November 14, 2003

Mr. Theodore Sullivan
Site Vice President
Entergy Nuclear Northeast
James A. FitzPatrick Nuclear Power Plant
Post Office Box 110
Lycoming, NY 13093

SUBJECT: FITZPATRICK NUCLEAR POWER PLANT - NRC INSPECTION REPORT 05000333/2003009

Dear Mr. Sullivan:

On October 2, 2003, the NRC completed a team inspection at the James A. FitzPatrick Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on October 2, 2003, with Mr. Brian O'Grady and other members of your staff.

This inspection examined activities conducted under your license as they relate to plant design activities and compliance with the Commission's rules and regulations. The inspection consisted of system walkdowns; examination of selected calculations, drawings, procedures, modifications, safety evaluations, surveillance tests and maintenance records; and interviews with personnel.

Based on the results of this inspection, the team identified three findings of very low safety significance (Green), two of which were determined to involve violations of NRC requirements. However, because of their very low safety significance and because the issues have been entered into your corrective action program, the NRC is treating the two issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy, issued May 1, 2000, (65FR25368). If you deny any of the non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the James A FitzPatrick facility.

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Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Systems Branch
Division of Reactor Safety

Docket No: 50-333
License No: DPR-59

Enclosure: Inspection Report 05000333/2003009

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REGION I

Docket No:	50-333
License No:	DPR-59
Report No:	050333/2003009
Licensee:	Entergy Nuclear Northeast Post Office Box 110 Lycoming, NY 13093
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	September 15 - October 2, 2003
Inspectors:	L. Scholl, Team Leader M. Barillas, Reactor Inspector F. Bower, Sr. Reactor Inspector C. Cahill, Sr. Reactor Inspector B. Norris, Sr. Reactor Inspector T. O'Hara, Reactor Inspector J. Tallieri, Reactor Inspector
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000333/2003-009, on 09/15 - 10/02/2003; James A. FitzPatrick Nuclear Power Plant, Plant Design Team Report.

The inspection was conducted by seven Region I inspectors. Two Green non-cited violations (NCVs) and one Green finding were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

Green. The team identified a non-cited violation (NCV) regarding the licensee’s failure to incorporate the assumptions of the battery loading calculations into the station’s operating procedures for a station blackout, as required by 10CFR50, Appendix B, Criterion III, Design Control.

This finding is more than minor since it is associated with the design control attribute of the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences. (Section 1R21.b.1)

Green. The team identified that the High Pressure Coolant Injection (HPCI) surveillance procedures failed to test four valves in the as-found condition because the valves were operated at least one time prior to performing the ASME in-service timing test.

This finding is more than minor since it is associated with the procedure quality attribute of the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences. (Section 1R21.b.2)

Green. The team identified a NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Actions, involving the licensee's failure to replace the 52STA switches in three of the four emergency diesel generator (EDG) output breaker cubicles in a timely manner.

This finding is more than minor since it is associated with the equipment performance attribute of the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences. (Section 4OA2.b)

B. Licensee Identified Violations.

None.

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

A pilot inspection was performed using inspection procedure (IP) 71111.DS, Plant Design. This procedure combined baseline inspection activities of IPs 71111.02, Evaluations of Changes, Tests, or Experiments; 71111.17, Permanent Plant Modifications; and, 71111.21, Safety System Design and Performance Capability. The results of this inspection are documented in a format consistent with the original individual IPs.

1R02 Evaluations of Changes, Tests, or Experiments (IP71111.02)

a. Inspection Scope

The team reviewed a sample of fourteen safety evaluations (SEs) required by 10 CFR 50.59 for changes to facility systems, structures, and components or procedures as described in the James A. FitzPatrick Updated Final Safety Analysis Report (UFSAR). The SEs were associated with recently implemented changes and with plant design activities associated with the plant systems reviewed as part of the pilot procedure. The review was conducted to verify that the changes to the facility or procedures as described in the UFSAR, and test and experiments not described in the UFSAR, were properly reviewed and documented by the licensee in accordance with 10 CFR 50.59.

The team also reviewed a sample of thirteen changes and tests for which Entergy determined that a safety evaluation was not required. This review was performed to verify that Entergy's threshold for performing safety evaluations was consistent with the requirements of 10 CFR 50.59.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed eight permanent plant modifications to verify that the design and licensing bases, and the performance capability of risk significant structure, systems, and components (SSCs) were not degraded through plant modifications. Additionally, in performing Enclosure 1 to the procedure, "Safety System Design and Performance Capability," numerous calculations, set point changes and engineering evaluations that meet the criteria for inclusion within the scope of "plant modification" were examined by the team. These documents are included in the Attachment to this report.

Plant changes were selected for review based on risk insights for the plant and included structures, systems and components associated with the initiating events, barrier integrity and mitigating systems cornerstones. The inspectors walked down selected plant systems and components, interviewed plant staff, and reviewed applicable documents including procedures, calculations, modification packages, engineering evaluations, drawings, corrective action documents, the UFSAR, technical specifications, and system design basis documents (DBDs).

The inspectors verified that selected attributes were consistent with the design and licensing bases. These attributes included component safety classification, energy requirements supplied by supporting systems, seismic qualification, instrument set-points, uncertainty calculations, electrical coordination, electrical loads analysis, and equipment environmental qualification. Design assumptions were reviewed to verify that they were technically appropriate and consistent with the UFSAR. For selected modifications the 50.59 screenings or safety evaluations were reviewed as described in section 1R02 of this report. The inspectors verified that procedures, DBDs, and the UFSAR were properly updated with revised design information and operating guidance. The inspectors also verified that the as-built configuration was accurately reflected in the design documentation and that post-modification testing was adequate to ensure that the SSCs were operable.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (IP 71111.21)

a. Inspection Scope

The team reviewed the design and performance capability of the emergency diesel generator (EDG) and high pressure coolant injection (HPCI) systems. The SDP worksheets and the individual plant examination (IPE) were reviewed to identify initiating events and core damage sequences where these systems were credited with performing accident mitigation functions. Components selected for detailed review included the HPCI turbine and pumps, the emergency diesel generators and the safety related batteries.

The capability of the HPCI system to mitigate a small break loss-of-coolant accident was reviewed in detail as was the use of the HPCI system for a source of high pressure coolant injection during loss-of-offsite power events, including station blackout conditions. The use of the HPCI system to mitigate an anticipated transient without scram (ATWS) event was also reviewed. The capability of EDGs to provide emergency alternating current (AC) power to the vital safety systems during accidents that are postulated to occur coincident with a loss-of-offsite power was reviewed. In particular, the ability of the EDGs to support the operation of emergency core cooling systems during a large break loss-of-coolant accident coincident with a complete loss-of-offsite power was examined.

Selected procedures, calculations and engineering evaluations associated with a station blackout event were also reviewed. These reviews included the abnormal operating procedures for station blackout, equipment room and suppression pool (torus) heatup calculations and vital battery capacity calculations. Battery capacity and associated load shedding actions during a station blackout are particularly important since the batteries are required to support all necessary DC loads, including the HPCI system, for a four hour duration.

In evaluating the design and functional capabilities of the selected systems the team reviewed the mechanical, electrical, and instrumentation design features of the primary system and its components. In addition, the team reviewed the adequacy of selected support systems and components that included DC power, service water, instrument air and EDG starting air.

Specific aspects of the mechanical design review included assessing the pressure and flow rate capabilities of the HPCI main and booster pumps. The team also reviewed the available net positive suction head to the pumps during alignment to the condensate storage tank (CST) or the torus under varying temperatures, and torus suction strainer debris loading conditions. The effects of elevated torus temperature conditions (during loss-of-coolant accident (LOCA) or station blackout events) on lubrication oil cooling were also reviewed to ensure bearing and pump seal temperature limits would not be exceeded. For the EDG system, the team reviewed mechanical design aspects associated with the diesel engine performance and the associated support systems that included jacket water cooling, lubricating oil, fuel oil supply and starting air supply and capacity. EDG loading calculations were reviewed to ensure the system had adequate capacity to operate the maximum expected loads under accident conditions.

Additional mechanical design aspects reviewed for both the HPCI and EDG systems included design documentation, drawings, operability determinations, and HPCI pump minimum flow protection. The team reviewed the adequacy of room heating and ventilation equipment to provide adequate equipment space environmental conditions during normal and accident conditions. The team performed field walkdowns of the accessible EDG and HPCI systems equipment to assess the material condition and verify that the installed configuration was consistent with design drawings, operating procedures and other design and vendor information. The team also assessed the adequacy of freeze protection measures to ensure that important components that were exposed to the elements would not freeze and prevent the systems from performing their safety functions.

The team reviewed the design and performance capabilities of the electrical, and instrumentation and control systems to support the operation of the EDG and HPCI systems during normal, accident and transient conditions. These reviews included verification that selected design requirements and commitments contained in the UFSAR, design documents, industry standards, and vendor information had been established and were being maintained. Documents reviewed included drawings, calculations (including instrument setpoint and loop uncertainty calculations), engineering analyses, accident analyses and design changes. The team reviewed associated component electrical testing as well as control circuit logic testing. Selected instrumentation calibration and functional tests were also reviewed. The team also reviewed the vital battery testing procedures to ensure they implemented the plant technical specification surveillance requirements, that they were consistent with IEEE Standards and that specific components, such as the HPCI turbine and EDG controls, would receive adequate voltage under all conditions. Operating experience information, including vendor information in the form of service information letters (SILs), was reviewed to ensure the licensee properly evaluated and incorporated applicable recommendations.

The team reviewed plant procedures used to operate and test the EDG and HPCI systems during normal, abnormal and accident conditions. Procedures reviewed included system operating procedures, abnormal and emergency operating procedures, alarm response cards surveillance procedures and maintenance test procedures. The team reviewed the last completed surveillance tests required by plant technical specifications for the selected systems to assess the adequacy of the procedures and to ensure data met procedure requirements or was properly dispositioned.

b. Findings

b.1 Station Blackout Battery Load Shed Assumptions not Translated into Procedures

Introduction: The team identified a non-cited violation (NCV) of very low safety significance (Green) regarding the licensee's failure to incorporate the assumptions of the battery loading calculations into the station's operating procedures for a station blackout, as required by 10CFR50, Appendix B, Criterion III, Design Control.

Description: The team identified that a battery load shedding action specified in the calculation for the "B" station battery was not properly translated into the abnormal operating procedure (AOP). Specifically, JAF-CALC-ELEC-02610, "125 VDC 'B' Station Battery Sizing & Voltage Drop," Rev. 2, assumed that operator actions to shed loads from the "B" station battery would be completed within 30 minutes of the onset of a station blackout. AOP-49, "Station Blackout," Rev. 10, stated that was actions to shed battery loads should be commenced within 30 minutes and completed expeditiously. Failure to shed the loads on the "B" station battery within 30 minutes would result in a reduction of the design margin available for the battery during the 4-hour coping period of a station blackout event. A procedure change request (PCR) had been submitted on September 12, 2002, to change the wording in the procedure and during this inspection AOP-49 was revised to be consistent with the calculation.

Analysis: This finding is more than minor since it is associated with the design control attribute of the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was evaluated using Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences.

Enforcement: 10CFR50, Appendix B, Criterion III (Design Control), requires that applicable design basis for structures, systems, and components be translated into procedures. Contrary to this requirement, the licensee failed to revise AOP-49 to be consistent with the calculation for station blackout. However, because of the very low safety significance, and because the issue was entered into the corrective action program (CR-JAF-2003-04556), it is being treated as a NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000333/2003009-01, Calculation Assumption for Station Blackout Battery Load Shed not Translated into the Procedure.

b.2 HPCI Valve Preconditioning

Introduction: While performing HPCI system valve surveillance testing, the licensee failed to test 4 valves in the "as-found" condition. Instead of timing the valve stroke on the first operation of the valve the licensee operated each valve at least once prior to timing the valve stroke. NUREG-1482 "Guidelines for Inservice Testing at Nuclear Power Plants," Section 3.5, notes that the as-found condition is generally considered to be the condition of the valve without pre-stroking or maintenance and that most inservice testing is performed in a manner that generally represents the condition of a standby component if it were actuated in the event of an accident (i.e., no preconditioning before actuation).

Description: Test procedure ST-4N, Rev. 49, HPCI Quick-Start, Inservice, and Transient Monitoring Test, operates motor operated valves 23MOV-14, 23MOV-16, 23MOV-20 and 23MOV-21 from their normal standby positions. After repositioning these valves, a subsequent step in the procedure directs the stroking and timing of the operation of the valves. In each case the second operation of the valve is recorded as the in-service test (IST) program credited test after the valve has already been operated. Thus, these valves are not stroke timed in their "as-found" position, from which they would be called upon to actuate in the event of an accident.

The inspector reviewed several copies of completed ST-4N where this preconditioning was performed, including the most recently completed test procedure performed in October 2002. In all instances reviewed, the valves operated within the required IST program times. However, the testing under this procedure, preconditions (exercises) each of these valves before the operating times are recorded.

Analysis: This finding is more than minor since it is associated with the procedure quality attribute of the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was evaluated using Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences.

Enforcement: NUREG-1482 "Guidelines for Inservice Testing at Nuclear Power Plants," Section 3.5, notes that the as-found condition is generally considered to be the condition of the valve without pre-stroking or maintenance and that most inservice testing is performed in a manner that generally represents the condition of a standby component if it were actuated in the event of an accident (i.e., no preconditioning before actuation). Contrary to the guidance of NUREG-1482, the licensee failed to test valves 23MOV-14, 23MOV-16, 23MOV-20 and 23MOV-21 in their "as-found" condition. Since NUREG-1482 is not a regulatory requirement, no violation of regulatory requirements occurred: FIN 05000333/2003009-02, Preconditioning of HPCI Valves Prior to Stroke Time Testing. This performance deficiency has been entered into the corrective action program as CR-JAF-2003-04566.

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of condition reports associated with the plant design issues, with an emphasis on the HPCI and EDG systems, to verify the licensee was identifying issues at an appropriate threshold, entering them into the corrective action program and implementing appropriate corrective actions. Condition reports reviewed are identified in Attachment 1.

b. Findings

Introduction: The team identified a NCV of very low safety significance (Green) regarding the licensee's failure to replace the 52STA switches in three of the four EDG output circuit breaker cubicles in a timely manner as required by 10CFR50, Appendix B, Criterion XVI, Corrective Actions.

Description: The team identified that actions associated with a 2000 modification to replace defective stationary switches in three of the four EDG breaker cubicles was still open, and not scheduled until the next refueling outage in 2004. Specifically, on June 8, 1999, the "A" EDG output circuit breaker tripped on reverse power. Condition report CR-JAF-1999-00976 was written to document the unexpected trip of the 4KV breaker.

The equipment failure evaluation determined the cause to be over-travel of a stationary switch (52STA) in the breaker cubicle, allowing contacts that should have remained closed to reopen. The 52STA switch was replaced on the “A” EDG. The extent of condition review identified fifty-two other 4KV breakers that used the 52STA switch, twenty-two of which were safety-related. The system engineer provided the schedule for the implementation of the modifications (the replacement of the 52STA switches). The schedule showed that many of the non-safety and all of the safety-related switches had already been replaced, with the exception of the switches for the three remaining EDGs, which were not scheduled until the next refueling outage in the fall of 2004. The inspectors found that the licensee had not taken advantage of prior opportunities to replace the switches for the remaining EDGs resulting in the untimely corrective actions for very risk significant components.

Analysis: This finding is more than minor since it is associated with the equipment performance attribute of the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was evaluated using Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” The issue was not a design or qualification deficiency that the licensee had evaluated in accordance with GL 91-18, and was determined to be of very low safety significance (Green) because it did not result in an actual loss of safety function of a single train for internal or external event initiated core damage sequences.

Enforcement: 10CFR50, Appendix B, Criterion XVI, requires that conditions adverse to quality be promptly identified and corrected. Contrary to this requirement, the licensee failed to replace the 52STA switches for the “B,” “C,” and “D” EDG breakers during the 2002 refueling outage. However, because of the very low safety significance, and because the issue was entered into the corrective action program (CR-JAF-200304556), it is being treated as a non-cited violation (NCV) consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000333/2003009-03, Untimely Replacement of Switches for EDG Output Breaker Cubicles.

4OA6 Meetings, including Exit

On October 2, 2003, the team presented the inspection results to Mr. B. O’Grady and other members of the licensee’s staff. The team verified this inspection report does not contain proprietary information.

ATTACHMENT

KEY POINTS OF CONTACT

Licensee Personnel

B. Marks	Electrical Engineering
R. Bell	Valve Test Engineer
V. Bhardway	Electrical Engineering Manager
J. Boyer	IST Supervisor
M. Brian	Engineering Support
G. Brownell	Licensing Engineer
D. Burch	Design Engineer
M. Clark	Mechanical Engineering
A. DeGracia	System Engineer
A. Holliday	Licensing Manager
D. Holliday	CR Coordinator
M. Kayhan	System Engineer
A. Khanifar	Design Engineering Manger
L. Leiter	System Engineer
O. Limpias	Engineering Director
F. Lukaczyk	Operations Support
A. Mitchell	Design Supervisor, Instrumentation and Controls
T. Page	Operations Support
J. Scranton	IST Engineer
J. Stead	Battery Engineer
P. Swinburne	Mechanical Engineering
J. Vania	System Engineer EDGs Electrical

NRC Personnel

L. Doerflein, Chief, Systems Branch
L. Cline, Senior Resident Inspector
D. Dempsey, Resident Inspector

LIST OF ITEMS OPEN, CLOSED, AND DISCUSSED

Opened and Closed

NCV 05000333/2003009-01	Calculation Assumption for Station Blackout Battery Load Shed not Translated into the Procedure (Section 1R21.b.1)
FIN 05000333/2003009-02	Preconditioning of HPCI Valves Prior to Stroke Time Testing (Section 1R21.b.2)
NCV 05000333/2003009-03	Untimely Replacement of Switches for EDG Output Breaker Cubicles (Section 4OA2.b.)

LIST OF ACRONYMS

AC	Alternating Current
AOP	Abnormal Operating Procedure
ATWS	Anticipated Transient Without Scram
CR	Condition Report
CST	Condensate Storage Tank
DBD	Design Basis Document
DC	Direct Current
EDG	Emergency Diesel Generator
HPCI	High Pressure Coolant Injection
IP	Inspection Procedure
IPE	Individual Plant Examination
IST	Inservice Test
LOCA	Loss-of-coolant Accident
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
PCR	Procedure Change Request
PI&D	Piping and Instrumentation Drawing
PRA	Probabilistic Risk Assessment
SBO	Station Blackout
SDP	Significant Determination Process
SE	Safety Evaluation
SIL	Safety Information Letter
SSC	Structures, Systems and Components
UFSAR	Updated Final Safety Evaluation Report

DOCUMENTATION REVIEWED

<u>10 CFR 50.59 Screens</u>

JAF-CALC-MISC-02875, Rev. 1, Suppression Pool Temperature Following a Small Break Accident with HPCI Operation
JAF-CALC-ELEC-02609, Rev. 2, A Station Battery Sizing/Duty Cycle & Voltage Drop Calculation
JAF-CALC-ELEC-02610, Rev. 2, B Station Battery Sizing/Duty Cycle & Voltage Drop Calculation
JAF-SE-98-025, Rev. 2, HPCI & RCIC Suppression Pool Suction Strainer Replacement
JAF-CALC-ELEC-01488, Rev. 4, 4 KV Emergency Buss Loss of Voltage, Degraded Voltage and Time Delay Relay Uncertainty and Setpoint Calculation
JD-01-146, Reserve Station Transformer T2 and T3 Tap Changes
OP-46A Procedure Change, 4160 and 600V Normal AC Power Distribution
JAF-CALC-RCIC-04472, RCIC Response Time Sensitivity Analysis for JAF During LOFW Event
JAF-CALC-DWC-04401, Establish the Design Temperature and Heat Loads in the Drywell and to Supercede Calculation 01891.01.
FSAR Change 03-009, Revise FSAR Description of Fire Pump Testing to Remove Incorrect and Unnecessary Detail
DCR 01-208, FSAR Change Request for Figure 4.7-1, Sheet 2
FSAR Change Request, Revise FSAR Figure 9.13-3 Rev. 2 to Incorporate D1-96-033
DCR 99-125, Document Change Request and UFSAR Change to Remove/Alter Detail for Quick-disconnect Fittings

<u>10 CFR 50.59 Nuclear Safety Evaluations</u>

JAF-SE-98-025, Rev 2, HPCI and Rx Core Isolation Cooling Suction Strainer
JAF-SE-93-026, Rev. 1, Modification F1-89-158, Power 27MAP from Station Batteries
JAF-SE-93-027, Rev. 0, Modification F1-89-159, Power RCIC Pump Enclosure Exhaust Fan from LPCI Inverter
JAF-SE-94-090, Rev. 0, AOP-49, Station Blackout, Rev. 3
JAF-SE-96-048, Rev. 2, Revision to FSAR to Raise Maximum Allowable Lake Temperature from 82°F to 85°F
JAF-SE-95-034, Removal of Various Containment Valves from AP-01.04, Rev. 1
JAF-SE-92-079, HPCI System Enhancements, Minor Modification M1-91-123, Rev. 1
JAF-SE-00-037, Evaluation of the Main Stack Location, Rev. 0
JAF-SE-01-009, Feedwater Heaters Maximum String Flow, Rev. 0, August 17, 2001
JAF-NSE-01-005, Rev. 0, MSIV Instrument Functional Test With Failed RPS Position Switch
JAF-SE-00-011, Rev. 0, Power Supplies - 15MOV-102(OP)/103(OP)
JAF-SE-00-003, Update FSAR to remove inconsistency concerning max. EDG room temp.
JAF-SE-01-004, Evaluation Of Control Rod Fast Withdrawal Velocity
JAF-SE-01-012, Clarification of FSAR Description of EDG Air Start System

<u>Design Changes</u>

JE-02-127, EDG Fuel Oil Transfer Pump Motor Replacement, Rev. 0
D1-94-112, HPCI Booster Pump Impeller Substitution, Rev. 2
JD-02-122, Final Feedwater Temperature Reduction
JD-02-188, HPCI/RCIC Electrical Overspeed Trip Test Equipment, Rev. 0
JD-01-146, Reserve Station Service Transformer T2 and T3 Tap Changes
MI-92-210, EDG speed controller 125V to 48 VDC isolated power supply Modification
MI-89-029, Overspeed Trip Setpoint Modification
FI-84-041, Second Level of Undervoltage Protection

Calibration Procedures

IMP-71.9, Voltage Relays Type NGV23A and NGV 23B Calibration, Rev. 6, April 17, 2001
IMP-71.9, Voltage Relays Type NGV23A and NGV 23B Calibration, Rev. 6, August 7, 2001
IMP-71.9, Voltage Relays Type NGV23A and NGV 23B Calibration, Rev. 6, September 24, 2001
ISP-90, 4KV Emergency Power (Buses 10500 and 10600) Undervoltage Relay (Loss of Voltage) Calibration, Rev. 11, October 12, 2002
ISP-90-1, 4KV Emergency Power (Buses 10500 and 10600) Undervoltage Relay (Degraded Voltage) Calibration, Rev. 12, October 19, 2002
ISP-91, 4KV Emergency Power (Buses 10500 and 10600) Undervoltage Timer (Loss of Voltage) Calibration, Rev. 8, October 11, 2002
ISP-91, 4KV Emergency Power (Buses 10500 and 10600) Undervoltage Timer (Loss of Voltage) Calibration, Rev. 8, October 16, 2002

Work Requests

98-03416-06	98-03418-05	02-07820-01	03-09197-00
98-03418-02	01-10168-02	03-09153-00	
98-03418-04	01-14245-00	03-09197-00	

Station Procedures

AOP-72, 115 KV Grid Loss, Instability, or Degradation, Rev. 2
OP-22, Diesel Generator Emergency Power, Rev. 46
OP-44, 115 KV System, Rev. 13
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