industrial safety infractions (safety glasses and hard hats not being worn) that were promptly corrected. No lost-time accidents or injuries resulted from the ongoing re-tube work. The inspectors found NYPA's response to the identified industrial safety concerns to have been appropriate. No additional industrial safety concerns were observed by the inspectors on this job or other site work during the outage.

5.4 Standby Gas Treatment Charcoal Filter Water Deluge System Concern

While conducting work items generated by a previous action plan, NYPA discovered that the piping for the standby gas treatment (SBGT) filter spray system contained debris. The quantity of this debris was of sufficient amount that the spray system may not have performed when manually actuated.

During the September 1993 time period, while conducting surveillance test

ST-76M, Nozzle Air Flow Test for Standby Gas Treatment System, the licensee discovered that eight of the eighteen fire protection water spray nozzles protecting the A and B SBGT charcoal filters were clogged. NYPA, at that time, concluded that the clogged nozzles could have resulted in the water spray system being ineffective in extinguishing a postulated charcoal filter fire. Short-term corrective actions for the September 1993 event included unclogging the nozzles. One of the long-term corrective actions included preparation of a work package to conduct additional inspections of the piping upstream of the flow control nozzles. It was during the 1994-95 outage inspection that NYPA discovered the accumulation of scale in the piping between the flow control valve and the ST-76M test connection.

By performing system walkdowns, NYPA was able to determine that the source of scale, in the normally dry piping, was the result of residual water trapped in an undrainable section of the piping (loop seal). They postulated that following an inadvertent actuation of the spray system, this section of piping did not drain and provided the water source for a long-term corrosion process. The corrective actions for this event are not yet complete. However, the piping was flushed and drained completely to minimize the chance of clogging the nozzles in the near future.

The inspector concluded that the corrective actions following the initial event could have been more aggressive and timely to identify the loop seal in the water spray line. However, the safety significance of the issue was minor, given manual fire suppression equipment was available in the area to combat a fire, had the water spray system failed as a result of clogged nozzles.

6.0 MANAGEMENT MEETINGS (30702,71707)

6.1 Exit Meetings

At periodic intervals during the course of this inspection, meetings were held with senior facility management to discuss inspection scope and findings. In addition, at the end of the period, the inspectors met with licensee representatives and summarized the scope and findings of the inspection as they are described in this report. The licensee did not take issue with any of the findings reviewed at this meeting.