August 31, 2007

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC COMPONENT DESIGN BASES INSPECTION REPORT NO. 05000333/2007006

Dear Mr. Dietrich:

On July 19, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the James A. FitzPatrick Nuclear Power Plant. The enclosed inspection report documents the inspection results, which were discussed on July 19, 2007, with you and other members of your staff.

The inspection examined 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. In conducting the inspection, the team examined the adequacy of selected components and operator actions to mitigate postulated transients, special initiating events, and design basis accidents. The inspection also reviewed Entergy's response to selected operating experience issues. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents two findings of very low safety significance (Green), both of which involved 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 issues as non-cited violations (NCVs), in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any NCV in this report, 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, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the James A. FitzPatrick Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report No. 05000333/2007006
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	05000333/2007006
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant (JAF)
Location:	268 Lake Road Scriba, New York 13093
Inspection Period:	June 11 - July 19, 2007
Inspectors:	J. Schoppy, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader L. Cheung, Senior Reactor Inspector, DRS B. Fuller, Reactor Inspector, DRS A. Ziedonis, Reactor Inspector, DRS O. Yee, Nuclear Safety Professional Development Program (NSPDP) Participant (Trainee) L. Hajos, NRC Electrical Contractor W. Sherbin , NRC Mechanical Contractor
Approved By:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000333/2007006; 6/11/2007 - 7/20/2007; James A. FitzPatrick Nuclear Power Plant; Component Design Bases Inspection.

This inspection was conducted by a team of four NRC inspectors and two NRC contractors. Two findings of very low risk significance (Green) were identified, both of which were considered to be non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using NRC Inspection Manual Chapter (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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The team determined that Entergy did not maintain appropriate design basis calculations to ensure that the safety-related motors for the emergency service water (ESW) and standby liquid control (SLC) pumps had adequate starting voltage.

  The finding is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the ESW and SLC systems to respond to initiating events to prevent undesirable consequences. This finding is of very low significance because it did not result in the loss of operability.

  This finding has a cross-cutting aspect in the area of human performance (Resources component) because Entergy did not ensure that adequate resources were available to maintain complete, accurate and up-to-date design documentation. (IMC 0305, aspect H.2.(c)) (Section 1R21.2.1.1)

- Green. The team identified a finding of very low safety significance involving a non-cited violation of 10 CFR 50, Appendix B, Criterion III, Design Control. The team determined that Entergy failed to properly identify and evaluate the potential for vortexing in the emergency diesel generator (EDG) fuel oil transfer pump (FOTP) suction inlet piping. Specifically, Entergy's EDG fuel oil storage tank (FOST) inventory calculation did not include any allowance for suction line submergence to prevent air entrainment resulting from the effects of vortexing.

  The finding is more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the EDGs to respond to initiating events to prevent undesirable consequences. This finding is of very low significance because it did not result in the loss of safety function.

This finding has a cross-cutting aspect in the area of problem identification and resolution (PI&R) (Self - and Independent Assessments component) because Entergy did not ensure that design basis self assessments were of sufficient depth, comprehensive, appropriately objective, and self-critical. (IMC 0305, aspect P.3.(a)) (Section 1R21.2.1.2)

B. Licensee Identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (IP 71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and operator actions for review using information contained in the James A. FitzPatrick (JAF) Probabilistic Risk Assessment (PRA) and the U.S. Nuclear Regulatory Commission's (NRC) Standardized Plant Analysis Risk (SPAR) model. Additionally, the JAF Significance Determination Process (SDP) Phase 2 Notebook, Revision 2.1, was referenced in the selection of potential components and actions for review. In general, the selection process focused on components and operator actions that had a Risk Achievement Worth (RAW) factor greater than 2.0 or a Risk Reduction Worth (RRW) factor greater than 1.005. The components selected were located within both safety-related and non-safety related systems, and included a variety of components such as electrical buses, pumps, motors, diesel generators, battery chargers, strainers, transmitters, controllers, and valves. The components selected involved 11 different plant systems.

The team initially compiled a list of 50 components and 8 operator actions based on the risk factors previously mentioned. The team performed a margin assessment to narrow the focus of the inspection to 17 components and 4 operator actions. The team's evaluation of possible low design margin included consideration of original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition/equipment reliability issues. The assessment included items such as failed performance test results, significant corrective action history, repeat maintenance, Maintenance Rule (a)1 status, operability reviews for degraded conditions, NRC resident inspector input of equipment problems, system health reports and industry operating experience (OE). Consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins. The margin review of operator actions included complexity of the action, time to complete action, and extent of training of the action.

The inspection performed by the team was conducted as outlined in Inspection Procedure 71111.21. This inspection effort included walk-downs of selected components, interviews with operators, system engineers and design engineers, and reviews of associated design documents and calculations to assess the adequacy of the components to meet design bases, licensing basis and beyond design basis requirements. A summary of the reviews performed for each component, operator action, operating experience sample, and the specific inspection findings identified are discussed in the following sections of the report. Documents reviewed for this inspection are listed in the attachment.

.2 Results of Detailed Reviews

.2.1 Detailed Component and System Reviews (17 samples)

.2.1.1 A Emergency Service Water Pump Motor (46P-2A)

a. Inspection Scope

The team reviewed calculations, drawings, maintenance procedures, and vendor data to ensure that the ”A” emergency service water (ESW) pump motor was adequately designed and maintained. Specifically, the team reviewed load flow and short circuit calculations to determine whether the motor had adequate voltage for running and starting. The team also reviewed the design of protective relaying for this motor to verify that equipment was properly protected, and not susceptible to spurious tripping under expected transient and steady state loading conditions. The team also conducted several detailed walkdowns to assess the material condition of the motor and its support systems and to ensure adequate configuration control. Based on an extent-of-condition concern identified during the ESW pump motor review, the team also reviewed the standby liquid control (SLC) pump motors to determine whether the motors had adequate voltage for starting and running.

b. Findings

Introduction. The team identified a Green non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control, associated with Entergy’s failure to maintain adequate design basis calculations to ensure that the safety-related motors for the ESW and SLC pumps had appropriate starting voltage.

Description. The team noted that the FitzPatrick Updated Final Safety Analysis Report (UFSAR) Section 8.6.6.h stated “The 4000V RHR and core spray pump motors will start and accelerate at 75% and 70% voltage respectively. Other 4000V and 600V load center motors will start and accelerate at 70% to 80% voltage.” The team compared this statement with industry standards which require all electric motors to run at a minimum of 90% of rated voltage and start at 80% of rated voltage (NEMA MG-1).

The ESW and SLC pump motors are rated 575 Volts (V). The team requested Entergy’s calculation and/or analytical method supporting adequate starting voltage at the A ESW pump motor, and determined that Entergy did not have any calculations or analytical data to facilitate this design basis review. Specifically, the team determined that Entergy assumed that the maximum voltage drop from the 600V load centers to the loads was 22 volts, but had no calculations to support that value. The team also noted that Entergy assumed that the maximum voltage drop from the 4160V switchgear to the loads was 20 volts, but did not have calculations to support that value for all loads, including transformers.

Additionally, the team determined that other information provided and available for review called into question the operability of the ESW and SLC pump motors. In particular, (1) JAF-SPEC-SWS-04013, dated July 30, 1970, indicated that the ESW pump motor required a minimum of 90% of rated voltage to start, and (2) General Electric Voltage Limit Study, dated December 6, 1976, indicated that the SLC pump motor also required 90% of rated voltage to start. The team also noted that following the NRC Safety System Design and Performance Capability engineering team inspection in August 2005, Entergy initiated several condition reports (JAF-2005-3275, 2005-3356, 2005-3371, 2005-3427, and 2005-3468) associated with electrical calculation deficiencies identified during the inspection; however, to date, Entergy had not allocated adequate resources to affect needed improvements in the electrical calculations.

Following the team's questions, Entergy initiated condition report (CR) JAF-2007-2503 and CR JAF-2007-2550 to evaluate the operability for the ESW and SLC pump motors, respectively. Subsequently, Entergy performed calculations to determine the required minimum starting voltage for these safety-related motors and determined that the motors remained operable. The team reviewed Entergy's calculations and operability determinations for the ESW and SLC pump motors, and concluded that Entergy had adequately assessed the continued operability of these safety-related pumps.

<u>Analysis</u>. The performance deficiency associated with this finding was that Entergy failed to develop and maintain an adequate, comprehensive and verifiable calculation to ensure that the safety-related motors for the ESW and SLC pumps had adequate starting voltage. The issue was reasonably with Entergy's ability to foresee and correct prior to July 2007. The team determined that the issue was more than minor because it was similar to NRC IMC 0612, Appendix E, Example 3.j. In this case, the lack of an engineering calculation resulted in a condition where there was a reasonable doubt as to ESW and SLC pump motor operability. In addition, this finding is associated with the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the ESW and SLC systems to respond to initiating events to prevent undesirable consequences. The team reviewed this finding using the Phase 1 SDP worksheet for Mitigating Systems and determined that the finding was of very low safety significance (Green), because it was a design deficiency confirmed not to result in loss of operability.

This finding had a cross-cutting aspect in the area of human performance (Resources component) because Entergy did not ensure that adequate resources were available to maintain complete, accurate and up-to-date design documentation.

<u>Enforcement</u>. 10 CFR 50 Appendix B, Criterion III, Design Control, requires, in part, that design control measures be established and implemented to assure that applicable regulatory requirements and the design basis for structures, systems, and components (SSCs) are correctly translated into specifications, drawings, procedures, and instructions. In addition, the design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable

testing program. Contrary to the above, from initial plant operation until July 2007, Entergy's design control measures failed to ensure that the design basis was correctly translated into motor specifications and that the safety-related ESW and SLC pump motors could start during worse case accident conditions applicable to their safety function. Because this issue is of very low safety significance, and it was entered into Entergy's corrective action program (CAP) (condition report (CR) JAF-2007-02503 and CR JAF-2007-2550), this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000333/2007006-01, Failure to Maintain Adequate Design Basis Calculations for Safety-Related Motors)**

.2.1.2 A Emergency Diesel Generator

a. Inspection Scope

The team reviewed the "A" emergency diesel generator (EDG) to assess whether the EDG would start and run as required during postulated accident conditions to meet design bases requirements. The review included the fuel oil storage and supply, starting air, room ventilation, and jacket water (JW) cooling systems. The team reviewed calculations, fuel oil transfer analyses, starting air capability analyses, heat exchanger performance analyses, system health reports, and condition reports to verify maintenance, testing and operation of the EDG systems satisfactorily met design basis requirements. The team reviewed periodic test results and procedures to verify fuel oil levels and transfer pump performance, starting air receiver pressures, and essential service flow rates were demonstrated and maintained within acceptable limits. The team also reviewed recent eddy current test results of the JW cooler tubes. The team walked down selective accessible components and areas associated with the EDG to verify proper component alignment and the absence of observed adverse material conditions that could potentially impact system operability. The team also witnessed portions of EDG surveillance tests (STs) to independently assess EDG performance and test control.

b. Findings

Introduction. The team identified a Green NCV of 10 CFR 50, Appendix B, Criterion III, Design Control, as Entergy's EDG fuel oil storage tank (FOST) inventory calculation did not include any allowance for fuel oil suction line submergence to prevent air entrainment resulting from the effects of vortexing.

Description. The team reviewed the FitzPatrick UFSAR, Section 8.6.2.9, which states "the fuel oil system for each of the emergency AC power sources has the capacity to supply fuel to its respective emergency AC power source to operate it continuously at full load for seven days." The team also reviewed Technical Specification (TS) Surveillance Requirement (SR) 3.8.3.1, which requires that each FOST contain greater than, or equal to, 32,000 gallons of fuel oil. The team noted that the TS Bases for SR 3.8.3.1, states the SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each EDG's operation for seven days at full load.

The team found that Entergy used JAF Technical Services Systems Engineering Memorandum No. JSEM-90-0033, dated June 18, 1990, to verify sufficient onsite fuel oil inventory. This evaluation considered 1447.7 gallons as unavailable because the intake pipe is 12 inches from the bottom of the tank. The team noted that there was no allowance for potential vortexing at the inlet piping because the unavailable volume was based solely on the amount of fuel oil below the pipe surface. The team concluded that Entergy did not properly identify and evaluate the impact of the potential for vortexing in the fuel oil transfer pump (FOTP) suction line in their EDG FOST inventory calculation.

On July 10, 2007, Entergy initiated CR 2007-02490 to evaluate the condition. Entergy performed an evaluation, and determined that 3.3 inches of fuel oil above the FOTP suction piping inlet would be required to prevent vortexing. This would result in a loss of 618 gallons of fuel oil available in each FOST to meet the TS requirements. The team also noted that on June 29 and July 2, 2007, Entergy had self-identified two separate concerns associated with FOST inventory deficiencies and initiated corrective action CRs (2007-02392 and 2007-02408) to evaluate. These issues also resulted in a reduction in available fuel oil inventory margin and represent missed opportunities to identify and evaluate the vortexing concerns. In response to the FOST inventory issues, Entergy took prompt action to add additional inventory to each FOST to establish margin to the TS limit. Entergy also entered the aggregate issue into their CAP and has an action to determine reportability under 10 CFR 50.73 if there was not sufficient FOST capacity to support seven days of EDG operation at full load as required by TS 3.8.3.

Entergy's failure to account for vortexing in their tank inventory calculations resulted in an additional loss of 618 gallons of margin, leaving little or no margin to the TS-required limit, and required a detailed engineering review to determine if the available margin was reduced to less than zero in some cases. Although the TS seven day capacity and EDG operability were called into question, the team determined that there was no loss of EDG safety function based upon the EDG mission time, tank transfer capabilities, existing fuel oil on site, and procedures for obtaining additional fuel oil supply.

Analysis. The performance deficiency associated with this finding was that Entergy failed to properly evaluate and document the unusable volume of the EDG FOST needed to prevent vortexing and ingesting air into the FOTP. The issue was reasonably within Entergy's ability to foresee and correct prior to July 2007, especially given the recent industry operating experience (OE) concerning vortexing concerns. The finding is more than minor, similar to NRC IMC 0612, Appendix E, Example 3.I, because Entergy had to re-perform a calculation to determine whether the existing condition was acceptable. In addition, this finding is associated with the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of the EDGs to respond to initiating events to prevent undesirable consequences. The team reviewed this finding using the Phase 1 SDP worksheet for Mitigating Systems and determined that the finding was of very low safety significance (Green), because it did not result in the loss of safety function and was not risk significant due to external events.

This finding has a cross-cutting aspect in the area of PI&R (Self - and Independent Assessments component) because Entergy did not ensure that design basis self assessments were of sufficient depth, comprehensive, appropriately objective, and self-critical.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, Design Control, requires, in part, that measures be established to ensure that the design basis for SSCs are correctly translated into specifications, drawings, procedures, and instructions. Contrary to this requirement, from initial plant operation until July 2007, Entergy had not correctly translated FOST design bases into specifications, drawings, procedures, and instructions necessary to prevent the onset of vortexing at the intake of the FOTPs. Because this issue is of very low safety significance, and it was entered into Entergy's CAP (CR JAF-2007-02490), this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000333/2007006-02, EDG FOST Capacity Calculation Did Not Account for Vortexing)**

.2.1.3 Division B 125 VDC Battery (71SB-2) and Charger (71BC-1B)

a. Inspection Scope

The team reviewed the design and testing documents of the Division B 125 Volts direct current (Vdc) battery (71SB-2) and charger (71BC-1B) to verify that these components were adequately designed and tested, and that they could perform their required safety functions. The team reviewed battery loading and charger sizing calculations, battery float and equalizing charging voltages, battery charger undervoltage (UV) relay setpoint calibration procedures, service and performance discharge test procedures, short circuit calculations, and breaker interrupting ratings to perform the evaluation. The team also reviewed electrical schematics for selected circuits to ensure that coordination existed between the downstream and the upstream breakers and fuses. In addition, the team conducted detailed walkdowns of these components to visually inspect the physical/material condition of the battery and the charger, and confirm that the battery room and charger room temperatures were within specified design temperature ranges. During the walkdowns, the team visually inspected the battery for signs of degradation such as excessive terminal corrosion and electrolyte leaks. Finally, the team reviewed battery test results to verify that battery condition and test acceptance criteria satisfied applicable TS requirements.

b. Findings

No findings of significance were identified.

.2.1.4 High Pressure Coolant Injection Auxiliary Oil Pump (23P-150)

a. Inspection Scope

The team reviewed General Electric (GE) design specifications and the high pressure coolant injection (HPCI) turbine vendor manual to determine the design requirements for the HPCI auxiliary oil pump (AOP). The team reviewed periodic HPCI pump test results and procedures to verify AOP performance met the design requirements. The team reviewed DC voltage drop calculations to ensure vendor electrical requirements for voltage at the motor terminals were met. The team also reviewed FitzPatrick Licensee Event Report (LER) 87-10, HPCI Inoperable Due to Auxiliary Oil Pump Low Pressure, to ensure Entergy appropriately implemented corrective actions from the LER into HPCI pump operating procedures. The team reviewed a lube oil filter modification to ensure filter requirements were in accordance with vendor specifications. Finally, the team walked down the HPCI AOP and oil system to verify proper alignment and acceptable material condition.

b. Findings

No findings of significance were identified.

.2.1.5 High Pressure Coolant Injection High and Low Reactor Water Level Switches

a. Inspection Scope

The team reviewed the design basis requirements, calibration procedures, setpoint, and instrument uncertainty calculations for the HPCI high (Level 8) and low (Level 2) reactor water level switches. For Level 2, the team reviewed four reactor water level transmitters (02-3LT-72A, B, C, D) which provide signals to four trip units (02-3STU-273A, B, C, D), and for Level 8, the team evaluated two reactor water level transmitters (02-3LT-83C, D) which provide signals to two trip units (02-3MTU-283C, D). The team reviewed the instrument loop and logic diagrams associated with these instruments to verify their design adequacy. The team also reviewed the calibration records of selected instruments to verify that Entergy maintained the instruments within the acceptable accuracies and performed the calibrations in accordance with the station procedures. In addition, the team conducted a walkdown of the transmitters mounted on the instrument racks to assess their physical/material condition.

b. Findings

No findings of significance were identified.

.2.1.6 4KV to 600V Transformer T13 (71T-13)

a. Inspection Scope

Transformer T13 (4kV/600V) supplies power to engineered safety feature low voltage buses. The team reviewed calculations, drawings, maintenance procedures, and vendor data to ensure that transformer T13 was adequately designed and maintained. Specifically, the team reviewed load flow calculations to verify that the voltage at the loads was applied within their design ratings. The team also reviewed the design of protective relaying for this transformer to ensure that equipment was properly protected, and not susceptible to spurious tripping under expected transient and steady state loading conditions. Finally, the team conducted a detailed walkdown to assess the physical/material condition of the accessible portions of the transformer and to ensure adequate configuration control.

b. Findings

No findings of significance were identified.

.2.1.7 Torus Purge Isolation Valve (27AOV-117)

a. Inspection Scope

The team reviewed the ability of the torus purge isolation air operated valve (AOV) to operate during design basis events, transient and accident conditions. This AOV has a safety function in the closed position to provide primary containment isolation, and can be manually open for venting of containment under severe accident conditions. The team reviewed design basis calculations and severe accident venting calculations to verify that the valve would operate in both directions as required. Additionally, the team reviewed drawings, inservice test (IST) results, applicable design basis documents (DBDs), and health reports to assess the capability of the valve to operate as designed. The team conducted detailed system walkdowns to assess the material condition of the AOV and vent piping and to ensure adequate configuration control. The team also discussed AOV performance and trending with the AOV engineer and design engineers to ensure that the AOV could function as designed. Finally, the team reviewed NRC Generic Letter (GL) 89-16, Installation of a Hardened Wetwell Vent, and Entergy's response to GL 89-16, to ensure that plant configuration and procedural controls were consistent with Entergy's current licensing basis.

b. Findings

No findings of significance were identified.

2.1.8 A Emergency Service Water Pump (46P-2A)

a. Inspection Scope

The team reviewed the ESW design basis document (DBD), drawings, calculations, procedures, STs and modifications. The team reviewed these documents to ensure that the pump was capable of meeting its design basis requirements, with consideration for allowable pump degradation, net positive suction head (NPSH) requirements, and strainer clogging affects. The team interviewed engineers, reviewed system health reports and related CRs, and performed detailed walkdowns to assess the current condition of the pump. The team reviewed room cooler thermal performance test procedures and results for room coolers supplied by the ESW system to ensure that emergency core cooling system (ECCS) room temperatures could be maintained within design limits.

b. Findings

No findings of significance were identified.

.2.1.9 High Pressure Coolant Injection Flow Controller (23FIC-108)

a. Inspection Scope

The team reviewed the design basis requirements and design documents associated with the HPCI pump discharge flow controller (23FIC-108) to verify its design adequacy. The flow controller provides a signal to the HPCI turbine steam inlet control valve for HPCI pump speed control. The team reviewed Calculation JAFG-CALC-PHCI-04140, HPCI Flow Instrument Loop Uncertainty Analysis, to verify that all uncertainty parameters for each instrument within the control loop were accounted for and the instruments met the accuracy requirements. The team also reviewed the calibration procedure, and the results of the last three calibrations to verify that all devices within the instrument loop were properly calibrated within the required interval, and in accordance with station procedures. Finally, the team conducted a detailed walkdown to assess the physical/material condition of the accessible portions of the HPCI flow controller and to ensure adequate configuration control.

b. Findings

No findings of significance were identified.

.2.1.10 Residual Heat Removal Heat Exchanger Service Water Outlet (10MOV-89B)

a. Inspection Scope

The team reviewed the ability of the residual heat removal (RHR) heat exchanger service water outlet motor-operated valve (MOV) to operate during design basis events, transient and accident conditions. This MOV is required to open to allow heat removal from the RHR system during certain accident conditions. The team reviewed calculations including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis to verify that the valve would operate as required. Additionally, the team reviewed drawings, IST results, associated DBDs, and RHR system health reports to assess the capability of the valve to operate as designed. The team conducted detailed system walkdowns to assess the material condition of the MOV and RHR service water piping and to ensure adequate configuration control. Finally, the team discussed MOV performance and trending with the MOV engineer and design engineers to ensure the valve could function as designed.

b. Findings

No findings of significance were identified.

.2.1.11 4KV Vital Bus 10500 Undervoltage Relays

a. Inspection Scope

The team reviewed drawings, logic diagrams, and vendor manuals to verify that the degraded grid undervoltage (UV) relays could perform their intended function. The team reviewed the adequacy and appropriateness of design assumptions and calculations related to the UV protection to assure proper operation. On a sample basis, the team reviewed maintenance and test procedures to verify the associated acceptance criteria ensured that the UV relays were capable of performing their intended function.

b. Findings

No findings of significance were identified.

.2.1.12 High Pressure Coolant Injection Valve (23MOV-19)

a. Inspection Scope

The team reviewed the ability of the HPCI injection MOV to operate during design basis events, transient and accident conditions. The MOV has a safety function in the open position to provide an injection path to the “A” feedwater header, as well as a safety function in the closed position for outboard containment isolation. The team reviewed calculations including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis to verify that the valve would operate in both directions as required. Additionally, the team reviewed drawings, IST results, associated DBDs, and HPCI system health reports to assess the capability of the valve to operate as designed. The team conducted several detailed system walkdowns, including torus room inspections, to assess the material condition of the MOV and HPCI pump discharge piping and to ensure adequate configuration control. The team also discussed MOV performance and trending with the MOV engineer and design engineers to ensure the MOV could function as designed. The team reviewed Entergy’s disposition and associated corrective actions for applicable industry operating experience (OE), including Regulatory Issue Summary (RIS) 01-015, Performance of DC-Powered Motor-Operated Valve Actuators, to independently assess Entergy's evaluation and corrective actions.

b. Findings

No findings of significance were identified.

.2.1.13 Residual Heat Removal Torus Cooling Isolation Valve (10MOV-39B)

a. Inspection Scope

The team reviewed the ability of the residual heat removal torus cooling isolation valve to operate during design basis events, transient and accident conditions. This MOV has a safety function in the closed position to ensure maximum low pressure coolant injection (LPCI) flow is initially provided to the reactor vessel upon receipt of a LPCI injection signal, and a subsequent safety function in the open position to provide adequate torus cooling during design basis events. The team reviewed calculations including required thrust, degraded voltage, maximum differential pressure, and valve weak link analysis to verify that the valve would operate in both directions as required. Additionally, the team reviewed drawings, IST results, associated DBDs, and RHR system health reports to assess the capability of the valve to operate as designed. The team also conducted several detailed system walkdowns to assess the material condition of the MOV and RHR discharge piping and to ensure adequate configuration control. Finally, the team discussed MOV performance and trending with the MOV engineer and design engineers to ensure that the valve could function as designed.

b. Findings

No findings of significance were identified.

.2.1.14 Diesel Driven Fire Pump (76P-4)

a. Inspection Scope

The team reviewed design basis documents, including hydraulic calculations and drawings, to ensure that the diesel driven fire pump was capable of meeting system functional and design basis requirements. The team independently assessed engineering input for modifications implemented to allow the fire pump to provide water for emergency reactor vessel cooling and EDG jacket water (JW) cooling for beyond design bases events. The team also reviewed pump test results, system health reports, and corrective action documents to verify that the fire pump design margins were maintained and to confirm that Entergy appropriately entered problems into their CAP. In addition, the team reviewed the diesel engine cooling system and pump minimum flow requirements to assess the ability of the pump to operate under design basis conditions. Finally, the team witnessed a fire pump performance test and performed several detailed walkdowns to assess the physical/material condition of the pump and its support systems and to ensure adequate configuration control.

b. Findings

No findings of significance were identified.

.2.1.15 4.16KV Circuit Breaker 71-10560

a. Inspection Scope

The team reviewed safety-related 4KV circuit breaker 71-10560 to verify it could provide auxiliary power to 600V Unit Substations L15 and L25, and associated 600V safety-related loads. The team reviewed calculations, drawings, maintenance procedures, and vendor data to verify that circuit breaker 71-10560 was adequately designed and maintained. Specifically, the team evaluated breaker rating protection, control, and operation. The team also conducted a detailed walkdown of the 4160V vital buses to verify that their material condition, operating environment, support equipment, and breaker alignments were consistent with the design basis.

b. Findings

No findings of significance were identified.

2.1.16 Residual Heat Removal Pump Torus Suction Strainer (10F-4B)

a. Inspection Scope

The team reviewed the ability of the strainer to operate during design basis events, transient and accident conditions. The strainer has a design basis function to permit adequate suction flow to the “B” RHR pump and stop foreign particles larger than its mesh openings to prevent pump damage. The team reviewed calculations including NPSH, debris generation and transport, pressure drop across the strainer, structural loading, hydrodynamic loading, and vortexing to verify that the strainer would perform as required. Additionally, the team reviewed drawings, associated DBDs, and RHR system health reports to assess the capability of the strainer to operate as designed. Finally, the team interviewed the system engineer and reviewed system health reports to assess the material condition of the strainer.

b. Findings

No findings of significance were identified.

2.1.17 Channel A Reactor Vessel Low Pressure for Core Spray and Low Pressure Coolant Injection Valve Permissive Interlocks

a. Inspection Scope

The team reviewed the design documents associated with channel A reactor vessel low pressure (pressure transmitter 02-3PT-52A and master trip unit 02-3MTU-252A) for the core spray (CS) and LPCI injection valve permissive interlocks to verify that the injection valves would function as designed. The team reviewed the instrument loop and logic diagrams of these instruments to verify their correctness, and conducted a walkdown of the transmitter mounted on the instrument rack in the reactor building to assess its physical/material condition. The team also reviewed calculation JAF-CALC-NBI-00204, Reactor Vessel Low Pressure (CS/LPCI Injection Permissive and Recirculation Discharge Valve Permissive) Instrument Uncertainties and Setpoint Calculation, to verify that all uncertainty parameters for the pressure transmitter and the master trip unit were accounted for and that the instrument setpoint met the accuracy requirements. In addition, the team reviewed calibration procedures, and the results of the last three calibrations to verify that the transmitter and the master trip unit were properly calibrated within the required interval, and in accordance with station procedures.

b. Findings

No findings of significance were identified.

.2.2 Review of Low Margin Operator Actions (4 samples)

The team assessed manual operator actions and selected a sample of four operator actions for detailed review based upon risk significance, time urgency, and factors affecting the likelihood of human error. The operator actions were selected from a PRA ranking of operator action importance based on RAW and RRW values. The non-PRA considerations in the selection process included the following factors:

- Environmental conditions or restrictions for preforming the actions;
- Personnel access to equipment;
- Plant procedures that address the actions;
- Need for additional personnel or equipment;
- Information available for diagnosing conditions and initiating actions;
- Ability of operator to recover from errors while performing task;
- Consequences of failure to complete action;
- Time to complete actions; and
- Task included in the Systematic Approach to Training (SAT) based training program and trained on.

The selected operator actions were generally characterized as having one or more of the following attributes:

- Low margin between the time required and time available to perform the actions;
- Reliability or redundancy of the components associated with the actions;
- Complexity of the actions; and
- Procedure or training challenges that may impact the operators' ability to perform the actions.

.2.2.1 Reactor Pressure Vessel Depressurization

a. Inspection Scope

The team selected the manual operator actions to depressurize the reactor pressure vessel during response to transients or accidents where the reactor water level cannot be maintained above the top of active fuel. These manual operator actions were identified in Entergy's PRA as risk significant based on the risk importance to prevent uncovering the reactor core, and the likelihood of an operator error associated with preforming the task under high stress. The team selected this sample because the risk significant, time critical manual actions were complex and appeared to have a low margin between the time required and the time available to perform the actions. The team interviewed licensed and non-licensed operators and reviewed operator training to determine the time required to perform the manual actions. The team performed field and main control room walkdowns to independently assess operator task complexity. The team evaluated the available time margins to perform the operator actions to verify the reasonableness of Entergy's operating and risk assumptions.

b. Findings

No findings of significance were identified.

.2.2.2 Containment Venting

a. Inspection Scope

The team selected the operator action to initiate containment venting when pressure cannot be maintained below the primary containment pressure limit. Safety relief valves (SRVs) are also expected to close due to high containment pressure. The potential consequence of failure of this action is primary containment over-pressurization and reactor vessel re-pressurization after the SRVs close. The team reviewed Entergy's incorporation of this action into site procedures, classroom training, and job performance measures (JPMs). The team also walked down procedures to remotely and locally operate the torus vent valves with a licensed reactor operator. Finally, the team walked down the alignment of the standby gas treatment system which comprises the vent flow path from the torus to the stack to ensure that the vent valves were accessible, properly maintained, and functional.

b. Findings

No findings of significance were identified.

.2.2.3 Provide Fire Protection Water to Emergency Service Water for Emergency Diesel Generator Cooling

a. Inspection Scope

The team selected the operator action to manually align the fire protection system to the ESW system in order to provide cooling water to the EDGs. Failure of the ESW pumps following a loss-of-offsite power leads to a station blackout (SBO) if cooling is not restored to the EDG JW coolers. The potential consequence of failure of this action is core damage after the EDGs overheat and batteries supplying ECCS systems are depleted. The team reviewed Entergy's incorporation of this action into plant operating procedures, classroom training, and JPMs. The team walked down the alignment procedure with a licensed reactor operator to ensure procedure clarity and proper equipment staging, and to assess the time dependency of actions.

b. Findings

No findings of significance were identified.

.2.2.4 Shedding DC Loads with a Station Blackout

a. Inspection Scope

The team selected the operator actions to manually shed DC loads during a SBO event. The potential consequence of failure of this action is faster depletion of station batteries. The team reviewed Entergy's incorporation of these actions into plant operating procedures, classroom training, and simulator training. The team walked down the procedure to manually shed DC loads and to align systems with a licensed reactor operator to assess procedure adequacy and clarity.

b. Findings

No findings of significance were identified.

.3 Review of Industry Operating Experience and Generic Issues (5 samples)

a. Inspection Scope

The team reviewed selected Operating Experience (OE) issues that had occurred at domestic and foreign nuclear facilities with apparent applicability to JAF. The team performed a detailed review of the OE issues listed below to verify that JAF had appropriately assessed potential applicability to site equipment, and, if required, the actions taken to address the OE were effective in correcting or preventing the issue from occurring at the site.

- Operating Experience Smart Sample FY 2007-002

  NRC Operating Experience Smart Sample (OpESS) FY 2007-02, Flooding Vulnerabilities Due to Inadequate Design and Conduit / Hydrostatic Seal Barrier Concerns, addresses internal flooding events and hydrostatic seal barrier concerns. The team evaluated internal flood protection measures for the EDG rooms, the 4kV switchgear rooms, and the relay room. The team walked down the areas to assess operational readiness of various features in place to protect redundant safety-related components and vital electrical components from internal flooding. These features included equipment floor drains, flood detection, floor barrier curbs, and wall penetration seals. The team conducted several detailed walkdowns of the turbine building, EDG rooms, 4kV switchgear rooms, relay room, and cable tunnels to assess potential internal flood vulnerabilities. The team also reviewed Entergy's internal flood analysis, engineering evaluations, alarm response procedures, and CRs associated with flood protection equipment and measures. Finally, the team interviewed FitzPatrick personnel regarding their knowledge of indications, procedures, and required actions during postulated water pipe ruptures in the areas discussed above.

- NRC Information Notice 97-90: Use of Non-Conservative Acceptance Criteria in Safety-Related Pump Surveillance Tests

  The team reviewed Entergy's disposition of NRC Information Notice (IN) 97-90, which discussed the potential for using non-conservative acceptance criteria in safety-related pump surveillance tests (STs). The team verified that Entergy entered IN 97-90 into its CAP for review and considered all actions listed within the IN. The team reviewed Entergy's actions, which included verifying that pump IST acceptance criteria for differential pressure and flow rate were conservative with respect to design bases requirements.

- NRC Information Notice 2005-04: Single Failure and Fire Vulnerability of Redundant Electrical Safety Buses

  The team reviewed Entergy's assessment of the potential impact of single failure and fire vulnerability on FitzPatrick's redundant 4160 Vac safety buses (10500 and 10600). Entergy completed an assessment on this topic in 2005 as documented in CR-JAF-2005-00487 and determined that the issue described in the IN did not apply to FitzPatrick because the bus transfer schemes at FitzPatrick are different from those at Crystal River Nuclear Station. The two safety buses at FitzPatrick are fed from two non-safety 4160 Vac buses (10300 and 10400) and the fast transfer (unit trip) and residual transfer (loss of voltage with 3 second time delay) take place at these non-safety buses. Entergy determined that a complete loss of these non-safety buses would not prevent the four EDGs from feeding the safety buses, and that the safety buses were not vulnerable to a single failure or a fire induced mal-operation. The team reviewed the one-line and transfer logic diagrams, and discussed the issue with the cognizant engineer, to independently assess these transfer schemes. Finally, the team conducted several detailed walkdowns of the vital switchgear rooms to assess the material condition of the fire barriers, detection equipment, and suppression systems and to ensure adequate configuration control.

- NRC Information Notice 2003-16: Icing Conditions Between Bottom of Dry Storage System and Storage Pad

  The team reviewed the applicability and disposition of dry spent fuel storage concerns described in NRC IN 2003-16. This IN apprised licensees of an icing condition at an independent spent fuel storage installation (ISFSI) that placed the dry spent fuel storage systems in an unanalyzed condition. The team reviewed Entergy's evaluation of the icing concern relative to their particular ISFSI and associated actions. The team selected IN 2003-16 for review because the icing condition could potentially occur at other ISFSI sites using free-standing ventilated concrete cask designs located in the northern regions of the country. The team conducted a walk down of the FitzPatrick ISFSI and discussed the issue with ISFSI engineers to ensure that Entergy took appropriate actions relative to potential icing concerns.

- Regulatory Issue Summary 01-015: Performance of DC-Powered Motor-Operated Valve Actuators

  The team reviewed Entergy's applicability review and disposition of NRC RIS 01-015, Performance of DC-Powered Motor-Operated Valve Actuators. The team reviewed associated engineering evaluations and calculations to ensure that DC-powered MOVs could function as designed. The team also reviewed a sample of condition reports associated with DC issues to verify that the problems identified in RIS 01-015 had not occurred.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution (PI&R)

.1 Corrective Action Review

a. Inspection Scope

The team reviewed a sample of problems that Entergy identified and entered into their corrective action program. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design or qualification issues. In addition, CRs written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

.2 Cross Reference to PI&R Findings Documented Elsewhere in this Report

Section 1R21.2.1.2 of this report describes a NCV associated with the area of PI&R because Entergy did not ensure that design basis self-assessments were of sufficient depth, comprehensive, appropriately objective, and self-critical.

4OA6 Meetings, Including Exit

On July 19, 2007, the team presented the inspection results to Mr. P. Dietrich and other members of JAF management. The team verified that no proprietary information is documented in the report.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel:

J. Abisamra	Manager, Design Engineering
T. Andersen	Senior Electrical Engineer
A. Barton	Senior Electrical Engineer
R. Bell	Component Engineering
S. Bono	Director, Engineering
D. Burch	Design Engineering
B. Burnham	System Engineering
J. Costedio	Manager, Regulatory Compliance
D. Deretz	Regulatory Compliance
P. Dietrich	Site Vice President
S. Juravich	Electrical Engineer
S. Kim	Senior Electrical Engineer
B. Marks	Instrumentation and Control Engineer
D. Ruddy	Supervisor, Electrical Engineering
D. Poulin	Assistant Operations Manager
T. Savory	Supervisor, Procurement Engineering
J. Scranton	IST Program Engineer
R. Sullivan	Reactor Operator
P. Swinburne	Design Engineering
T. Yost	Instrumentation and Control Engineer

NRC Personnel:

G. Hunegs	Senior Resident Inspector
D. Dempsey	Resident Inspector
W. Schmidt	Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Open and Closed

05000333/2007006-01	NCV	Failure to maintain adequate design basis calculations for safety-related motors. (Section 1R21.2.1.1)
05000333/2007006-02	NCV	EDG FOST capacity calculation did not account for vortexing. (Section 1R21.2.1.2)

LIST OF DOCUMENTS REVIEWED

Audits and Self-Assessments

LO-JAFLO-2007-0001, CDBI Pre-Inspection Snapshot, dated 4/30/07

Calculations

73-6, Screenwell Building Ventilation Fan Selection, Rev. 0
98-019, Emergency Core Cooling and Reactor Core Isolation System Pump Suppression Pool NPSH, Rev. D
02268-C-5012-002, Loop Uncertainty Calculation for ESW Flow 46FE-101A1 & A2 and 46FE-101B1 & B2, Rev. 0
02268-M-5016-11, Establish Maximum EDG Cooler Tube Side Velocity and Flow Rate, Rev. 0
12966-E-81-1, Safety Loads Terminal Voltage Calculation, Rev. 0
12966-E-81-3, Motor Starting Voltage Profile - Emergency Buses, Rev. 0
14620-E-72-2, 600V Class 1E Cable Size Review, Rev. 2
14620-E-9013-4, Fast Bus Transfer, Rev. 0
14620-E-9016-2, Second Level (Degraded Grid) Undervoltage Relay Set Point Determination for Emergency Buses, Rev. 0
14620-E-9017-4, Momentary and Interrupting Short Circuit Duties at Normal and Emergency 4.16kV Buses, Rev. 0
14620.9011-US(N)-002, Pressure Drop Across Drywell Isolation Valves (27AOV-113 and 27AOV-114) and Torus Isolation Valves (27AOV-117 and 27AOV-118) for Severe Accident Venting, Rev. 0
14620.9011-US(N)-004-0, Suppression Chamber (20") and Drywell (24") Vent and Purge Butterfly Valves Evaluation Based on Relaps/Mod2 56 PSIG and 62 PSIG Results, Rev. 0
14620.9033-US(N)-002-1, Total Available Tube Plugging Margin for the Emergency Diesel Generator (EDG) Jacket Cooler Heat Exchangers, Rev. 2
93179, MOV Thrust Capacity Calculation, Instruction 3, Rev. 2
93719-C-06, Valve Thrust Assessment 10MOV-89A and B, Rev. 3
93179-C-12, Valve Thrust Assessment 10MOV-39B, Rev. 2
93179-C-33, Valve Thrust Assessment 10MOV-19 and 20, Rev. 2
93719-C-37, Valve Thrust Assessment 13MOV-21, Rev. 1
93179-C-45, Valve Thrust Assessment 14MOV-11A and B, 14 MOV-12A and B, Rev. 0
A384.F02-07, FitzPatrick ECCS Strainer Replacement, Rev. 2
DRN-06-2217, Minor Calculation Change for Calculation 90-018, Rev. D
E-43A, Motor Feeder Cable Sizing Calculation 10P-1A (RHR Service Water Pump Motor), Rev. 0
E-43C, Motor Feeder Cable Sizing Calculation 10P-1C (RHR Service Water Pump Motor), Rev. 0
E-43E, Motor Feeder Cable Sizing Calculation 10P-3A (RHR Pump Motor), Rev. 0
E-43F, Motor Feeder Cable Sizing Calculation 10P-3B (RHR Pump Motor), Rev. 0
E-43I, Motor Feeder Cable Sizing Calculation 14P-1A (Core Spray Pump Motor), Rev. 0
E77-01, Emergency Diesel - Generator Load Review, Rev. 1
F1-86-094, Fire Protection Diesel Fire Pump Injection into RHRSW, Calc Sets 2 and 3

F1-97-031, Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement, Rev. 0
GE-NE-T2300766-00-01, James A. FitzPatrick Containment Analysis, October 1999
JAF-04-36054, Fabricate Stem-to-Yoke Coupling for Bettis Actuator HD-732-SR80(CW) on 27AOV-117/118, Rev. 0
JAF 89-031, EDG Underground Storage Tank 93 TK-6A, B, C, D Level - Volume Calculation, Rev. 0
JAF-CALC-05-00117, Perform 600 Volt MCC Control Voltage Drop Calculation to Verify the Minimum Pickup Voltage for Selected Contactor Circuits, Rev.1
JAF-CALC-BNI-00198, Reactor Vessel Level 2 HPCI/RCIC Initiating Setpoint Calculation, Rev. 5
JAF-CALC-CAD-4481, Design Basis Calculations for the Inner and Outer Exhaust AOVs 27AOV-117 and 27AOV-118 at FitzPatrick NPP, Rev. 0
JAF-CALC-DGV-02026, EDG SWGR Rooms Temp. Following HELB in Turbine Building, Rev. 2
JAF-CALC-DGV-04251, Revised Heat Release and EDG Room Temperature for POT-92A Data, Rev. 0
JAF-CALC-ELEC-02610, 125 VDC Station Battery B Sizing and Voltage Drop, Rev. 2
JAF-CALC-ELEC-04343, Calculation for the JAF Plant Going from Full Load to a Trip with Estimated LOCA and with Operator Action, Rev.0
JAF-CALC-ELEC-04554, Emergency Bus Voltage Profile for RSST's T2 and T3 Tap Setting @113kV, Rev. 0
JAF-CALC-FPS-0170, Fire Pump Performance, Rev. 0
JAF-CALC-HPCI-00275, 23LS-74A, 23LS-74B, 23LS-75A, 23LS-75B, Condensate Storage Tank (CST) Low Level Switches Setpoint Calculation, Rev. 3
JAF-CALC-HPCI-00840, Vortexing Concerns in the CST During HPCI Operation, Rev. 1
JAF-CALC-HPCI-2094, Reduced Voltage Analysis for 23MOV-19, Rev. 3
JAF-CALC-HPCI-2133, Thrust and Torque Limits Calculation for 23MOV-19, Rev. 3
JAF-ICD-RHR-3181, GL 89-10 Degraded Voltage Calculation Input Data Verification for 10MOV-89B, Rev. 0
JAF-CALC-MISC-00287, Retrofit of Fire Damper, Rev. 0
JAF-CALC-NBI-00209, Reactor Vessel Water Level 8 HPCI and RCIC Trip Setpoint, Rev. 5
JAF-CALC-RCIC-02122, Thrust and Torque Limits Calculation for 13MOV-39, Rev. 3
JAF-CALC-RHR-1124, Reduced Voltage Analysis for 10MOV-39A and 10MOV-39B, Rev. 0
JAF-CALC-RHR-1712, Thrust and Torque Limits Calculation for MOV-39B, Rev. 2
JAF-CALC-RHR-4372, Thrust and Torque Limits Calculation for MOV-89B, Rev. 6
JAF-CALC-SWS-03026, Minimum ESW Flow Requirements for the EDG Jacket Water Coolers with Elevated Lake Temperatures Up to 85°F, Rev. 0
JAF-CALC-SWS-03337, Evaluation of IST Instrument Accuracy for 46PI-112A and B When Used in Conjunction With Screenwell Level Instruments, Rev. 0
JAF-CALC-SWS-04350, Bases for In-Service Test Acceptance Criteria for ESW Pumps 46P-2A and 46P-2B, Rev. 1
JAF-ECAF-H06-BUSCOORD, Breaker Coordination, Rev. 1
JAF-RPT-05-00021, FitzPatrick 115kV Off Site Supply Steady State Voltage Analysis, Rev. 0
JAF-RPT-MULTI-03000, ECCS and RCIC Suction Strainer Replacement Modification Supplement to the Plant Unique Analysis Report, Rev. 2

JAF-SE-98-013, Residual Heat Removal and Core Spray Suppression Pool Suction Strainer Replacement, Rev. 3
Pipe Stress Re-analysis Program - Problem Number 900, Emergency Service Water Line Nos. 6-WES-151-5A, 5B and 18-WES-151-4A, Rev. 0

Completed Surveillance Test Procedures

IMP-23.3, High Pressure Coolant Injection System Flow Indication Calibration (IST), dated 5/2/07, 2/22/05, and 5/12/03
ISP-90, 4KV Emergency Power (Buses 10500 and 10600) Undervoltage Relay (Loss of Voltage) Calibration, dated 8/16/06, 8/30/06, 10/20/06 and 10/28/06
ISP-90-1, 4KV Emergency Power (Buses 10500 and 10600) Undervoltage Relay (Degraded Voltage) Calibration, dated 9/6/06, 10/20/06 and 10/25/06
ISP-90-1, 4KV Emergency Power (Buses 10500 and 10600) Undervoltage Relay Degraded Voltage) Timer Instrument Calibration, dated 9/6/06 and 10/25/06
ISP-91, 4KV Emergency Power (Buses 10500 and 10600) Undervoltage Timer (Loss of Voltage) Calibration, dated 10/24/06
ISP-100A-RPS, RPS Instrument Functional Test/Calibration (ATTS), dated 6/7/07
ISP-100B-RPS, RPS Instrument Functional Test/Calibration (ATTS), dated 3/28/07
ISP-100C-RPS, RPS Instrument Functional Test/Calibration (ATTS), dated 6/7/07
ISP-100D-RPS, RPS Instrument Functional Test/Calibration (ATTS), dated 3/29/07
ISP-175A1, Reactor and Containment Cooling and ATWS Instrument Functional Test/Calibration (ATTS), dated 6/1/07, 3/5/07, 9/21/06, and 4/7/06
ISP-175A2, Reactor and Containment Cooling and ATWS Instrument Functional Test/Calibration (ATTS), dated 5/29/07
ISP-175B1, Reactor and Containment Cooling and ATWS Instrument Functional Test/Calibration (ATTS), dated 6/11/07
ISP-175B2, Reactor and Containment Cooling and ATWS Instrument Functional Test/Calibration (ATTS), dated 6/11/07
ISP-275A, Reactor Pressure and Drywell Pressure Transmitter Calibration, dated 10/18/06, 10/12/04, and 10/11/02
ISP-276A, Reactor Level (ECCS) Transmitter Calibration, dated 10/22/06, 10/13/04, and 10/9/02
MST-071.12, 125 VDC Battery and Charger Weekly Surveillance Test, dated 6/6/07 and 6/13/07
MST-071.13, 125 VDC Station Battery Quarterly Surveillance Test, dated 3/21/07 and 12/28/06
MST-071.20, 125 VDC Station Battery Service Test, dated 10/25/06 and 11/7/06
MST-071.24, Station Battery B Modified Performance Test, dated 10/1/04
MST-071.26, Station Battery A Modified Performance Test, dated 10/12/06
MST-071.27, Station Battery Charger Performance Test, dated 10/27/06
ST-1C, Primary Containment Isolation Valve Exercise Test (IST), dated 8/22/05, 11/14/05, 2/5/06, 2/26/06, 4/30/06, 10/1/06, 10/21/06, 10/18/06, 10/21/06, 10/21/06, 10/28/06, and 3/11/07
ST-2AL, RHR Loop A Quarterly Operability Test, dated 3/23/07
ST-2AM, RHR Loop B Quarterly Operability Test, dated 4/4/04, 11/4/04, 9/02/05, 1/31/06, 7/01/06, 11/30/06, and 3/8/07

ST-2XB, RHR Service Water Loop B Quarterly Operability Test (IST), dated 8/01/05, 03/24/06, 6/15/06, 9/07/06, 10/28/06, 11/30/06, 12/01/06, 1/24/07, and 4/03/07
ST-4N, HPCI Quick-Start, Inservice, and Transient Monitoring Test, dated 5/16/05, 8/29/05, 11/1/05, 1/25/06, 4/17/06, 7/10/06, 11/5/06, 1/25/07, and 5/4/07
ST-8Q, Testing of the Emergency Service Water System (IST), dated 5/16/07, 4/27/07, 2/14/07, and 8/5/05
ST-9BA, EDG A and C Full Load Test and ESW Pump Operability Test, dated 5/29/07 and 4/30/07
ST-9BB, EDG B and D Full Load Test and ESW Pump Operability Test, dated 5/17/07 and 4/16/07
ST-9LA, EDG A and C Fuel Oil Transfer Pump Operability Test, dated 6/30/07 and 7/13/07
ST-9LB, EDG B and D Fuel Oil Transfer Pump Operability Test, dated 6/17/07
ST-9QA, EDG A and C Full Load Test (8 Hour Run), dated 5/1/06
ST-9QB, EDG B and D Full Load Test (8 Hour Run), dated 4/17/06
ST-34A, PCIS Group 2 Logic Functional and Simulated Automatic Actuation Test, dated 1/8/2007
ST-38B-X205, Type C Leak Test of Torus Purge Exhaust Line Valves (IST), dated 10/21/06 and 10/27/06
ST-39BX-9A/B, Type B and C LLRT of Containment Penetrations (IST), dated 3/87, 9/88, 6/90, 4/92, 9/92,5/93,4/94,1/95, 11/96, 10/98, 10/00, 10/02, 10/04, and 10/16/06
ST-50, Floor Drain Flow Test, dated 10/1/03
ST-76AD, East Diesel Fire Pump 76P-4 Performance Test, dated 1/19/07, 3/10/05, 3/1/04, and 1/21/02
ST-76B, Electric Fire Pump 76P-2 Operational Check, dated 5/19/07
ST-76C, West Diesel Fire Pump 76P-1 Operational Check, dated 5/27/07
ST-76AC, East Diesel Fire Pump 76P-4 Operational Check, dated 5/29/07, 4/6/07, and 3/9/07

<u>Condition Reports (CR-JAF-)</u>

1993-00688	2001-03224	2003-02484	2005-03427	2007-02201*
1994-00273	2001-03521	2003-02485	2005-03468	2007-02202*
1995-00046	2001-03619	2003-02488	2006-00040	2007-02203*
1995-00344	2001-03980	2003-02604	2006-01229	2007-02205*
1995-01195	2001-04884	2003-02609	2006-02976	2007-02213*
1996-00428	2002-00385	2003-03169	2006-04660	2007-02243
1996-01594	2002-01145	2003-04208	2007-01374	2007-02265*
1998-00227	2002-01342	2003-04343	2007-01579	2007-02297
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FE-1B Sh. 2, Main Online Diagram Station Service Transformers, Rev. 12
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LIST OF ACRONYMS

AC	Alternating Current
ADAMS	Agency-Wide Documents Access and Management System
AOP	Auxiliary Oil Pump
AOV	Air Operated Valve

ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CR	Condition Report
CS	Core Spray
CST	Condensate Storage Tank
DBD	Design Basis Document
DC	Direct Current
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ESW	Emergency Service Water
FOST	Fuel Oil Storage Tank
FOTP	Fuel Oil Transfer Pump
GE	General Electric
GL	Generic Letter
HPCI	High Pressure Coolant Injection
IMC	Inspection Manual Chapter
IN	Information Notice
IPEEE	Individual Plant Examination of External Events
ISFSI	Independent Spent Fuel Storage Installation
IST	Inservice Test
JAF	James A. FitzPatrick Nuclear Power Plant
JAFLO	James A. FitzPatrick Learning Organization (process improvement tool)
JPM	Job Performance Measure
JW	Jacket Water
KV	Kilovolt
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MOV	Motor Operated Valve
MR	Maintenance Rule
NCV	Non-Cited Violation
NEMA	National Electrical Manufacturers Association
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OE	Operating Experience
OpESS	Operating Experience Smart Sample
PARS	Publicly Available Records
PI&R	Problem Identification and Resolution
PRA	Probabilistic Risk Assessment
RAW	Risk Achievement Worth
RHR	Residual Heat Removal
RIS	Regulatory Issue Summary

RPS	Reactor Protection System
RRW	Risk Reduction Worth
SAT	Systematic Approach to Training
SBO	Station Blackout
SDP	Significance Determination Process
SLC	Standby Liquid Control
SPAR	Standardized Plant Analysis Risk
SR	Surveillance Requirement
SRV	Safety Relief Valve
SSC	Structure, System, and Component
ST	Surveillance Test
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UV	Undervoltage
V	Volt
Vac	Volts Alternating Current
Vdc	Volts Direct Current