U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No.: 93-24

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: October 10, 1993 through November 20, 1993

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

RESULTS: See Executive Summary

TABLE OF CONTENTS

TABLE OF CONTENTS (CONT'D)

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

EXECUTIVE SUMMARY

James A. FitzPatrick Nuclear Power Plant

Inspection Report No. 50-333/93-24

Plant Operations:

The plant operated at 100% power until a shutdown was commenced on October 23 to support a planned maintenance outage. A reactor vessel notching event was observed during the shutdown, NYPA installed a modification during the outage to preclude this type of event from recurring. Temporary operating procedures used to support the outage were reviewed and no concerns were noted. Most outage work proceeded without incident, however some performance deficiencies were noted which led to a resin spill and a shutdown scram signal. These events involving procedural noncompliance are being cited as violations of regulatory requirements **(VIO 93-24-01)**.

Maintenance:

Numerous maintenance and surveillance activities were observed and found to be adequately conducted. Though the majority of all work done during the outage was performed safely and correctly, there were several instances where performance deficiencies were identified. An inadvertent traversing in-core probe over-withdrawal event was reviewed. The cause of the event was inadequate knowledge of the system, however, NYPA's event evaluation was thorough. A review of teflon tape use in the plant found that while guidance and controls were weak, no actual misapplications were identified.

Engineering:

NYPA identified and resolved a reactor protection system hot short vulnerability. Follow up on Information Notice 93-79 found that NYPA is closely following industry information involving shroud cracking. FitzPatrick's shroud was made by the same manufacturer, during the same time period and of the same material as Brunswick's shroud. NYPA intends to perform a full inspection at the earliest opportunity (currently planned for the January 1995 refueling outage). NYPA identified additional deficiencies with their control room ventilation system. Compensatory measures were put in place prior to restart while a long term action plan is being executed. Inspector review found that the system can still perform its safety function. A detailed review of the reactor vessel water level backfill modification was conducted. The inspector found all aspects of the design, installation and testing to be adequate. However, the modification introduced a fatigue failure mode to the reference leg nozzles which could occur in 5.4 years. An additional modification is planned to eliminate this failure mechanism during the next refueling outage.

Executive Summary

Plant Support:

The inspectors performed a tour of the drywell during the outage. Radiological controls were good and only minor housekeeping deficiencies were noted. A review of a non-reportable security event involving the violation of NYPA's 5 hour pre-work alcohol abstinence period found that the site Fitness for Duty program is being adequately implemented. A review of fire protection Technical Specifications revealed that a once per three year manual hose station flow test was not being performed. The flow tests were subsequently satisfactorily performed. The failure to perform the required flow test is being cited as a violation of Technical Specifications **(VIO 93-24-02)**.

Safety Assessment/Quality Verification:

A review was conducted of NYPA's recently established nuclear programs assessment section (NPAS). The NPAS is providing an independent safety engineering group (ISEG) type function that previously did not exist at FitzPatrick. Based on limited evidence, the NPAS appears to be satisfactory and a good initiative. FitzPatrick conducted a brief safety stand-down on November 5 to stress safety and attention to detail, following several personnel performance incidents. This stand-down was a good initiative by station management. However, a few subsequent events occurred that indicated continued management attention was necessary. NYPA management agreed to examine further human performance initiatives.

DETAILS

1.0 SUMMARY OF FACILITY ACTIVITIES

1.1 NYPA Activities

At the beginning of the assessment period, the plant was operating at 100% power. On October 23, FitzPatrick commenced a shutdown for a planned maintenance outage. Major activities during the outage included reactor building closed loop cooling service water piping replacement and installation of a reactor vessel level reference leg backfill modification. Most work proceeded without incident, but some performance deficiencies were noted and NYPA held a brief safety stand-down on November 5 to reemphasize safety consciousness and attention to detail. At the end of the inspection period, the FitzPatrick staff commenced a normal reactor startup on November 20.

1.2 NRC Activities

The inspection activities during this report period were done during normal, backshift and weekend hours by the resident staff. There were 72 hours of backshift (evening shift) and 23 hours of deep backshift (weekend, holiday and midnight shift) inspections during this period. There were 422 hours onsite during this inspection period.

A region based team completed an Operational Safety Team Inspection during the week of October 11, 1993.

A region based inspector conducted an inspection of the environmental monitoring program during the week of October 18, 1993.

A region based inspector conducted a review of the outage radiation protection activities during the week of November 1, 1993.

A public meeting was held at the FitzPatrick Training Center on November 2 to present the findings of the Operational Safety Team Inspection.

A region based inspector conducted a review of the logic system functional testing program during the week of November 15, 1993.

Region based inspectors conducted an inspection of the fire protection program during the week of November 15, 1993.

2.0 PLANT OPERATIONS (71707,71710,93702,40500,62703)

2.1 Followup of Events Occurring During Inspection Period

2.1.1 October 1993 Planned Shutdown

On October 23 the FitzPatrick staff commenced a planned load reduction at 5:05 p.m. and removed the generator from the grid at 3:58 a.m. on October 24. All control rods were fully inserted by 2:18 p.m. on October 24 and a plant cooldown was commenced. During this plant shutdown and cooldown (as in the past few plant shutdowns) a special reactor vessel level watch was established to closely monitor vessel level during the depressurization. The purpose of this special watch was to monitor for any indications of level notching (transient false high level readings) caused by non-condensible gases coming out of solution in the condensing chamber reference legs. The inspector noted that to closely monitor reactor vessel level during the cooldown, the FitzPatrick staff had to use an uncalibrated extended range of the normal hot calibrated narrow and wide range reactor vessel level transmitters. Consequently the levels and level changes observed may not be accurate, but are still sufficient to monitor for the notching phenomena.

At approximately 8:20 p.m. (at a reactor vessel pressure of 25 psig), 9:00 p.m. (11 psig), and 11:00 p.m. (7 psig) reactor vessel level notches of 10, 22, and 5 inches, respectively, were observed by the special vessel level watch on only one channel of reactor vessel level (the wide range level associated with condensing chamber 2A). All other reactor vessel level instrument channels tracked normally. Control room operators closely monitored the 2A wide range channel and minimized any evolutions potentially impacting vessel level until it restored itself to the normal level band. The inspector noted that the wide range 2A condensing chamber was one of two chambers (2A and 2B) that exhibited a cooling-down as detected by the installed condensing chamber thermocouples (reference inspection report 93-12, section 4.1).

The inspector verified that the operations staff response to this event was appropriate. At no time was the control room staff without accurate level indication and proper vessel level inventory was maintained throughout the level indication transient. As discussed in section 4.5 of this report, the FitzPatrick staff installed a reactor vessel water level backfill modification that will preclude this type of event from recurring.

2.1.2 Shutdown Scram Signal

On November 5, NYPA made a 10 CFR 50.72 non-emergency notification that the plant had received a reactor scram, group 2 and shutdown cooling isolation signal. The plant was shut down at the time. Shutdown cooling remained isolated for 40 minutes and resulted in an approximate 4°F rise in reactor coolant temperature. The scram and isolation signals were caused by a false low reactor vessel water level sensed at one of the narrow range level instruments when the refueling level instrument high side drain was opened.

The events that led to this condition are as follows. As a result of NRC Bulletin 93-03, NYPA was performing a modification to the reactor water level reference legs to provide continuous backfill capability. As part of the pre-operational test of this modification, an additional temporary level transmitter was connected to the refueling level instrument high and low test drain valves. After the temporary transmitter was installed, a leak test was performed on November 5 to verify system integrity. The refueling high and low test drain valves were tagged shut during this leak test. After the leak test, the temporary transmitter sensing lines were depressurized and during the depressurization were partially drained. When the tags for the refueling instrument high and low test drain valves were released, the high side drain valve was opened, effectively venting the variable leg into the partially drained temporary transmitter piping. Since the refueling level instrument shares a common variable leg with one of the narrow range instruments, the vented variable leg caused the indication of an erroneously low reactor level on the narrow range instrument, which caused the isolation and scram signals.

This event was further complicated when the refueling low side test drain valve was opened. Because the isolation and bypass valves on the temporary transmitter remained open after the leak test, when the low and high side test drain valves were opened, a flow path was created which allowed the refueling reference leg to drain to the vessel which caused the refueling level instrument to read high.

There were a number of elements that contributed to this event. Specifically, as a result of the inadequacies of the required sketch for the hydrostatic test, the hydro hose was not installed at the correct connection and caused instrument tube draining when it was disconnected. Also, neither the leak test procedure nor the modification installation procedure addressed the possibility of draining the tubing after the leak test or required venting and filling the tubing prior to unisolating the temporary instrument. Finally, Administrative Procedure (AP)-12.01, Equipment and Personnel Protective Tagging, requires that the order of instrumentation valving be reviewed for its impact on a plant trip or system initiation. The failure to perform this review adequately is an example of procedural non-adherence and is a violation of Technical Specification 6.8(A). **(VIO 93-24-01)**

2.1.3 <u>Resin Spill</u>

On November 13, the spent resin tank was inadvertently overfilled and resulted in spilling approximately 4 cubic feet of resin in the radwaste building. This event occurred due to a failure to follow procedures. Earlier in the day, resin had been transferred from the waste demineralizer to the spent resin tank in accordance with Operating Procedure (OP)-34, Resin Transfer, Regeneration and Cleaning. The last step of this procedure shuts 20 AOV 311, resin inlet to spent resin tank valve, but this action was not taken due to personnel error. Therefore, when a subsequent transfer of resin from the mixed resin tank to the waste demineralizer was attempted, the resin was actually misdirected to the spent resin tank and overflowed it. The error was not discovered until later when 20 AOV 311 was noted to be open. No significant radiological exposure occurred as a result of this event and the spill was subsequently cleaned up. This event is a second example of procedural non-adherence and is a violation of Technical Specification 6.8(A). **(VIO 93-24-01)**

2.1.4 Temporary Operating Procedures Review

To support two major outage work activities (service water piping replacement and reactor vessel water level backfill modification) the inspector reviewed the procedural controls and implementation of three specific temporary operating procedures: TOP-153, Temporary Reactor Building Component Cooling Water System; TOP-156, Temporary Turbine Building Component Cooling Water System; and TOP-157, Reactor Pressure Vessel Level Control Using Condensate Transfer Keep Full. TOP-153 utilized a tractor trailer mounted chiller unit to supply cooling to selected closed loop cooling loads via temporary hose connections. TOP-156 utilized two submersible pumps in the screenwell and temporary hoses to supply cooling to the service air compressors. TOP-157, provided a temporary means of reactor vessel makeup via the condensate transfer system while both the normal feed and condensate systems were removed from service for repairs and the control rod drive system was out of service for the reactor vessel level backfill modification tie-ins.

The inspectors found these TOPs to be appropriately written and properly implemented. Walkdown of each temporary system identified no concerns. Control room operators were knowledgeable of the TOPs and temporary equipment and were familiar with abnormal system response guidance. The inspectors had no outstanding questions or concerns with the use of these TOPs.

2.1.5 Review of Residual Heat Removal Pump Motor High Temperature

During the morning tour of the control room on October 28, the inspector learned that earlier in the day the A residual heat removal (RHR) pump was secured due to high motor winding temperature and the C RHR pump was started to maintain shutdown cooling flow. The inspector was concerned about the potential for overheating the C RHR pump motor because the crescent area unit coolers were secured due to the service water systems outage to replace service water piping to/from the reactor building closed loop cooling heat exchangers.

Followup by the inspector and review of a November 4 memorandum (JTS-93-0688) by the system engineer identified that the A RHR pump motor windings generally run warmer. This condition has been known for several years, but motor winding temperatures have remained below the maximum winding design temperature of 302°F, as specified by General Electric. Since the completion of RHR pump performance testing in July 1991, NYPA has planned to perform RHR pump motor maintenance in the 1995 refuel outage.

Consequently, the lack of unit cooler operation in the reactor building crescent areas does not adversely impact RHR pump operability under the current shutdown conditions. In addition, operation of the C RHR pump motor was closely monitored and motor winding temperatures remained below the alarm setpoint. The inspector had no further questions.

2.2 Engineered Safety Features System Walkdown

The inspector conducted partial control room and in-plant walkdowns of the following systems:

- A and C emergency diesel generators
- A and B core spray
- A and B standby liquid control

No discrepancies were noted during the walkdowns of the above systems.

A detailed walkdown was conducted in the accessible portion of the residual heat removal service water (RHRSW) system. No discrepancies were noted. A review of RHRSW surveillance testing verified compliance with Technical Specification requirements. Minor deficiencies were noted with some surveillance tests and were discussed with responsible plant representatives. A review of the Final Safety Analysis Report determined that part of the system's safety design basis is that it provides an additional source of water for post-accident containment flooding via a cross-tie to the RHR system. However, the cross-tie piping is not QA category I piping or subject to inservice inspection. NYPA stated that no credit is taken for the cross-tie in any accident analysis and that it is therefore not required to be QA category I. Further inspector review substantiated this conclusion, however, the inspector noted that the FSAR could be more clear in discriminating between actual safety design basis of systems and additional design features or capabilities. The licensee acknowledged the inspector's concern. This issue has been turned over to the corporate licensing group for further evaluation.

3.0 MAINTENANCE (62703,61726)

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 activities were observed:

- Work Request (WR) 93-2913 and WR 93-2912 issued to control the replacement of 03CRD-90B and 89B per modification MI-90-201 and protective tag 93-1866, on October 28.

- WR 93-365-00, Fit-up and welding of a new flange assembly downstream of 46 MOV 102B, on October 29.

- Jumper 93-133, initiated to support temporary installation of a corrosion monitoring rack on the reactor building closed loop cooling system (reference JAF-SE-92-178), on October 29.

- Various aspects of the installation of modification F1-93-075, Reactor vessel water level backfill modification, performed during the October 24 to November 20 maintenance outage.

- Various aspects of the replacement and installation of new piping per modification F1-90-171 and F1-89-066, Replacement of service water piping to/from the reactor building closed loop cooling heat exchanges, performed during the maintenance outage.

The inspector noted in review of the field work performed per F1-90-171, Installation Procedure (IP) #1, that a number of procedural steps in section 8.1, Prefabrication, and section 8.2, Field Installation, were not signed off as completed. Followup with the construction services and site engineering staffs determined that these prefabrication steps (work to be performed in the shop prior to field work), had some element that could not be completed until work was completed in the field. Similarly, a section 8.2 step had multiple elements that precluded signoff until scaffolding was completely removed and new rigging brackets painted. Although the inspector could not identify any specific problems with the installation processes for this modification, the inspector concluded that the installation procedural controls have a high potential for human error in that a given activity could be inadvertently missed. In addition, multiple activities on one step may limit the quality control and auditability of the process. The FitzPatrick Construction Services and Site Engineering staffs acknowledged this observation and indicated that the installation procedural control process was being reviewed for improvement and that the inspector's comments would be considered in the revision.

- WR 50651, Service water system screenwash isolation valve replacement.

- WR 93-2219, Replace seal on A turbine building closed loop cooling pump, on October 22.

- WR 93-0854, Welding of support for reactor vessel level instrumentation modification.

Other than noted, no concerns were identified during inspector review of the above activities.

3.1.1 Maintenance Observation Summary Findings

Though the vast majority of all work done during the outage was performed safely, correctly, and in accordance with procedures, there were a number of instances where performance deficiencies were identified.

On October 28, work was performed with unapproved work instructions and without required Quality Assurance (QA) inspections. The work in question was one of five packages performed to replace valve handwheels in the crescent area. Three of the five jobs were on category I designated valves and required QA inspections. Two of the three category I jobs were properly reviewed. However, the third was not reviewed by the maintenance supervisor or the QA inspector as required due to an inadvertent oversight by the individuals. This was contrary to Administrative Procedure 10.01, Problem Identification and Work Control and is a third example of procedural noncompliance. **(93-24-01)**

On October 28, during the performance of Instrument Maintenance Procedure 12.6, Reactor Water Cleanup System High Temperature (12 TIS-99) Test/Calibration, 12 MOV 69, the reactor water cleanup (RWCU) system return line containment isolation valve, was inadvertently isolated. While restoring from the surveillance test, the I&C technicians erroneously marked step 9.2.38 (which resets primary containment isolation signals), as not applicable. Additionally, the technicians did not fully complete step 9.2.39, which verifies the isolation signal is reset, before requesting the operations staff to reclose the breaker for 12 MOV 69. When the breaker was shut and the isolation signal remained present, the valve shut. Reactor water level was being controlled through the RWCU system at the time of the valve closure, and prompt operator response prevented a level transient. This event is a fourth example of procedural non-compliance and is a violation of Technical Specification 6.8.1. **(93-24-01)**

On November 3, the CO_2 system was inadvertently charged when a painter pulled the latch pin and moved a manual CO_2 hose nozzle from its wall mount. The painter acted without authorization and removed the CO_2 nozzle to allow sufficient space for him to move his scaffold cart through. The control room operators received an alarm when the system charged. Control room operators responded properly and no CO_2 was actually discharged.

Problems were also experienced with the protective tagging process. On November 2, a switchyard breaker that was tagged in the shut position to provide ground continuity to an adjacent work site, was inadvertently tripped open in violation of the tag out. Additionally, on November 12, the wrong control power circuit breaker was tagged open to provide isolation for work on the 345 KV disconnect 10031.

Conclusion

As stated in the preceding paragraphs, a number of poor personnel performance events occurred during this period of high maintenance activity. Many activities were properly planned and executed. However, four specific events involving procedural noncompliance have been cited as violations of regulatory requirements. Additional NYPA management attention is warranted to reverse this declining performance trend.

3.1.2 Traversing In-Core Probe (TIP) Over-withdrawal

On November 3, NYPA discovered that TIP detectors A and B were found retracted from their normal in-shield housed position of -1 inch to a position of greater than -400 inches at the TIP drive units outside of the TIP room. Due to NYPA's use of titanium gamma detectors rather than fission chambers, no significant radiation exposures resulted from this event. The discrepancy was identified when I&C was preparing to repair a threaded connection on the C TIP tubing. During the previous day, the A and B TIP ball valves had been replaced following a local leak rate testing (LLRT) failure. The TIP ball valves were subsequently retested satisfactorily. The time or cause of the TIP retraction was not readily apparent and NYPA convened a root cause team to investigate.

Through interviews, document reviews, and field testing (event recreation) the team was able to determine a plausible scenario. When the TIP central control unit (CCU) is deenergized, it loses its resident program and assumes default values for the TIP position (i.e., 0 inches). When the CCU is subsequently reenergized, it reboots and performs operational diagnostic checks. This process takes approximately ten minutes. While the system is rebooting, it senses the TIPs at the 0 inch position, vice the desired -1 inch position, and demands a retraction of the TIPs. Thus, when the protective tagging request (PTR) for replacing the ball valves was cleared and the CCU powered up, the A and B TIP withdrew to the drive units. The C TIP did not withdraw because the manual "on/off" switch remained tagged in the "on" position per the LLRT. When in the "on" position, this switch disables the drive unit and allows for manual handcranking of the TIPs. Historically, the manual handcrank and the manual switch have been tagged in addition to the TIP master power supply, but for the ball valve replacement only the power supply was tagged. Overall, the event was caused by inadequate understanding of the system. Additionally, there were missed opportunities to identify the overwithdrawn condition during the post work testing of the A and B ball valves. However, once the condition was discovered, NYPA took actions to assess the radiological consequences and performed a thorough root cause evaluation.

3.1.3 Use of Teflon Tape

During the conduct of plant tours this inspection period, the inspector noted the broad application of teflon tape on threaded connections in the control rod drive system. Because of the potential for non-compatibility with stainless steel and the potential for introduction into close tolerance fluid systems, the inspector reviewed the acceptability of teflon tape use and the plant controls, in place, to prevent is misapplication. Initial response by the FitzPatrick staff identified there were no specific administrative controls for the use of teflon tape. However, a 1987 Technical Services Department memorandum (JTS-87-0519) did address the use of teflon tape pipe thread sealant on hydraulic control unit piping only and made recommendations controlling its future use.

Having identified essentially no established controls of the present use of teflon tape, the FitzPatrick staff developed a detailed action plan (JCM-93-001) to identify potential uses in the plant, to determine known restrictions for its use, to develop corrective actions to resolve any misapplications, and to establish procedural controls for its future use. The inspector reviewed the results of this action plan with responsible station managers and determined that there was no improper use of teflon tape in safety related systems based upon known application restrictions. Specifically, prohibitions on its use on systems or components subjected to high temperatures and/or high radiation. Both conditions result in material decomposition and the release of fluorides which contribute to austenitic stainless steel intergranular stress corrosion cracking (IGSCC).

Corrective actions to ensure proper future use of teflon tape were still being finalized at the conclusion of the inspection period. However, a revision to Work Activity Control Procedure (WACP)-10.1.13, Chemical Material Control Program, was being drafted, as well as warehouse issuance instructions and training of plant staff to ensure their familiarity with its restrictions. The inspector concluded these actions were appropriate to ensure proper application of teflon tape on safety related systems.

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:

- STP-10AN, In-situ design basis differential pressure test of 10 MOV-89A, on November 9.
- STP-10AP, In-situ design basis differential pressure test of 10 MOV-89B, on October 25.
- ST-4N, HPCI flow rate and inservice test (IST), on November 20.
- ST-9C, Emergency AC power load sequencing test and 4 kV emergency power system voltage relays instrument functional test, on November 19. The surveillance test had been revised to verify the undervoltage tripping of the 10514/10614 breakers. These functions were tested satisfactorily.

- WACP-010.1.34, Hydrostatic testing following repair, replacement, or modification activities, performed for the post-installation leak testing of the modification F1-93-075 test transmitter, on November 5.

- ST-39J, Leak testing of RHR and core spray testable check valves (IST), on October 26.

- ISP 82-1, Standby liquid control system temperature instrument calibration.

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

3.2.1 <u>Emergency Diesel Generator Surveillance Testing</u>

On October 28, 1993, NYPA documented the results of a logic system functional testing adequacy review for the emergency power system. This review identified some potentially significant discrepancies. Note 5 for Technical Specification Table 3.2-2 states that one of the functions of the emergency bus undervoltage timers is to trip the normal/reserve tie breakers in conjunction with 75 percent emergency diesel generator voltages. One of the discrepancies noted was that the logic to trip the bus tie breaker on loss or degraded voltage was not being exercised by surveillance test (ST)-9C, Emergency AC Power load Sequencing Test and 4 kV Emergency Power System Voltage Relays Instrument Functional Test. Specifically, two of the breakers in series connect the safety to the non-safety bus. The ST initiates the loss of voltage condition by opening one of these tie breakers (10304/10404). Opening one of these two breakers automatically trips its companion tie breaker (10514/10614) and precludes the demonstration of the loss of voltage trip of the tie breaker. NYPA's initial review of this finding concluded that while not all individual contacts were being tested, all of the actual relays were being tested and that the current testing provided adequate assurance that the circuit would perform its intended function. Also, NYPA decided to revise the ST to test the contacts in question as a procedural enhancement prior to the next scheduled performance of the test.

The inspector questioned whether this approach was adequate. TS table 4.2.2 does not require a logic system functional test for the emergency bus logic. However, it does require an instrument functional test for the emergency bus under-voltage relays and timers. The TS definition of a functional test requires the injection of a signal where possible to verify the proper instrument channel response, alarm, and/or initiating action. The inspector questioned whether the initiated action, in this case the tripping of the tie breakers, was being adequately verified. This issue remained under NYPA and specialist inspector review at the conclusion of this inspection period, and is further addressed as an unresolved issue in report 93-16.

4.0 ENGINEERING (37700,93702,92700,92701)

4.1 Reactor Protection System Hot Short Vulnerability

On October 20, NYPA determined that their reactor protection system (RPS) was vulnerable to a postulated hot short. Specifically, FitzPatrick has RPS group status lights on three panels in the control room. The status lights all have isolation resistors, but for one of the panels, the isolation resistors are not in a protected enclosure. So a potential short to the upstream side of the unprotected isolation resistor could prevent a scram of the associated group (25% of all rods). Additionally, NYPA determined that the wiring between the scram contactors in a control room panel for rod groups 3 and 4 was not in conduit (50% of all rods). NYPA performed a modification to resolve these discrepancies prior to restarting the plant. The inspector had no unresolved questions and this issue was further reviewed in specialist inspection report 93-16.

4.2 Information Notice 93-79 Followup

Information Notice (IN) 93-79, Core Shroud Cracking at Beltline Region Welds in Boiling-Water Reactors, was issued by the NRC on September 30, 1993. As the title states, this IN alerted boiling-water reactor (BWR) utility owners of the potential for intergranular stress-corrosion cracking developing in core shroud beltline welds. The most recent discovery of this problem was at Brunswick Unit 1 (Carolina Power and Light Company) where the cracking was identified during an in-vessel visual inspection conducted during a planned refueling outage.

Followup by the inspector determined that NYPA was aware of the Brunswick Unit 1 discovery prior to the issuance of IN 93-79 via General Electric and the BWR Owner's Group. NYPA had reviewed a video-taped vessel internals inspection conducted during the previous refueling outage. Although the video-taped internals inspection did examine the core shroud, it did not concentrate on the areas of concern and was of insufficient resolution to identify the types of surface cracking first identified by the Brunswick staff. The NYPA engineering and non-destructive inspection staffs have deemed this inspection inconclusive. Consequently, NYPA plans to conduct a detailed inspection of the core shroud during the next refueling outage (currently planned for January 1995).

The inspector learned that the FitzPatrick core shroud manufacturer is the same (Sun Ship and Dry Dock, Inc.) as the Brunswick Unit 1 and 2 core shrouds. In addition, the time period of fabrication and material composition (304 stainless steel) are similar. Both General Electric and the NRC staff continue to evaluate this potentially generic core shroud cracking concern. As stated in the IN, no immediate safety concern or specific action on the part of BWR owners is required at this time.

4.3 Review of Scram Discharge Volume Pipe Welds

On November 19, at the request of the NRC staff, NYPA conducted a comprehensive review of all pipe welds in the scram discharge volume (SDV) and scram discharge instrument volume (SDIV) to verify the adequacy of these pipe welds. A detailed action plan was developed and implemented to review the SDV and SDIV pipe welds and related welding activities. At the conclusion of the work day, NYPA had completed the action plan and concluded that the SDV and SDIV, and associated pipe welds were properly installed and non-destructively examined per industry codes and standards.

The inspector monitored NYPA's followup of this issue and reviewed the action plan and its results. The action plan, in addition to reviewing all weld records for SDV and SDIV pipe welds, reviewed the following: all safety related welding conducted during the 1993 maintenance outage; non-destructive examination (NDE) records for a 1983 modification (Mod. No. 82-18) of the SDV and SDIV; all AQCRs, DERs, and known weld related deficiencies back to 1982 on the control rod drive system; hydrostatic testing results associated with inservice inspection program and modification No. 82-18; and interviews with responsible supervisors and inspectors involved with welding activities. The inspector noted that even though the SDV and SDIV is ASME Code Class 2/3 piping, the modification 82-18 installation procedures required 100% NDE of all welds on the system. The NDE included 100% magnetic particle (MT) and dye penetrant testing (PT) of all socket welds and 100% radiographic examination (RT) of all eight and ten-inch diameter piping butt welds. All of these NDE records were previously examined by certified ASME Code Level III NYPA inspectors earlier this year and a sampling (five) were reviewed during an earlier NRC inspection (reference inspection report 93-06, section 7.2). No problems were identified at that time. As stated above, the balance of the action plan items identified no additional welding concerns.

The inspector concluded that NYPA's action plan to review the SDV and SDIV welding activities was comprehensive and thoroughly executed. The NDE records clearly demonstrated the adequacy of these pipe welds, and system testing supports continued re-verification of system integrity. The inspector noted that NYPA will submit the results of their review to the NRC in separate correspondence. The inspector had no further questions or concerns at this time, pending review of NYPA's letter.

4.4 Control Room Ventilation Update (URI 93-14-03)

On October 30, NYPA made a non-emergency notification to the NRC to report that control cables for both trains of the emergency supply fans for control room ventilation are located within the same conduit. Additionally, several of the safety related ventilation dampers were found to be supplied by non-safety related cables and/or non-safety related power. Compensatory measures have been taken with the ventilation system to allow restart. The control room ventilation system has been tagged in the isolate position (one emergency supply fan running, normal intake and exhaust MOVs and dampers closed) and the emergency make-up and modulating dampers have been failed in their safety function positions. This places the system in the emergency lineup and since no damper actuations are required, supplying the dampers with non-safety cables and/or power does not pose an operability problem. The control cable separation issue concerning the emergency fans was a result of the wiring configuration for the auto start circuits that cause the second fan to start if the first fan fails. To correct the cable separation deficiency, NYPA implemented a minor modification to disable the auto start function and disconnected the cables. To compensate for the loss of the auto start function, NYPA now procedurally requires starting the second fan in an emergency so that the auto start function will not be necessary.

In order to justify restarting the plant with all of the outstanding identified ventilation system deficiencies, NYPA performed a reasonable assurance of safety evaluation. The inspector reviewed this document and concluded that in spite of the significant number of deficiencies, the system was capable of performing its safety function. In order to consolidate and address the various issues associated with the ventilation system, NYPA created a dedicated project team. Three technical services department engineers with no additional duties are assigned to the team. They are augmented by two ventilation specialist contractors and received considerable support from the corporate office in the areas of licensing and generating a design basis document. NYPA appears to have bounded the problem, but considerable effort is still required to resolve the various issues. This item will remain open pending full restoration of the system.

4.5 Modification Review

The inspectors conducted a detailed review of modification F1-93-075, Reactor Vessel Water Level Backfill Modification, including: walkdown of new hardware; observation of installation work activities; review and assessment of the safety evaluation; review of the installation control procedures; and review of the preoperational test. This modification was mandated by the NRC per NRC Bulletin 93-03 which identified the potential for inaccuracies in the reactor vessel water level monitoring/indication systems at boiling water reactors (BWRs). The accuracy of these level monitoring systems has been challenged by the potential presence of non-condensible gases dissolved in the reference leg of BWR level instrumentation. During vessel depressurization these dissolved gases, if present, come out of solution resulting in false high level indication (possibly a step increase or notching) as the gas bubble migrates up the reference leg piping.

On several different occasions the inspectors witnessed modification hardware installation activities during the outage. No specific deficiencies were directly observed by the inspectors. One problem with system installation was identified by the FitzPatrick staff which involved the incorrect installation of a test transmitter. This discrepancy was identified and corrected prior to placing the transmitter in service. The apparent cause for this mistake was inattention to detail and insufficient self-checking. A second problem involved the improper removal of protective tags and this event is discussed in section 3.1 of this report. Generally, control over the various stages of modification installation was good.

Inspector review of the nuclear safety evaluation (JAF-SE-93-072, revision 1) supporting this modification identified no significant problems. However, a few aspects of the safety evaluation were not clear and the NYPA staff agreed further clarification was warranted to support the technical adequacy of the evaluation. The most significant clarification to the safety evaluation involved the adequacy of the thermal-hydraulic stress analysis summary. This inspector observation was also shared by the Plant Operations Review Committee (PORC) who conditionally approved the safety evaluation for hardware installation, but maintained a PORC open item to review the final thermal-hydraulic stress analysis when completed.

The final thermal-hydraulic stress analysis concluded that the installed condensing chamber piping configuration satisfies code (ANSI B31.1) requirements. However, a 1992 ASME Boiler and Pressure Vessel Code, section III, fatigue evaluation per NB-3653.1 through NB-3653.7 demonstrated that due to the as-built configuration of the vessel nozzle N-12A steam leg piping to the 2A and 3A condensing chambers, the fatigue life of the N-12A pipe tee (a vertical rise to the 2A condensing chamber, from a near horizontal pipe run to the 3A condensing chamber) has a prorated fatigue life of 5.4 years. Consequently, start-up and operation with this piping configuration is satisfactory, but a modification to the nozzle N-12A pipe tee is required for continued operation beyond the next scheduled refueling outage. In consultation with the NRC Region I and NRR staff, the inspector found the resolution of this thermal-hydraulic stress concern appropriate. NYPA's pursuit of this issue was noteworthy, in that, reactor vessel level instrumentation piping construction code (ANSI B31.1) did not require this type of stress analysis, but it proved to be prudent to do so.

The inspectors reviewed the adequacy of the modification preoperational test and various aspects of its implementation. No concerns were identified.

In summary, the inspectors found the design, installation, and testing of this modification to be appropriate. Aside from the thermal-hydraulic stress analysis issue discussed above and the editorial clarifications made by the NYPA staff to the safety evaluation, the overall engineering support and control of this modification were appropriate.

5.0 PLANT SUPPORT (64704,71707,83750,40500)

5.1 Radiological Controls

On November 11, the inspectors inspected all levels of the drywell accompanied by a radiation protection technician. General material condition and housekeeping was satisfactory. The majority of drywell work activities were completed, but final cleanup and closeout inspections by the plant staff had not yet been performed. Some miscellaneous items (flashlights, paperwork, anti-contamination clothing articles, power cables and tubing, etc.) remained to be removed from the drywell and the equipment hatch reinstalled. Radiation dose rates and contamination levels encountered during the inspections were considered normal as expected. The inspectors were provided with a thorough briefing of radiological conditions by the responsible radiation protection supervisor prior to entry and the technician accompanying the inspector practiced good ALARA and contamination control techniques. The inspectors identified no radiological or safety problems.

5.2 Security

5.2.1 Fitness for Duty Event Review

On November 17, the inspector was informed by NYPA management that a non-reportable fitness for duty (FFD) event (reference 10 CFR 26.73) had occurred. A station employee reporting to work (selected at random) tested 0.025 percent blood alcohol concentration (10 CFR 26 and station policy limit is 0.04 percent) by alcohol breath test. The individual admitted to consuming a beer during the pre-work five-hour abstinence period which is contrary to 10 CFR 26.20 guidance and the FitzPatrick FFD Policy. The individual was denied access to the facility for work on the mid-shift crew. Inspector followup identified the following actions were taken or planned by NYPA management: the individual's site access badge was pulled, pending supervisor and management interviews with the individual; retest of the individual to determine FFD and return to work; and broad dissemination of the facts and corrective actions associated with this event to ensure all plant staff are reminded of the FFD policy and their obligation to abide by it. Based upon the actions described above and the fact that the NYPA FFD Program identified this individual's failure to meet the FFD Policy, the inspector concluded NYPA's actions were appropriate and the FFD Program was being adequately implemented.

5.3 Fire Protection

During a review of fire protection Technical Specifications, the inspector noted that the once per three year "flow/hydrostatic test" listed in surveillance Table 4.12.3 was not captured in any operations surveillance tests. Further review revealed that the hydrostatic test was being properly conducted by maintenance surveillance test (ST) 76.9, Fire Hose Inspection and Hydrostatic Test, but that no flow test was being conducted. NYPA's initial response was that no flow testing was required because the surveillance test program was based on the National Fire Protection Association (NFPA) codes and NFPA does not require hose station flow testing. Inspector review found that both General Electric and Westinghouse standard Technical Specifications have required both a station flow test and a hose hydrostatic test every three years. The inspector then concluded that FitzPatrick's "flow/hydrostatic test" requirement would require a station flow test despite the lack of guidance in the NFPA codes. This information was provided to the licensee and after several days of review, on November 16, NYPA concluded that the flow test was required. The licensee then generated and performed the required surveillance test satisfactorily. The failure to perform flow tests at fire hose stations is a violation of Technical Specification 4.12.3. **(VIO 93-24-02)**

6.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (40500)

6.1 Review of ISEG Function

To support a request by the NRC headquarters staff to determine to what extent an independent safety engineering group (ISEG) or ISEG-like function is provided by the licensee, a review of the recently established nuclear programs assessment section (NPAS) was conducted. Historically, FitzPatrick has not had an ISEG (as defined in Technical Specifications for more recently licensed nuclear facilities) or an ISEG-like function. As of July 1993, a senior nuclear assessment engineer provides this function at FitzPatrick. This senior engineer independently (outside of the site line management and Quality Assurance Organizations) reviews and monitors activities at the FitzPatrick site and reports directly to the Vice President Nuclear Operations and Maintenance. This individual has conducted reviews alone, to date, but has authority to supplement his efforts with outside consultants, as warranted.

The individual serving as the senior nuclear assessment engineer brings considerable operations experience with him as he was a former Resident Manager and long-time staff member and supervisor at FitzPatrick. Based upon a review of the monthly reports issued to date (two) and discussions with site managers, the observations and recommendations provided in these reports have been of high quality, objective, focused on a diverse and appropriate cross-section of in-plant activities, and appear to be well received. Evidence shows that NYPA management is initiating some of the recommendations contained in these reports. A mechanism is available to site line management to request specific reviews by the senior nuclear assessment engineer. Based upon reports issued, to date, this mechanism has not been exercised.

In summary, an ISEG-like function has been established at the FitzPatrick facility via the NPAS (one individual) and has been functioning for a few months. The inspector's assessment of the NPAS's independent review, evaluation ability and successes is based upon limited tangible evidence, but appears to be a good initiative by NYPA management. The long term success of this section can not yet be determined at this time.

6.2 Safety Stand-down

On November 5, FitzPatrick management conducted a one-hour safety stand-down. The stand-down consisted of a discussion period between department managers and supervisors with their staffs to reemphasize the need for attention to detail and self-verification. The safety stand-down was prompted by a station management assessment and demonstrated need to reexamine these areas based upon a recent increase in human performance related events. The one-hour shop discussions were facilitated by a handout which summarized thirteen recent events involving equipment and material use, worker preparation, and error detection. The handout also provided copies of Plant Standard (STD)-2.160, Protection of Plant Equipment, STD-2.600, Attention to Detail, and STD-2.800, Self-Verification, to each station worker.

In light of the recent human performance problems encountered during the maintenance outages, the inspectors considered this safety stand-down a good initiative by NYPA station management. It clearly demonstrated that station management was trending human performance related events and that a threshold had been met to take corrective action before a more safety significant event occurred. However, based upon a few events subsequent to this stand-down (reference sections 2.1.2 and 2.1.3) the effectiveness of this effort was questionable. Further attention to human performance improvement initiatives appears to be warranted. In response, the licensee developed a comprehensive action plan to address this issue.

6.3 QA Exit

On October 28, the inspector attended a QA exit of a Technical Specification required audit of plant staff qualifications. No significant findings were discussed, but the level of detail of the three recommendations indicated that a thorough review had been conducted.

7.0 REVIEW OF WRITTEN REPORTS (92700,90712,90713)

7.1 LER Review

The inspectors reviewed the following Licensee Event Reports (LERs) and found them to be well written, concise, accurate, and properly submitted for NRC staff review within the guidelines of 10 CFR 50.73:

- LER 93-08-01, Incomplete functional testing of carbon dioxide fire suppression system, Supplement 1. LER 93-18 and this event were previously reviewed in inspection reports 93-17 and 93-20.
- LER 93-19, Potential design inadequacies in the control room ventilation system. This issue was reviewed in inspection report 93-20, section 4.1.
- LER 93-021, Motor operator valve failure due to inadequate brake design.
- LER 93-020, Reactor high pressure scram due to turbine bypass valve partial closure.

The inspector identified no additional concerns or problems with NYPA's response to these events.

8.0 MANAGEMENT MEETINGS (30702,71707)

8.1 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.