U.S. NUCLEAR REGULATORY COMMISSION

Region I

Report No.:	90-08
Docket No.:	50-333
License No.:	DPR-59
Licensee:	New York Power Authority Post Office Box 41 Lycoming, New York 13093
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, New York
Dates:	November 4 through December 22, 1990
Inspectors:	W. Schmidt, Senior Resident Inspector R. Plasse, Jr., Resident Inspector

Approved by: Glenn W. Meyer ____________________ 2-12-91

Glenn W. Meyer, Chief   Date
Reactor Projects Section No. 1B

Inspection Summary:

This inspection report discusses routine and reactive inspections of plant activities during day and backshift hours including: plant operations, radiological protection, surveillance and maintenance, emergency preparedness, security, engineering and technical support, and quality assurance and safety verification.

Results:

An Executive Summary and an Outline of Inspection follow.

RESIDENT INSPECTOR OFFICE
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
INSPECTION REPORT NO. 90-08

EXECUTIVE SUMMARY

Operations

Control room operators continued to perform well during routine and scram recovery activities. However, a high startup flux (>15 % reactor power) reactor scram was caused when operators opened a feed pump discharge valve injecting a cold slug of water and causing a reactivity increase. This method of placing a feed pump in service was done in an attempt to restore vessel level as it neared the low level scram setpoint.

Radiological Protection

The Radiological and Environmental Services (RES) department performed well in dealing with an unsecured High Radiation Area gate (> 1000 mr/hr). The RES department also performed well in dealing with the spill of low activity condensate in the reactor building and in the corrective action from a previously identified violation of access requirements for entry to a posted High Radiation Area.

Surveillance and Maintenance

NYPA has not taken adequate action to preclude short duration pressure transients in reactor vessel level sensing lines from causing reactor scrams. The December 12 scram was the second instance, in the last year, where valving in a feed flow level control transmitter caused a reactor scram.

NYPA failed to enter the LCO for inoperable primary containment isolation valves when reactor water cleanup outboard isolation valve breakers were racked out during a surveillance test with the valves open. Because these breakers were racked out, the valves did not go closed during the December 12, 1990 reactor scram as did the other Group II isolation valves. Operators responded correctly and closed the valves. Further, the FitzPatrick TS does not specify any out of service time allowance for testing instrumentation, and the applicable operability LCOs are not applied when calibration forces the number of instruments to be less than that required. These surveillance issues were classified as an Unresolved Item.

The inspector concluded that the corrective actions from the identified problem with the low feedwater flow control valve had been ineffective and had lead to the reactor scram on December 15. During observed maintenance activities, NYPA displayed good performance.

Emergency Preparedness

The annual NRC observed emergency exercise was conducted on December 19. The NRC was a partial participant, with a site response team, Region I and portions of Headquarters. NRC observation team assessments of exercise performance will be provided in Inspection Report No. 50-333/90-24.

Security

The inspector reviewed the security measures at the new NYPA warehouse and found them to be acceptable.

Engineering and Technical Support

NYPA continued to have difficulty with the operation of the boundary check valves between the safety related emergency service water and non-safety related service water systems. Three valves failed to close during IST on November 15, due to corrosion or silt buildup.

NYPA has issued a procedure that deals with the identification and review of FSAR and design basis deviations. The procedure allows for the development of a Reasonable Assurance of Safety document for such instances, which must be completed within 30 days, if a 10 CFR 50.59 evaluation can not be completed within that time.

Safety Assessment/Quality Verification

NYPA's Safety Review Committee was found to perform an adequate review of operations for FitzPatrick during the November 1990 meeting.

The FitzPatrick QA program procedures for identification, control and correction of adverse conditions appeared to be adequate. However, implementation of the AQCR program was found to be weak with respect to resolution of identified concerns. This was considered an Unresolved Item pending further NRC review.

RESIDENT INSPECTOR OFFICE
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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OUTLINE OF INSPECTION

1.0 Operational/Event Summary

2.0 Operations (MC 71707, 93702)

a. December 12, reactor scram due to false low reactor vessel level signal during surveillance testing.
b. December 15 reactor scram due to high startup flux scram, due to cold water injection. Review of operating philosophy for use of procedures.
c. December 17 rupture of a condensate transfer system line.
d. Review of immediate corrective actions for inoperable service water check valves.

3.0 Radiological Protection (MC 71707, 92702)

a. Unsecured High Radiation Area Gate, Licensee Identified Violation 90-08-01.
b. Radiological consequences of the condensate transfer line rupture.
c. (Closed) Violation 90-05-01. Entry by security guards into a Posted High Radiation Area without proper precautions.

4.0 Surveillance and Maintenance (MC 61726, 62703, 92702, 92703)

a. Inadequate action in review of the January 1990 reactor scram due to pressure spiking on the RPS low level instrument sensing line, Unresolved Item 90-08-02. Closed Item F-2 from Inspection Report 89-12.
b. Entry into LCOs for equipment during time when it is inoperable during testing. FitzPatrick TS do not specify any time limit for instrument testing. Unresolved Item 90-08-03.
c. Ineffective corrective action on low feedwater flow control valve following the December 12 scram.
d. (Closed) Unresolved Item 90-04-05. Installation of jumpers.
e. Review of maintenance.

5.0 Emergency Preparedness (MC 71707)

a. December 19 annual observed emergency exercise.

Outline of Inspection (Continued)

6.0 Security (MC 71707)

a. Review of security measures at the new warehouse.

7.0 Engineering and Technical Support (MC 90712, 92702)

a. (Open) Unresolved Item 90-02-06; Review of LER 90-25, Failure of Service Water Check valves to shut during IST.
b. Review of Nuclear Generation Procedure - 38.

8.0 Safety Assessment/Quality Verification (MC 40500, 92720)

a. Review of the November 1990 Safety Review Committee meeting.
b. Review of Adverse Quality Condition Report System. Weaknesses in the review of AQCRs were identified, Unresolved Item 90-08-04.
c. (Closed) Unresolved Item 88-29-07.

9.0 Other Inspections

10.0 Exit Interview

Attachment A - Acronyms

DETAILS

1.0 OPERATIONAL/EVENT SUMMARY

On November 15, three of the swing check valves that act as boundaries between the safety related emergency service water (ESW) and the non-safety related service water (SW) systems failed to close during in-service testing (see Sections 2.d and 7.a below). The unit operated at rated power until December 12, when an automatic scram occurred because of false low vessel level signals generated during surveillance testing on a feed water control reactor vessel level instrument (see Sections 2.a and 4.a below). During the subsequent reactor startup, on December 15, the reactor automatically scrammed due to high startup neutron flux (> 15 % of full power) following a cold water injection from the feed system (see Sections 2.b below). The unit was restarted on December 16, and returned to full power on December 17. On December 18, a low activity water (< 1 x 10-6 uc/ml) spill, of approximately 2000 gallons, occurred when a condensate transfer line to the spent fuel pool cooling system ruptured (see Sections 2.c and 3.b below). The annual emergency preparedness exercise was conducted, with partial NRC participation, on December 19 (see Section 5 below).

2.0 OPERATIONS

a. Operators performed well during recovery from the December 12 reactor scram, which resulted during a transmitter calibration and is discussed in Section 4.a. Shift Supervisor (SS) control and monitoring was excellent, as was the flow of information from the operators to the SS. Operators had difficulty in restarting a feed pump and feeding the vessel through the low flow control valve, after the feed pumps tripped as expected on high vessel level following the scram. This led to a second scram condition on an actual low vessel level. Operators performed well evaluating the feed system problems and regaining feed flow, without the need of HPCI or RCIC.

b. On December 15, a reactor scram occurred during reactor startup when a steam flow/feed flow mismatch occurred at low power, which caused reactor vessel level to decrease. Operator actions to restore level resulted in a reactor scram on a high startup flux signal. Specifically, reactor vessel level was being controlled with the low flow control valve (bypass control valve around the closed feed pump discharge valves) in automatic, with one condensate and one booster pump in operation. At approximately 490 psig reactor pressure and 6% reactor power, while pulling control rods and increasing the EHC system pressure set to maintain one turbine bypass valve open, steam flow exceeded the feed water flow causing a decrease in vessel water level. The operators noticed that the demand signal for the low flow control valve had gone to 100% (the low flow control valve controller only has the demand signal and does not have position indication). Prior to this the signal had been at approximately 60%. This indicated to the operators that the feed flow demand exceeded the low flow control valve's capacity and the resulting steam flow/feed flow mismatch causing the decreasing level. (Later evaluation of the low flow control valve revealed a mechanical restriction that prevented full stroking of the valve; see Section 4).

At that point operators attempted to reduce the mismatch by: reducing power through insertion of control rods and trying to increase the flow through the low flow control valve by starting additional condensate and booster pumps; lowering the EHC system pressure set to approximately 440 psig; and starting a feed pump and bringing it up to 600 psig discharge pressure (discharge valve closed). None of these actions were fully effective in stopping the decrease in water level.

When the level reached 179 inches (two inches above the low level scram setpoint), operators jogged open the feed pump discharge valve three times. This increased feed flow began to restore vessel level to the normal range; however, the large injection of cold water increased reactivity and a power increase resulted. The reactor scrammed on high startup flux (15% power setpoint).

NYPA concluded that the root cause of the scram was weak operational control of the startup process, which had resulted in the steam flow/feed flow mismatch. As corrective action NYPA revised the operating procedure for controlling vessel level during startup to emphasize the review of the demand indication on the low flow control valve. NYPA, including the PORC, concluded that operators had taken technically acceptable actions in response to the low vessel level.

The inspector reviewed the NYPA evaluations and corrective actions and agreed that the root cause was weak operational control of the startup and that the operator actions in response had been technically acceptable. Further, in discussions with plant management, the inspector noted that the system operating and abnormal procedures did not specifically address the situation that the operators had faced and that some of the operators' actions had adjusted various procedure steps to the applicable situations. The inspector concluded that the operators took acceptable actions; nonetheless, the inspector stated that this event provided an opportunity for plant management to re-emphasize to the operating staff that a cautious, deliberate approach that must be used when operators are confronted with conditions not specifically addressed by existing procedures. The plant management agreed to discuss this issue as part of continuing operator training.

c. The operations staff performed well when the 4 inch line to fuel pool skimmer surge tank from the condensate transfer system ruptured. Approximately 2000 gallons spilled on an upper level of the reactor building and cascaded to the crescent area. Operators took immediate actions to secure the condensate transfer pumps to stop the leak and enter EOP-5, Secondary Containment Control on high reactor building sump levels. Because the condensate transfer system had been supplying RHR keep-fill the operators placed both RHR sub-systems in torus cooling to maintain RHR full. After identification and isolation of the leak, NYPA restored the condensate transfer system and RHR keep-fill, and proceeded to replace the section of Bondstrand pipe that ruptured with steel piping.

The loss of RHR keep-fill was the only safety-related effect of this spill. NYPA has installed RHR keep-fill pumps that take a suction from the torus which will allow securing of the condensate transfer cross-connect to RHR. TS Amendment No. 166 reflected the containment isolation valves associated with this system and NYPA was completing the procedure changes to allow use of this system.

d. The inspector found that the operations staff took proper action when the three service water check valves were found to be inoperable during IST on November 15. These actions included starting the ESW pumps to supply the cooling loads and isolation of the service water supplies to the three unit coolers that were effected. Further, the entry into a TS 3.0.C, 24 hour LCO when the ESW and SW flow was secured to these coolers to allow repair was conservative.

3.0 RADIATION PROTECTION

a. Adequate measures were taken by NYPA when an operator discovered a high radiation area gate unlocked. TS 6.11(A) requires that areas with radiation field greater than 1000 mr/hr be locked to prevent unauthorized entry. On December 19, during an emergency drill a licensed operator determined that the high radiation area gate at the steam tunnel was unlatched. The operator informed the RES supervisor, the gate was immediately shut, and a survey was conducted. Radiation fields in the steam tunnel at that time were greater than 1000 mr/hr.

The gate had been checked shut during daily radiological technician rounds on the previous day. NYPA determined the possible causes were either failure to ensure that the gate was shut after exiting the area, or a faulty latching mechanism. NYPA could not determine the last individual to exit the area. The latch was found to be operable but difficult to secure and was repaired. In addition NYPA upgraded the lock core to prevent operators from performing routine entries to the area without receiving a key from the RES department.

Based on this occurrence and previous failures, NYPA planned to perform a quarterly surveillance to check proper operation of all radiologically controlled gate latching mechanisms. A Notice of Violation was not issued as allowed by the NRC Enforcement Policy, 10 CFR Part 2, Appendix C, Section V.G.1., since the problem was self-identified and the corrective actions were adequate. Assignment of an open item number identified this non-cited violation solely for tracking purposes. LI NCV 90-08-01.

b. While the radiological significance of the condensate (activity < 1 x 10-6 uc/ml) spill was minor (Section 2.b), the RES department took adequate actions to evaluate the spread of existing contamination and to prevent possible personnel contamination events. Extensive cleanup efforts on various levels of the reactor building were necessary to reestablish radiological controls due to washout of existing contamination by the spill. Actions taken by the RES staff in response to this event appeared to be correct and were completed satisfactorily.

c. (Closed) Violation 90-05-01: NYPA agreed with this violation in their October 15, 1990 response. The cause of the violation was the failure of security guards to follow the requirements for access to posted high radiation areas. The inspector found that the corrective actions of retraining security guards on RWP and high radiation area access control requirements, and including discussions of access control requirements at meetings with general plant personnel were adequate. There have been no further instances of individuals entering high radiation areas without the use of an RWP. The inspector concluded that corrective actions were acceptable and that this item was closed.

4.0 SURVEILLANCE AND MAINTENANCE

a. It does not appear that NYPA took sufficient action to fully evaluate the effects of variable water leg pressure spikes on RPS low vessel level instruments. The December 12 scram was the second scram caused in 1990 by an apparent pressure transient on the reactor water level sensing line supplying two of the four low level instruments. Each of these instruments provided a scram signal to one of the two RPS channels. This and the previous scram were caused during return to service, following calibration, of the same feed control level instrument. NYPA was unable to determine the cause of the January 1990 scram and planned, based on the second scram to continue evaluation of the effects of returning the feed control level instrument to service. It was noted in Inspection Report 89-12 that the Rosemount transmitters and trip units used in RPS do not have any electronic dampening built in. Thus any very short pressure spike above the trip point activates the trip unit. The inspector considered that the adequacy of the corrective action was an Unresolved Item pending review of NYPA's root cause analysis, and closed F-2 from Inspection Report 89-12 administratively. UNR 90-08-02

b. During review of the December 12 reactor scram, the inspector found that NYPA had not entered in the LCO for containment isolation valves which had been rendered inoperable as part of a surveillance test. The LCO requires that the valves be returned to service within four hours or the other isolation valves in that line be closed. Specifically, I&C technicians were performing calibration of the RWCU outboard containment isolation valve steam leak detection circuitry prior to the scram. As a prerequisite to the testing, the outboard RWCU isolation valves were left open with their breakers racked out to prevent inadvertent isolation of RWCU; this caused the valves to be inoperable. Following the scram, these valves did not close as part of the Group II isolation on low vessel level; subsequently, the operators restored the valves to operability by reshutting the breakers, at which time the valves went shut. While the inspector found that the valves were returned to operability within four hours, NYPA had not entered into the TS LCO for inoperable primary containment isolation valves, when the valve breakers were racked out. NYPA stated that it is their current policy that when safety systems or components are made inoperable during surveillance testing that the applicable LCO need not be entered.

Further, the FitzPatrick TS does not specify an allowable time that the instrument may be taken out of service for calibration. In this case the LCO for RWCU steam leak instrument requires all of the instruments to be operable or RWCU be isolated. The calibration procedure required the RTD leads to be disconnected from the instruments thus making it inoperable while it was being tested. This issue remains unresolved pending further review by the NRC staff. UNR 90-08-03.

c. The inspector found that corrective actions following the December 12 scram were weak and did not ensure proper operation of the low flow control valve. During recovery from that scram, operators had identified that there was a problem with either the A feed pump discharge check valve or the low flow control valve. After finding no apparent deficiency with the check valve, following disassembly, NYPA management specified that the low flow control valve be stroked to ensure it was not sticking. The operator gave the valve an open signal and verified that the stem moved in the open direction and then closed the valve. There was no attempt to verify the valve stroked properly through its full travel. A stuck open check valve and a sticking flow control valve were the most likely causes of the problem based on past experience. As those were shown to be acceptable, the problem was not further evaluated.

Following the difficulties associated with vessel level which resulted in the December 15 scram, NYPA identified that the low flow control valve would not stroke through its full travel due to a ruptured diaphragm in the air operator. The ruptured diaphragm had resulted in a limited stroke for the valve and had caused a steam flow/feed flow mismatch when feed demand through the valve was high. The inspector concluded that NYPA's corrective actions for the problems with the valve identified following the December 12 scram had been ineffective. Although the initial approach had been reasonable, it appe hat corrective actions were terminated too soon when this approach did not provi explanation for the problem.

d. (Closed) Unresolved Item (90-04-05): The inspector reviewed NYPA's corrective actions to prevent installation of temporary jumpers effecting component operability without SS permission. Additional training was conducted with contract services supervisors to ensure that any jumper installation necessary to support work, receives SS permission. In addition, whenever a hose, spoolpiece, flange or electrical leads are connected into a system to support a modification, contract services personnel were trained to check with the Work Control Center to determine if the activity is required to be controlled in accordance with WACP 10.1.3, Control of Jumpers, Lifted Leads, and Temporary Modifications. Further, this would allow the establishment of a fire watch if the jumper disabled a fire barrier. The inspector closed this item.

e. Generally, the inspector found that NYPA's controlled use of LCOs to perform planned maintenance continued to be effective and a good initiative. The prioritization and identification of work to be completed following the two scrams was also well controlled.

5.0 EMERGENCY PREPAREDNESS

a. On December 19, the annual NRC observed emergency preparedness exercise was conducted. This was a partial participation exercise including the State of New York, the County of Oswego and portions of the NRC. A NRC site team responded and participated along with an NRC Region I base team and Headquarters. Details of the NRC observation team's findings made during the exercise will be discussed in Inspection Report 90-24.

6.0 SECURITY

a. The inspectors reviewed the security precautions at the new NYPA warehouse. The system for providing entry of stores into the protected area was adequate to ensure proper controls.

7.0 ENGINEERING AND TECHNICAL SUPPORT

a. (Open) Unresolved Item 90-02-06; LER 90-25; Service Water Check Valves Fail to Close During Testing. NYPA took corrective actions regarding the recent failure of the three swing check valves to perform as designed. Specifically, NYPA had committed to the enhanced testing of these check valves following the 1990 refueling outage when operability concerns because of silting and corrosion in ESW and its interfacing SW check valves were identified. The testing was conducted to mimic the design condition of a rupture upstream of the service water check valves with ESW flow through the cooled component. Two of the three valves each supplied one of the divisional electric bay unit coolers, the other valve supplied the division I cable tunnel unit cooler. These valves were found to be inoperable during the 1990 refueling outage, were repaired and satisfactorily tested during the subsequent four monthly surveillance tests.

The LER stated that the result of the valves not closing could have been less than the desired flow to the specific unit coolers and to other ESW cooled components, because of back flow to the service water system. The justification for not considering this condition of serious safety significance was well founded. However, this reasoning was mitigation to the fact that the valves would not have performed as designed. NYPA increased the testing frequency to twice every month on these ESW to SW boundary valves. Further, the swing arms and pivots on these three valves were replaced with stainless steel in hopes of minimizing the effects of corrosion. The inspector concluded that these corrective actions were appropriate, for this instance. However, this item remains open pending review of NYPA's revised submittal to address Generic Letter 89-13 and review of corrective actions to address long term corrosion and silting effects on the ESW system.

b. The inspector reviewed a new NYPA Nuclear Generation Procedure (NGP- 38), dated August 27, 1990, which dealt with the identification of deviations from the FSAR, and other design basis documents. This procedure made it clear what NYPA considered to be design basis documents and included NYPA submittals to the NRC and any subsequent NRC SERs. This procedure directs that NYPA shall perform a 10 CFR 50.59 safety evaluation or a reasonable assurance of safety (RAS) evaluation which allows operation of the plant until a 10 CFR 50.59 evaluation can be conducted. The procedure specified a thirty day time period to complete the evaluation. This appeared to be too long for cases were safety related equipment was involved. While the procedure did address the reportability aspects of such a deviation, it did not specify an appropriate method to ensure proper tracking and resolution documentation for identified deficiencies.

The inspector identified a concern involved with this procedure. When the procedure was issued in August 1990, NYPA apparently did not review their corrective actions systems then to ensure that any existing appropriate open items were reviewed under this process. This was evident when the inspector reviewed AQCR 90-95, dealing with the containment hydrogen monitoring system (see Section 8.b below).

8.0 SAFETY ASSESSMENT/QUALITY VERIFICATION

a. The inspector observed that NYPA's Safety Review Committee (SRC) was effective in the review of safety evaluations and plant operating status. The NYPA personnel on the SRC were self-critical and took recommendations from other members of the committee freely. The members of the SRC from both Con Edison (Indian Point Unit 2) and Niagara Mohawk (Nine Mile Point Units 1 and 2) gave useful comments and insights. The individual contracted to sit on the SRC for his nuclear experience provided exceptionally well defined comments on every aspect of plant operations.

b. The inspector conducted a review of the Adverse Quality Condition Report (AQCR) system. The inspector reviewed the procedures used to identify, control and correct adverse conditions, and the acceptability of several actual AQCRs. The inspector concluded that the AQCR procedures were adequate to outline an effective program. However, the implementation of the program was weak because of ineffective review of adverse conditions by QA.

The implementation of the QA program at FitzPatrick was controlled by Administrative Procedure 1.7, which specified that the QA department controlled the AQCR system. The identification, correction and evaluation of adverse quality conditions was controlled by three Site Quality Assurance Procedures (QAPs), QAP 15.2 Identification, Control and Resolution of Adverse Quality Conditions, QAP 16.1, Corrective Actions, and QAP 18.3, Trend Analysis Program. The procedures specified adequate review and determination of the significance of the nonconforming conditions. There were three significance categories; standard, indeterminate and significant. The procedure for determining the appropriate category was clear. Reasons for categorizing an AQCR as significant included; a potential NRC violation, reportability to the NRC, recurring inoperable equipment, a negative trend, or QA management discretion. Root cause analysis was specified for significant and indeterminate AQCRs and the procedure appeared to be adequate. The controls for escalating inadequate or late responses, ultimately to the NYPA President were adequate. The inspector found that these procedures were adequate to outline an effective program; however, the procedures were cumbersome and somewhat repetitive.

The inspector noted two weaknesses with this program:

-- The guidelines for when a non-QA employee must use an AQCR to document an adverse condition were vague. While QA personnel must always use the AQCR, other plant personnel may use the AQCR if the condition can not be addressed by other plant systems or procedures acceptably. There was no indication of what other systems or procedures were acceptable to QA for documenting an adverse condition.

-- The interaction between AQCRs and Occurrence Reports was not written in the procedure. The OR system functions to allow the determination of the effects of adverse conditions on the operability of systems and in determining the reportability of such instances. The AQCR program does not direct initiation of an OR to allow operability determination to be made.

The inspector reviewed twenty-one (21) of the approximately one-hundred-eighty (180) AQCRs written from the beginning of the year until November 1990. The inspector noted the following in his review of these selected AQCRs:

-- The process for acceptance of corrective actions did not ensure a multi-person review of the actions and their effectiveness. Twelve (12) of the twenty-one (21) AQCRs did not receive multi-person review because the reviewer and the supervisory reviewer were the same person (i.e., either the QA or QC supervisor). This practice did not appear to meet the intent of QAP 15.2, section 6.8, which stated that the QA department supervisor and a reviewer should either accept or reject the corrective actions.

-- Review of these AQCRs led the inspector to identify two significant, one indeterminate and one standard AQCR (as follows) which reflected that QA review was not always sufficient to ensure adequate corrective actions.

a. Significant AQCR 90-95 was written on May 7 to document a deficiency between the as-built condition of the containment hydrogen monitoring system and NYPA's commitments to meet TMI Action Plan Item II.F.1.6. This AQCR was made significant based on the probability that the NRC would consider it a violation. The monitoring system was supposed to allow each train to sample the drywell, torus, and secondary containment. It was determined that one train could not sample the torus air space because its sample location was in the drywell downcomer ring header in the torus, and thus was sampling drywell atmosphere.

Because this was classified as a significant AQCR, a root cause analysis was conducted. This analysis stated that the engineers and reviewers had not properly reviewed the modification and that it was an isolated case. There were no corrective actions specified. Further, review by the Technical Services system engineering group stated that there was no requirement to sample the torus, so there was no deficiency with the system as it was. A letter dated May 9, 1990, was issued by the Resident Manager to NYPA corporate engineering requesting that they review this issue. This review was to be completed by December 1, 1990, and was being tracked as an open item by the Superintendent of Power on the Action/Commitment Tracking System. Based on the engineering review and on the issuance of the letter to corporate, the QA department closed this AQCR on May 21.

The inspector reviewed the AQCR on November 19, and discussed it with NYPA staff members. NYPA issued OR 90-315 documenting the situation. NYPA did not formally report the apparent design deficiency to the NRC but the NRC Project Manager (PM) was contacted by the NYPA licensing group and was told of the situation. The PM had previously been briefed on the situation by the resident inspector. It should be noted that such contact with the NRC does not constitute any form of official notification.

The process used to evaluate and close this issue should have been completed before the unit restarted from the 1990 refueling outage, because NYPA knew that they would be operating with a system that was not as described in the design basis.

b. AQCR 90-116 was written on May 18 to document removal of the wrong snubber during the snubber testing program. This AQCR was initially classified as a significant item because QA management felt that this represented a trend, since on May 1 AQCR 90-83 was written to document the same type of problem. As allowed by procedure the maintenance department responded to this AQCR and stated that they did not agree with the significance, because the same type of issue had been raised during an audit of the snubber program. The QA supervisor then approved this rationale and down graded the AQCR's significance to standard. This allowed the maintenance department not to address why the initial corrective action from the first AQCR was not effective. While the AQCR remained open the inspector found that the downgrading of significance was inappropriate.

c. AQCR 90-112 was written on May 15, and was classified as indeterminate because it could not be determined if the condition was reportable or if it would have resulted in an NRC violation. A Technical Services engineer identified a potentially overstressed condition in the normal reactor water sampling line due to an inappropriately designed pipe support downstream of the outboard containment isolation valve (02-2SOV-40). In the as-found condition the engineer believed that pipe and/or penetration damage could occur due to the thermal expansion and the location of the support.

The engineering response to this condition was that rework was needed. The response to the root cause analysis, was that the initial design was not adequate. However, this same section of piping had been modified in 1985 and the initial design inadequacy was not discovered. The corrective action of performing the rework was symptomatic to this specific instance and did not address the need for any further review of other piping supports. Further, why the design error was not identified during the 1985 modification and how similar situations would be prevented in the future were not addressed.

d. AQCR 90-166 was written on June 21 and classified as an item of standard significance. The inspector's assessment is that this classification was questionable since the cause of the apparent difficulty was failure to conduct safety related work, outside the skills of the trade, with an approved procedure. In this case the opening and closing of the torus hatch (X-200A) was completed without a procedure. This resulted in the bolting (grade B8) being over-torqued, because the maintenance planner, who wrote the work request (WR), assumed a wrong torque value. The procedure that had existed, in part, to address this penetration had been modified to delete the applicable portions in early 1990. It had included specific QC engineering hold points and QC observation points. The torque value specified on the WR was taken from a generic maintenance procedure for bolting and was not correct for grade B8 bolts. QC monitored the evolution of opening and reclosing the hatch and considered the evolution to be within the skills of the trade.

Once it was determined that the bolting had been over-torqued, it was replaced with grade B7 bolting. Again, this was done without a procedure (i.e., the work request stated to install the B7 bolting and to torque it to a specified value). The B8 bolting was specified on the approved plant drawing for this penetration. Further, this WR did not recognize that the replacement of the B8 bolting with the B7 bolting was a modification. A temporary modification to allow the installation of the B7 bolting was performed on June 23, five days after the bolting had been installed. This TM 90-145 stated that the bolting was marginally satisfactory for use until B8 bolting could be procured, based on a miscellaneous calculation JAF 90-084. This calculation only addressed the torquing requirements for the B7 bolting, it did not clearly address why they were only minimally satisfactory. Discussions with technical support engineering led the inspector to believe that the bolting was acceptable, but that the reasoning was not justified on the TM or the calculation.

The inspector determined that corrective actions taken were purely symptomatic, and did not cause heightened awareness of personnel to what should be done with a procedure and when a temporary modification was appropriate.

Based on the above examples, the inspector concluded that NYPA's review and corrective actions for identified problems were not consistently timely and effective. The inspector considered this ineffectiveness to be an Unresolved Item (UNR 90-08-04) pending NRC review of these AQCR concerns.

c. (Closed) Unresolved Item 88-29-07: The inspector had questioned whether review of 10 CFR 21 reports by upper NYPA management was needed. Based on further review and discussions with NYPA corporate personnel, the inspector concluded that the Resident Manager of each nuclear facility has the authority to determine and report conditions under 10 CFR 21. NYPA administrative procedures suggest that these conditions be reviewed by the Executive Vice President of Nuclear Generation prior to reporting the condition. However, the inspector concluded that this review was optional and at the discretion of the Vice President. This item was closed.

9.0 OTHER INSPECTIONS AND ENFORCEMENT CONFERENCES

a. Radiological Controls, November 5-9, 1990; Inspection Report 90-22.

b. Radiological Effluents, November 26 - 30, 1990; Inspection Report 90-23.

c. Emergency Preparedness Exercise Observation, December 18 and 19, 1990; Inspection Report 90-24.

10.0 EXIT INTERVIEW

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.

APPENDIX A

James A. FitzPatrick Nuclear Power Plant

Acronyms

AQCR	-	Adverse Quality Condition Report
EHC	-	Electro Hydraulic Control
EOF	-	Emergency Operations Facility
EOP	-	Emergency Operating Procedures
ESW	-	Emergency Service Water
HPCI	-	High Pressure Coolant Injection System
IST	-	In-Service Testing
LCO	-	Limiting Condition for Operation
LER	-	Licensee Event Report
NRC	-	Nuclear Regulatory Commission
NYPA	-	New York Power Authority
OR	-	Occurrence Report
OSC	-	Operations Support Center
QA	-	Quality Assurance
QC	-	Quality Control
RCIC	-	Reactor Core Isolation Cooling System
RES	-	Radiological and Environmental Services
RHR	-	Residual Heat Removal System
RPS	-	Reactor Protection System
RTD	-	Resistance Temperature Detector
RWCU	-	Reactor Water Cleanup System
RWP	-	Radiation Work Permit
SW	-	Service Water
TS	-	Technical Specification
TSC	-	Technical Support Center
WR	-	Work Request