April 3, 2001

Mr. Theodore Sullivan
Vice President - Operations
Entergy Nuclear Northeast
James A. FitzPatrick Nuclear Power Plant
Post Office Box 41
Lycoming, NY 13093

SUBJECT: NRC'S FITZPATRICK REPORT 05000333/2000-013

Dear Mr. Sullivan:

On February 17, 2001, the NRC completed an inspection at the James A. FitzPatrick Nuclear Power Plant. The results of this inspection were discussed on March 2, 2001, with you and other members of your staff. The enclosed report presents the results of that inspection.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The NRC identified two findings that were evaluated under the risk significance determination process and were determined to be of very low safety significance (Green). These findings have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report. Furthermore, these findings were determined to involve violations of NRC requirements, but because of their very low safety significance, the violations are non-cited.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room). Should you have any questions regarding this report, please contact me at 610-337-5211.

Sincerely,

/RA/

Glenn W. Meyer, Chief
Projects Branch 3
Division of Reactor Projects

Docket No. 05000333
License No.: DPR-59

Enclosure: Inspection Report 05000333/2000-013

cc w/encl:
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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	05000333
License No.:	DPR-59
Report No.:	2000-013
Licensee:	Entergy Nuclear Northeast Post Office Box 41 Lycoming, NY 13093
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Post Office Box 41 Scriba, New York 13093
Dates:	December 31, 2000 to February 17, 2001
Inspectors:	R. A. Rasmussen, Senior Resident Inspector R. A. Skokowski, Resident Inspector J. Jang, Sr. Reactor Inspector, Radiation Safety and Safeguards Branch H. Gray, Sr. Reactor Inspector, Systems Branch R. Bhatia, Reactor Inspector, Electrical Branch D. Dempsey, Reactor Inspector, Systems Branch D. Silk, Sr. Emergency Preparedness Inspector
Approved by:	G. W. Meyer, Chief Projects Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000333/2000-013, on 12/31/2000-02/17/2001; New York Power Authority, James A. FitzPatrick Nuclear Power Plant; Operability Evaluations.

The report covers a seven-week inspection by resident inspectors, an engineering baseline inspection, a specialist review of the reactor building crane, the baseline inspection of emergency preparedness, and a specialist review of the radiological effluent performance indicators. These inspections identified two Green issues that were noncited violations (NCVs). The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process (SDP) in Inspection Manual Chapter 0609 (see Appendix 1).

Mitigating Systems

GREEN. Entergy did not properly evaluate a potentially risk significant common mode failure of the residual heat removal service water (RHRSW) and emergency service water (ESW) systems. A weld repair associated with the hinge pin in an RHRSW check valve failed and prevented proper alignment of the valve disc on the seat. The weld repair had been performed on three other RHRSW check valves and two ESW check valves, but these valves were not evaluated.

This failure to adequately implement the corrective action system for a potential common mode failure of two risk significant safety systems was evaluated using the SDP and determined to be Green (of very low safety significance) because after actual inspection of the similar valves, the systems remained operable. (Section 1R15)

GREEN. Following the evaluation of erroneous flow indications on the high pressure coolant injection system, Entergy did not adequately consider potential venting issues with other safety systems. The inspectors concluded that similar conditions to those noted on HPCI could have reasonably existed on other safety systems.

This failure to adequately implement the corrective action system for a potential issue that could reasonably impact multiple safety systems was evaluated using the SDP and determined to be Green (of very low safety significance), because after actual inspections and system venting checks, the other systems remained operable. (Section 1R15)

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Report Details

SUMMARY OF PLANT STATUS

The reactor operated at full power for the majority of the inspection period. A downpower was necessary on January 6 due to environmental fouling of the main condenser. The reactor was returned to full power the following day, but required a second, unplanned power reduction due to fouling caused by intake system cleaning operations. The movement of flow control gates in the intake structure caused loose shells to wash into the condenser, which necessitated the unplanned downpower.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors reviewed the most recently completed cold weather preparation checklist as contained in administrative procedure (AP) 12.04, "Cold Weather Preparations." The inspectors also reviewed the final safety analysis report (FSAR) for potential concerns associated with cold weather conditions to verify adequate incorporation into the procedure. In addition, the inspectors reviewed the corrective action program database and identified no significant issues associated with cold weather conditions over the last two years.

b. Findings

No findings of significance were identified.

1R02 Changes to License Conditions

a. Inspection Scope

The inspectors reviewed a sample of safety evaluations (SEs) performed by the FitzPatrick staff to verify that changes at the FitzPatrick station related to systems, structures, or components (SSCs) and procedures, as described in the FSAR, were reviewed and documented in accordance with 10 CFR 50.59. The SEs were selected from the changes performed during the last year, taking into consideration the risk significance of the change and the impact on the three reactor safety cornerstones. The inspectors also reviewed a sample of the safety reviews (SRs) or 50.59 screens associated with changes to SSCs and procedures for which the plant staff determined that a SE was not required. The review was to verify that the threshold for performing SEs was consistent with 10 CFR 50.59. The inspectors' review included eight SEs and eleven SRs. Portions of other SEs and SRs were evaluated while picking the specific sample for review and as input to determine the effectiveness of the problem identification and resolution process in the 50.59 area. The specific documents reviewed are listed in Attachment 1 to the inspection report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments

a. Inspection Scope

The inspectors performed a partial system walkdown of emergency service water system (ESW) train B and the associated emergency diesel generators (EDGs) during a planned unavailability of ESW A.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors toured several plant areas and observed conditions related to fire protection. These included:

- Emergency diesel generator rooms.
- Screenwell safety pump rooms.
- Areas with inoperable emergency lights.
- Key fire doors in the reactor building.

On January 16 Entergy had ten emergency lights out-of-service for either scheduled maintenance or emergent work, and the inspectors assessed the impact of this condition on the operators ability to safely shutdown the plant in the event of a fire.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

Based on a review of the FSAR, and the individual plant evaluation (IPE) and individual plant evaluation of external events (IPEEE), external flooding was not considered a credible event; therefore, the inspectors focused on recent deficiency and event reports

(DERs) related to flood protection equipment. Specifically, the inspectors reviewed the issues and corrective actions associated with the following DERs:

- DER 01-00440, Problems with reactor building perimeter sump pumps.
- DER 01-00609, Leakage into the west crescent room from outside groundwater.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program

a. Inspection Scope

The inspectors observed licensed operators' performance in the simulator during training scenarios, and observed the Entergy evaluation of performance following the training.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed ongoing actions and progress related to the performance improvement action plan for the ESW system. This review was prompted by a return of problems associated with system check valves. The inspectors also reviewed the scoping and risk significance classification of the safety relief valve electric lift function with respect to the maintenance rule.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work

a. Inspection Scope

The inspectors reviewed scheduling and maintenance risk associated with emergent work on the ESW system. The inspectors also reviewed previously planned work activities during a period with concurrent challenges to the high pressure coolant injection (HPCI) system and the residual heat removal service water (RHRSW) system.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

1. Residual Heat Removal Service Water System Operability

a. Inspection Scope

The inspectors reviewed the operability evaluation associated with the discovery of a failed RHRSW system check valve.

b. Findings

Entergy did not properly evaluate a potentially risk significant common mode failure of the RHRSW and ESW systems. A weld repair associated with the hinge pin in an RHRSW check valve failed and prevented proper alignment of the valve disc on the seat. The weld repair had been performed on three other RHRSW check valves and two ESW check valves, but these valves were not evaluated.

On January 10, 2001, a control room annunciator prompted operators to identify a failed RHRSW pump discharge check valve. Valve 10RHR-14B, that was designed to prevent backflow through an idle RHRSW pump and is required for system operability, was leaking past the seat. The check valve was subsequently opened and inspected, and found to have a displaced hinge pin that allowed the valve disc to twist and not seat properly.

Following the valve inspection on January 11, 2001, maintenance determined that a weld repair performed in 1993 to capture the hinge pin had worn and allowed the hinge pin to move within the valve body. Maintenance determined that the weld repair, that was recommended by the valve manufacturer, was performed on all four RHRSW discharge check valves and the two ESW pump discharge check valves at approximately the same time in 1993.

On January 18 the inspectors obtained a copy of the hinge pin failure evaluation and determined that the issue had a common mode failure possibility to the other RHRSW trains and the ESW trains. Although the maintenance engineer had been aware of this for several days, the information had not been reported to the control room or documented in the corrective action system. Once identified by the NRC, Entergy took actions to schedule and inspect the other valves. Although degradation on several of the valves was worse than expected, no other valves were to the point of failure.

The inspectors reviewed the actions taken by Entergy following the discovery of the displaced hinge pin. Although the initial discovery and repair of 10RHR-14B was completed promptly, the evaluation for the potential risk significance of a common mode failure was inadequate. The facts related to the failure should have been entered into the corrective action system and reported to operations shortly after their discovery on January 11th.

This failure to adequately implement the corrective action system when faced with a potential common mode failure of two risk significant safety systems was evaluated using the SDP and determined to be Green (of very low safety significance), because

after actual inspection of the valves, the systems remained operable. However, this failure to identify the potential common mode failure of the hinge pins is a violation of 10CFR50, Appendix B, Criterion XVI, "Corrective Actions." This violation is being treated as a non-cited violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issues associated with this violation are in the Entergy corrective action system as DER 01-948. (NCV 05000333/2000-013-01)

2. High Pressure Coolant Injection Flow Indication

a. Inspection Scope

The inspectors reviewed the operability evaluation performed following a failure of the HPCI flow controller which resulted in HPCI indicating 1400 gpm flow in standby and shutdown.

b. Findings

Following the evaluation of erroneous flow indications on the HPCI system, Entergy did not adequately consider potential venting issues with other safety systems. The inspectors concluded that similar conditions to those noted on HPCI could have reasonably existed on other safety systems.

On January 24, 2001, operators noted that HPCI indicated 1400 gpm flow with the system shutdown. An investigation revealed air in the instrument lines, and the transmitter was vented and returned to service. On January 25 the condition returned with an indication of 1100 gpm. The transmitter was vented again and more air was discovered in the sensing lines.

Entergy performed an evaluation and determined that the air was introduced into the system during the refueling outage and that the system venting had been inadequate due to the system configuration, i.e., the sensing lines were not properly installed in that they had air traps. The plant drawings called for a positive slope that would have been self venting. The transmitter venting procedures performed a minimal vent based on the self venting design of the system. Entergy also determined that although this condition had occurred following previous outages, no efforts were taken to correct the slope of the tubing. Although the air in the sensing lines produced an erroneous indication with the system in standby, Entergy concluded that as system pressure increased due to system operation, the effect of the air would have been minimal and the HPCI system would have functioned.

Following the evaluation of HPCI, Entergy did not adequately consider potential venting issues with other safety systems. In the case of HPCI, Entergy determined that a history of not identifying and correcting inadequately sloped tubing, combined with a maintenance procedure that was inadequate for use on improperly sloped tubing, resulted in inadequate venting. The inspectors concluded that similar conditions to those noted on HPCI could have reasonably existed on other safety systems. Entergy responded to the inspectors' concerns and expanded their evaluation. Although several other instances of improperly sloped tubing were identified, the instruments proved to be adequately vented.

Although factors were identified that suggested that other systems could have had similar venting issues, no Entergy actions were taken to broaden the scope of the investigation until prompted by the inspectors. Improperly vented instruments could lead to indication and control problems that could impact safety system performance. The inspectors concluded that this represented a failure to adequately implement the corrective action system when faced with a potential issue that could reasonably impact multiple safety systems. This was evaluated using the SDP and determined to be Green (of very low safety significance), because after actual inspections and system venting checks, the other systems remained operable. However, this failure to adequately evaluate the extent of system venting deficiencies is a violation of 10CFR50, Appendix B, Criterion XVI, "Corrective Actions." This violation is being treated as a non-cited violation, consistent with Section VI.A of the Enforcement Policy, issued on May 1, 2000 (65FR25368). The issues associated with this violation are in the Entergy corrective action system as DER 01-950. (NCV 05000333/2000-013-02)

3. Other Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability determinations associated with the following plant equipment challenges:

- Repetitive failures of the emergency service water system keepfill check valves.
- Numerous HPCI system discrepancies identified during scheduled system maintenance.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed permanent plant modifications selected from approved changes that had been completed or prepared for installation within the last year to verify the modification did not degrade any SSC or place the plant in an unsafe condition. The selection was based on risk significance, impact on the reactor safety cornerstones, and a representative sample of engineering disciplines and plant activities. The review included design (analyses, calculations and technical evaluations), as-installed implementation, and post-modification testing of the changes, and completeness of the documentation. As needed, discussions were held with the responsible design and system engineers, and other personnel familiar with the changes. The inspectors' review covered ten modifications which included permanent plant changes, design changes, set point changes, procedure changes, equivalency evaluations, calculations, and commercial grade dedications. Portions of other modifications were evaluated while picking the specific sample for review and as input to determine the effectiveness of the problem identification and resolution process in the

plant modification area. The specific documents reviewed are listed in Attachment 1 to the inspection report.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors observed post maintenance testing for work performed on the RHRSW system check valves and for work performed during a planned HPCI maintenance outage.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed procedures and observed portions of testing related to the following surveillance tests:

- Standby liquid control system quarterly testing.
- Emergency service water system functional testing.
- Residual heat removal pump inservice testing.
- Control rod scram time testing.

b. Findings

No findings of significance were identified.

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspectors reviewed Entergy's actions related to alert and notification system testing. Since neighboring Nine Mile Point is primarily responsible for the siren notification system, the inspector verified that Entergy was receiving, reviewing, and retaining documentation regarding the siren maintenance and testing program. The inspector also verified that Entergy was monitoring the efforts of offsite officials to implement and maintain the tone alert radio program for residents within the emergency planning zone that are outside of the siren coverage area.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation Testing

a. Inspection Scope

The inspector reviewed Entergy's commitments for ERO staffing and facility activation. Several training and qualification records were reviewed for newly qualified ERO members. Random records for previously qualified ERO members were also reviewed. Staff depth for key ERO positions was reviewed to ensure that sufficient numbers of responders were available. The procedure for initiating ERO call-in was reviewed and walked through with control room and security personnel. Documentation from weekly pager tests and recent call-in drills were reviewed for timeliness and consistency. Classroom training for activating the pagers and the tele-notification system provided to security personnel was observed by the inspector. Proposed corrective actions to enhance the reliability of the notification process was reviewed.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes

a. Inspection Scope

The inspector reviewed recent emergency plan and implementing procedure changes to determine if the changes resulted in a decrease of effectiveness of the emergency plan. The 10 CFR 50.54(q) review process was assessed. No EAL changes have been implemented since the last inspection; however, the inspector reviewed documentation to ensure that annual EAL training was provided to offsite officials.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector reviewed corrective actions identified by Entergy during quality assurance audits, surveillances, communication drills, drill reports, and regular self-assessments. DERs assigned to the EP department were also reviewed to determine significance of the issues and to determine if repeat problems were occurring. The inspector reviewed the reports for the 1999 and 2000 10CFR50.54(t) reviews to assess that the reviews met the requirements and if any repeat issues were identified.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

Safety System Unavailability, High Pressure Injection and Heat Removal Systems

a. Inspection Scope

At FitzPatrick the high pressure injection PI monitors HPCI performance and the heat removal system PI monitors RCIC. The inspector interviewed the system engineer responsible for monitoring these systems and reviewed the PI tracking records. The inspector also reviewed operating logs to verify unavailability time was appropriately recorded. The inspector reviewed records and PI reporting for the past four quarters.

b. Findings

No findings of significance were identified.

RETS/ODCM Radiological Effluent Occurrences

a. Inspection Scope

The inspector reviewed the following documents to ensure Entergy met all requirements of the performance indicator from the first quarter 1999 to the fourth quarter 2000 (8 quarters):

- monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases;
- quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases;
- associated procedures.

b. Findings

No findings of significance were identified.

Emergency Preparedness

a. Inspection Scope

The inspector reviewed the process for identifying the data that is utilized to determine the values for the three emergency preparedness PIs which are: 1) Drill and Exercise Performance, 2) Emergency Response Organization Participation, and 3) Alert and Notification System (ANS) Reliability. The review assessed data collected in 2000. Classification, notification and protective action opportunities were verified by reviewing selected scenarios. Attendance records for drill and exercise participation was reviewed. ANS data was not reviewed since this data is shared with Nine Mile Point, which was reviewed in Inspection Report 50-220&410/2000-011.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

1. Inspection findings presented in Section 1R15 of this report also had implications regarding Entergy's implementation of the corrective action system. In two instances conditions were identified on risk significant safety systems that could have reasonably co-existed on redundant trains and similar equipment. For the purpose of this inspection, these issues were dispositioned as individual violations of 10 CFR 50, Appendix B, "Corrective Actions". However, these issues add to a trend of performance issues in this area. Performance of the corrective action system will be reviewed as part of the baseline inspection program in an upcoming inspection.

   Additional items associated with the corrective action program were reviewed without findings.

2. Engineering Team Evaluation of Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed the FitzPatrick problem identification and resolution program related to selected plant modifications and safety evaluations. The review was conducted to verify that Entergy identified issues at the proper threshold and entered them in the corrective action program, and to evaluate the adequacy of the resultant corrective actions. DERs related to the 50.59 process and to design changes were reviewed to assess the scope of identified problems. Additionally, the quality assurance audits and surveillance reports for these topics were sampled. The manager of corrective actions was interviewed to determine the level at which DERs were initiated and how DERs were prioritized for action and cause analysis. Items reviewed for this area are listed in Attachment 1 to the inspection report.

b. Findings

No findings of significance were identified.

4OA5 Other

.1 Independent Spent Fuel Storage Installation (ISFSI) (IP60851)

a. Inspection Scope

The inspector reviewed activities regarding the Reactor Building Crane (RBC), which is planned to be used for the handling of fully loaded spent fuel casks. A new single failure proof trolley was installed on the existing RBC to meet the criteria set forth in NUREG-0554, Single-Failure Proof Cranes for Nuclear Power Plants.

The inspector reviewed the current annual inspection of the RBC. This included a detailed review of procedure MP-088.02, Reactor Building Crane Inspection 88CR2*, Revision 6, which described the annual, monthly, and daily inspection criteria of the RBC components, to verify the procedure followed the industry guidelines established by the American National Standards Institute (ANSI) and ASME B 30.2, Overhead and Gantry Cranes, 1983.

The inspector reviewed magnetic particle surface examinations of the RBC 125 and 20 ton hooks to verify they were performed and to determine if recordable indications were found. In addition, the inspector verified that the main and the auxiliary hoist running ropes were inspected for unusual wear, fraying, kinking, and tightness of end clamps and rope clips.

The inspector also reviewed corrective actions in response to identification of abnormal wear on the drum pinion and the main hoist bull gears during implementation of the RBC inspection procedure visual examination. This included verification that Entergy documented the issue in a DER (DER-01-00199), that the DER was properly prepared and correctly classified, and that Entergy would perform a root cause investigation.

b. Findings:

No findings of significance were identified.

4OA6 Meetings

Exit Meeting Summary

On March 2, 2001, the inspectors presented the inspection results to Mr. Sullivan and other members of his staff, who acknowledged the findings presented. Additionally, at the conclusion of the engineering inspection on February 8, 2001, the inspectors presented the results of the inspection to Mr. Sullivan, V. P. of Operations, and other members of Entergy management and staff.

The inspectors asked whether any materials examined during the inspection should be considered proprietary. Where proprietary information was identified, it was returned to Entergy after review.

During the exit, two findings of very low safety significance were discussed, which were determined to be non-cited violations (NCVs). Should Entergy elect to contest these NCVs, a written response within 30 days of the date of this Inspection Report, with the basis for the denial, should be sent to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, and the NRC Resident Inspector at the FitzPatrick facility.

PARTIAL LIST OF PERSONS CONTACTED

N. Avrankatos	Emergency Planning Coordinator
M. Colomb	General Manager
W. O'Malley	General Manager Operations
T. Sullivan	VP Operations
K. Szeluga	Emergency Planning Coordinator Assistant
G. Tasick	Licensing Manager
G. Thomas	Director Design Engineering
A. Zaremba	Director of Safety Assurance

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened
None

Opened and Closed
NCV 05000333/2000-013-01: Failure to implement the corrective action system.
NCV 05000333/2000-013-02: Failure to implement the corrective action system.

Closed
none

LIST OF BASELINE INSPECTIONS PERFORMED

71114-02	Alert and Notification System Testing
71114-03	Emergency Response Organization Augmentation Testing
71114-04	Emergency Action Level and Emergency Plan Changes
71114-05	Corrections of Emergency Preparedness Weaknesses and Deficiencies
71151	Performance Indicator Verification

LIST OF ACRONYMS USED

ANS	Alert and Notification System
ANSI	American Nuclear Standards Institute
AOV	Air-operated Valve
AP	Administrative Procedure
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
DER	Deficiency and Event Report
EAL	Emergency Action Level
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
ERO	Emergency Response Organization
ESW	Emergency Service Water
FSAR	Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
IPE	Individual Plant Evaluation
IPEEE	Individual Plant Evaluation of External Events
IR	Inspection Report
LER	Licensee Event Report
MR	Maintenance Rule
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NYPA	New York Power Authority
PI	Performance Indicator
RBC	Reactor Building Crane
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
SDP	Significance Determination Process
SE	Safety Evaluation
SR	Safety Review
SSC	System, Structure, Component

ATTACHMENT 1

Documents Reviewed during the FitzPatrick Mods & 50.59 Inspection 2/5-2/8/01

Plant Change Records

D1-99-052, “Use of Namco EA 170-121200 Limit Switches in lieu of EA-170-11100", Rev. 0
JE-00-016, “Chem Pump and Motor Replacement,” Revision 0
JE-99-086, “Power range Neutron Monitoring System 20 Volt Power Supply Replacement,” Revision 0
JE-99-039, “Circuit Modification to increase voltage to EHC Master Trip Solenoid Valves,” Revision 0
JE-00-126, “ESW Check Valve Replacement - 46ESW-13B,” Revision 0, dated September 20, 2000
JE-00-021, “46SES-2A/B/C 24" Butterfly Valve Replacements,” dated February 10, 2000
M1-97-070, “SRV Electric Lift Feature/ATWS Level 2 Setpoint Change,” Revision 0, dated April 12, 2000
JD-99-134, “Replace 12MOV-18 and Feeder Breaker for 12MOV-18,” dated April 25, 2000

Technical Evaluations

JAF-ECCF-239, “Voltage Drop Evaluation for the Main Turbine Master Trip Solenoid Valves,” Revision 1

Commercial Grade Dedication Evaluations

CG-00-024, “Upgrade Air Filter Regulators”
CG-00-023, “MOV Ball Nut Coupling, Pressure Seal Protective Ring, Lower Bearing Thrust Ring”

Design Analysis

91-024, “Addendum 3C, Load List for 4160 Volt and 600 Volt Buses,” Revision 3

Calculations

JAF-CALC-ELEC-00757, “RPS Electrical Protection Assembly for Normal Supply Feeder Trips Channel Uncertainty and Setpoint Calculation,” Revision 10
JAF-CALC-NBI-03415, “Reactor Vessel Level 2 ATWS (Anticipated Transient Without Scram) Initiation Setpoint Calculation,” Revision 0, dated January 14, 2000
JAF-CALC-NBI-03434, “Reactor Vessel High Pressure Electrical Lift Initiation Uncertainty and Setpoint Calculation,” Revision 0, dated January 31, 2000
JAF-CALC-NBI-04130, “Required Dropout Voltage For GE Type HFA Relays Used In The SRV Electric Lift Circuitry,” dated October 12, 2000
JAF-CALC-RWCU-02196, “Power Uprate Stroke Time Analysis For 12MOV-18,” Revision 2, dated October 24, 2000
JAF-CALC-RWCU-04028, “Qualification of the 12MOV-18 Valve Replacement,” Revision 0, dated March 12, 2000
JAF-CALC-RWCU-02118, “Thrust and Torque Limits Calculation for 12MOV-18,” Revision 4, dated October 27, 2000

JAF-CAL-Elec-02428, “12MOV-18 RWCU Pumps Suction Outboard Isolation Valve Motor Replacement,” Revision 1, dated September 26, 2000
JAF-CALC-NBS-02899, “02E-K5A,B - ADS Interlock Timer Relays Uncertainty and Setpoint Calculation,” Revision 1, dated October 26, 2000
JAF-CALC-NBS-04073, “SRV Electric Lift Power Supply and Indicating Light Spec. Calculation,” Revision 0, dated October 14, 2000

Setpoint Change Request

S1-00-0002, “345 kV Line Protection Tone Direct Trip Auxiliary Timer Setpoint”
A1-00-0008, “RPS Instrument Power Buses Electrical Protection Assemblies settings of Overvoltage, Undervoltage and under frequency Relays”, dated 4/26/00.
A1-00-0053, “ADS Time Delay for ADS Interlock”
A1-00-0045, “ATWS RPT/ARI Initiation Transmitters”
S1-00-0019, “Reactor Vessel High Pressure SRV Electrical Lift Initiation Setpoint”

10CFR 50.59 Safety Evaluations

JAF-SE-99-021, “The Installation of Permanent Temporary Power Jumpers between safety and non-safety MCCs during refueling and outage periods Use,” Revision 0
JAF-SE-99-027, “Digital Upgrades To The H2/O2 Containment Atmosphere Monitoring (CAM) System,” Revision 1
JAF-SE-99-043, “ Long Term Operation In Single Element Feedwater Level Control”, Revision 0
JAF-SE-00-025, “ESW Chemical cleaning,” Revision 3
JAF-SE-00-038, “RPV Nozzle N-9, Inconel 52 Overlay,” Revision 0
JAF-SE-00-049, “Core spray "B" indication evaluation,” Revision 0
JAF-SE-00-039, “12MOV-18 Actuator Closing Logic,” Revision 0, dated August 4, 2000
JAF-SE-00-018, “Replace 12MOV-18 and Feeder Breaker for 12MOV-18,” Revision 0, dated April 25, 2000
JAF-SE-99-033, “MSIV Enhancements,” Revision 0, dated January 17, 2000

10CFR 50.59 Nuclear Safety Screen Evaluations

<u>Temporary Modifications</u> (All were 50.59 screenouts)

TMod 00-084, Disable CR annunciator for eccentricity detector
TMod 01-001, Instrument Air Drier 39AD-4A2 Heater Substitution, Rev. 0, dated 1/4/01
TMod 01-002, Jumper of failed EHC cooling fan
TMod 01-003, Install blank flanges until valve repair complete
TMod 00-072, Monitor 31MCU-1A Panel Actuation of the 86 Lockout Relay
TMod 00-074, Disable Audible Alarm LPCI Indep. P/S
TMod 00-075, Jumper Added to Utilize unused Buffer Card to Display Control Rod 30-47 Position 48 Indication, dated November 21, 2000
TMod 00-079, Time Delay for Main Leads Drain Alarm and Actuation of 29MOV-101D
TMod 00-029, Capping ESW vent line, cooler 66UC-22J
TMod 00-032, Installation of Pressure Gauge on Sensing Line to 23PCV-50

LCOs (Both were 50.59 screenouts)

PLCO 01-0100, “REO, Core Spray Closing 14cps-61b and 14cps-64b”
LCO A010116 HPCI REO, and venting of flow instrumentation.

Modifications

JD 99-092, “Installation of Permanent Temporary Power Jumper,” Rev. 0.
JD 99-095, “RHRSW strainer packing”

Deviation Event Reports (DERS)

DER-01-00556, “Inappropriate Procedural Requirement,” dated February 8, 2001
DER-01-00511, “Component Upgrade Effect on Design Calculation,” dated February 6, 2001
DER-00-01227, “Component/Structure Upgrade Affects on Design Calculations,” dated April 4, 2000
DER-00–01504, “Component Upgrade Effect on Design Calculation,” dated April 24, 2000

Procedures
AP-03.03, Rev. 17, Deviation and Event Report Analysis
AP-03.11, Rev 5, Operability Determinations
AP-16.11, Rev 3, Design Change Process
MCM-4, Rev 9, Nuclear Safety and Environmental Impact Screens and Nuclear Safety Evaluations
ST-4N, “HPCI Quick Start, Inservice Test, and Transient Monitoring Test (IST)*,” Revision 45, dated June 23, 2000
OP-68, “Automatic Depressurization System*,” Revision 14, dated October 18, 2000
ST-16J1, “Control Room and Relay Room Emergency Lighting Test*,” Revision 6, dated October 29, 2000
AOP-28, “Operation During Plant Fires*,” Revision 10, dated October 29, 2000
AOP-36, “Stuck Open Relief Valves*,” Revision 11, dated October 24, 2000
ARP 09-4-2-37, “SRV Elec Lift Initiated or Bypassed,” Revision 3, dated October 25, 2000
ARP 09-5-1-46, “ECCS Div I ATTS TU Inop or Gross Fail,” Revision 3, dated October 25, 2000
OP-46A, “4160V and 600V Normal AC Power Distribution,” Revision 40
IMP-7.6, “ Neutron Monitoring System LPRM, APRM and RBM Instrument Maintenance,”Rev.31
ST-21H, “Main Turbine 24 VDC Master Trip A and B Solenoid Test,” Revision 4
ISP-175A3, “Rx Press ATWS Instrument Functional Test/Calibration (ATTS)*,” Revision 1, dated December 13, 2000
AP-01.04, “Tech Spec Related Requirements, Lists, and Tables*,” Revision 23, dated January 23, 2001
ST-22C, “ADS Logic System Functional Test*,” Revision 41, dated November 2, 2000
POT-02-3H, “Pre-Operational Test For New SRV Panels 09ECCS1-EP and 09ECCS2-EP*,” Revision 0, dated October 11, 2000
POT-02R, “Pre-Operational Test For Rx Level RPT/ARI Logic (Div I)*,” Revision 0, dated October 4, 2000
ST-39B-X14, “Type C Leak Test Of RWCU Suction Line Vlvs (IST)*,” Revision 8, dated October 20, 2000

Procedure Changes

CMM 5.1,“Configuration Management Manual-Software Quality Assurance Program,” Revision 2
ISP-94.8, “Main Turbine Generator EHC System Power Supplies Calibration,” Revision 5
ISP-94, “Reactor Protection System Electrical Protection Assembly Functional Test and Calibration,” Revision 26

Other

List of DERs written in 2000 on 50.59s and Type 1 Design Changes
Lab Reports on ESW system internal pipe deposits dated 11/24/99, 12/22/99, 6/30/00 &11/6/00.
QA Audits A98-01J, A00-02J, -05J, -06J, -14J and surveillances 2127, 2137 and 2156.
11825-1.66-128, “Elementary Diagram of Power Range Neutron Monitoring System,” ST-3,
RvC ESK-8ER,”345 kV Line EDIC No 1 Transfer Trip Receiver,” Sheet 1, Revision 13
EE-1R, ”600 Volt One Line Diagram 71MCC-131,141,252 & 262,“ Sheet 7, Revision 26.
20.3-233,”8 Inch Control Valve 21,000 Series, ”Vendor Drawing, Revision B
7.65-320B,”6 Inch Control Valve 40,000 Series, ”Vendor Drawing, Revision B
6.49-12D, “24" VVF-15B Dimensional Outline, Allis-Chalmers Stream seal BF Valve”
6.37-125, “Check Valve 2-1/4" - 18" Velan”
6.37-349, “Check Valve 3"-150 Swing Check Valve 46ESW-13A/B”
1.83-38, “ADS Elementary Diagram”
6.35-27, “Main Steam Isolation Valve Detail, Sh. 2 of 6"
6.35-47, “Packing Chamber Retrofit”
6.35-49, “Main Steam Isolation Valve Configuration Matrix”
6.35-48, “Articulated Coupling Assembly”
NUS-A017MA, “Operations and Maintenance Manual of Dual 20 Volt Power Supply,” Revision 0
JAF Vendor Manual No. N430-0159,”Dual 20 Volt Power Supply-NUSA017MA Operation and Maintenance Manual,” Revision 0
JAF Vendor Manual No. H041-0051,”Kemp Oriad Type Dryer Multi-Cam Timer,” dated 1/15/80
NEDC-32121P, “Pressure Switch/Transmitter For Two-Stage Target Rock Safety/Relief Valve,” Revision 1, dated August 1995
NEDC-31984P, “Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,” Supplements 1 and 2, dated July 1991
NEDC-31697P, “Updated SRV Performance Requirements For The James A. Fitzpatrick Nuclear Power Plant,” Revision 1, dated October 1991
NEDC-32616P, “James A. Fitzpatrick Nuclear Power Plant Anticipated Transients Without Scram (ATWS) Analysis for Recirculation Pump Trip (RPT) Setpoint Changes,” dated August 1996
NEDO-10349, “Analysis of Anticipated Transients Without Scram,” dated March 1971

APPENDIX 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	Radiation Safety	Safeguards
● Initiating Events	● Occupational	● Physical Protection
● Mitigating Systems	● Public	
● Barrier Integrity		
● Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margins.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margins and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margins but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.