U.S. NUCLEAR REGULATORY COMMISSION

Region I

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| License No.: | DPR-59 |
| Licensee: | New York Power Authority<br>P.O. Box 41<br>Lycoming, New York 13093 |
| Facility: | James A. FitzPatrick Nuclear Power Plant |
| Location: | Scriba, New York |
| Dates: | April 18, 1993 through May 22, 1993 |
| Inspectors: | W. Cook, Senior Resident Inspector<br>J. Tappert, Resident Inspector<br>P. Drysdale, Senior Reactor Engineer<br>L. Scholl, Reactor Engineer |

Approved by: Richard J. Urban 6/18/93

for Peter W. Eselgroth, Chief, Reactor Projects Section 1B, DRP — Date

**INSPECTION SUMMARY:** Routine NRC resident inspection of plant operations, maintenance, engineering and technical 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**

**NRC Region I Inspection Report No. 50-333/93-10**

**04/18/93 - 05/22/93**

### Plant Operations

During this inspection period, power operations were interrupted by a low reactor vessel level scram on April 20, and a shutdown to repair the HPCI injection check valve on May 18. As a result of the low vessel level scram review, the inspectors determined that effective action had not been taken to preclude recurring feedwater check valve failures. Additionally, the inspectors identified a need for further review of two issues noted in the post-trip review related to excessive cool down and heat up rates and inadequate logic testing of the HPCI pump discharge valve **(URI 93-10-01)**. On May 19, while initiating shutdown cooling, the plant experienced a spurious high reactor pressure isolation of the shutdown cooling system. This is the third automatic isolation this year, and this issue remains unresolved pending further NYPA and NRC review **(URI 93-10-02)**. A thorough review of the icing events in February and March 1993 was conducted and NYPA actions were determined to have been appropriate. NYPA's self-identification of a missed fire protection commitment resulted in a non-cited violation.

### Maintenance

During the two outages this period, numerous maintenance tasks were worked. Overall, maintenance activities were well supervised and executed. Specifically, troubleshooting and repair of the feedwater master controller was well controlled and performed. Contingency planning for the HPCI check valve steam leak included the formation of a multi-disciplined team to identify and assess various options. The entire repair effort was thoroughly planned and professionally conducted. Repair of the RCIC turbine inlet isolation valve was professionally conducted and the system outage was utilized to weld repair a previously identified defect.

### Engineering and Technical Support

NYPA's root cause analysis for the leaking recirculation jet pump riser decontamination connection was reviewed. The root cause analysis was generally acceptable, however, the inspector was concerned by a finding that indicated an engineering walkdown was signed off without being performed. Subsequent review determined that a post-installation modification walkdown was performed, but that it was inadequate. The root cause analysis will be revised with this additional information and further corrective actions will be taken.

Safety Assessment/Quality Verification

NYPA has implemented a 24-hour outage manager position. This new position has resulted in better interdepartmental communications and improved work scheduling and execution. A special inspection was conducted to review NYPA's operator training in regard to potential vessel level errors introduced following a depressurization transient. The inspector concluded that NYPA was making operators aware of the potential effects of non-condensible gases, but that additional training and procedural guidance may be warranted after the BWR Owners Group testing results are available.

## DETAILS

### 1.0 SUMMARY OF FACILITY ACTIVITIES

#### 1.1 NYPA Activities

At the beginning of the inspection period, the reactor was operating at 100% power. On April 20, the plant experienced a reactor feed pump and vessel level transient that resulted in a reactor low level scram and emergency core cooling system actuations. After correcting the mechanical problems which led to the scram, NYPA restarted the unit on April 26 and achieved 100% power on May 3. On May 11, steam was observed issuing from the HPCI injection check valve. After monitoring the leakage and exploring various options, NYPA commenced a reactor shutdown on May 18 to repair the valve. The unit remained shut down through the end of the inspection period.

#### 1.2 NRC Activities

The inspection activities during this report period included inspection during normal, backshift and weekend hours by the resident staff. There were 14 hours of backshift (evening shift) and 15 hours of deep backshift (weekend, holiday and midnight shift) inspections during this period.

During the week of May 10, 1993, a region based inspector reviewed NYPA's training of operators regarding potential reactor vessel level errors resulting from depressurization transients. The inspector's observations are documented in section 5.3 of this report.

### 2.0 PLANT OPERATIONS (71707,62703,61726)

#### 2.1 Routine Plant Operations Review

During the inspection period the inspectors observed control room activities including operator shift turnovers, shift crew briefings, panel manipulations and alarm response, and routine safety system and auxiliary system operations conducted in accordance with approved operating procedures and administrative guidelines. The inspectors made independent verifications of safety system operability by review of operator logs, system markups, control panel walkdowns and component status verifications in the field. Discussions were held with operators and technicians in the field to assess their familiarity with current system status and personnel response to events during the inspection period. In addition, during plant tours, inspectors reviewed routine radiological control practices. The activities inspected were acceptable.

### 2.1.1 Operational Safety Verification

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

-- A and C emergency diesel generators

-- A and B standby liquid control

-- Reactor core isolation cooling

-- High pressure coolant injection

No significant discrepancies were noted.

### 2.1.2 Reactor Startup

The inspectors witnessed various aspects of the reactor startup commenced on April 26. The mode switch was placed in STARTUP at 2:32 a.m. and the reactor reached criticality at 5:10 a.m. Satisfactory testing of the high pressure coolant injection system, per ST-4N, was observed prior to the unit exceeding 150 psig.

During heat-up of the reactor, the K safety relief valve (SRV) tail piece temperature was observed to be higher than the other SRV tail piece temperatures. The inspector noted that the station staff drafted and carefully executed an action plan to address this issue. The action plan included a detailed walkdown of the K SRV and associated piping in the drywell and stroking the valve with reactor pressure at 940 psig. The valve stroke apparently cleared what was on the valve seating surface and the tail piece temperature came down within the range of the other SRV tail piece temperatures. The inspector verified that the K SRV was stroked in accordance with ST-22B. The inspector concluded that this operational issue was appropriately addressed by the station staff.

Following satisfactory resolution of the K SRV tail piece temperature issue and reactor feedwater pump thrust bearing wear detector replacements, the unit was synchronized with the grid at 11:11 a.m. on April 29. The reactor achieved 100% power on May 3, 1993.

## 2.2 Review of Daily Surveillance and Instrument Check

On May 7, the inspector reviewed the Daily Surveillance and Instrument Check, ST-40D, maintained by the control room and auxiliary operators. The inspector noted that all of the out-of-specified range instrument readings were properly red-circled, but few of them were explained in the remarks section of the surveillance sheets. The inspector questioned the shift supervisor (SS) on the out-of-specification readings that were not easily accounted for. The SS satisfied the inspector's questions and stated that the surveillance sheets would be properly annotated in the remarks section for those out-of-specification readings. The inspector considered the SS's response to this performance observation appropriate. During subsequent reviews of ST-40D during this inspection period, the inspector noted good annotation of red-circled readings on the surveillance sheets.

2.3 Low Reactor Water Level Scram

On April 20, 1993, the plant was operating at 100% power and at nominal reactor vessel level of 202 inches. At 4:54 p.m., a loose electrical connection on the A reactor feed pump (RFP) turbine control governor caused a reduction in speed in the A RFP with a corresponding lowering of reactor vessel level. In response to the low reactor water level alarm, plant operators took manual control of the reactor feed pumps. However, operators were unable to recover the level due to a gross internal failure of the A RFP discharge check valve. A portion of the B RFP flow went through the failed check valve and the A RFP minimum flow valve, which had opened as a result of the A RFP stalling, to the main condenser. This diversion of flow caused reactor level to continue to decrease. A low level reactor scram was received at 177 inches. 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 128.4 inches. With HPCI and RCIC injecting, water level rapidly increased until the RCIC, HPCI, 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 operators response to the transient was appropriate and within the bounds of the plant procedures. The inspector reviewed various aspects of the post-trip review process and found that it was improved since issuance of a violation in April 1993. Several issues were raised as a result of NYPA's post-trip review. Due to turbulent flow from the discharge of the RFPs, the discharge check valves (34 FWS-4A and 4B) have failed ten times over the life of the plant. The majority of these failures were by 34 FWS-4A due to more turbulent flow through this valve. A number of design enhancements were made to the valve over the years to increase hanger bolt strength, prevent hanger bolt loosening, and prevent disc contact with the valve body in the open position. These modifications achieved only limited success.

The inspector determined that a January 1991 engineering memorandum recommended replacement of these valves during the next refueling outage with check valves specifically designed for turbulent flow. Although this proposal had been presented at recent modification prioritization meetings, it had not been scheduled. In 1990, a preventive maintenance (PM) program for the feedwater discharge check valves was created. Although the mean time between failures for 34 FWS-4A was 25 months, the PM interval for inspection of the valve was set at two years. This was thought to be conservative since recent design enhancements were expected to increase the valve's service life. Valve 34

FWS-4A was last inspected in May 1992, but this inspection was unable to predict or prevent the failure in April 1993. Overall although NYPA has done a good job analyzing and evaluating the failure mechanisms, effective action has not been taken to prevent recurring valve failures. In response to this event, the maintenance department is reevaluating their short and long term corrective actions to prevent future failures.

There were other significant issues identified in NYPA's post-trip review. specifically, concerns were raised regarding the reactor vessel cool down and heat up rates in the reactor bottom head area concerns were raised regarding the reactor vessel during the scram recovery and with the logic testing of 23 MOV 19, the HPCI pump discharge isolation valve. LER 93-09, which discusses these issues, was issued at the end of this inspection period and has not been adequately assessed by the inspectors. Therefore, these issues will be reviewed in a future inspection report and are an unresolved item. **(URI 93-10-01)**

2.4 Shutdown Cooling Isolation

At 4:25 p.m. on May 19, 1993, the unit operators were conducting a normal reactor shut down and cool down. At a reactor pressure of one 1 psig, operators were attempting to initiate shutdown cooling. When the B residual heat removal (RHR) pump was started, valves 10 MOV 17 and 18, the inboard and outboard suction isolation valves, automatically closed on a high reactor pressure (>75 psig) isolation signal. The high pressure isolation signal was promptly reset. Shutdown cooling was initiated without incident on the second attempt. Initial indications, based on a drop in reactor vessel level, were that the suction line may not have been completely filled. Collapsing of the void upon start of the RHR pump most likely caused the pressure spike. This is the third isolation of shutdown cooling during system initiation this year. This last incident occurred despite the fact that the procedure had been changed to require filling the suction piping prior to initiation. NYPA continues to investigate the cause of these isolations and formulate appropriate corrective actions to prevent recurrence. An unresolved item has been assigned pending the completion of NYPA's root cause analysis and review by the inspector. **(URI 93-10-02)**

2.5 Operating Events Followup

2.5.1 Ice Accumulation and Blockage in the Circulating Water Intake Structure and Screenwell

Background

NRC inspection report 50-333/93-04 documented an initial follow-up to a plant shutdown on February 25, 1993, after an apparent ice blockage at the entrance of the circulating water system (CWS) intake in Lake Ontario caused a restriction in flow to the screenwell structure. The screenwell structure provides cooling water for the CWS, the normal service water system (SWS), the emergency service water (ESW) system, the residual heat removal service water (RHRSW) system, and the plant fire protection system. Subsequent to the first icing event, two other similar ice blockage events occurred on March 13 and 22, 1993.

February 25, 1993 Event

Early on the morning of February 25 during extremely cold weather (15°F, 15 mph winds) with the plant operating at 100% power, operators noticed an increase in screenwell temperature, increased CWS pump operating current, and instances of auto starts of the fire pumps. An operator was dispatched to the screenwell structure and reported that the water level was 10 feet below normal. The plant was manually scrammed. See NRC inspection report 50-333/93-04 for additional detail.

March 13, 1993 Event

On March 13 with the plant shut down, operators were running two CWS pumps in anticipation of an early return to power operations. There were extreme and widespread storm conditions through the Northeastern U.S. that resulted in gale-force winds over Lake Ontario. These conditions produced up to 15 foot waves on the lake. Floating ice on the lake was pulled into the intake structure and accumulated on the trash racks upstream of the travelling screens. When the high differential pressure (d/p) alarm for the travelling screens was received in the control room, operators were dispatched to the pump house to investigate. Operators found a 6 inch water level difference across the travelling screens. Since tempering flow was not in operation, a large ice buildup occurred on the trash racks that the screen wash could not remove. Operators used the installed rakes to break up the ice. After the CWS pumps were secured and intake flow was reduced, the remaining ice at the traveling screens was removed by the screen wash.

March 22, 1993 Event

On the night of March 22, the plant was starting up, power was less than 10%, and operators were running two CWS pumps. The air temperature in the vicinity of the unit was extremely low and floating ice on the lake was drawn into the screenwell structure and again accumulated on the trash racks upstream of the travelling screens. At about 11:00 pm, operators received a high d/p alarm for the travelling screens. Operators reduced CWS flow by securing one CWS pump to alleviate the high d/p condition. Tempering flow was in operation but was not sufficient to prevent ice accumulation in the screenwell structure.

Event Comparison

The notable differences in environmental and plant conditions prior to each event were as follows:

February 25: The plant was operating at 100% power; three CWS pumps were operating with a total flow of approximately 370,000 gpm; CWS tempering flow (partial recirculation) was in operation; relatively stable weather conditions persisted for several days prior to the event.

March 13: The plant was shutdown; two CWS pumps were operating with a total flow of approximately 240,000 gpm; CWS tempering flow was not in operation; significant snowfall and severe weather conditions existed for several hours prior to the event.

March 22: The plant was starting up, with power less than 10%; two CWS pumps were operating with a total flow of approximately 370,000 gpm; CWS tempering flow was in operation; temperatures were extremely low.

Some similar environmental conditions that existed for several days prior to each event include: 1) The lake water temperature was approximately 33°F; 2) Ambient air temperature was continuously below freezing; and 3) High thermal losses from the lake surface occurred each night. NYPA concluded after a review of meteorological data, that these environmental conditions were appropriate for the formation of large masses of "frazil" and "grease" ice. On February 15, the blockage occurred at the intake entrance in the lake, whereas, on March 13 and 22, the blockage occurred in the screenwell structure at the traveling screens. The March 13 event was exacerbated by the additional presence of pack ice, slush, and snow that accumulated over the CWS intake on the lake surface during the storm.

Effect of Low Screenwell Level

Of the three events, only the February 25 event caused a low water level in the screenwell structure. When this occurs, net positive suction head (NPSH) becomes a concern depending where pump elevations are located in relation to the water level. Operators observed high operating currents on the CWS pumps and concluded the pumps operated for 15 to 30 minutes with suction water level below the minimum required for adequate NPSH. The fire pumps had also run with inlet level below their minimum NPSH. Consequently, an investigation was initiated to evaluate the effects of low water level operation on the CWS and fire pumps. NYPA's evaluation concluded that the screenwell level dropped 10 feet to the 236-237 ft. elevation, but that the water level was maintained above the minimum required for adequate NPSH for all safety-related pumps (235 ft.) in the screenwell. Although the operating current increased in all CWS pumps during the event, all CWS pumps continued to operate normally and the licensee increased the pump monitoring frequency temporarily to assure continued reliability. The fire protection system pumps were unable to maintain their required discharge head due to operation with the pump inlet water level below the NPSH. However, operation of the fire pumps with screenwell level below their NPSH during this event did not cause them to fail. Independent inspector review of the performance data for the SWS pumps, ESW pumps, and RHRSW pumps concluded that they did not experience performance degradation as a result of the low water level.

NYPA was not able to conclusively determine the precise cause of the February 25 intake flow blockage. However, the inspector considered that their analysis and corrective actions were adequate to provide for prudent and safe plant operations or unit shutdown, if conditions warrant. A NYPA consultant indicated in a published report that a large accumulation of frazil ice could not completely block the intake when operating only ESW and RHRSW pumps because the flow velocity would be too low to capture ice and pull it down from the surface to the intake. If any significant ice accumulation occurred at the intake, a resulting high flow velocity would erode the ice enough to permit sufficient flow for minimum shutdown plant ESW and RHRSW needs.

Corrective Actions

NYPA's final root cause analysis and corrective actions to the February 25 event were detailed in a plant Technical Services Department memorandum (JTS-93-0124), dated March 11, 1993. Plant computer points were set to alarm when a 5°F/hr CWS temperature increase occurred in the screenwell intake and the main condenser waterbox to give operators an early warning of ice blockage. Based upon temperature data from the February 25 event, operators would have about 45 minutes to respond before conditions require a plant shutdown. A new Abnormal Operating Procedure (AOP)-64 was written to provide guidance to the operators when the screenwell level drops below normal or when the CWS temperature alarms. Initial actions directed by the procedure are to immediately investigate the situation, reduce plant power, secure one CWS pump, and then observe the water level. If the water level continues to decrease, immediate shutdown of the reactor is required and the second CWS pump was to be secured. NYPA also issued a new surveillance test (ST-8T) to require operators to monitor intake level and temperature when CWS tempering flow is in operation and when lake temperature is ≤33°F. The procedure also requires the Radiological and Environmental Services (RES) department to determine if conditions are favorable for frazil ice formation. If so, the control room operators are to be notified and hourly monitoring of the screenwell level is to be conducted. These actions were taken prior to startup.

Additional corrective action taken included installing water level indicators in the screenwell structure. This can provide operators with a quick reference to the actual level. A permanent modification for a water level instrument in the screenwell structure with control room indication and annunciation was initiated. The licensee plans to install this modification during the Fall 1993 maintenance outage.

In response to the March 13 event, NYPA took action to revise the RES procedure for evaluation of icing conditions. The procedure was revised to require a daily assessment of the potential for frazil ice formation between the period from November 15 to April 15 (when lake temperatures are typically low). In addition, AOP-64 was revised to provide more specific guidance for operators to secure CWS pumps under various plant conditions. Immediate shutdown is required if screenwell level reaches the 240 foot elevation. This guidance was also added to AOP-56 which is used for a high d/p condition at the traveling screens or the trash racks. The CWS operating procedure, OP-4, was also revised to require operation of tempering flow if the intake water temperature drops to ≤34°F (tempering flow will maintain screenwell water temperature at approximately 40°F). Lastly, procedure ST-40D was revised to record lake temperature once per shift and to ensure tempering flow is established if lake temperature is ≤34°F. NYPA also accelerated the design process for the permanent water level instrument in the screenwell structure.

On March 18, the PORC deleted the surveillance procedure which required hourly level readings if frazil ice conditions were present, and directed that a continuous watch be instituted to visually observe the screenwell water level and to record the reading every 15 minutes. The inspector reviewed the intake level watch instructions, issued as Operator Aid No. 582 prior to plant startup, and found them to be appropriate. The inspector noted that the instructions state that any change in water level is to be immediately reported to the control room.

The day after the March 22 icing event, the temperature was expected to reach a record low with a high probability of more frazil ice. NYPA's consultants provided recommendations to reduce CWS flow temporarily, to keep the traveling screens in continuous motion, and to open the tempering gate further in an effort to raise the water temperature slightly in the screenwell.

These additional corrective actions helped prevent ice from clinging to the trash racks so that it could be removed and washed out by the screen wash. No further icing problems were noted.

Conclusions

The inspector reviewed NYPA's long term plans for modifications and the engineering evaluations which model the hydraulics in the CWS flow path. Several alternatives were being considered by NYPA to improve tempering flow and to provide reverse flow capability at power. NYPA also intends to examine the possibility of modifying flow patterns around the nozzle in the lake to reduce its susceptibility to pulling in ice. The inspector considered the above actions adequate for plant startup, for prudent monitoring of ice conditions in the CWS intake, and for taking appropriate actions to assure safe plant operation or shutdown under icing conditions.

2.5.2 Missed Fire Protection Commitment

On May 14, station management informed the inspector that a recently completed review identified that NYPA had not been satisfying their commitment to maintain the smoke detection system operable in the East and West cable tunnels. By NYPA letter dated June 26, 1992, which requested schedular exemptions to 10 CFR 50, Appendix R, NYPA committed to provide interim compensatory actions while the East and West cable tunnel fire suppression systems were being replaced. An NRC letter dated September 10, 1992, granted these Appendix R exemptions. The compensatory actions included: a continuous fire watch in each tunnel; daily walkdowns to ensure transient combustibles are kept to a minimum; backup manual fire suppression will be available via installed hose stations in and adjacent to the tunnel areas; portable carbon dioxide fire extinguishers installed throughout each tunnel; and the existing automatic ionization smoke detection system will remain operable and provide early indication of a fire to the control room operators.

During followup of a malfunctioning smoke detector, the FitzPatrick staff recognized that the surveillance testing on the cable tunnel smoke detection systems had not been performed since the cable tunnel fire suppression systems were removed from service and compensatory firewatches were posted. The original surveillance test covered both the fire suppression and smoke detection systems and the failure to revise or develop a separate surveillance test for the smoke detection systems was an oversight.

The inspector verified that the surveillance test was revised and both East and West cable tunnel smoke detection systems were satisfactorily tested and declared operable per Technical Specifications (TS) on May 15. Since posted firewatches were in both cable tunnels for the time period the smoke detection systems were not TS operable (as required by TS 3.12.E.1.b.) and the detectors were found to be functional when tested, the safety consequence of this event were minimal. Failing to recognize that the smoke detection system was not TS operable within the 14 day limiting condition for operation of TS 3.12.E.2, NYPA did not submit a special report concerning this event to the NRC within 30 days. This is a violation of TS 3.12.E.2. However, as previously stated the safety consequences of this event were minimal, the oversight was identified by the FitzPatrick staff, the corrective actions were prompt and thorough, and this problem was not of a recurring nature. Consequently, the criteria of 10 CFR 2, Appendix C, Section VII.B.2 have been satisfied and this violation is not cited. The discovery of this missed commitment by the FitzPatrick staff was positive. The inspector raised his concerns with NYPA's program to track and ensure compliance with commitments to the NRC to site management during the exit meeting.

2.5.3 Service Water Check Valve Failures

On May 17, during the performance of quarterly surveillance test ST-8R, Emergency Service Water Check Valve and Strainer Test (Inservice Test), operators identified that two check valves (46ESW-40B and 46SWS-67A) failed their seat leakage tests. Both valves were subsequently disassembled, inspected, and cleaned. The failure of both carbon steel check valves to seat properly was attributed to microbiological influenced corrosion (MIC). Small nodules were found on the seating surfaces which inhibited good valve closure against system pressure. The failure of 46ESW-40B was of minor significance. The inspector observed that compensatory actions taken for the failure of 46SWS-67A (service water supply to safety-related unit cooler UC-16A) were well planned and executed. The failure of 46SWS-67A to close had the potential for short-circuiting emergency service water around the unit cooler. Until repairs were affected, operations personnel isolated service water flow to UC-16A and ran the A emergency service water pump to supply the unit cooler. The inspector concluded that the inservice testing was properly performed and the problems encountered appropriately resolved by the station staff. The inspector had no concerns.

## 3.0 MAINTENANCE (IP 62703)

### 3.1 Observation of Maintenance Activities

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 summarized below and in the following sections were observed:

-- The inspector reviewed radiation work permits (RWP #93-156 and #93-159) and protective tagouts associated with the replacement of the 1A reactor water cleanup pump per work request #110721 and modification #F1-90-202. No discrepancies were noted.

-- Portions of the West cable tunnel cooling coil replacement per work request #116914 were observed. No deficiencies were noted.

### 3.2 Reactor Feedwater Pump Oscillations

While operating at full reactor power, the station staff identified that the reactor feedwater pump speeds were oscillating approximately 100 rpm around their normal operating speed of approximately 4000 rpm. Initial troubleshooting identified that the speed oscillations occurred while the feedwater master controller was in automatic. With the individual feedwater pumps in manual control, the speed oscillations dampened out. The inspector learned that the master controller had been replaced during a recent outage and that a controller output gain adjustment would be necessary to remedy this problem.

The operations and instrumentation and controls (I&C) staffs developed a detailed action plan to perform the gain adjustment at full power. The action plan called for a practice run on the simulator with a designated crew of operators and I&C technicians. After successful adjustments on the simulator, the crew performed the gain adjustment on the control panel feedwater master controller on May 7. Several adjustments had to be made before the desired feedwater pump performance was achieved. The evolution was executed without incident. The inspector noted good command and control by the operators involved, good three-point communications, and when an EPIC computer fault was experienced between adjustments, further work was halted by the shift supervisor until the computer problems were resolved. Overall performance by the station staff for the evolution was good.

### 3.3 HPCI Check Valve Body to Bonnet Leakage

A body to bonnet leak was identified on 23 HPI-18, HPCI injection check valve, on March 16, 1993. The leak was stopped by retorquing the bonnet, but reappeared on May 4. The leakage was reduced by hot torquing the bonnet on May 6. On May 11, steam was observed issuing from the valve (hot feedwater flowing back through the HPCI injection line and flashing to steam). The immediate concerns were possible grounding of MOVs in the vicinity of the leak and exceeding the primary system allowable leakage rate (the allowable leak rate was not immediately known). NYPA management immediately formed a multi-disciplined team to identify and assess various options. Initial actions taken were to increase ventilation in the area of the valve and monitor the leakage rate (the initial rate was approximately .75 gpm). An engineering evaluation was done and determined that a leakage rate of up to 5 gpm was within licensing commitments (primary pressure boundary leakage not exceeding 10 CFR Part 100 release limits in the event of a LOCA inside containment). On May 18, with the leak rate approaching 2 gpm, NYPA management decided to commence a shutdown and repair the valve rather than continue to operate with this degraded condition. Upon disassembly of the valve, the mechanical pressure seal was found cocked. NYPA determined that the most probable cause was personnel error while installing the seal during reassembly of the valve by maintenance personnel. The seal was replaced and the valve was successfully local leak rate tested and hydrostatically tested. The entire repair effort was thoroughly planned and professionally conducted. NYPA's approach to resolution of the leak was appropriately conservative at all times.

### 3.4 RCIC Steam Isolation Valve

On May 18, NYPA commenced a forced shutdown to repair the HPCI injection check valve. In addition to the HPCI check valve, approximately 25 priority one work requests (WR) were worked. Among them was WR 116776, repair of 13 MOV 131, RCIC turbine steam inlet isolation valve. The inspectors observed portions of the repair and post work testing. All work was professionally conducted with appropriate levels of supervision. No discrepancies were noted. Additionally, NYPA took advantage of the system outage by conducting a weld repair on the upstream valve to pipe weld on 13 MOV 131. The weld defect was identified during the radiography review conducted in December 1992, but was not worked immediately because the system was not safety related.

## 4.0 ENGINEERING AND TECHNICAL SUPPORT (93702)

### 4.1 Review of NYPA's Root Cause Analysis for Recirculation Jet Pump 'J' Riser Decontamination Connection

Background

During the reactor pressure vessel 1000 psig inspection on March 7, 1993, a leak from the threaded cap downstream of the J recirculation riser one inch tap isolation valve (02-2-RWR 715) was discovered. This isolation valve in conjunction with a threaded end cap was one of ten such configurations that were installed during the 1992 refueling outage for the purpose of providing a means to connect decontamination equipment to the reactor water recirculation piping. Following the discovery of this leak, separate mechanistic and human performance root cause analyses were performed by the FitzPatrick staff.

Inspector Observations

The final human performance root cause analysis was completed on April 26, 1993, as documented by site engineering department memorandum, JSED-93-0330. The inspector reviewed this analysis and discussed the results with the author and responsible station management. The inspector noted that several performance problems were identified, however, one finding was of particular concern to the inspector. Specifically, the failure to perform a final modification walkdown by the responsible engineer. The facts associated with this root cause finding were that "Although a step was signed off in Installation Procedure, IP-1, that a final dedicated walkdown was performed, no evidence validating this was discovered. A search of drywell entry records was performed and no entries were recorded in the time period in question." This aspect of the analysis was discussed with the author and station management on May 13. Station management stated that this was unacceptable performance. This type of work was clearly not their expectation or in accordance with Revision 2 of the Modification Control Manual, MCM-19, Modification Closeout. MCM-19, section 6.1.5 states, in part, that the responsible engineer shall ensure that a field walkdown occurs after modification installation. This walkdown shall provide additional assurance that all affected components are covered by acceptance tests.

Review by the inspector determined the following chronology of events: the new decontamination connections were installed in early February 1992; these new connections were used for the recirculation piping decontamination evolution performed on February 26, 27, and 28; the threaded end caps were installed in early March 1992; the piping thermal insulation was installed in late September 1992; the vessel hydrostatic leak test was performed between September 28 and October 2, 1992; and the responsible engineer signed off for a walkdown of the installation on October 12, 1992. As stated above, during the root cause investigation by the FitzPatrick staff, no evidence could be found that the responsible engineer performed a walkdown during the October 1992 time period.

After discussions with station management on May 13, 1993, to assess the appropriateness of the responsible engineer's October 12, 1992 procedure sign-off, NYPA broadened their review of radiation work permit (RWP) records. They determined that the responsible engineer had made drywell entries under RWP 60.11, Decontamination Connection Modification, on January 23, 1992 and March 26, 1992. Each entry was approximately 30 minutes in duration. When compared with the above chronology, the March 26 entry by the engineer coincides with a post-installation walkdown after removal of the decontamination equipment and installation of the threaded end caps.

Conclusions

The inspector determined that contrary to the April 26, 1993 root cause analysis, it was reasonable to conclude that the responsible engineer performed a post-installation modification walkdown. The adequacy of this walkdown was clearly unsatisfactory, as evidenced by the subsequent discovery of inadequate cap thread engagement (8 of 10 less than .596 inches) and an undocumented elbow fitting on the B riser connection, among other concerns identified by the April 26 root cause analysis. Based upon the above, station management stated the root cause analysis would be revised and additional corrective actions taken. The inspector plans to review this revision to the root cause analysis in conjunction with follow-up of a previously identified violation (**93-06-01**) involving this issue.

## 5.0 SAFETY ASSESSMENT/QUALITY VERIFICATION (71707, 93702)

### 5.1 Review of Licensee Event Reports (LERs) and Special Reports

The following LER was reviewed and found satisfactory:

LER 93-08, Failure to Perform Required Offgas Sampling.

### 5.2 Outage Manager Role

The inspectors noted that during the unit outages this inspection period, NYPA implemented a 24-hour outage manager position. These three eight-hour shift, outage manager positions are filled by experienced station supervisors who are designated in advance of the outage. They are temporarily relieved of their normal staff duties and report directly to the General Manager of Operations. Their principle duties are to oversee all planned outage work activities and to facilitate the smooth execution of those activities. Should work conflicts or unforeseen problems arise, the outage manager is the principle liaison between the responsible departments to resolve the issue.

In addition to the outage managers, NYPA has exercised close control over the work scope of the recent outages. Outage work activities are planned in advance and any emergent work items are carefully screened via scope control meetings held daily. These new outage planning and control initiatives has resulted in improved inter-department communications and improved work scheduling and execution in the field.

5.3 Review of Temporary Instruction 2515/119 - WATER LEVEL INSTRUMENTATION ERRORS DURING AND AFTER DEPRESSURIZATION TRANSIENTS (GL 92-04)

Background

The objective of this inspection was to determine NYPA's implementation of operator guidance and training to address potential reactor vessel level responses following rapid depressurization transients, and to ensure that this guidance and training was consistent with current plant Emergency Operating Procedures (EOPs).

NRC Information Notice (IN) No. 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization" and Generic Letter (GL) No. 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)" discussed concerns that noncondensible gases may become dissolved in the reference leg of boiling water reactor (BWR) water level instrumentation and lead to a false high level indication after a rapid depressurization event. The dissolved gases, which accumulate over time during normal operation, can rapidly come out of solution during depressurization and displace water from the reference leg resulting in a false high vessel water level indication. This potential transient is safety significant because water level signals are used for actuating automatic safety systems and for guiding operator actions during and after an event.

Level Instrumentation Systems Review

This problem affects only cold reference leg water level instruments, and not heated reference leg instruments known as Yarways. The FitzPatrick plant originally utilized Yarways for some of the instruments. However, Yarways were replaced and only cold reference legs are currently used for the reactor vessel level instrumentation. There are five condensing chambers to supply reference legs. Chamber 1 supplies the reference leg for the refueling level instrument. Condensing chambers 2A and 3A are supplied by a common steam connection to the reactor vessel. The 2A condensing pot provides makeup for the reference leg associated with half of the wide range level instrumentation and one of the fuel zone instruments. The 3A provides the makeup for the reference leg associated with half of the narrow range level instrumentation. A similar arrangement exists for the 2B and 3B condensing pots in that they share a common steam supply line and the 2B chamber is associated with the remaining wide range and fuel zone level instrumentation and the 3B is associated with the remaining narrow range instrumentation.

The reactor vessel level instrumentation provides signals for the following safety related functions:

Narrow range:

- Level indication and high and low level alarms
- Permissive signal for the automatic depressurization system (ADS)
- Reactor core isolation cooling (RCIC) system high level isolation
- Reactor protection system (RPS) logic for low level scram
- Primary containment isolation
- High pressure coolant injection (HPCI) system high level trip
- Standby gas treatment (SGT) initiation

Wide range:

- Level indication
- Primary containment isolation
- Emergency core cooling systems (ECCS) initiation
- RCIC and HPCI low level initiation
- Alternate rod insertion (ARI)
- Anticipated transient without scram (ATWS)
- Emergency diesel generator start
- Reactor water recirculation pump trip

Fuel Zone:

- Level indication
- Containment spray permissive

Refuel:

- Level indication during refueling operations

Operator Guidance and Training Review

The inspector determined that licensed operators had received pertinent training on several occasions. Prior to the startup in December 1992, the system engineer conducted training with all of the licensed operators to ensure they were familiar with the noncondensible gas problem and how it could potentially affect level instrumentation performance. Industry experience with the problem was also reviewed including specific problems experienced at the Peach Bottom, Pilgrim and LaSalle facilities. The effects of leaks in the instrument lines and level "notching" during plant cool downs was also reviewed at this time. The operations supervisor briefed the operators on this issue. All licensed operators were required to read the October 16, 1992 letter from the BWR Owners Group (BWROG) which contained guidance on how to determine reactor vessel level following a rapid depressurization that results in erroneous level indications. The Operations Supervisor also uses the Night Order

Book to keep the operators informed of new information as it becomes available. A videotape presentation was shown to all licensed operators during a training cycle. The presentation included a thorough review of the guidance contained in the October 16, 1992, BWROG letter. The videotape was well done and made excellent use of graphics to reinforce the principles contained in the letter. The inspector also learned that a formal classroom training module will be developed to address the GL 92-04 issues upon completion of testing by the BWROG.

The system description lesson plan for reactor vessel level instrumentation (SDLP-02B) does not currently contain information on the noncondensible gas issue. However, the lesson plan is being revised to contain Generic Letter 92-04 as a reference, and to require viewing the videotape discussed above. This revision was scheduled to be completed by June 30, 1993.

The inspector determined that licensed operators have not received specific simulator training relative to the potential reactor vessel level problems. Also, specific simulator modeling has not been performed. NYPA currently plans to provide simulator training when the BWROG completes its study of the issue and can better quantify the expected instrument responses during depressurization transients. NYPA was also in the process of collecting temperature data on the condensing chambers to better apply the BWROG findings to their specific configuration. Current simulator training, for scenarios where reactor vessel level is considered to be indeterminate, is based upon factors other than the noncondensible gas problem.

Procedures Review

NYPA has reviewed the plant Emergency Operating Procedures (EOPs) to ensure that they are consistent with the current BWROG guidance on responding to a rapid depressurization with concurrent reactor vessel level indication inaccuracies due to noncondensible gases in the reference legs. NYPA concluded that no changes to the EOPs were necessary at this time. EOPs specify that when reactor vessel level is undetermined, emergency depressurization and reactor vessel flooding is required.

Operating procedure OP-65, "Startup and Shutdown Procedure," has been revised to caution the operators that during a plant depressurization and cool down, noncondensible gases coming out of solution in the reference legs could cause alternating step increases and decreases (notching) in indicated reactor pressure vessel water level. The inspector also determined that procedure ODSO-17, "Auxiliary Operator Plant Tour and Operating Logs," has been revised to alert the operators to inspect for any instrumentation leakage and to immediately report any leaks to the shift supervisor.

Plant Computer Review

The plant computer displays the various reactor water level channels and also computes and displays an average level. If the individual channel indication exceeds the normal range of the instrument the computer display turns a magenta color and the point does not indicate any specific level (question marks replace the normal indication). This display alerts the operators to a potential problem which requires investigation and corrective action, as necessary.

Summary

The inspector concluded that NYPA was taking appropriate actions to make the operators aware of the potential effects of noncondensible gases in the reference legs and was closely involved with the BWROG's efforts to resolve this issue. As discussed in a previous report, 50-333/93-06, NYPA has initiated temperature monitoring of the reactor vessel level condensing chambers to better understand the noncondensible gas phenomena and its impact on the FitzPatrick vessel level instruments. The FitzPatrick staff was collecting and evaluating temperature data at the time of the inspection. Additional training, procedural guidance and plant modifications may be warranted when the BWROG testing results are available.

## 6.0 MANAGEMENT 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.