U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-333
License No:	DPR-59
Report No.	50-333/98-05
Licensee:	New York Power Authority (NYPA) Post Office Box 41 Scriba, New York, 13093
Facility:	James A. FitzPatrick Nuclear Power Plant
Dates:	July 27, 1998 through August 14, 1998
Team Leader:	E. Harold Gray, Senior Reactor Engineer, Division of Reactor Safety (DRS), Region I
Inspectors:	Doug Dempsey, Reactor Engineer, DRS Ken Jenison, Sr. Resident Inspector, Susquehanna Station Paul Kaufman, Sr. Reactor Engineer, DRS Omar Mazzoni, NRC Contractor for Electrical and I&C Tom Shedlosky, Senior Reactor Analyst, DRS Tom Burns, Reactor Engineer, DRS
Approved by:	Glenn Meyer, Chief Civil, Mechanical and Materials Engineering Branch Division of Reactor Safety

9810060027 980928
PDR ADOCK 05000333
G PDR

EXECUTIVE SUMMARY

FitzPatrick
NRC Inspection Report 50-333/98-05

The engineering inspection team judged the engineering activities at the FitzPatrick plant to be generally good, including input to modifications, surveillance testing, and post-maintenance testing. The engineering methods and results were generally sound and correct, and engineering backlogs have been at an acceptable level. Overall, vertical slice inspections of RCIC and ADS found acceptable design and licensing bases with adequate operating and testing procedures. Problems were identified with operability determinations, evaluations, and calculations in that they were frequently narrowly focused, limited in scope, and not thorough, although none of those sampled had reached an incorrect operability conclusion. The operability procedure had been revised just prior to this inspection, and the new revision appeared to address the NRC concerns.

The team concluded that NYPA had maintained an accurate design and licensing bases for the ADS and reactor pressure relief systems. The design basis was appropriately derived from the nuclear steam system supply design, and system modifications had maintained congruity with that design. Engineering calculations supported critical system parameters and test procedure acceptance criteria. Safety evaluations were effective during system modifications. Plant operating procedures, surveillance procedures, and maintenance procedures were accurate and consistent with the design and licensing bases. NYPA personnel were aware of generic industry issues and had taken action to minimize S/RV malfunctioning. (E1.1)

Procedures pertaining to the RCIC system specified alignments, operational limits, and alarm responses that were consistent with the existing plant configuration and design bases as described in the UFSAR and design specifications. Test programs provided adequate assurance of system condition and functionality. Design calculations were consistent with applicable licensing, design, and operations documents. Modifications to the system had implemented vendor recommendations, reflected industry experience, and enhanced system reliability. Design basis performance margins have been maintained adequately. NYPA's evaluations of industry information pertaining to motor-operated valve issues were timely and acceptable. High availability and low corrective maintenance backlog evidenced effective maintenance of the system. With minor exceptions, the UFSAR accurately described the design and operation of the RCIC system. One noncited violation of NRC design control requirements of 10 CFR 50, Appendix B, Criterion III was identified. (NCV 50-333/98-05-01) (E1.2)

The team evaluated leakage past the HPCI steam admission valve and concluded that HPCI remained operable. The team's concerns regarding valve operation and temperature effects were resolved by NYPA without affecting HPCI operability, but appeared to indicate that prior NYPA reviews had been adequate but not thorough. (E2.1)

Executive Summary

The team reviewed the planned corrective actions associated with the slow pressurization of the LPCI system and concluded that while these were adequate, previous corrective actions had not resolved the problem over the previous 7 years. (E2.2)

The team concluded that NYPA had been slow to recognize and evaluate the effects of an abnormally configured system (long term operation of control room ventilation system in the accident mode) and an offnormal system condition (SLC high room temperature). No actual adverse effects were identified, although NYPA was evaluating further. (E2.3)

The team concluded that NYPA's response to an emerging plant issue (MSIV leaking into SBGT) was properly evaluated, including a well-supported operability determination for SBGT. (E2.4)

The team reviewed TS LCO entries and concluded that NYPA had resolved the various concerns adequately, including ODs when appropriate. (E2.5)

NYPA's post-trip review of a September 1996 inadvertent plant trip appeared to have missed opportunities to provide better guidance on UPS (uninterruptible power supplies) that could have aided in a 1998 UPS event. (E2.6)

NYPA adequately evaluated the environmental qualification concerns and sealed the conduits entry after water intrusion into the MCC. (E2.7)

The safety evaluation procedures provided appropriate guidance and had been revised to incorporate both current industry and NRC 10 CFR 50.59 guidance. Annual plant 50.59 reports had been properly submitted to the NRC. (E3.1)

Some nuclear safety evaluation analyses were not thorough supporting the conclusions. These observations are similar to a QA audit and the recently completed self-assessment findings in the 10 CFR 50.59 safety evaluation area. (E3.2)

The modifications reviewed by the team were acceptable and indicated high quality engineering work. (E3.3)

The team concluded that NYPA's program to address fuse control issues had been slow in implementation, that engineering work was adequate, and that fuse-related events were infrequent. (E3.4)

The team found that NYPA had undertaken proactive measures in the reassessment of their protection approach for containment electrical penetrations. However, in existing assessments, NYPA had not considered the effects of operation under LOOP+LOCA conditions, and the consequential lower short circuit currents. The failure to analyze the effects of a LOCA plus LOOP on containment penetration protection degradation due to short circuiting is a level IV violation of 10 CFR50, Appendix B , Criterion III where no written response to NRC is required (NOV 50-333/98-05-02). Corrective actions were acceptable, including modifications to install fuses and plans for modification during the upcoming outage. (E3.5)

Deviation Event Reports (DERs), Problem Identification Reports (PIDs), action plans and Work Requests (WRs) properly identified, categorized and tracked plant problems. Operability determinations were adequately performed. Engineering support for DERs, PIDs, action plans and WRs was weak in some instances. (E7.1)

NYPA QA audits were detailed, comprehensive and resulted in the identification of areas for improvement. Deficiencies were appropriately documented using the DER process, and problem identification was effective. Audit personnel were properly qualified. (E7.2)

Nuclear safety evaluations were being performed by technically qualified individuals who had received training regarding the preparation and review of safety evaluations. (E7.3)

The team concluded that there has been improvement in the self assessment process. The more recent self assessments were critical, comprehensive and sufficiently broad to identify problems and areas for improvement. Strong management endorsement and employee involvement in this program appeared to exist. (E7.4)

The NYPA process for the evaluation and distribution of industry operating experience information was timely, thorough and effective. Information applicable to NYPA was identified, documented and tracked thorough distribution, evaluation, action, and closure. (E7.5)

The team determined that NYPA properly identified and categorized issues related to human performance error. In the past year, NYPA has increased management attention on human performance errors and has a specific action plan to address human performance. Thus far, NYPA's actions have not reduced the level of human performance errors over the long term. Based on the events reviewed, the team determined that the present impact of human error on the safe operation of the unit has been acceptable. (E7.6)

NYPA adequately documented and tracked backlogs of work, including corrective maintenance and engineering support to operations. The backlogs appeared to be manageable. The team concluded that the open design engineering items had minimal safety implications and that NYPA was managing the design engineering corrective action backlog effectively. (E7.7)

TABLE OF CONTENTS

Report Details

III. Engineering

E1 Conduct of Engineering - Vertical Slice Inspections

E1.1 Automatic Depressurization System (ADS) and Reactor Pressure Relief Vertical Slice Inspection

a. Inspection Scope (93809)

The team reviewed the design and licensing bases for the automatic depressurization system (ADS) to determine the functional requirements for the system and each active component during accident and abnormal conditions. The scope included the ADS and its support systems, DC electric power and its pneumatic operating supply. The design and licensing bases of the reactor safety relief valves were reviewed.

The team reviewed the ADS and reactor pressure relief system designs and appropriate engineering calculations. The team also reviewed the acceptance criteria specified in test procedures for adequacy of supporting design calculations and engineering documents. The team reviewed selected safety evaluations associated with design modifications for conformance with the design and licensing basis.

The team reviewed the normal and emergency operation procedures to determine their consistency with the design and licensing bases, surveillance test procedures to determine that the procedures satisfied the technical specification requirements and properly reflected the system design basis, and maintenance procedures and system problem report history for Maintenance Rule system performance.

The team reviewed NYPA's actions concerning a recently discovered single failure which potentially could have resulted in loss of the redundant functions of HPCI (high pressure coolant injection) and ADS during a small break LOCA (loss of coolant accident) (LER 50-333/98-003).

b. Observations and Findings

Design and Licensing Bases

The team reviewed applicable FSAR (Final Safety Analysis Report), technical specification (TS), and General Electric Co. (GE) design documents. The values stated in TS 4.6.E.1 for safety/relief valve (S/RV) lift settings had been acceptably developed within GE NEDC-31697P.

The team reviewed the detailed system descriptions in the FSAR, the technical specification basis and GEK-34628A, which agreed with plant design information. The system design information and the team found that the design bases for ADS and reactor pressure relief were consistent with the licensing basis documents.

The team found calculations were available to confirm specific design parameters and test procedure acceptance criteria. The team sampled design engineering calculations and also reviewed the engineering calculations that support the acceptance criteria of tests and inspections. These included calculations of S/RV solenoid operated valve (SOV) coil resistance, S/RV flow rate and discharge line back pressure, S/RV discharge line vacuum breaker unseating force, correlation between S/RV leakage and suppression chamber water heatup, S/RV discharge pipe thermocouple alarm setpoint and uncertainty and S/RV pneumatic accumulators design sizing and leak rate. This sampling of calculations included a range of time periods.

Safety Evaluations

The team found that selected safety evaluations appropriately reflected the design and licensing bases for the scope of the modifications, based on review of the six safety evaluations listed at the end of the report.

Normal and Emergency Operation Procedures

The team found that the normal and emergency operating procedures for ADS and reactor pressure relief were consistent with design and licensing bases. The team reviewed two normal operating procedures, two abnormal operating procedures (AOPs), and four emergency operating procedures (EOPs). The sampled procedures conformed with the system design and licensing basis. The procedures contained additional insights from the contemporary studies of the Mark I containment, including a preferred operating sequence for the S/RVs to minimize local heating of the suppression chamber.

The team found that the ADS surveillance test procedures accurately implemented the technical specifications surveillance requirements. The team reviewed nine surveillance test procedures and found that the surveillance procedures conformed with the system design basis and implemented the technical specifications surveillance requirements.

Maintenance

The team reviewed four maintenance procedures and a technical manual. Maintenance procedures conformed with system licensing and design bases, and technical manuals.

The team reviewed a thirteen year history of Licensee Event Reports and a five year history of deviation event reports (DERs) for the ADS and the S/RVs. The plant has a history of S/RV lift setpoint drifts that reflects industry experience with these components. In addition to this generic issue, an incident occurred on February 26, 1996, in which foreign material, small metallic particles, contaminated the instrument nitrogen system used as the pneumatic supply to the containment. These particles were transported to and fouled a S/RV SOV. The issue was addressed by DERs-96-0234, 96-0239 and 96-0247.

The team found that the S/RV maintenance procedures reflected the vendor technical requirements, that the ADS and S/RVs are being monitored as required by the maintenance rule, and that the action plan for resolving S/RV SOV problems was appropriate. NYPA personnel cleaned the nitrogen system and categorized the S/RVs and the instrument nitrogen system as Maintenance Rule performance category (a)(1). The team reviewed the action plan for resolving S/RV solenoid valve problems, JTS-APL-96-010, ACTS No. 19543, Revision 1, dated February 29, 1996. The team also reviewed the Maintenance Rule basis document for the nuclear boiler system which includes the ADS. The team discussed the industry generic issues on S/RV setpoint drift with the system engineer, who was aware of the current status and findings of boiling water reactor owners' group and vendor work in this area. NYPA has made initiatives to quantify total S/RV leakage based on suppression chamber water temperature increase. NYPA has also developed a process to estimate the leakage rate through an individual S/RV through analysis of the valve discharge line temperature history. The estimate is intended to identify a leaking valve prior to its malfunction.

The team reviewed the NYPA analysis which postulated a loss of ADS through a fault and cable raceway failure and found the analysis to be effective. NYPA has implemented a design change to install a fuse in a non-1E circuit and thus prevent the potential ADS cable failure. The equipment postulated to fail included the residual heat removal (RHR) system containment cooling motor operated valves (MOVs), because their control power cables were routed through the subject cable raceway.

NYPA reported the results of their analysis as LER 50-333/98-003 and also communicated the issue to General Electric Company as a potential 10 CFR Part 21 issue. General Electric issued a service information letter, SIL No. 615, concerning the issue.

NYPA made a risk analysis of the issue which confirmed that it had a very low significance. This is because of the combination of events, all with very low probability of occurrence, needed for the postulated event. These include a small break LOCA, sustained loss of off site power and a sustained electrical fault on a 120 volt cable. The team noted that the risk analysis was conservative in that it did not credit several plant systems which are expected to be available to operators to mitigate the effects of the event. These include reactor water makeup from RCIC (reactor core isolation cooling) or control rod drive, and operation of the S/RVs through their Appendix R safe shutdown circuit. This control power circuit, although non-1E, is independent of the normal ADS power supply, control circuit, cable raceways and S/RV SOVs. Including credit for this equipment would further reduce risk contribution by this postulated failure.

The team found that NYPA's identification of a complex system interaction that potentially resulted in a loss of redundant ECCS function of HPCI and ADS required for a small break LOCA represented an insightful generic concern. Actions have been taken to modify the electrical circuit that was postulated to be a potential cause of an ADS failure. LER 50-333/98-003 is closed.

Miscellaneous Design Evaluations

The team found that NYPA's ADS system single failure evaluations constituted an in depth examination of the plant conditions as designed and provided for constructive re-examination of plant conditions. The team verified that the ADS valves' minimum voltage was adequate. The team reviewed the engineering actions related to DER-97-00701 and DER-97-01763, in terms of the analyses of an ADS single failure. The team also reviewed the adequacy of voltage to the ADS valves, by review of calculation JAF-CALC-ELEC-02609, Rev. 0, "125 VDC Station Battery "A" Voltage Drop".

The team found that the calculations of ADS valve minimum voltage did not take into account the farthest valve from the source. However, NYPA demonstrated that the addition of the cable length to the farthest valve caused a negligible voltage drop; therefore, the minimum voltage was still adequate.

c. Conclusions

The team concluded that NYPA had maintained an accurate design and licensing bases for the ADS and reactor pressure relief systems. The design basis was appropriately derived from the nuclear steam system supply design, and system modifications had maintained congruity with that design. Engineering calculations supported critical system parameters and test procedure acceptance criteria. Safety evaluations were effective during system modifications. Plant operating procedures, surveillance procedures, and maintenance procedures were accurate and consistent with the design and licensing bases. NYPA personnel were aware of generic industry issues and had taken action to minimize S/RV malfunctioning.

E1.2 Reactor Core Isolation Cooling (RCIC) Vertical Slice Inspection

a. Inspection Scope (93809)

The team reviewed and assessed the operation, testing, and maintenance of the reactor core isolation cooling (RCIC) system. The review included pertinent sections of the UFSAR, Technical Specifications (TS), system design basis document (DBD), flow diagrams and other system drawings, calculations, modifications, operating (normal and emergency), and in-service and surveillance test procedures and results. The review was multi disciplinary (mechanical, electrical, and instrumentation and control) and included a review of analyses that support system performance during normal and accident conditions, and discussions with system and design engineers. The review also included: (1) verification of the appropriateness and correctness of design assumptions; (2) confirmation that design bases were consistent with the licensing basis; and (3) verification of test adequacy.

b. Observations and Findings

Procedures

The team found that operating procedure OP-19, "Reactor Core Isolation Cooling*," specified system alignments and operational limits that were consistent with the existing plant configuration and design bases as described in Sections 4.7, and 14.5 of the Updated Final Safety Analysis Report (UFSAR) and the GE design specifications.

Alarm response procedures (ARPs) provided adequate guidance to operators in response to anticipated operational anomalies. The team identified one example, ARP 09-04 Window 22, "RCIC Isolation Trip Logic Initiated," in which the listed high ambient temperature trip settings were not consistent with the existing setpoints. NYPA initiated a DER to correct the condition. The team determined that this erroneous setpoint would not have affected NYPAs response to a trip of this annunciator. Failure to update the ARP constitutes a violation of minor significance and is not subject to formal enforcement action.

Calculations indicated that RCIC system operation during a station blackout (SBO) event would cause compartment ambient temperature to exceed the isolation trip setpoints and cause automatic isolation of the RCIC steam supply line. To prevent the loss of cooling water flow to the reactor, abnormal operating procedure AOP-49, "Station Blackout," directed operators to place the HPCI/RCIC steam line break detection circuits into the test condition, defeating the automatic isolation function. (This action had been approved by the NRC in Safety Evaluation Reports dated November 13, 1991 and June 9, 1992). TS Table 3.2.1, Action F, requires the HPCI/RCIC steam lines to be isolated manually within one hour if the automatic isolation system is inoperable. The team considered that ARP-49 could be enhanced by providing additional guidance regarding the potentially conflicting requirements to minimize delays or confusion in responding to the event. NYPA initiated DER 98-01728 to review and address the condition.

System Testing

The team reviewed engineering actions regarding consideration of instrument accuracy in surveillance test acceptance criteria, specifically for the RCIC steam, supply pressure and injection flow instrument loops. The review included issues affecting extension of the plant operating cycle to 24 months. The team determined that procedures ST-24A, "RCIC Monthly Operability Test*," and ST-24J, "RCIC Flow Rate and Inservice Test (IST)*," properly included instrument uncertainties in their acceptance criteria.

Except for containment or high energy line break isolation RCIC system functions, components were not included in the FitzPatrick Inservice Test (IST) Program. The team reviewed the accident analyses contained in the UFSAR and confirmed that RCIC is not credited for accident mitigation, although use of the system is described in the discussions of certain anticipated transients (e.g. loss of feedwater and loss of auxiliaries). The team thus agreed that the RCIC system is not within the IST program scope as defined in Section XI of the ASME Code of record for FitzPatrick. Notwithstanding, the team found that the surveillance test program applied to the RCIC system in accordance with plant technical specifications met the intent of the ASME Code (i.e. identification of component degradation prior to failure).

For those RCIC system components whose functions fell within the scope of the Code, NYPA's test methods, test periodicity, and deferred test justifications were acceptable.

The team confirmed that the limiting values of full stroke time of steam line isolation valves 13MOV-15 and 13MOV-16 were consistent with the assumptions made in NYPA's high energy line break calculations.

The limiting value of full stroke time for valve 13MOV-27 specified in procedure ST-24J, "RCIC Flow Rate and Inservice Test (IST)*," was changed from 5.0 seconds to 6.25 seconds in calculation JAF-CALC-RCIC-02103 to account for degraded voltage and differential pressure effects on motor speed. FSAR Table 7.3-1 was revised to reflect the change. The team identified that the acceptance criterion for the valve in procedure ST-1W, "HPCI and RCIC Primary Containment Isolation Valve Test (IST)*," was not revised. However, the error had been previously identified by NYPA during an IST program improvement review, an ACTS item was opened to correct the discrepancy prior to using the procedure, and no actual testing errors had resulted. This violation of minor significance is not subject to formal enforcement action.

The team reviewed aspects of the RCIC system, including surveillance test instrument uncertainty calculations, monthly operability test instrumentation, and d.c. power distribution, and found these areas to be acceptable.

<u>Design Calculations</u>

The RCIC system design calculations reviewed by the team were consistent with applicable licensing, design, and operations documents. Inputs were justified adequately, methodologies were acceptable, and design outputs were reasonable, with the following minor exception.

Calculation 98-019, "ECCS & RCIC Pump Suppression Pool NPSH," Revision B, was performed to support the planned suppression pool suction strainer replacement project. The calculation showed that the RCIC pumps have a large net positive suction head margin under worst case (SBO) conditions. However, the assumed system flowrate (400 gallons per minute - gpm) did not account for the additional 16 gpm flowrate to the lubricating oil cooling system. Nonetheless, the additional flowrate did not affect the conclusions of the calculation.

Modifications

Modifications to the RCIC system since FitzPatrick commenced commercial operation were implemented to enhance reliability or in response to generic industry issues identified in GE SILs. The team reviewed a representative sample of the modifications (See Attachment 1 of this report) and verified that design basis performance margins have not been reduced by the modifications over time.

Modification F1-89-159 re-powered RCIC enclosure exhaust fan 13FN-2A from the original nonsafety-related source to a safety-related low pressure coolant injection (LPCI) system battery and inverter. To meet the single failure criterion, selective coordination of the power circuit was required to protect the LPCI bus from a postulated fault on this nonsafety-related fan. Assuming that the other LPCI bus loads were electrically coordinated, the original modification package concluded that the existing 15 ampere circuit breaker for the fan was adequate. NYPA had identified that the assumption was incorrect, and an electrical circuit change form (ECCF-61) was added to the package to replace the existing breaker with a fused disconnect switch. However, the ECCF was not implemented when the modification was installed in 1994. Subsequently, NYPA identified the discrepancy during a design control audit and DER 95-01381 was opened to correct the technical aspects of this problem. Similar issues were not identified for other modifications of the RCIC system. Circuit coordination was achieved in 1996 when fuses were installed under modification M1-95-116.

The team reviewed the coordination study and concluded that the discrepancy was resolved acceptably. Failure to implement ECCF-61 in 1994 degraded the ability of the LPCI bus protection system to withstand a single failure of the nonsafety-related RCIC enclosure fan. The safety consequences were mitigated by the fact that: (1) LPCI bus loads consist of MOVs that function during an accident for only a brief period of time, and (2) the redundant LPCI battery bus and inverter were not subject to the postulated single failure. This represented a favorable example of problem identification and resolution by NYPA engineering. However, this nonrepetetive, licensee-identified and corrected violation of the design control requirements of 10 CFR 50, Appendix B, Criterion III is being treated as a non-cited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy.

Use of Industry Information

NRC Information Notice 96-48, "Motor-Operated Valve Performance Issues," dated August 21, 1996 informed licensees that use of published motor-actuator run (versus pullout) efficiency values in MOV sizing calculations may result in underpredicting actuator output capability. In response to the Notice, NYPA reviewed its MOV calculations and identified several valves that, while remaining operable, needed modification to restore full design capability under degraded voltage conditions.

NYPA was evaluating the impact on MOV operability of the information contained in Limitorque Technical Update 98-01, "Actuator Output Torque Calculation'" during the inspection. NYPA's evaluations were on-going, and operability determinations were being performed as required. NYPA also was taking steps to modify susceptible MOVs either with larger motors, or by reducing torque switch settings.

Recommendations regarding the RCIC system contained in GE SILs were adequately addressed through modifications, calculation updates, and operating procedure revisions. SILs reviewed by the team are listed in Attachment 1 of this report.

Maintenance

While not safety-related at FitzPatrick, the RCIC system is within the scope of the maintenance rule (10 CFR 50.65) in that: (1) some system valves are safety-related, (2) RCIC is relied upon to mitigate certain anticipated plant transients, and (3) the system is credited in emergency operating procedures. NYPA established system performance criteria, and system performance (high availability) was well within the allowable range. There were are no outstanding DERs of operational significance, control room deficiencies, or operator workarounds associated with the system, and the corrective maintenance backlog was very low (i.e. 15 items such as small packing leaks). The team reviewed selected maintenance procedures and historical work orders, and concluded that the procedures and practices were consistent with the recommendations contained in the vendor manuals.

Updated Final Safety Analysis Report

The RCIC system has been designed, installed, and operated consistent with the descriptions in Sections 4.7 and Chapter 14 (Accident Analyses) of the UFSAR. Two minor discrepancies were identified by the team and entered into NYPA's corrective action system:

- UFSAR Table 4.7-1 and Figure 4.7-3 needed to be updated to reflect an increase in analyzed suppression pool (RCIC pump suction) temperature from 140 degrees Fahrenheit to 144 degrees Fahrenheit that resulted from the reactor core power uprate to 2535 Megawatts-thermal.

- Section 4.7.5 of the UFSAR erroneously stated that suppression pool temperature did not exceed 112 degrees Fahrenheit.

This violation of minor significance is not subject to formal enforcement action.

Surveillance Testing

The team reviewed the engineering actions related to equipment surveillance and instrumentation accuracy associated with RCIC pressure and flow instrumentation loops, including a review of issues affecting the 24 month cycle extension. Documents included in the review were JAF-RPT-MISC-02577, "Assessment of Surveillance Test Instrument Uncertainty", dated 11/16/96 (page 20, 13PI-93 and

13FI-91), ST-24A, RCIC Monthly Operability Test", and ST-24J, "RCIC Flow Rate and Inservice Test (IST).

The team reviewed the NYPA assessment of surveillance testing and found the assessment and followup actions to be acceptable and appropriate. The team review of the monthly operability test acceptance criteria determined that the instrument uncertainties had been properly included. The team found that equipment surveillance and instrumentation accuracy associated with RCIC pressure and flow instrumentation loops were acceptable.

c. Conclusions

Procedures pertaining to the RCIC system specified alignments, operational limits, and alarm responses that were consistent with the existing plant configuration and design bases as described in the UFSAR and design specifications. Test programs provided adequate assurance of system condition and functionality. Design calculations were consistent with applicable licensing, design, and operations documents. Modifications to the system had implemented vendor recommendations, reflected industry experience, and enhanced system reliability. Design basis performance margins have been maintained adequately. NYPA's evaluations of industry information pertaining to motor-operated valve issues were timely and acceptable. High availability and low corrective maintenance backlog evidenced effective maintenance of the system. With minor exceptions, the UFSAR accurately described the design and operation of the RCIC system. One noncited violation of NRC design control requirements of 10 CFR 50, Appendix B, Criterion III was identified. (NCV 50-333/98-05-01)

E2 Engineering Support of Facilities and Equipment

E2.1 High Pressure Coolant Injection (HPCI) System Operability

a. Inspection Scope

In the control room the team observed operators respond to an alarm for the HPCI drain pot level. The team reviewed the HPCI system for the effect of this condition on operability.

b. Observations and Findings

The operators responded appropriately to the HPCI drain pot high level alarm. Subsequently, the team determined that the alarm had a frequency of approximately once an hour and that there was a NYPA action plan in place for the problem. The HPCI action plan noted that the drain pot alarms resulted from a leak on the HPCI steam admission valve (approximately forty times the design leak rate) and provided the scheduled corrective maintenance for the upcoming outage to correct the valve leakage.

The team addressed the following concerns.

- Long term erosion effect on the steam admission valve - NYPA determined that the valve remained operable, in that the safety function (opening stroke) of the steam admission valve was unaffected by the potential erosion.

- Elevated temperature affect on component parts (instrumentation, oil quality, bearings, etc) - The lubricating oil surrounding the HPCI shaft bearings had recently increased to a temperature (140 degrees F) in excess of the temperature that it would experience during HPCI operation. NYPA determined that this was a localized condition and would not affect the ability of the oil reservoir to lubricate components during a HPCI actuation. To better address the concern, NYPA established a preventive maintenance item to flush the HPCI shaft bearing with fresh oil to prevent excess oil breakdown and increased the number of oil analysis samples to be taken.

The team noted that the leakage and resulting alarms had existed for approximately two years and that the alarms represented a distraction of operators, which could not be resolved without a reactor outage.

The team reviewed the involvement of engineering in the resolution of the HPCI leakage problem, including a review of the corrective maintenance performed on the system during the prior two year period. Engineering provided a 1996 system engineer memo which identified that steam admission valve leakage was above the design level for the valve. There was no indication that engineering had completed detailed evaluations or measurements to resolve problems discussed in the engineering memo or subsequent problems identified by the team.

c. Conclusions

The team evaluated leakage past the HPCI steam admission valve and concluded that HPCI remained operable. The team's concerns regarding valve operation and temperature effects were resolved by NYPA without affecting HPCI operability, but appeared to indicate that prior NYPA reviews had been adequate but not thorough.

E2.2 Low Pressure Coolant Injection (LPCI) System Operability

a. Inspection Scope

In the control room the team observed that the LPCI system was pressurizing and that operators were responding by venting the residual heat removal (RHR) heat exchanger. The team evaluated the effect of the slow pressurization on LPCI system operability.

b. Observations and Findings

The operators appropriately responded to the high pressure condition in the LPCI system by venting the RHR heat exchanger. The team determined that the frequency of the alarm was approximately three times in an eight hour period, that the condition had existed for approximately seven years, and that a NYPA action plan existed, including planned corrective actions. The action plan noted that the high pressure conditions were most likely from a steam condensing line between the HPCI exhaust and the RHR heat exchanger through a leaking isolation valve. Because of the steam leak, system pressure was cycling up to approximately 200 psig; below the 350 psig relief valve setting.

The team reviewed the operability determination (OD) and action plan for this condition, which had existed in various forms for approximately 7 years. The OD concluded that the LPCI system was operable under this condition. In addition, NYPA resolved concerns raised by the team regarding potential system interactions in this condition.

The team evaluated engineering involvement in the resolution of the LPCI system pressurization problem, including a review of the corrective maintenance performed on the system during a three year period prior to the inspection. A 1996 system engineering memo addressed the above problem, and there were various maintenance-related activities to address the problem; yet, NYPA had been unable to resolve or greatly reduce the problem.

c. <u>Conclusions</u>

The team reviewed the planned corrective actions associated with the slow pressurization of the LPCI system and concluded that while these were adequate, previous corrective actions had not resolved the problem over the previous 7 years.

E2.3 Extended Abnormal Conditions

a. <u>Inspection Scope</u>

In the control room, the team noted that the control room ventilation system was operating in its post accident configuration and had been operating in this configuration since June 15, 1998. Also, the team noted that room temperature for the standby liquid control (SLC) system had exceeded the high temperature alarm setting for weeks. The team reviewed the effects of these extended abnormal conditions, i.e., long term operation on the control room ventilation system in its accident mode and the high room temperature on SLC system.

b. <u>Observations and Findings</u>

Although an acceptable operating mode, the team questioned whether evaluations or analyses had been performed to address any potential adverse effects from the extended operation of the ventilation system in this mode, e.g., on the preventive maintenance activities and schedule. The team did not identify any actual adverse effects. NYPA initiated a DER to fully evaluate this operating mode.

In the SLC instance NYPA performed an OD to confirm that the SLC system had remained operable during the extended high room temperature condition. The team reviewed the OD and found it to be acceptable.

The team noted that NYPA did not appear to have a process to recognize and evaluate possible adverse consequences of long term, offnormal system conditions and abnormally configured systems.

c. Conclusions

The team concluded that NYPA had been slow to recognize and evaluate the effects of an abnormally configured system (long term operation of control room ventilation system in the accident mode) and an offnormal system condition (SLC high room temperature). No actual adverse effects were identified, although NYPA was evaluating further.

E2.4 Plant Shutdown due to Steam Leak

a. Inspection Scope

On August 10, 1998, while the unit was at power, both trains of SBGT system were declared inoperable as a result of observed SBGT flow indications while the SBGT system was not in operation. Eventually, NYPA determined that a MSIV (main steam isolation valve) had a steam leak which needed a plant shutdown to repair. The plant shutdown occurred on August 11, 1998. The team evaluated engineering support, including root cause evaluation and corrective actions, associated with the evaluation of the steam leak.

b. Observations and Findings

NYPA operators responded to a control room alarm that both trains of SBGT system had flow indications while the SBGT system was not in operation. NYPA established a response team, wrote a DER, and performed an OD for the SBGT system. Subsequently, a second alarm occurred. On August 11, 1998, NYPA received a third alarm, staffed a team to enter the steam tunnel and discovered that the A MSIV had a moderate packing leak; the unit was then shut down to repair the packing leak.

The NRC team reviewed the OD on the SBGT system following the initial flow indications and determined that NYPA evaluations were responsive and conservative. The OD was technically valid and well supported with an appropriate assumed leakage rate. The NRC team determined that NYPA's actions to determine the source and the extent of MSIV degradation were adequate responses to a developing problem that resulted from a MSIV packing leak outside of primary containment.

c. Conclusions

The team concluded that NYPA's response to an emerging plant issue (MSIV leaking into SBGT) was properly evaluated, including a well-supported operability determination for SBGT.

E2.5 Operating Log Review

a. Inspection Scope

The team reviewed a selection of TS LCO (Limiting Conditions for Operation) entries to determine if operability determinations (ODs) were appropriate and to assess operator corrective actions.

b. Observations and Findings

Of the LCO entries reviewed and observed by the team, each of the problems was adequately resolved by NYPA, including ODs when appropriate. Nonetheless, the team identified a concern regarding the manner in which operators complied with TS 3.5.B following the NYPA discovery of missed leak testing surveillances on RHR on July 9, 1998. This concern was documented in DER 98-01517 and will be addressed by the resident inspectors as part of the review of LER 98-006.

c. Conclusions

The team reviewed TS LCO entries and concluded that NYPA had resolved the various concerns adequately, including ODs when appropriate.

E2.6 September 1996 Plant Trip Review

a. Inspection Scope

The team reviewed the engineering aspects of a post-trip review (DER-96-01060) for an inadvertent shorting of terminals that caused an automatic plant trip on September 16, 1996. The team review of the incident focused on investigation of the initiating event, the failure of RHR pump D to start, the lack of the fast transfer scheme operation, and the operator problems in getting the uninteruptible power supply (UPS) system back on line. The team review included main one line ac and dc diagrams. Engineering support of operations was reviewed for the issue of plant load rejection and run back to auxiliary load, which included the review of operating procedures.

The plant was tripped on September 16, 1996, due to an accidental jumpering of trip contacts of a protective relay. Maintenance was being performed on the relay with the plant on line and at full power. Following the tripping of the plant there was a delay in restoring power to the UPS buses, a failure of RHR pump D to start, and a lack of fast transfer of auxiliary loads.

b. Observations & Findings

Though the plant trip occurred in accordance with the plant design, the team questioned the need for an automatic trip function. The team found that the need for a trip initiation by the ground protection relay did not appear to have been reviewed by the NYPA team investigating the event.

In reviewing the function of the protective relay that initiated the plant trip, the team found that it provides for detection of ground faults when the generator step up transformer is used to feed the auxiliaries, with the main generator out of service (this is a possible plant operating mode, but seldom, if ever, used). Under these conditions, a ground fault would be of a very low level, since the system neutral is ungrounded. The ground fault magnitude would be given by the unbalance of the system charging capacitance, which the team estimated would possibly be a fraction of one ampere. This low level fault could be tolerated for a rather long period of time without incurring damage to equipment, and thus, it could be cleared by operator action. Thus, the ground fault relay did not fall under the basic tenant that automatic tripping should be initiated on system faults which could cause damage to equipment. It appeared that the function of the relay could be changed to an alarm only, thereby avoiding an accidental automatic trip. The accidental trip had safety implications, in that it challenged the performance of safety systems. NYPA agreed to review the issue and initiated ACTS number ACT-98-34881, dated August 13, 1998, to add an agenda item to the September 30, 1998, Scram Frequency Reduction Committee.

Following the plant trip, the UPS system bus was restored after an interruption of about one hour and thirteen minutes, due to poor understanding of the system on the part of the operators. However, as disclosed by a subsequent loss of power in 1998, the UPS system had poor documentation, and a readable one line diagram did not exist. Therefore, the team concluded that the evaluation performed in 1996 appeared to have missed an opportunity to provide better documentation and understanding of the UPS system.

The RHR pump D failure to start occurred due to failed circuit breaker cell switches. NYPA investigation revealed that the switches were not required in the circuit and that they could be bypassed (reference DER-96-1060). The team found that the response to this failure issue was effective. The team did not identify any concerns regarding operating procedures.

c. Conclusions

NYPA's post-trip review of a September 1996 inadvertent plant trip appeared to have missed opportunities to provide better guidance on UPS (uninterruptible power supplies) that could have aided in a 1998 UPS event.

E2.7 Water Intrusion into a Motor Control Center (MCC)

a. Inspection Scope

On July 31, 1998, HPCI was declared inoperable due to a direct current (dc) ground on the station battery bus. The HPCI auxiliary lube oil pump breaker opened due to water intrusion into MCC 71 BMCC-4 which provided power to HPCI auxiliaries. Operator response to this event was documented in NRC Inspection Report 50-333/98-04. During this inspection the team reviewed NYPA's engineering involvement and response to this event.

b. Observations and Findings

Modification 84-073 dated January 9, 1985, had sealed the top entry conduits for several safety related MCCs, including MCC 71 BMCC-4, due to environmental qualification (EQ) concerns. However, the EQ concerns were later determined by NYPA to be unwarranted.

NYPA's corrective actions to the water intrusion event was to provide new sealing for the top entry conduits, for MCC 71 BMCC-4, which took into account the previous information that the EQ concerns were no longer valid. Engineering was involved in providing input to the operability evaluation and to the sealing material specification.

c. Conclusions

NYPA adequately evaluated the environmental qualification concerns and sealed the conduits entry after water intrusion into the MCC.

E3 Engineering Programs and Procedures

E3.1 10 CFR 50.59 Safety Evaluation Program

a. Inspection Scope (37001)

The team evaluated NYPA's engineering performance implementing the 10 CFR 50.59 requirements for proposed changes, tests and experiments (CTEs), including procedures for conducting safety evaluations.

b. Observations and Findings

Procedures for Conducting Safety Reviews and Evaluations

Procedure Modification Control Manual MCM- 4, Revision 7, dated June 30, 1998, provided the guidance related to 10 CFR 50.59 implementation at FitzPatrick. The procedure was recently revised to incorporate both industry and NRC guidance related to 10 CFR 50.59 safety evaluations and clearly controlled the safety evaluation, review, and approval process. Review of NYPA's 50.59 annual report submittals to the NRC for 1994, 1995, and 1996 revealed that NYPA was submitting the 50.59 annual reports as required by 10 CFR 50.59(b)(2).

c. Conclusions

The safety evaluation procedures provided appropriate guidance and had been revised to incorporate both current industry and NRC 10 CFR 50.59 guidance. Annual plant 50.59 reports had been properly submitted to the NRC.

E3.2 Safety Evaluations

a. Inspection Scope (37001)

The quality of safety evaluations was evaluated in accordance with 10 CFR 50.59 to determine if the safety evaluations for plant modifications and procedure changes addressed all safety issues pertinent to the associated change, test, or experiment (CTE) and did not involve an unreviewed safety question (USQ). Review consisted of selected safety evaluations from the listing in the annual summary of plant 50.59s for 1996 for FitzPatrick, temporary modifications, and permanent design changes. Self-assessments and QA audits of nuclear safety evaluations were also reviewed. Plant Operations Review Committee (PORC) and Safety Review Committee (SRC) meetings were also observed where 50.59 evaluations were presented and discussed.

b. Observation and Findings

The team did not identify any significant technical errors in the safety evaluations. Safety evaluations and applicability determinations (safety evaluation screens) for CTEs were performed satisfactorily in accordance with plant procedures. No unreviewed safety questions were identified, appropriate conclusions were reached, and safety evaluations and applicability determinations were adequately made.

Even though the safety evaluations met the NYPA's procedural requirements, some minor problems existed with four safety evaluations. The team determined that these safety evaluation analyses did not thoroughly support the conclusions of the safety evaluations. The team noted that similar deficiencies were also observed in QA audit A98-01J, dated February 5, 1998 and the recently completed NYPA self-assessment of 10 CFR 50.59 safety evaluations. NYPA issued DERs to address the QA and self-assessment findings.

Lack of thoroughness in supporting conclusions was observed in the following safety evaluations:

- Minor modification safety evaluation for MI-89-036, Revision 0, dated March 21, 1990, "Emergency Diesel Generators Governor Shutdown Solenoid Control Circuit Changes." The safety evaluation failed to properly assess the potential for new failure modes introduced by the addition of a new timer, which was not in the original circuit. NYPA initiated a DER (DER-

98-01913, dated August 13, 1998) to fully evaluate the new timer's failure modes and later concluded that the failure of the timer did not introduce additional failure modes to the circuit.

- Type 1 design change safety evaluation for design change D1-95-012, Revision 0, dated January 27, 1995, "Evaluation of Pipe Supports Adjacent to 23 MOV-60." The safety evaluation was lacking in that the degraded condition of the pipe support was judged to be adequate to perform its design function based upon comparison with other pipe supports installed on seismic piping systems. The safety evaluation did not fully assess the loading conditions of the pipe support.

- Temporary modification 97-045. The safety evaluation only assessed the flow rate of one condensate pump operating. The safety evaluation did not assess the maximum flow rate if two condensate pumps were operated simultaneously.

- Torus temperature monitoring system (TTMS) modifications made in 1981 and in 1986 upgraded digital monitoring equipment associated with the TTMS. NYPA had reviewed the prior safety evaluations for these modifications in response to Generic Letter (GL) 95-02 and had found that the safety evaluations did not fully address all issues associated with the modifications. NYPA issued the DER 98-01697, dated July 27, 1998, to reevaluate the safety evaluations and to fully address the potential for software common mode failures, failure modes and effects of the software based components, effects of the human-machine interface, electromagnetic interference effects, and control equipment used to control and modify software configuration. The team determined that Revision 1 to the evaluation dated July 29, 1998, fully addressed all applicable concerns in an acceptable manner.

Even though minor problems were identified, none resulted in a safety significant deficiency in the safety analysis. Therefore, the safety evaluations were sufficient to support the plant modifications.

c. Conclusions

Some nuclear safety evaluation analyses were not thorough in supporting the conclusions. These observations are similar to a QA audit and the recently completed self-assessment findings in the 10 CFR 50.59 safety evaluation area.

E3.3 Plant Modifications

a. Inspection Scope

The team evaluated the quality of modifications based on review of nine modifications, as listed at the end of the report.

b. Observations & Findings

The team found the engineering reviews and analyses, primarily in the electrical and instrument area, to be acceptable; the safety evaluation for F1-86-029, performance monitoring system upgrade, was particularly well performed. Nonetheless, the team found that the new electrical loads added by this modification were not included in the plant loading analysis. However, this loading was of small magnitude and not connected to the safety buses, which eliminated an issue with plant safety bus loading conditions. There appeared to have been appropriate interfacing among the engineering disciplines involved.

c. Conclusions

The modifications reviewed by the team were acceptable and indicated high quality engineering work.

E3.4 Fuse Control Program

a. Inspection Scope

The team identified several corrective maintenance documents and DERs that referred to fuse problems or fuse documentation problems. The team reviewed the 1995 fuse control program action plan JSED-APL-95-008 ACTS 16386 SSC, Fuse Control. Additionally, the team reviewed Procedure AP-05.12, JSED-APL-95-008, Rev. 5, dated April 1, 1998, DES 2.3, Rev. 0, "Fuse Data Base Changes."

b. Observations and Findings

NYPA implemented its current action plan with resulting DERs in the later part of 1995 and early 1996. The team reviewed nine associated DERs and determined that the support supplied to operations by engineering for the specific items identified in the DERs was adequate. As of May 1997 the action plan was not fully implemented in the field, with significant actions to be completed. The team judged that the overall response of engineering to fuse control issues, identified in the industry during the late 1980s and early 1990s, had been generally slow. Nonetheless, based on a lack of significant fuse related events at FitzPatrick, NYPA's control over safety related fuses was determined to be adequate.

c. Conclusions

The team concluded that NYPA's program to address fuse control issues had been slow in implementation, that engineering work was adequate, and that fuse-related events were infrequent.

E3.5 Design Calculations

a. Inspection Scope

The team reviewed the design calculations in protective device and electrical penetration coordination studies. As part of a corporate initiative and in accordance the single failure criterion of IEEE 279, NYPA undertook a review of protective features for the containment electrical penetrations. The team reviewed the Containment Electrical Penetration Study, JAF-CALC-ELEC 02769.

b. Observations & Findings

The team found that the calculations for the containment electrical penetrations were generally well performed, except that they lacked the consideration of all possible plant operating modes. As a result of this, the short circuit faults that could be present during a LOCA + LOOP (loss of offsite power) event were not evaluated. The team pointed out that due to the lower capability, there would be lower magnitude short circuit faults under operation with the EDGs as sole power sources. Also, the inspector found that conductors outside the penetration were assumed to melt and open up, thereby causing the circuit to be interrupted within a time that will coordinate with the penetration withstand capability. The team noted that the conductor melting assumption was consistent with the FitzPatrick licensing basis.

NYPA revised the calculations and found that six nonsafety-related circuits would be minimally protected under the newly considered scenario of LOOP + LOCA, with an assumed single failure. NYPA follow up actions included DER 98-01910, which initially declared the affected penetrations inoperable and reported the event via ENS notification number 34640 on August 13,1998. Subsequently, an operability evaluation, dated August 18, 1998, was performed to provide for plant restart and operation until the next scheduled refueling outage (RO-13) without changes to a few of these circuits. Modifications were implemented for the remaining circuits by the addition of safety-related fuses in series. The team judged these corrective actions to be acceptable.

Further, the team found that engineering calculations for short circuit consistently lacked the evaluation of minimum short circuit levels. The shortcoming was deemed important to safety, because it prevented knowing whether or not the protective devices had adequate sensitivity to protect against all levels of short circuit faults that could occur in the electrical system. While this was of importance for faults on the safety systems, it was of even greater importance for faults in the non-safety system, since design non-safety loads are fed off safety buses. Therefore, as a minimum, the analysis should have ensured that failures in non-1E systems would not result in any perturbation to the safety systems. To address the team's questions, NYPA initiated DER actions in these areas.

The protection coordination calculations had several inconsistencies; an important one was the unsubstantiated assumption of the magnitude of impedance of branch circuit breakers. The team noticed that the coordination depicted in the calculations was, in multiple instances, dependent only on the large magnitude of breaker impedance assumed. The team found that small variations from the assumed impedance magnitude would render incorrect conclusions of the coordination evaluation. This issue affected most of the coordination studies reviewed by the team. NYPA agreed to review the studies and issued DER-98-01812.

The team was also concerned in that the protection studies were performed for conditions of maximum short circuit only. The evaluation of the maximum short circuit is always required in order to establish the ability of fault interrupting devices to adequately clear and withstand the effects of short circuits. However, as pointed out in IEEE 141, the authoritative source document for the design of industrial systems distribution, the evaluation of minimum short circuit is also required in order to establish whether or not the protective devices are set such that they will provide interruption of the fault currents within clearing times that are consistent with the protective criteria. The lack of evaluation of minimum short circuit faults prevented the assessment of the requirement that faults would be cleared within the time necessary to prevent serious damage to the protected equipment, and/or consequential damages due to fire, and to provide protection of personnel. To address this issue NYPA prepared DER-98-01907. The failure to analyze the effects of a LOCA plus LOOP on containment penetration protection degradation due to short circuiting is a level IV violation of 10 CFR50, Appendix B , Criterion III where no written response to NRC is required. (NOV 50-333/98-05-02)

The team reviewed the DC battery sizing calculations and found them adequate.

c. Conclusions

The team found that NYPA had undertaken proactive measures in the reassessment of their protection approach for containment electrical penetrations. However, in existing assessments, NYPA had not considered the effects of operation under LOOP+LOCA conditions, and the consequential lower short circuit currents. The failure to analyze the effects of a LOCA plus LOOP on containment penetration protection degradation due to short circuiting is a level IV violation of 10 CFR50, Appendix B , Criterion III where no written response to NRC is required (NOV 50-333/98-05-02). Corrective actions were acceptable, including modifications to install fuses and plans for modification during the upcoming outage.

E7 Assurance of Quality in Engineering Activities

E7.1 Plant Deficiencies

a. Inspection Scope

The team reviewed a sample of completed Deviation Event Reports (DERs), Problem Identification Documents (PIDs), NYPA action plans and Work Requests

(WRs) to evaluate the effectiveness of NYPAs root cause and operability determinations (ODs) for identified deficiencies.

b. Observations and Findings

Overall, the team determined that NYPA adequately identified and documented plant deficiencies when the PID, NYPA action plan and DER systems were considered together. However, several instances of weak implementation were identified by the team.

Engineering support for ODs, DER resolution, plant operations, corrective maintenance, work planning, and repair/replacement part equivalency calculations was reviewed and found to be mixed. For example, the engineering activities related to an August 10, 1998, SBGT system LCO entry for dual train operability were responsive with respect to establishing SBGT system operability (see Section E2.4). Nonetheless, in the same SBGT event, engineering did not determine the root cause until a steam tunnel entry revealed the MSIV leak.

The team determined that the conclusions reached in ODs were adequate but narrowly scoped. Nonetheless, the team noted that there was no evidence of any ODs performed by operations that was later reversed by engineering or plant management.

c. Conclusions

Deviation Event Reports (DERs), Problem Identification Reports (PIDs), action plans and Work Requests (WRs) properly identified, categorized and tracked plant problems. Operability determinations were adequately performed. Engineering support for DERs, PIDs, action plans and WRs was weak in some instances.

E7.2 Quality Assurance Audits

a. Inspection Scope

The team reviewed the Quality Assurance (QA) Division audit report for Inspection Test and Operating Status (A97-03J), Industry Operating Experience (OER) Review Program (A97-04J and A98-03J) and Design Control (98-06J). The qualifications of the audit team members were reviewed to verify their preparation and training to perform the audit function.

b. Observations and Finding

Audit reports were found to be thorough, deep, critical and comprehensive. The audit findings in report A98-06J were particularly insightful in the area of operability determinations. The results of this audit were indicative of a need to provide an adequate basis for operability determinations. These findings were documented using the DER process. Numerous recommendations were made for process improvement and were documented for action and tracking by initiating an Action

Commitment Tracking System (ACTS) item. The team judged that the NYPA QA audit process provided good problem identification.

The team reviewed qualification records of four NYPA auditors who participated in the above audits for compliance with appropriate audit standards. All personnel were found to be certified in accordance with ANSI N45.2.23 (Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants). The team noted that the audit personnel were also qualified in the areas of operations, inspection techniques and non-destructive testing.

c. Conclusion

NYPA QA audits were detailed, comprehensive and resulted in the identification of areas for improvement. Deficiencies were appropriately documented using the DER process, and problem identification was effective. Audit personnel were properly qualified.

E7.3 Training and Qualification for Safety Evaluations

a. Inspection Scope

The training and qualification program for individuals involved in performing 50.59 safety evaluations was reviewed.

b. Observations and Findings

Training records of six individuals of approximately 158 individuals that prepare and review safety evaluations were reviewed. These individuals had completed the required reading of MCM-4, Revision 7 for performing safety evaluations. Since Revision 7 involved minor changes, individuals performing safety evaluations were only required to read and sign a form that they had reviewed the changes contained in the revised safety evaluation procedure. The training was considered appropriate to support the 50.59 safety evaluation process. Individuals were satisfactorily trained to perform safety evaluations.

c. Conclusions

Nuclear safety evaluations were being performed by technically qualified individuals who had received training regarding the preparation and review of safety evaluations.

E7.4 Self Assessment Program

a. Inspection Scope (40500)

NYPA self assessments are is governed by administrative procedure (AP) 03.07, Internal Appraisal. The program was reviewed to determine if adequate guidance was provided for a critical self-evaluation of departmental activities. The team reviewed seven (7) self-assessment reports from the period March 1996 through April 1998 and twelve (12) reports covering the period of April 1998 to the present to determine the effectiveness of the self assessment function. The primary focus of the review was to evaluate the depth and detail of these assessments. The self assessments selected for review were predominately of engineering activities, including closeout of an NRC IFI on modification quality.

b. Observations and Findings

The team determined that the administrative procedure provided adequate guidance to personnel for the implementation of this program. Personnel are provided observation and assessment program guidelines so that each assessment will focus on how well each department is executing its critical functions.

A review of the self assessment reports covering the period from 1996 to April 1998 revealed significant weaknesses. The self assessment process was lacking the needed depth, criticality and probing to produce meaningful results. The assessment produced minimal items requiring corrective action, change or improvement in performance, standards or practices. The team concluded, as previously found by NYPA, that assessments performed during this period added little value to the NYPA effort to identify and correct deficiencies.

The team noted that from 1997 to the present a concerted effort had been made by NYPA management to enhance the self assessment program. The effort included program changes, presentations, and small group meetings with employees. Considerable emphasis was made on the need for critical assessments to identify those conditions that can be improved.

An improvement in the self assessments was noted from the period of April 1998 to the present. This improvement was evidenced by the detail with which the assessments were conducted and the identification of issues. Twelve self assessments were conducted during this period. The depth and extent of the assessments resulted in a number of significant findings which were documented with DERs. Constructive recommendations for process improvement were made using the ATS process.

c. Conclusion

The team concluded that there has been an improvement in the self assessment process. The more recent self assessments were critical, comprehensive and sufficiently broad to identify problems and areas for improvement. Strong management endorsement and employee involvement in this program appeared to exist.

E7.5 Industry Operating Experience Review

a. Inspection Scope

The team reviewed the industry operating experience (OE) program, two NYPA QA audit reports (A96-02J and A97-04J), seven quarterly assessments and NYPA evaluations of two NRC Information Notices (IN 97-78 on Crediting Operator Actions and IN 98-07, on Offsite Power Reliability Challenges from Industry Deregulation). Three OE related Deviation Event Reports (DER) were reviewed from receipt of information through closure. The team evaluated the performance of the Operations Review Group (ORG) to screen reports received for applicability to FitzPatrick, assign responsibility, and distribute the DER for disposition.

b. Observations and Findings

The team determined that NYPA's review of OE information was generally timely and effective and that [illegible] information was distributed to the responsible departments for disposition. ORG used the DER and ACTS process to assure tracking and capture of evaluation results and any corrective action. A review of DER 97-0293 (Liner Plate Corrosion in Concrete Containments), DER 96-0925 (TIP Ball Valve Roll-Pin Cracking) and DER 96-00032 (Target Rock SRV Main Disc Spring) indicated that the identification, screening, distribution, evaluation and closure process was well documented and tracked through closure.

c. Conclusion

The NYPA process for the evaluation and distribution of industry operating experience information was timely, thorough and effective. Information applicable to NYPA was identified, documented and tracked through distribution, evaluation, action, and closure.

E7.6 Human Performance

a. Inspection Scope

The team reviewed a sample of events, problems, and conditions to determine the contribution of human error. For those problems that resulted from human performance errors, the team attempted to determine if NYPA adequately identified the root cause of the errors and obtained adequate corrective action.

b. Observations and Findings

NYPA had initiated DER-98-01059 to document and trend human performance errors. The team reviewed this DER and a selection of other DERs and PIDs that resulted from human performance errors. The team determined that a relatively constant level (approximately 14 per month) of human performance errors existed in operations, maintenance, and engineering. Approximately half of the errors were historical in nature and were discovered though various NYPA corrective action process reviews. FitzPatrick management, PORC, and SRC were aware of the human performance error level and included DER-98-01059 as part of the NYPA human performance action plan. The team further determined that NYPA's sensitivity to human performance errors appeared to have increased over the previous year, which may have affected the number of incidents identified and reported to the DER system.

The team attempted to establish the significance of the human errors and identified only two of significance, both involving radiation control. The team reviewed NYPA's specific corrective actions for the two errors and the actions addressed in the NYPA DER, and determined that NYPA's actions were adequate.

The team determined that NYPA properly identified and categorized issues related to human performance error. NYPA management, PORC and SRC had reviewed detailed corrective action plans to reduce human error including the areas of HP. Based on the sample of issues reviewed, the team determined the impact of human performance errors on the safe operation of the unit had been acceptable. However, the NYPA action plan has not been successful in producing a long term reduction in human performance errors.

c. Conclusions

The team determined that NYPA properly identified and categorized issues related to human performance error. In the past year, NYPA has increased management attention on human performance errors and has a specific action plan to address human performance. Thus far, NYPA's actions have not reduced the level of human performance errors over the long term. Based on the events reviewed, the team determined that the present impact of human error on the safe operation of the unit has been acceptable.

E7.7 Corrective Action Backlog

a. Inspection Scope

The team evaluated the levels of corrective action work awaiting completion.

b. Observations and Findings

There were moderate backlogs in corrective maintenance, engineering support, and corrective actions for DERs. NYPA management was aware of the backlogs and had established goals to reduce the backlogs. The team observed PORC and SRC discussing and evaluating line management's progress and effectiveness in reducing the backlogs.

The team determined that a trend existed of a backlog of engineering support for plant operations, maintenance, and procurement, without a detailed performance indicator for NYPA to track. The bulk indicators suggest that the amount of work outstanding is approximately equal to six months backlog in most of the categories.

Resolution of outstanding conditions in the plant was determined by the team to be slow. In some instances corrective action plans have existed years without complete resolution of the problems. The team did not identify any significant safety impacts due to backlogged engineering support and the backlog appeared to be manageable.

(Closed) Inspector Followup Item 50-333/97-04-01: Adequacy of corrective action program for engineering issues. In 1996, based on NRC and QA audit findings, NYPA initiated an ACTS item concerning the adequacy of the corrective action program for design engineering issues. At that time 13 out of 21 QA-related ACTS items assigned to design engineering had remained incomplete for greater than one year, two of which concerned ineffective corrective actions. The team reviewed the current list of open ACTS items assigned to design engineering and discussed their status with the design engineering manager.

Design engineering is responsible for approximately 41% of the corrective action items associated with DERs that are greater than one year old. Monthly status reports indicated that this percentage was slowly being reduced, and none of the items were overdue. Most of these items greater than one year old tracked long-term commitments. The team observed that the items largely were administrative in nature (e.g. update various databases, enhance procedures, etc.) and did not affect safety system functionality.

c. Conclusions

NYPA adequately documented and tracked backlogs of work, including corrective maintenance and engineering support to operations. The backlogs appeared to be manageable. The team concluded that the open design engineering items had minimal safety implications and that NYPA was managing the design engineering corrective action backlog effectively.

E8 Miscellaneous Engineering Issues

E8.1 (Closed) Violation 50-333/96-07-04: 10 CFR 50, Appendix B, Criterion XI, Test Control. This violation involved inadequate test methods or acceptance criteria related to: (1) establishment of reactor protection system electrical protection assembly (RPS/EPA) calibration period, (2) surveillance testing of residual heat removal service water pumps, and (3) performance of station battery service tests. The team verified that NYPA revised the affected calculations and procedures and provided training on the violation, its causes, and the lessons learned. To prevent recurrence, procedure MCM-11A, "Preparation, Review and Approval of Modification Test Requirements," was revised to ensure that test instrument accuracy is accounted for in developing acceptance criteria. NYPA also performed an evaluation of current technical specification surveillance requirements to ensure that instrument accuracies were considered appropriately. The team also verified that instrument accuracy will be considered in establishing safety-system functional requirements in the proposed improved technical specifications for FitzPatrick. The above corrective actions were acceptable, and this violation is closed.

E8.2 (Closed) Violation 50-333/96-07-06: 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. This violation involved NYPA's failure (1) to evaluate promptly 54 design basis document open items (DDOIs) pertaining to the residual heat removal (RHR) system, and (2) to resolve recurring APRM flow bias flow transmitter calibration failures. NYPA reviewed the DDOIs and determined that none had an adverse impact on RHR system operability. About 45% of the open items already had been addressed and closed, and the balance were prioritized and entered into ACTS. The team had no concerns regarding NYPA's disposition of the remaining DDOIs. After reviewing APRM flow bias flow transmitter equipment history, NYPA reduced the instruments' calibration frequency to twelve month intervals. The team considered this corrective action to be acceptable. To prevent recurrence, NYPA conducted training on the violation, its causes, and lessons learned, issued nuclear engineering administrative procedure NEAP-38, "Design Basis Document Validation Procedure," to define responsibilities and interfaces for design basis document (DBD) validations, and assigned ownership of the DBDs to specific engineers to establish accountability for DBD content and open item closure. NYPA's corrective actions for this violation were acceptable, and this violation is closed.

E8.3 (Closed) Inspector Followup Item 50-333/97-80-01: In accordance with 10 CFR 50.65 (a)(3), the periodic assessment for balancing reliability and availability must be completed once every refueling cycle not to exceed 24 months. During the Maintenance Rule inspection conducted on September 29 through October 3, 1997, that NYPA had not yet completed the initial periodic assessment.

NYPA completed the initial periodic assessment on December 15, 1997. The assessment covered the period October 31, 1995 thru October 31, 1997. The assessment was performed using the guidance provided in NYPA's Engineering Standard 13 (ES-13), Maintenance Rule Program Periodic Assessment. The ES-13 standard followed the guidance provided in Regulatory Guide 1.160 and NUMARC 93-01 for optimizing reliability and availability of systems, structures and components (SSC).

The NYPA assessment presented data which confirmed the effectiveness of the maintenance program in minimizing system unavailability. A sampling of performance indicators for three risk significant systems (RHR, HPCI and EDG) indicated the effectiveness of the current maintenance program in minimizing unavailability.

NRC found the assessment generally acceptable. This item is closed.

E8.4 (Closed) Violation 50.333/96007-03: 10 CFR 50, Appendix B, Criterion III, Design Control. On or before October 25, 1996, the design basis was not correctly translated into procedures, the adequacy of design was not varified, and design changes were not subjected to design control measures as evidenced by review of five calculations. In three examples, NYPA used unverified assumptions in the performance of design calculations regarding spare battery capacity, effect of air injection on HPCI pump operability and use of a calculation to model a plate heat exchanger as a shell and tube heat exchanger. One example involved an error in a calculation as a result of application of wrong electrical loads to safety related batteries associated with modification F1-89-158. The fifth example was performance of a calculation with the incorrect assumption that a non-safety related component could be assumed as a limiting single failure.

Immediate corrective action was taken by NYPA to address each example identified. NYPA performed operability assessments to review the adequacy of unverified judgements. NYPA reviewed, revised and prepared new calculations to properly document the basis for the judgements made (ELEC-00426, 00427, 01417, 01418, HPCI-00840 and DHR-02535). The team reviewed the corrective action for the five examples cited and found that revision of calculations and preparation of new calculations addressed the deficiencies identified.

Long term corrective action was taken by NYPA to develop an action plan to track additional improvements in design calculations. An increased emphasis was placed on training to increase staff sensitivity to the process for utilizing and controlling computer generated calculations and analysis. Training was scheduled for engineering staff to reinforce managements expectations regarding documentation of engineering judgement.

Based on the actions taken in response to the violation, the team found this approach to be acceptable, and this violation is closed.

E8.5 (Closed) Violation 50-333/96-07-05: This violation of 10 CFR 50.59 concerned a temporary high efficiency particulate air (HEPA) filter and blower installed in the A RHR heat exchanger room without a safety evaluation. NYPA agreed that the installation should have been treated as a temporary modification, and that the appropriate reviews should have been performed and documented. The HEPA filter/blower has been removed.

The corrective actions documented in NYPA's revised violation response letter JAFP-97-0061, dated February 21, 1997, were reviewed by the team. NYPA concluded the violation was due to personnel error. The team found that the proposed change to Administrative Procedure AP-05.02, "Control of Temporary Modifications," was not accomplished because NYPA determined that the appropriate method to control temporary ventilating systems was by RES department procedure RP-RESP-02.15, "Portable Ventilation Systems," which was revised to include a prerequisite that all portable ventilation systems installed in steam leak detection system areas will require initiation of a temporary modification. The team concluded that actions taken were appropriate to address the concern, and this violation is closed.

E8.6 (Closed) Violation 50-333/96-08-01: This 10 CFR 50.59 violation concerned the operation of all four residual heat removal (RHR) system pumps in the suppression pool cooling mode for ten hours, on November 7, 1995, prior to performing a safety evaluation. Frequent, long-term operation of the RHR system either in the suppression pool cooling or test modes was determined by the NRC to constitute an unreviewed safety question (per 10 CFR 50.59) due to the increased likelihood of a malfunction due to a water hammer event.

NYPA determined that the violation was due to less than adequate supervisory methods. NYPA did not believe that the ten hour extended run of four RHR pumps constituted a special test or a procedure change which required a safety evaluation. Corrective actions documented in NYPA's letter JAFP-97-0085, dated March 7, 1997, were reviewed by the team. NYPA safety evaluation JAF-SE-95-063, "Operation of Four RHR pumps In Torus Cooling Flow Mode," concluded that the operation of the system by existing operating procedures was consistent with the license and design bases and did not pose an unreviewed safety question. Operating procedures OP-13 and OP 13B were revised to include precaution statements regarding potential water hammer and equipment damage when starting RHR pumps in an RHR loop that is not full. The team concluded that the satisfactory actions were taken, and this violation is closed.

E8.7 (Closed) Violation 50-333/97-01-02: This 10 CFR 50.59 violation concerned the installation without a safety evaluation of tie-wraps on the torus-to-drywell vacuum breaker isolation and containment vent and purge isolation valves to restrain the air operator disengaging levers. NYPA's corrective actions documented in NYPA letter dated April 28, 1997, were reviewed by the team. NYPA concluded the violation was due to the lack of adequate administrative guidance for performing 10 CFR 50.59 reviews. The team confirmed the corrective actions implemented by NYPA, including the completed nuclear safety and environmental impact screen, which concluded that no safety evaluation was required, adequately address the issue. This violation is closed.

E8.8 (Closed) Licensee Event Report (LER) 333/98-003. Refer to part Section E1.1 of this report for details on closing of this LER.

E8.9 (Closed) Inspector Followup Item 50-333/96-07-07:

Review phase two of engineering assurance assessment of modification quality. In 1996 NYPA's engineering assurance (EA) group conducted a phase one quality audit of modifications that had been implemented at FitzPatrick since 1992. The audit considered such attributes as design interfaces, test requirements, design specifications, design verification reviews, and document control, and identified 92 discrepancies. In reviewing a sample of the modifications, the NRC found additional discrepancies that had not been identified by NYPA, and questioned the depth of the reviews. This followup item was opened to review phase two of the modification quality assessment.

In phase two of the audit, NYPA reviewed safety and nonsafety-related modifications scheduled for implementation during the 1996 refueling outage (RO12). The results of the assessment were summarized in memorandum KM-96-27, dated October 30, 1996. EA identified weaknesses in field walkdowns, modification proposal forms, preliminary engineering packages, design inputs, administrative followup, and interdisciplinary reviews. ACTS items were generated for these items to track corrective actions.

To assess the effectiveness of the corrective actions, EA performed an assessment of modifications scheduled for the 1998 refueling outage (RO13). The team reviewed memorandum KM 98.28, which summarized the results of the assessment. NYPA found significant improvement in the documentation of design inputs and in attention to detail. Based on the number and type of the findings, the team concluded that NYPA's reviews were comprehensive and that the corrective actions were effectively addressing weaknesses in the development of modifications. Also, NRC inspection of modifications in this inspection found acceptable quality.

V. Management Meetings

X1 Exit Meeting Summary

The team presented the preliminary inspection results to members of NYPA staff and management at the conclusion of the inspection on August 14, 1998.

The team asked NYPA whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST PERSONS CONTACTED

NYPA
*M. Abramski, Licensing, Technical Lead
*D. Ackley, ORG Manager
*J. Boyer, Lead Systems Engineer
*P. Brozenich, Operations Manager
*R. Converse, Manager Maintenance
*T. DelGaizo, Engineering Consultant
*B. Drain, Supervisor Design Engineering
*M. Durr, I&C Supervisor Design Engineering
*K. Fetterman, Acting DE Support Manager
*T. Herrmann, Sr. Mechanical Design Engineer
*T. Landers, Technical Services Manager
*G. Lozier, Ast. Technical Services Manager
*D. Lindsey, General Manager, OPS
*J. Maurer, Manager, Support Services
*G. Mavrikis, WPO Engineering - EQRT
*H. Salmon, Vice President - Engineering
*G. Tasick, Manager, Design & Analysis
*P. Teater, Administration
*G. Thomas, Engineering Consultant
*D. Wallace, Manager, Design Engineering Support, NYPA Team Leader
*A. Zaremba, Licensing Manager

NRC
*G. Meyer, Chief, CMMEB, DRS
*E. H. Gray, Team Leader, DRS
*G. Hunegs, Sr Resident Inspector, FitzPatrick Plant
*R. Fernandes, Resident Inspector, FitzPatrick Plant

* Indicates present at the exit meeting of 8/14/98. Additional individuals were involved with the inspection or may have been at the exit meting but are not listed above.

INSPECTION PROCEDURES AND DOCUMENTS REVIEWED

AP-03.11, Revision 3, Operability Determination and AP-03.02, Deviation and Event Reporting

Safety Evaluations

JAF-SE-84-006, modification to remove the high drywell pressure permissive for ADS and to install manual inhibit switches, including panel NYPA's proposed change to the technical specifications, dated July 25, 1984 and Amendment No. 84, dated October 11, 1984

JAF-SE-86-166, Revision 5, to meet the long term, 100 day, operability requirements for ADS regarding the pneumatic supply to the accumulators

JAF-SE-90-064, replacement of S/RV pneumatic supply check valves with a valve that operates with lower differential pressure

JAF-SE-91-081, to modify the S/RV acoustic monitoring system to filter out background noise and restore its signal sensitivity

JAF-SE-92-086, to install an auxiliary shutdown panel to isolate S/RVs from the control room in the event of a fire

JAF-SE-92-095, for the pre-operational test of the S/RVs following the installation of the auxiliary shutdown

Maintenance

MP 002.04, Reactor Vessel (RV) Safety/Relief Valve (S/RV) Maintenance, Revision 16, dated April 30, 1998

MST-102.04, RV S/RV Inspection, Revision 3, dated April 2, 1996

MST-102.05, RV S/RV Setpoint Verification, Revision 4, June 4, 1997

MP-200.8, Maintenance and Replacement of Model V526 Valcor Solenoid Valves

Plant Modifications

M1-84-073, Minor Modification, 600 VAC and 125 VDC Motor Control Center Sealing of Top Entry of Conduits, 1/9/85

F1-86-029, "Performance Monitoring System Upgrade", Rev. 0, 6/24/94

D1-91-250, MSIV 90% Limit Switch Actuating Lever Replacement", Revision 0, 10/23/91

M1-93-101, Motor Replacement for 13MOV-15. DER 96-0276 and ACTS 19992 0.67 HP, 1.7 FLA, 575 VAC (Reviewed, motor data for motor rated 440/220V.) JAF-Calc-Elec-02430, Rev. 0, 10/31/97

D1-95-047, "Elec. Dist. Coord. 71MCC-331,97DS-1", Rev. 0, 3/30/95. Replace 40 A Bkr. And 30 A. fuses by 70 A bkr. And 45 A fuses, Q.A. Cat II/III

95-127, Temporary Modification, dated 8/17/95, installed 8/29/95, "Install ultrasonic FW flow equipment on FW pipes loops A & B"

F1-96-145, "Power Uprate Implementation", Rev. 0, 12/6/96, (review of electrical equipment sizing criteria)

M1-98-082, Minor Modification, "Secondary Protection for 71ACA3 Circuit 8, Rev. 0, 5/4/98

D1-98-092, "Replacement Breaker for MG set UPS AC generator", Rev. 0, 5/9/93

Design Calculations

Protective Device Coordination Study, JAF-CALC-ELEC-02016, Rev. 0, dated 1-13-93.

Primary Containment Electrical Penetration Coordination Study MV & LV Power Penetrations, JAF-CALC-ELEC-02769- Rev. 0, dated 9/25/97

125V DC Battery B Sizing and Duty Cycle, JAF-CALC-ELEC-02548, Rev 0