U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No.:	95-18
Docket No.:	50-333
License No.:	DPR-59
Licensee:	New York Power Authority P.O. Box 41 Lycoming, New York 13093
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, New York
Dates:	August 6, 1995 through September 23, 1995
Inspectors:	G. Hunegs, Senior Resident Inspector R. Fernandes, Resident Inspector T. Kenny, Senior Operations Engineer

Approved by: C J Cowgill III 10/20/95
Curtis J. Cowgill, Chief Date
Reactor Projects Branch 2, DRP

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

RESULTS: See Executive Summary

EXECUTIVE SUMMARY

James A. FitzPatrick Nuclear Power Plant

Inspection Report No. 50-333/95-18

Plant Operations: Actions to reduce the adverse effects of a degraded residual heat removal service water system strainer demonstrated good system awareness and understanding.

On September 5, 1995, a personnel error resulted in control signal failures to both the A and B reactor feedwater pumps with a corresponding lowering of reactor vessel level and a low level reactor scram. Plant safety systems responded as designed and the operators' response to the transient was appropriate given the speed at which the event occurred. The pre-start briefing package and tailgate topic review were comprehensive. The post-transient review package was in depth and of good detail. Identification and recommendation of an engineering evaluation of the potential single failure of the fuse in the feedwater control circuit was noted by the review group. With a few exceptions, the operators responded well to the transient.

There have been several protective tagging record (PTR) related events recently, which taken individually, have minor safety significance, but taken collectively, indicate that greater attention is warranted in this area. The personnel error which resulted in the reactor scram showed that the procedure for processing protective tagging requests was not followed **(VIO 95-18-01)**.

The vessel pressure temperature requirements were exceeded following the reactor scram and require further review **(URI 95-18-02)**.

Maintenance: Good coordination of work activities involving on-line maintenance for the A residual heat removal system resulted in a net safety benefit by making the equipment more reliable.

Improvement in work flow process and communications was noted during the outage.

A maintenance programmatic review showed that management was aware of the corrective maintenance backlog and appeared to be making an effort to reduce it, and that self assessment was being accomplished with positive results. Good depth and detail in the licensee's periodic review of equipment history documentation was noted.

The failure of a primary containment isolation system reset switch was the result of improper switch configuration at installation **(URI 95-18-03)**.

Plant Support: Tours of the protected area (PA) revealed no degraded conditions of the PA boundary, adequate lighting, and satisfactory conditions in the isolation zones.

(Executive Summary Cont'd)

Safety Assessment/Quality Verification: The licensee's outage self assessment in the area of radiological controls was a positive step taken to enhance the radiological controls program.

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DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

1.1 NYPA Activities

At the beginning of the inspection period the unit was operating at full reactor power. On September 5, 1995, the unit scrammed due to low reactor vessel water level. The low water level was caused by a problem with the main feedwater systems following a personnel error involving improper fuse removal. Following the five-day forced maintenance outage the reactor was made critical on September 10 and returned to full power September 17.

During the five-day shutdown the FitzPatrick staff implemented a pre-planned forced outage schedule to complete various maintenance tasks. Significant work completed by the licensee included replacement of the A station battery, repair of the B reactor recirculation pump mechanical seal, and replacement of three safety relief valve pilot assemblies.

On August 18 the Nuclear Advisory Committee to the Board of Trustees of the New York Power Authority met at the James A. FitzPatrick site. The committee was established to advise and assist the Board of Trustees in the discharge of its responsibilities. Areas of discussion included current plant status, review of recent events, and operation goals for both NYPA plants.

Near the end of the inspection period senior management conducted a safety stand down to emphasize to plant staff the need for self-verification.

1.2 NRC Activities

A region based specialist inspector conducted a review of the radiological controls program during the week of August 14, 1995 (reference NRC inspection report 50-333/95-17).

The Division of Reactor Projects Chief for Branch No. 1 was on site August 16-17, 1995 to tour the facility and conduct interviews.

On August 29-30, 1995, the Region I Division of Reactor Projects Director and Project Branch Chief were on site to discuss recent plant performance with station and corporate management personnel and to tour the facility.

On August 30, 1995, the James A. FitzPatrick Executive Plant Performance Review public meeting was held at the FitzPatrick training center (see section 6.1 of this report).

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

2.1 Operational Safety Verification

The inspectors observed plant operation and verified that the facility was operated safely and in accordance with procedures and regulatory requirements. Regular tours were conducted of the plant with focus on safety related structures and systems, operations, radiological controls and security. Additionally, the operability of engineered safety features, other safety related systems and on-site and off-site power sources was verified. No safety concerns were identified as a result of these tours and good performance was observed. Plant housekeeping was also assessed as good.

During the inspection period, the inspectors provided on-site coverage and followup of unplanned events. Plant parameters, performance of safety systems and licensee actions were reviewed. The inspectors confirmed that required notifications, when warranted were made to the NRC. During event followup, the inspectors reviewed the corresponding documentation, including the event details, root causes, and corrective actions taken to prevent recurrence.

2.2 Followup of Events Occurring During Inspection Period

2.2.1 Residual Heat Removal (RHR) Service Water Strainers

On August 15, the inspector observed surveillance test (ST)-2X, RHR Service Water Flow Rate, Strainer, and Inservice Test (IST). During the test, the discharge strainer basket for the B and D RHR service water pumps failed to isolate.

The RHR service water system provides cooling water to the RHR heat exchanger for post-accident containment heat removal. The system consists of two 100% capacity, independent supply loops. The RHR service water pumps discharge water into a common line and then through a dual basket type strainer. The strainer design allows the plant operators to clear debris from one strainer while the other strainer is in service which allows the system to remain operable.

During the surveillance test, the operators were unable to isolate the number two strainer. The effect of this condition would be that operators would be unable to clean out the number two basket and would have to rely on the number one basket not clogging up.

The inspector noted that a plant identified deficiency (PID) was written, and that the system engineer was notified appropriately. In discussion with the licensee, the inspector learned that the most probable cause of failure was a misaligned O-ring (gasket). The licensee indicated that a modification was in process to change the O-ring configuration during the next refueling outage to eliminate this problem. The licensee plans to investigate/repair the strainer during the next system outage. A special condition tag was installed on the strainer to ensure the number one (isolable) strainer remains in service to facilitate cleaning. The inspector considered that the licensee had taken prudent interim actions. The inspector had no further questions on this issue.

2.2.2 Low Reactor Water Level Scram

On September 5, 1995, at 1:03 p.m., a personnel error resulted in control signal failures to both the A and B reactor feedwater pumps (RFP) with a corresponding lowering of reactor vessel level and a low level reactor scram. Initial conditions at the time of the scram were that the plant was operating at 100% power and at normal water level of 202 inches. In response to the low reactor water level alarm, plant operators took manual control of the reactor feedwater pumps. However, operators were unable to recover the level due to the "lockout" of both feedwater control circuits because of the control signal failure. A low level reactor scram was received at 177 inches approximately 15 seconds after the fuse was removed. Level continued to decrease until high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) initiated. Alternate rod insertion also initiated and both reactor water recirculation pumps tripped. Minimum recorded level was 123.5 inches. With HPCI, RCIC and RFPs injecting, water level rapidly increased until the HPCI, RCIC, and RFP turbines tripped on high reactor vessel level. When the high level alarm cleared, a RFP was restarted and used to control reactor vessel level.

All control rods fully inserted on the scram and plant safety systems responded, as designed, to the decreasing and increasing reactor water level. The operator's response to the transient was appropriate given the speed at which the event occurred and within the bounds of plant procedures. The inspector observed control room activities during the shutdown following the scram and reviewed various aspects of the licensee's post-transient evaluation.

The post-transient review group identified equipment deficiencies and personnel performance issues and developed corrective actions. The inspector reviewed the post-transient review recommended corrective actions and based on the review, observations and discussions with personnel, determined that they were implemented prior to the subsequent startup. Equipment problems were repaired and/or evaluated. The pre-start briefing package and tailgate topic review were comprehensive and to the point. The post-transient review package was in depth and of good detail. Identification and recommendation of an engineering evaluation of the potential single failure of the fuse in the feedwater control circuit was noted by the review group. With a few exceptions, the operators responded well to the transient.

2.2.3 Personnel Error Causing Reactor Scram

As described above, when operators removed a power supply fuse in error during preparation for maintenance, a reactor scram on low water level occurred (see section 2.2.2). The inspectors reviewed the operations staff critique, JOPS-CRT-95-017, to evaluate the root cause analysis and corrective actions.

The critique summarized that two licensed operators, when confronted with a discrepancy between the fuse labelling in an electrical panel and the protective tag request (PTR), incorrectly reconciled the difference by concluding the panel label was wrong or faded by aging. They removed the incorrect fuse which induced the feedwater control problem and subsequent reactor scram.

Guidance on protective tagging is outlined in administrative procedure (AP)-12.01, Equipment and Personnel Protective Tagging, and in the operations department night orders on dual-concurrent verification. The inspector reviewed the circumstances surrounding the event and independently determined that several errors were made that contributed to the transient and low water level scram. As stated in the critique the control room supervisor had performed two distinct functions in the process: the PTR independent reviewer; and indirectly by way of dual-concurrent verification, the tagging operator. This action, although not prevented by procedure, compromised the independence of the tagging operator (person removing the fuse) from the person who second-checked the PTR. The dual-concurrence policy requires two people to verify the initial positioning of components when hanging a PTR. In this instance the tagging operator's decision was affected by the CRS who had previously erroneously identified the fuse.

The inspector also determined that neither the shift manager nor the PTR controller were informed of the label discrepancy as required by AP-12.01 when a tagging discrepancy or labelling problem exists. The inspector noted that this error was not identified in the critique.

Corrective actions included: counseling and briefing of operations personnel; revision of the PTR procedure to require specifying both system fuse number and panel fuse number; and establishing operator aids for fuse identification in panels containing critical circuits. The inspector noted that in an earlier revision of the critique, recommendation was given to change the PTR procedure to prohibit the PTR preparer or second checker from actually applying the PTR tags. Operations staff communicated to the inspector that this was not implemented because of potential difficulty during backshift PTR activities.

Listed below are other instances of PTR related events, which taken individually, may have minor safety significance but taken collectively indicate that greater attention is warranted in this area:

- NRC inspection report 94-29 documents a PTR error in which lack of self checking and self verification resulted in a loss of shutdown cooling.

- DER 95-005, dated January 12, 1995, involved the hanging of a tag on a circuit breaker labeled 71ACA4 CKT#8 vice the PTR required circuit breaker 71DCA4 CKT#8. The error was discovered by an electrician who was performing a walkdown prior to working the job.

- DER 95-660, dated March 21, 1995, involved the operation of a circulating water tempering gate while maintenance was being performed, without the PTR tag holders permission. No personnel were injured or equipment damaged. By procedure, the work was being performed with special condition tags in place. However the communications and controls in place did not prevent the near miss.

- DER-1129, dated July 21, 1995, discusses the discovery of a pump circuit breaker in the off position when the PTR released it in the on position.

The sequence of events in this most recent event is of concern to the NRC because two licensed operators made an incorrect decision, when procedure controls were in place to reduce the chance of mistakes being made, that resulted in a significant plant transient and challenge to the operators.

Technical Specification 6.8(A) states, in part, that written procedures and administrative policies shall be established, implemented, and maintained that meet or exceed the requirements and recommendations of Section 5, Facility Administrative Policies and Procedures, of ANSI 18.7-1972. ANSI 18.7-1972, Section 5.1.2 states, in part, that procedures shall be followed, and the requirements for use of procedures shall be prescribed in writing. Administrative procedure AP-12.01, section 6.9 requires the operators to notify the controller if any discrepancies exist between the label and the component identified on the PTR when hanging tags. Contrary to this on September 5, 1995, the discrepancy between the fuse label and the PTR was not brought to the attention of the controller. Subsequent operator actions resulted in a plant transient and scram. This is a violation. **(VIO 95-18-01)**

2.2.4 Exceeding Technical Specification Pressure - Temperature Curve Requirements

As a result of the scram, in conjunction with the recirculation pump trips, the temperature in the bottom head of the reactor dropped low enough such that the pressure - temperature (P-T) curves in the technical specifications (TS) were exceeded.

Licensee analysis indicated that the P-T limits for the bottom head were not exceeded as the bottom head curve is less restrictive than the more conservative pressure temperature curves included in TS. The P-T curves in the TS were developed to bound the conditions for the entire reactor vessel. In addition to exceeding the P-T curves for the lower head, the 100 degree per hour heatup and cooldown rate limits for the bottom head were exceeded during recovery from the scram. Exceeding the heatup and cooldown rate limits for the bottom head was determined from the temperature indication on the reactor vessel drain line which was isolated at the time. The licensee considers the temperature indication from the bottom head drain to be the most conservative as it was the most limiting. The 100 degree per hour heatup rate for the reactor vessel was exceeded when reactor water cleanup was reinitiated. Licensee analysis indicates that exceeding these limits is bounded by previous analysis. This issue is unresolved pending completion of the licensee's review and corrective actions and subsequent NRC review **(URI-95-18-02)**.

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 quality assurance/quality control (QA/QC) involvement, and appropriate equipment alignment and retest. The following maintenance work requests (WR) activities were observed:

- WR No: 94-10062-01 Replacement of Conoflow electric/pneumatic converter, 27E/P-103B, for the containment atmospheric dilution system flow control valve FCV-103B

- WR No: 95-05257-00, Replacement of the B control rod drive water pump, 03P-16B

- WR No: 95-04978-03, Contingency Plan If Battery Charger 71BC-1A Fails During Replacement of A Station Battery, reviewed on September 9, 1995

No concerns were identified during inspector review of the above activities. The activities observed and reviewed were properly conducted. In addition to the routine maintenance observations, the inspectors reviewed various maintenance activities and meetings associated with the outage. The inspectors noted, in general, improvement in work flow process and communications.

3.1.1 LCO Maintenance, Residual Heat Removal System

On August 7 the licensee entered a seven-day Technical Specification limiting condition for operations (TS LCO) to conduct on-line maintenance on the A train of the residual heat removal system (RHR). Maintenance planned included replacement of the C RHR pump seal, replacement of a torque switch on a motor-operated valve; replacement of a cooling water solenoid valve, service water strainer packing replacement, and other items. The inspectors reviewed the activity to evaluate the impact on safety and the licensee's use of procedures and maintenance practices regarding the removal of safety-related equipment from service for maintenance while the plant is at power.

The licensee utilizes an administrative procedure (AP)-5.13, Maintenance During LCO's, to define the plant policy for a planned LCO entry to conduct maintenance. The procedure defines the responsibilities for the various site managers and the assigned LCO coordinators for the maintenance task. The procedure lists precautions and limitations including requirements for operations management review of any scheduled maintenance and testing which has a potential for causing ECCS actuation, RPS trip, or automatic isolation of redundant ECCS trains. LCO screening and preparation checklist are utilized to document and obtain various reviews and approvals by department managers including the site executive office approval. The screening checklist ensures the task can be completed within 60% of the available time for the LCO and the evolution will result in less than five percent unavailability for the affected equipment. This task was completed within 19 percent of the available time and resulted in a 2.75 percent unavailability.

The licensee utilizes a detailed bar schedule to track work progression on the various items to be performed during the maintenance period. The progress was discussed at several meetings throughout the LCO period. The inspector determined that the extent of knowledge and awareness of supervisory and working level personnel concerning the schedule and maintenance activities was appropriate.

The inspector observed various work activities throughout the LCO and noted good coordination of work activities. The inspector concurred with the licensee's determination that the work completed resulted in a net safety benefit by making the equipment more reliable. The inspector performed system walkdowns, component tag-out reviews, and log book reviews to evaluate operability of the redundant train and found no adverse conditions.

In conclusion, the inspector determined that the evolution had minimal impact on overall safety and the planned maintenance was completed with minimal delays and problems.

3.1.2 A Review of Corrective Maintenance Backlog

The inspector conducted a review of the corrective maintenance backlog (repairs that can be made while the plant is operating) and concluded that there were no outstanding corrective maintenance items identified that were detrimental to the safe operation of the plant. Although the backlog list numbers 975 at this inspection, the most important items were being corrected promptly and plant operation was not hampered by the remaining backlog. Discussions with operating personnel indicated that maintenance does a good job addressing broken or malfunctioning equipment so the operators can operate the plant effectively and safely. The majority of the backlog was attributed to mechanical maintenance numbering 775, while electrical and I&C combined number around 200.

Of the 975 outstanding items there were 348 that were safety related, with the oldest dating back to 1988. The inspector determined that the older items were of a lower priority regarding plant operations such as, "A screw missing in a cabinet door," or "A sign painted over on a fire door." The more urgent repairs were being given the necessary priorities to maintain the plant operation in good order. Plant walkdowns by the inspector did not identify any obvious needed repairs such as water leaks or steam leaks. The plant was found to be clean and orderly.

The maintenance inspection of April 10-21, 1995 (NRC Inspection Report 50-33/95-09) identified that about 1100 corrective maintenance tasks were outstanding. At the close of that inspection, plant management indicated that the established goal was to have less than 400 corrective maintenance items and stated that they intended to achieve this goal in several months. The inspector noted that the work off rate was slower than expected. Since April the number of outstanding items has decreased by about 200.

Discussions with maintenance management disclosed that some new initiatives were being considered as a means to further reduce the backlog:

- All items over one year old have been assigned to the appropriate component engineer to determine if any can be moved to a minor maintenance list.

- After September 4, a group of experienced first line supervisors will walk down all open items to determine their validity.

The above two initiatives were intended to eliminate duplication as well as eliminate tasks that may have already been completed. The maintenance manager also indicated that these initiatives should aid in the work planning effort.

Plant managers were aware that the progress of working down the backlog was slow. However, the maintenance manager stated that the new initiatives should improve the progress on the backlog reduction.

The station was maintaining a monthly report that is reviewed by station management to keep them abreast of the corrective backlog developments. The report tracks and trends non outage as well as outage-related corrective maintenance and delineates: the goal (currently 600 non outage), the current backlog, the trend for the last seven months, and the number attributed to each department.

NYPA performed self assessment on the corrective maintenance program. The inspector reviewed an audit performed in 1994 (94-03J) and found it to be self critical with recommendations that led to improvement in the planning process and the reduction of the backlog. The inspector also noted that the audit concentrated on the I&C department. The inspector discussed his observation with station management at the conclusion of the inspection. The licensee acknowledged the inspector's observation.

3.1.3 Periodic Preventive Maintenance Reviews

The inspector reviewed plant procedures and conducted interviews with station personnel to determine if the licensee periodically reviews equipment history to identify repetitive failures or other adverse trends which may indicate inadequate or ineffective maintenance. Administrative procedure (AP)-5.05, Preventive Maintenance, requires periodic reviews of all corrective maintenance work requests and entry of adverse conditions and failure mechanisms into a database known as LONDON. Maintenance department standing order (MDSO)-6, Maintenance History, specifically requires a biennial review by the appropriate engineer of a list of all work requests performed since the last review. The review, in part, compares or trends corrective maintenance with previous periods, compares component PM effectiveness with its relative impact on plant overall maintenance, and identifies and evaluates components with multiple work requests. The licensee also notes that the system engineers review maintenance history in preparation of the system presentations given by them at periodic intervals.

The inspector reviewed the component history report for October 1993. The report included a review, by equipment area, of various safety-related equipment. The review period in general, covered the 1992 time frame and included recommendations from each engineer based on the findings of the review. The inspector concluded that the licensee performs periodic reviews of equipment history and noted that the 1993 report was of good depth and detail.

3.1.4 Primary Containment Isolation System Switch Replacement

On September 5, while performing instrument surveillance procedure (ISP)-100C, PCIS, PCIS Instrument Functional Test/Calibration, it was determined by maintenance personnel that one of the two primary containment isolation system reset switches had failed in the group one reset position. The switch was replaced that day; however, two weeks later the switch was found to not be working properly when another ISP was performed.

The function of the primary containment isolation system (PCIS) is to provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and the reactor coolant pressure boundary. The PCIS performs this by initiating automatic isolation of appropriate process lines which penetrate the containment, when monitored variables exceed certain limits. The purpose of the instrument surveillance procedure was to verify proper operation of the equipment which provides the initiating signals to the PCIS circuitry. After testing a specific initiation signal, the reset switch is rotated to the group reset positions to control the appropriate reset relays.

The inspector learned that the switch had failed to function properly in two distinct ways. The first event involved a loose screw in the internals of the rotary knob switch which prevented the switch from returning to the neutral position and de-energizing the reset relays. The second event involved the improper switch configuration installation. This resulted in the reset relays being energized when in the normal position and Group I reset position, and de-energizing when in the Group II/III reset position. At this time the inspector determined that in the first event the safety significance of the failure was minimal. In review of station prints, the Group I valves (main steam isolation valves) would have closed on an actual signal and would remain closed. By virtue of the reset relays being energized the subsequent opening of the MSIVs would have been possible for post-transient heat removal. This was also true for the second event. At the close of the inspection period the licensee was continuing to investigate the event and as such, not all the information was available for the inspectors. This issue will remain unresolved pending completion of NYPA evaluation and subsequent NRC review (URI 95-18-03).

3.2 Surveillance Observation

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-2X, RHR Service Water Flow Rate, Strainer, and Inservice Test (IST)
- ST-4B, HPCI Pump and MOV Operability Test
- ST-4N, HPCI Quick-Start, Flow Rate and Inservice Test
- ST-24J, RCIC Flow Rate and Inservice Test
- ST-9D, EDG, 115KV Reserve Power, Station Battery, or ESW Pump Inoperable

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

4.0 PLANT SUPPORT (71707,40500,92904)

4.1 Security

The inspectors performed a tour of the protected area (PA) during this inspection period to assess the integrity of the PA barriers. No openings or degraded conditions were found in the PA barrier and no concerns were identified by the inspector in this area.

5.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (40500,37551,92700,90712)

5.1 Review of Licensee Radiological Self Assessment

The inspectors reviewed outage self assessments by various licensee personnel in the area of radiological controls. The outage ALARA critique identified that the actual exposure for the outage was two thirds less than the projected outage exposure. Radiological and environmental services (RES) management and the critique attribute this to better communications; limiting of scope control; emphasizing first time quality; pre-staging of radiological work packages and ALARA review packages; and minimal steam leak repairs required on restart. Areas for improvement were identified and included methods for improving the work preparation process and review of the task which exceeded projected exposure. The outage manager's critique cited examples of improved teamwork between the radiation protection technicians and plant staff in completing work items. Additionally the inspectors reviewed an RES personnel evaluation of the radiological controls program at Indian Point 3. The evaluation was performed by RES technicians and concentrated on areas of concern identified by the IP3 RES staff at JAF on a previous visit.

The inspectors concluded that the cross plant reviews were a positive step taken by the RES staff to enhance the radiological controls program at JAF. Review of the deficiency event reports (DERs) for the outage period revealed a lack of any major radiological issues. Observations by the inspectors and interviews with plant personnel during the outage were consistent with the above self assessments.

6.0 MANAGEMENT MEETINGS (30702,71707)

6.1 Executive Plant Performance Review Meeting

On August 30, 1995 the FitzPatrick Executive Plant Performance Review Meeting was held at the FitzPatrick training center. The meeting was open to the public. Attachment 1 is the licensee's meeting handout which summarizes licensee corrective actions. Meeting attendees are included as Attachment 2. Four major areas were discussed: the control of contractors, Quality Assurance program effectiveness (corrective actions, thoroughness and implementation), radiation worker radiation protection practices, and inconsistent performance during unique or complex evolutions. The NRC concluded that the meeting was an effective forum to discuss plant performance issues.

6.2 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.

ATTACHMENT 1

Executive Plant Performance Review (EPPR) Meeting Handout

New York Power Authority
James A. Fitzpatrick
Nuclear Power Plant

WPO JAF IP3

Executive Plant Performance Review

Public Meeting

August 30, 1995

PERFORMANCE IMPROVEMENT AREAS

- Control of Contractor Work Activities
- Performance During Unique or Complex Evolutions
- Radiation Protection Program
- Quality Assurance Program Effectiveness (Corrective Action Thoroughness and Implementation)

PERFORMANCE IMPROVEMENT AREAS

CONTROL OF CONTRACTOR WORK ACTIVITIES:

- Training/Supervisory Oversight
  - Skills Demonstrated
  - Supervisor Qualifications
  - Expectations Communicated
- Project Managers
  - Incorporate Lessons Learned

PERFORMANCE IMPROVEMENT AREAS

CONTROL OF CONTRACTOR WORK ACTIVITIES:
(continued)

- Design Engineering Reorganization
  - Improved Communication
  - Modification Teams
  - Engineering Assurance Program
- Pre-Outage Milestones

New York Power Authority
James A. Fitzpatrick
Nuclear Power Plant

WPO
JAF IP3

PERFORMANCE IMPROVEMENT AREAS

PERFORMANCE DURING UNIQUE OR COMPLEX EVOLUTIONS:

- Planned Package Improvement
  - Chemistry Review
- Procedure Improvements
  - Need for Control Clear

PERFORMANCE IMPROVEMENT AREAS

PERFORMANCE DURING UNIQUE OR COMPLEX EVOLUTIONS: (continued)

- **Writers Guide Change**
  - Position/Person Notified
  - Reason Communicated
  - Expected Results Clear

PERFORMANCE IMPROVEMENT AREAS

PERFORMANCE DURING UNIQUE OR COMPLEX EVOLUTIONS: (continued)

- Use of Additional Oversight/Control
  - Senior Line Manager Responsible
  - Pace Controlled
  - Management Expectations Clear
  - Conservative Actions Necessary
  - Problems Resolved

PERFORMANCE IMPROVEMENT AREAS

PERFORMANCE DURING UNIQUE OR COMPLEX EVOLUTIONS: (continued)

- Use of Preoperational Tests
  - Recognition of Changed Operation
- Outage Preparation Improvements
  - Briefings to Raise Awareness/Sensitivity
  - Lessons Learned Emphasized
  - All Contractors Included
  - Training in Special Areas
  - Improved Planning

PERFORMANCE IMPROVEMENT AREAS

RADIATION PROTECTION PROGRAM:

- Improved Staff Accountability
  - Radiological Hold Process
  - Enhanced Radiation Worker Training
  - Management Observation Program Emphasis
  - Deviation Event Report (DER) Trending
  - Tailgate Meetings
  - Standards

PERFORMANCE IMPROVEMENT AREAS

RADIATION PROTECTION PROGRAM: (continued)

- Improved Radiation Protection Staff Performance
  - Conflict Resolution Skills
  - Coaching Skills

New York Power Authority
James A. Fitzpatrick Nuclear Power Plant

IMPROVEMENT WPO JAF IP3 THROUGH TEAMWORK

PERFORMANCE IMPROVEMENT AREAS

RADIATION PROTECTION PROGRAM: (continued)

- Quality Assurance Value Added
  - QA/RES Assistance
  - Radiation Protection Self-Assessments
  - Technician Involvement
- Radiation Procedure Upgrade Program
  - Consistent Direction
- Procedure Compliance
- Independent Team Reviews

PERFORMANCE IMPROVEMENT AREAS

QUALITY ASSURANCE PROGRAM EFFECTIVENESS:

- QA Oversight/Audit Process
- Deviation Event Report (DER) Trending
- Performance Enhancement Review Committee (PERC) Responsibilities
- Plant Operating Review Committee (PORC)
- Comprehensive Corrective Actions

PERFORMANCE IMPROVEMENT AREAS

QUALITY ASSURANCE OVERSIGHT/AUDIT PROCESS:

- QA Program Assessments
- Improvement Initiatives
  - Use of Technical Specialist
  - Enhanced Training for Auditors
  - Improved Audit Procedures
- Management Support
- Audit Finding Resolution

PERFORMANCE IMPROVEMENT AREAS

QUALITY ASSURANCE PROGRAM EFFECTIVENESS:

- QA Oversight/Audit Process
- Deviation Event Report (DER) Trending
- Performance Enhancement Review Committee (PERC) Responsibilities
- Plant Operating Review Committee (PORC)
- Comprehensive Corrective Actions

SALP RATINGS

Category 1

Licensee attention and involvement have been properly focused on safety and resulted in a superior level of safety performance. Licensee programs and procedures have provided effective controls. The licensee's self-assessment efforts have been effective in the identification of emergent issues. Corrective actions are technically sound, comprehensive, and thorough. Recurring problems are eliminated and resolution of issues is timely. Root cause analyses are thorough.

ATTACHMENT 2

EPPR Meeting Attendees

NRC

C. Carpenter, Project Manager, NRR
R. Cooper, Director, Division of Reactor Projects
C. Cowgill, Chief, Projects Branch 1, Division of Reactor Projects
G. Hunegs, FitzPatrick Sr. Resident Inspector
T. Marsh, Director, Project Directorate I-1, NRR

NYPA

W. Cahill, Chief Nuclear Officer
M. Colomb, General Manager, Operations
D. Lindsey, General Manager, Maintenance
R. Patch, Director, QA
D. Ruddy, Manager, Site Engineering
H. Salmon, Site Executive Officer
D. Topley, Acting General Manager, Support Services