UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BOULEVARD, SUITE 100
KING OF PRUSSIA, PENNSYLVANIA 19406-2713

August 23, 2013

Mr. Lawrence M. Coyle
Site Vice President
James A. FitzPatrick Nuclear Power Plant
Entergy Nuclear Northeast
P. O. Box 110
Lycoming, NY 13093

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000333/2013007

Dear Mr. Coyle:

On July 11, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick). The enclosed inspection report documents the inspection results, which were discussed on July 11, 2013, with Mr. Dave Poulin, Acting General Manager, Plant Operations, 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 to mitigate postulated transients, initiating events, and design basis accidents. The inspection involved field walkdowns, examination of selected procedures, calculations and records, and interviews with station personnel.

This report documents two NRC-identified findings that were of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance of the violations and because they were entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCVs) consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, 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 Senior Resident Inspector at FitzPatrick. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis of your disagreement, to the Regional Administrator, Region I, and the NRC Senior Resident Inspector at FitzPatrick.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for the public inspection in the NRC Public Docket Room or from the Publicly Available Records component of 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/

Paul G. Krohn, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure:
Inspection Report 05000333/2013007
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	05000333/2013007
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, New York
Inspection Period:	June 10 – July 11, 2013
Inspectors:	J. Schoppy, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader S. Chaudhary, Senior Reactor Inspector, DRS T. O'Hara, Reactor Inspector, DRS J. Patel, Reactor Inspector, DRS J. Chiloyan, NRC Electrical Contractor M. Yeminy, NRC Mechanical Contractor
Approved By:	Paul G. Krohn, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000333/2013007; 06/10/2013 - 07/11/2013; James A. FitzPatrick Nuclear Power Plant (FitzPatrick); Component Design Bases Inspection.

The report covers the Component Design Bases Inspection conducted by a team of four U.S. Nuclear Regulatory Commission (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." Cross-cutting aspects associated with findings are determined using IMC 0310, "Components Within the Cross-Cutting Areas." 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.

Cornerstone: Mitigating Systems

- Green: The team identified a finding of very low safety significance (Green) involving a non-cited violation of Technical Specification (TS) 5.4, "Procedures." Specifically, following EDG maintenance, operators did not restore the 'A' and 'C' EDG ventilation systems in accordance with operating procedure OP-60, "Diesel Generator Room Ventilation." In particular, operators failed to correctly position the 'A' and 'C' EDG room ventilation temperature controllers to automatic as required by Entergy procedure OP-60. Following discovery, operators promptly restored controllers to automatic, performed additional extent-of-condition control panel walkdowns throughout the plant, and entered the issue into their corrective action program to evaluate and address causal factors.

  The performance deficiency was determined to be more than minor because it was associated with the Configuration Control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding in accordance with IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings at Power," Exhibit 2 – Mitigating Systems Screening Questions. The team determined that the finding was of very low safety significance because it was not a design qualification deficiency resulting in a loss of functionality or operability and did not represent an actual loss of safety function of a system or train of equipment.

  The team determined that this finding has a cross-cutting aspect in the area of Human Performance, Work Practices Component, because Entergy did not adequately ensure supervisory and management oversight of EDG ventilation system restoration activities such that nuclear safety was supported. (IMC 0310, Aspect H.4(c)) (Section 1R21.2.1.9.b.1)

- Green: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," because Entergy had not verified the adequacy of the existing design analyses for the available net positive suction head (NPSH) to the EDG fuel oil transfer pumps. Specifically, the team identified several non-conservative design assumptions indicating that Entergy did not adequately account for NPSH in their calculation for the 7-day onsite supply of fuel oil to the EDGs. Entergy performed an operability evaluation, implemented appropriate compensatory measures, and entered the issue into their corrective action program to evaluate and resolve the design deficiency.

  The performance deficiency was determined to be more than minor because it was similar to Example 3.j of NRC IMC 0612, Appendix E, and was associated with the Design Control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding in accordance with IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings at Power," Exhibit 2 – Mitigating Systems screening questions. The finding was determined to be of very low safety significance because it was a design deficiency confirmed not to result in a loss of operability.

  This finding was not assigned a cross-cutting aspect because it was a historical design issue not indicative of current performance. Specifically, the performance deficiency had occurred outside of the nominal three year period for evaluating present performance as defined in IMC 0612. (Section 1R21.2.1.9.b.2)

Other Findings

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 for review using information contained in the FitzPatrick Probabilistic Safety Assessment (PSA) model and the U. S. Nuclear Regulatory Commission's (NRC) Standardized Plant Analysis Risk (SPAR) model for the James A. FitzPatrick Nuclear Power Plant. Additionally, the team referenced the Plant Risk Information e-Book (PRIB) for the James A. FitzPatrick Nuclear Power Plant in the selection of potential components for review. In general, the selection process focused on components that had a risk achievement worth (RAW) factor greater than 1.3 or a risk reduction worth (RRW) factor greater than 1.005. The components selected were associated with both safety-related and non-safety related systems, and included a variety of components such as pumps, breakers, inverters, diesel engines, batteries, room coolers, transformers, and valves.

The team initially compiled a list of components based on the risk factors previously mentioned. Additionally, the team reviewed the previous component design bases inspection (CDBI) reports (05000333/2010006 and 05000333/2007006) and excluded the majority of those components previously inspected. The team then performed a margin assessment to narrow the focus of the inspection to 16 components and 4 operating experience (OE) items. The team selected a main steam isolation valve (MSIV) and the suppression pool for large early release fraction (LERF) implications. The team's evaluation of possible low design margin included consideration of original design issues, margin reductions due to modifications, and margin reductions identified as a result of material condition/equipment reliability issues. The assessment also included items such as failed performance test results, corrective action history, repeated maintenance, Maintenance Rule (a)(1) status, operability reviews for degraded conditions, NRC resident inspector insights, system health reports, and industry OE. Finally, consideration was also given to the uniqueness and complexity of the design and the available defense-in-depth margins.

The inspection performed by the team was conducted as outlined in NRC Inspection Procedure (IP) 71111.21. This inspection effort included walkdowns 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 basis and licensing basis requirements. Summaries of the reviews performed for each component and OE sample, and the specific inspection findings identified are discussed in the subsequent sections of this report. Documents reviewed for this inspection are listed in the Attachment.

.2 Results of Detailed Reviews

.2.1 Detailed Component Reviews (16 samples)

.2.1.1 Feedwater Check Valve (34FWS28B)

a. Inspection Scope

The team reviewed applicable portions of Fitzpatrick's TSs, the Updated Final Safety Analysis Report (UFSAR), and system design basis documents (DBD) to identify design basis requirements for feedwater check valve 34FWS28B. The team reviewed drawings and vendor documents to verify that the installed configuration supported the design basis function under accident conditions. The team reviewed the check valve orientation and its distance from elbows and other valves. The team interviewed the system engineer, the check valve engineer, and the in-service test (IST) engineer to discuss the valve analysis and operational and maintenance history and to verify that potentially degraded conditions were being appropriately addressed. The team reviewed test procedures and recent test results against design bases documents to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that individual tests and analyses served to validate component operation under accident conditions. The team reviewed vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents in order to verify that potential degradation was monitored or prevented and that scheduled component inspections or replacements were consistent with vendor recommendations.

b. Findings

No findings were identified.

.2.1.2 'C' Emergency Diesel Generator (Electrical)

a. Inspection Scope

The team inspected the 'C' emergency diesel generator (EDG) to verify that it was capable of meeting its design basis requirements. The design function of the C EDG is to provide standby power to safety-related 4160 volt (V) bus 10500 when the preferred power supply is not available. The team reviewed the EDG loading study to ensure that it was consistent with the actual loading expected in response to a design basis accident. The team reviewed completed TS surveillance tests (ST) to ensure that the EDG met all applicable test acceptance criteria. The team performed independent electrical calculations to verify that the EDG's neutral grounding scheme met the design bases requirements affecting ground fault protection coordination. The team reviewed the EDG protective trips and relay setpoint calculations and calibration test results to assess the adequacy of the EDG protection during testing and emergency operations and to verify that any relay setpoint drift was acceptable and adequately addressed. The team interviewed system and design engineers to answer questions that arose during document reviews and walkdowns to evaluate the adequacy of maintenance and configuration control. In addition, the team visually inspected the physical and material condition of the EDG and its grounding transformer and reviewed associated corrective action documents to verify that Entergy identified and addressed any adverse conditions or trends.

b. Findings

No findings were identified.

.2.1.3 Suppression Pool

a. Inspection Scope

The team inspected the suppression pool to verify that it was capable of performing its design function. The team reviewed the DBDs pertaining to the suppression pool (torus) and the applicable sections of the UFSAR to determine the design requirements. The team reviewed the root cause report and the results of the laboratory analyses from the torus cracking event from 2005. The team also reviewed a sample of past American Society of Mechanical Engineers (ASME) IWE inspection results (visual and ultrasonic testing) of the torus wall thickness and coating conditions conducted during the most recent refueling outage to assess the material condition and structural integrity. The team reviewed recent pressure suppression chamber to drywell vacuum breaker and pressure suppression chamber to reactor building vacuum breaker test results to verify that the vacuum breakers remained operable and capable of performing their design function supporting suppression pool integrity. The team also reviewed operating logs, associated corrective action condition reports (CR), emergency core cooling system (ECCS) health reports, and applicable in-service inspection and test results to determine if there were any adverse trends and to ensure that Entergy adequately identified and addressed any adverse conditions. The team conducted a walkdown of the accessible portions of the torus structure to assess the material condition (including evidence of leakage), structural supports, potential hazards, and configuration control.

b. Findings

No findings were identified.

.2.1.4 High Pressure Coolant Injection Steam Supply Valve (23MOV16)

a. Inspection Scope

The team inspected the high pressure coolant injection (HPCI) turbine steam supply outboard containment isolation (23MOV16) to verify that it was capable of performing its design function in response to transients and accidents. The normally closed 23MOV16 valve is required to open for the HPCI system to perform its ECCS function and required to close to isolate the main steam line and reactor vessel to prevent depressurization in the case of a HPCI steam line break. The team reviewed applicable portions of FitzPatrick's TSs, the UFSAR, and the HPCI system DBD to identify design basis requirements for 23MOV16.

The team reviewed the original design calculations, including seismic qualification, valve specifications, and operating history to verify that the valve was acceptable for HPCI service, and to verify that it met the applicable ASME Code requirements. The team reviewed a sample of ST results to verify that valve performance met the acceptance criteria and that the criteria were consistent with the design basis. Also, the team reviewed operator training lesson plans to verify the technical adequacy and details of the plans. The team interviewed the motor-operated valve (MOV) program engineer and reviewed valve diagnostic test results and trending to assess valve performance capability, design margin, and susceptibility to recent industry OE. The team reviewed a sample of HPCI system corrective action CRs, the HPCI system health report, and applicable test results to determine if there were any adverse operating trends and to ensure that Entergy adequately identified and addressed any adverse conditions. The team also performed a walkdown of the valve, adjacent areas, accessible portions of the HPCI system, and associated control room instrumentation to assess the material condition, operating environment, and configuration control.

b. Findings

No findings were identified.

.2.1.5 'C' Inboard Main Steam Isolation Valve (29AOV80C)

a. Inspection Scope

The team inspected the 'C' inboard MSIV to ensure that the valve was capable of performing its intended safety function to close upon receiving an isolation signal. The team reviewed IST stroke time data and air-operated valve (AOV) program requirements to ensure that Entergy appropriately monitored valve performance and that the valve was capable of performing its required functions under accident conditions. The team reviewed required thrust, accumulator sizing, maximum expected differential pressure, and margin analyses. Additionally, the team reviewed air supply test results to verify that acceptance criteria were met and that performance degradation would be identified. The team reviewed MSIV instrument and solenoid valve testing requirements and test results to ensure that individual component performance was in accordance with the design requirements and specifications. In addition, the team reviewed the environmental qualification (EQ) design requirements and EQ test results of the MSIV solenoid valves and associated instrumentation to verify that the MSIV was capable of performing its intended safety function under postulated design conditions. The team also reviewed the maintenance and operating history of 29AOV80C, associated corrective action CRs, the system health report, and applicable test results to determine if there were any adverse operating trends and to ensure that Entergy adequately identified and addressed any adverse conditions.

b. Findings

No findings were identified.

.2.1.6 'A' Emergency Diesel Generator 4KV Breaker (71-10502)

a. Inspection Scope

The team inspected the 'A' EDG 4KV output breaker (71-10502) to verify that it was capable of performing its design function. The team reviewed the one-line diagrams, control schematics, and the design basis as defined in the UFSAR to verify the adequacy of the 'A' EDG 4KV output breaker to meet loading and switching requirements under the worst loading conditions. The team reviewed the breaker closing permissive and interlocks to verify that the breaker opening and closing control circuits functioned as designed. The team reviewed samples of preventive and corrective maintenance test results to verify that the applicable test acceptance criteria and test frequency requirements were satisfied. The team interviewed system and design engineers to answer questions that arose during document reviews to determine the adequacy of maintenance and configuration control. The team performed several walkdowns of the 4KV breaker and its associated 4KV bus to assess the installed configuration, material condition, operating environment, and potential vulnerability to hazards. The team also reviewed the maintenance and operating history of the breaker, associated corrective action CRs, the EDG system health report, and applicable test results to determine if there were any adverse operating trends and to ensure that Entergy adequately identified and addressed any adverse conditions.

b. Findings

No findings were identified.

.2.1.7 Crescent Area Unit Coolers (Common Cause Failure)

a. Inspection Scope

The team inspected safety-related crescent area unit coolers to verify that they were capable of performing their design function. The team reviewed applicable portions of FitzPatrick's TSs, the UFSAR, and the emergency service water (ESW) DBD to identify design basis requirements for the crescent area unit coolers. The team focused on potential common cause failure (CCF) mechanisms and events. The team discussed the CCF probablities assigned to the crescent area coolers with an Entergy PSA engineer to gain an understanding of Entergy's risk assessment methodology and to probe potential CCF vulnerabilities. The team reviewed plant drawings of the crescent area coolers and the ESW system to verify that they were consistent with the as-installed configuration and to identify potential CCF vulnerabilities. Additionally, the team reviewed the ESW system operating procedures and recent ESW system test results to verify that system flow rates and heat removal capability met design requirements. Also, the team reviewed unit cooler tube cleaning documentation and visual and eddy current tube inspection results to verify that Entergy properly maintained the crescent area coolers.

The team reviewed voltage drop calculations for the power supplies to the crescent area cooler fan units to verify that sufficient power was available to ensure cooler fan operation under normal and accident conditions. The team reviewed maintenance and test scheduling to verify that Entergy minimized the potential for multiple coolers to be out of service simultaneously during plant operation. The team performed several walkdowns of the accessible portions of crescent area coolers and fan units in the reactor building to assess the material condition, operating environment, and potential hazards. The team also reviewed the maintenance and operating history of the crescent area coolers, associated corrective action CRs, ESW system health reports, and applicable test results to determine if there were any adverse operating trends and to ensure that Entergy adequately identified and addressed any adverse conditions.

b. Findings

No findings were identified.

2.1.8 High Pressure Coolant Injection Inverter (23INV-79)

a. Inspection Scope

The team inspected the HPCI inverter (23INV-79) to verify that it was capable of performing its design basis function. The inverter is designed to provide essential power to the HPCI flow controller (23FIC-108) and other related essential devices for control of the HPCI injection rate. The team reviewed the loading documentation for the design basis maximum load to verify that adequate margin existed. The team also reviewed voltage drop calculations to ensure that adequate voltage was available for normal and design basis conditions. The team also performed several walkdowns of accessible portions of the HPCI inverter and instrumentation in the control room and in the HPCI room to assess the material condition, operating environment, and configuration control. The team also reviewed a sample of HPCI-related equipment preventive maintenance (PM) evaluation reports, corrective action CRs, the HPCI system health report, and applicable test results to determine if there were any adverse operating trends and to ensure that Entergy adequately identified and addressed any adverse conditions.

b. Findings

No findings were identified.

.2.1.9 'C' Emergency Diesel Generator (Mechanical)

a. Inspection Scope

The team reviewed the UFSAR, FitzPatrick's TSs, the EDG DBD, and piping and instrument diagrams (P&ID) to establish an overall understanding of the design bases of the EDG with special emphasis on the air starting system, the fuel oil supply and delivery system, and the room ventilation. The team reviewed design calculations and procedures to verify that Entergy appropriately translated the design bases and assumptions into these documents. The team performed EDG and support equipment walkdowns to verify that the installed configurations would support their design bases function under accident and loss-of-offsite power (LOOP) conditions and that Entergy properly maintained the associated components consistent with design assumptions. The team also directly observed portions of a monthly EDG ST to independently assess EDG performance, Entergy test control, and the operating environment (including ventilation performance). The team reviewed operating procedures to verify that component operation and alignments were consistent with design and licensing requirements and assumptions. The team reviewed test procedures and results against the design bases to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that tests and analyses served to validate component operation under accident and LOOP conditions.

The team reviewed vendor documentation, EDG health reports, PM and corrective maintenance history, and corrective action CRs to verify that potential degradation was monitored or prevented and that component rework and replacement was consistent with equipment qualification life. The team reviewed design modifications of components important to safety to verify that the quality and adequacy of components as well as the safety margin were not compromised.

b.1 Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of TS 5.4, "Procedures." Specifically, following EDG maintenance, operators did not restore the 'A' and 'C' EDG ventilation systems in accordance with operating procedure OP-60, "Diesel Generator Room Ventilation." In particular, operators failed to correctly position the 'A' and 'C' EDG room ventilation temperature controllers to automatic as required by Entergy procedure OP-60.

Description: During the 'A' and 'C' EDG monthly ST on June 25, 2013, the team observed that the 'C' EDG room ventilation exhaust damper did not open following the EDG start (the 'A' EDG room exhaust damper appeared full open and the 'C' EDG room temperature was elevated but < 120˚F). The team promptly shared this observation with a nuclear plant operator (NPO) and the EDG system engineer. Nuclear plant operators, with support from engineering, subsequently identified that both the 'A' and 'C' EDG room ventilation temperature controllers were positioned to "manual" versus "automatic" as required by OP-60. The 'A' controller was in manual and set for maximum cooling which caused the 'A' EDG room ventilation exhaust damper to open. The 'C' controller was in manual and not set for maximum cooling causing the damper to remain closed. Following identification of this configuration control deficiency, the NPO promptly repositioned the controllers to “automatic” and the team noted that the ‘C’ EDG room ventilation exhaust damper opened and then modulated to control EDG room temperature as designed.

Following discovery of the condition, Entergy performed a Human Performance Error Review (HPER) prompt investigation, performed additional extent-of-condition control panel walkdowns throughout the plant, and entered the issue into their corrective action program (CAP) (CR 2013-3354) to evaluate and address causal factors. The team also performed additional walkdowns of the EDG control panels and a sample of other important-to-safety control panels throughout the plant (including control room instrumentation) to independently assess Entergy’s configuration control and corrective actions. The team noted that several short duration ‘C’ EDG runs between May 22 and June 25 and daily NPO rounds in the EDG rooms represented missed opportunities for Entergy to identify and correct this adverse condition prior to June 25.

Entergy’s apparent cause evaluation (ACE) concluded that operators had likely taken the ‘A’ and ‘C’ EDG temperature controllers to manual in response to an EDG room ventilation trouble alarm during a prolonged ‘A’ and ‘C’ EDG surveillance on May 20, 2013. The ventilation trouble alarm was caused by an EDG switchgear room fire damper (92FD-1) deficiency. Following the surveillance, Entergy tagged both EDGs out-of-service, declared both EDGs inoperable, and initiated corrective maintenance on the failed fire damper. As part of the tagging process, operators tagged the associated ventilation supply fans, but did not hang tags on the temperature controller switches. Following the fire damper work, operators cleared the associated EDG tags and proceeded to perform a post maintenance test (PMT). Following the successful PMT, operators declared the EDGs operable, and exited the associated TS limiting condition for operation (LCO) on May 22. However, even though the fire damper tagout (92-005-92FD-1) contained restoration instructions to restore the EDG ventilation per the applicable section of OP-60, the operations Field Support Supervisor (FSS), a senior reactor operator qualified watchstander, did not provide clear direction to the NPO regarding the appropriate OP-60 section to verify during restoration activities. The FSS directed the NPO to perform Section G.3 of OP-60 which provided guidance on supply fan restoration, but did not verify the entire EDG ventilation system standby alignment. However, Section D.2.4 of OP-60 requires the operator to ensure the temperature controllers are in automatic and set to 80˚F. The FSS direction to perform Section G.3 contributed to operators declaring the EDGs fully operable with the ‘A’ and ‘C’ ventilation temperature controllers still positioned to manual. Entergy’s ACE determined that the apparent cause of operators’ failure to properly restore the ventilation controllers to automatic was lack of adequate supervisory oversight, noting that the NPO was fairly new to the position. Specifically, the FSS did not ensure that he understood the tasks required to restore the EDG ventilation system to a normal line-up and did not maintain an appropriate questioning attitude regarding the applicable section of OP-60 to perform.

On July 24, 2013, Entergy completed a past operability review and determined that the ‘A’ and ‘C’ EDGs remained operable from May 22 to June 25, 2013. Entergy based their operability evaluation on a safety-related EDG ventilation trouble alarm in the main control room that alarms on high room temperature (120˚F); the associated alarm response procedure (ARP) directs operators to the local ventilation alarm panel; and the local panel ARP directs operators to check the ventilation line-up and take the temperature controllers, located on that particular local panel, to manual to control temperature as needed. Entergy determined that the manual operator action was a simple action requiring no diagnosis, was described in an approved procedure, operators were trained on the procedure, and sufficient time and staffing were available to perform the action without impacting the safety function. The team reviewed Entergy’s operability evaluation and determined that it was reasonable and aligned with guidance provided in NRC IMC Part 9900 Technical Guidance, “Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety.”

Analysis: The team determined that Entergy’s configuration control deficiency was a performance deficiency that was reasonably within Entergy’s ability to foresee and correct and should have been prevented. The performance deficiency was determined to be more than minor because it was associated with the Configuration Control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e. core damage). Specifically, operators failed to properly restore the ‘A’ and ‘C’ EDG ventilation temperature controllers to automatic following fire damper maintenance and it adversely impacted EDG and 4KV switchgear room cooling during EDG operation, especially during the warmer summer months. The team evaluated the finding in accordance with IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings at Power,” Exhibit 2 – Mitigating Systems screening questions. The team determined that the finding was of very low safety significance because it was not a design qualification deficiency resulting in a loss of functionality or operability and did not represent an actual loss of safety function of a system or train of equipment.

The team determined that this finding had a cross-cutting aspect in the area of Human Performance, Work Practices Component, because Entergy did not adequately ensure supervisory and management oversight of EDG ventilation system restoration activities such that nuclear safety was supported. (IMC 0310, Aspect H.4(c))

Enforcement: TS 5.4, “Procedures,” states, in part, “Written procedures shall be established, implemented, and maintained covering . . . the applicable procedures recommended in NRC Regulatory Guide 1.33, Appendix A, November 1972.” Regulatory Guide 1.33, Appendix A, November 1972, Section D, “Procedures for Startup, Operation, and Shutdown of Safety Related BWR [boiling water reactor] Systems,” includes the onsite emergency power sources (EDGs) as a safety-related system. Contrary to the above, from May 22, 2013, to June 25, 2013, operators failed to follow operating procedure OP-60, “Diesel Generator Room Ventilation,” Revision 8, Section D.2.4 regarding the normal standby alignment of the ‘A’ and ‘C’ EDG ventilation systems. Because this issue was of very low safety significance, and it was entered into

Entergy's CAP (CR 2013-3354), this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000333/2013007-01, Failure to Correctly Position EDG Room Ventilation Temperature Controllers in Automatic)**

b.2 Findings

Introduction: The team identified a finding of very low safety significance (Green) involving a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," because Entergy had not verified the adequacy of the existing design analyses for the available NPSH to the EDG fuel oil transfer pumps (FOTP). Specifically, the team identified several non-conservative design assumptions indicating that Entergy did not adequately account for net positive suction head (NPSH) in their calculation for the 7-day onsite supply of fuel oil to the EDGs.

Description: Based on a review of FOTP NPSH calculations, the team noted that there was very little margin between the vendor's required NPSH and Entergy's calculated available NPSH. Specifically, a NPSH calculation update in June 2008 (Calculation 93.6, EC 5827 Markup) indicated that the margin was approximately 0.5 psi and a subsequent update in May 2010 (Calculation 93.6, EC 22069 Markup) to bound an EDG frequency of 61.2 hertz (Hz) reduced the margin to 0.13 psi. Entergy calculated the NPSH based on a FOTP flow rate of 14.5 gpm at 60 hertz (which translated to 14.79 gpm at 61.2 Hz). However, the team noted that Entergy's use of this flow rate was non-conservative with respect to NPSH based on a review of the FOTP vendor manual. In particular, JAF Manual R374-C005, "Roper Pump Installation and Operating Instructions," included a statement that the FOTPs deliver 0.009 gallons per revolution. The team checked the motor nameplate for the installed FOTPs and noted that the older motors' speed was 1760 revolutions per minute (RPM) and the newer ones were rated at 1750 RPM. This resulted in a pump flow rate of 15.84 gpm at 1760 RPM at 60 Hz (and 16.16 gpm at 61.2 Hz) resulting in greater friction losses thereby reducing the absolute pressure (NPSH available) at the pump suction.

To address the team's concerns, Entergy reanalyzed the FOTP NPSH and determined that it resulted in a loss of 12.75 inches of suction head due to the increased pump speed (CR 2013-3196). As part of their evaluation, Entergy noted that they had also used the same non-conservative FOTP flow rate in calculation JAF-CALC-07-00020 to evaluate the potential for vortex formation at the inlet to the fuel oil suction pipe in the EDG fuel oil storage tanks (FOST). Entergy determined that they needed to add an additional 38 gallons of fuel oil to their 7-day and 6-day EDG fuel oil supply procedural limits (limits of > 30,000 for a 7-day supply and > 27,000 for a 6-day supply) to preclude the formation of air-entraining vortices (CR 2013-3196). Additionally, the team identified that Entergy's original NPSH calculation used the pressure drop across a clean FOTP suction strainer rather than a loaded particulate strainer (CR 2013-3342). As a result of the aggregate head losses, Entergy determined that all eight FOTPs were operable but non-conforming with compensatory measures needed (changes to the required EDG

FOST levels in the applicable procedures). Entergy promptly changed the associated EDG procedures resulting in an operable but non-conforming determination. Entergy initiated actions within their CAP to evaluate and resolve the design deficiency. The team reviewed Entergy's associated operability evaluations and short-term corrective actions and determined that they were reasonable and appropriate. Based on a review of EDG operating and FOST level monitoring procedures, a limited sample of operating logs, and numerous direct observations of FOST level indications (local and remote); the team concluded that the very small adjustments in the EDG fuel oil supply procedural limits (38 gallons compared to pre-existing limits of > 30,000 for a 7-day supply) did not represent a past operability concern due to the conservative margins maintained regarding FOST fuel oil inventory.

Analysis: The team determined that Entergy's design control deficiency was a performance deficiency that was reasonably within Entergy's ability to foresee and correct and should have been prevented. The team determined that the issue was more than minor because it was similar to IMC 0612, "Power Reactor Inspection Reports," Appendix E, Example 3.j which states that if "the engineering calculation error results in a condition where there is now a reasonable doubt on the operability of a system or component" the performance deficiency is not minor. Specifically, the FOTPs would not have had sufficient NPSH available to satisfy the manufacturer's requirement for proper operation under all design and licensing bases conditions. In addition, this finding was more than minor because it was associated with the Mitigating System Design Control attribute and affected the cornerstone's objective to ensure the availability, reliability, and capability of systems (EDG FOTPs) that respond to initiating events to prevent undesirable consequences (i.e. core damage). The team evaluated the finding in accordance with IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings at Power," Exhibit 2 – Mitigating Systems screening questions. The finding was determined to be of very low safety significance (Green) because it was a design deficiency confirmed not to result in a loss of operability.

The team determined that this finding did not have a cross-cutting aspect because it was a historical design issue not indicative of current performance. Specifically, the performance deficiency had occurred outside of the nominal three year period for evaluating present performance as defined in IMC 0612.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. 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, prior to June 2013, Entergy failed to verify the adequacy of the design to assure that the FOTPs were capable of taking suction from the bottom of the FOSTs under all design and license bases conditions. Specifically, Entergy failed to perform an adequate NPSH calculation that accounted for the actual pump speed and that assumed appropriate pressure losses. Because this finding was of very low safety significance and because it was entered into Entergy's CAP (CRs 2013-3196 and 2013-3342), this violation is being treated as an NCV consistent with

Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000333/2013007-02, Failure to Verify Adequacy of the Fuel Oil Transfer Pump NPSH)**

.2.1.10 600 Volt Bus 11500

a. Inspection Scope

The team inspected 600V bus 11500 to verify that it was capable of performing its design function. The team reviewed the one-line diagrams, control schematics, and the design basis as defined in the UFSAR to verify the adequacy of the 600V bus capability to supply adequate voltage and current to the loads. The team reviewed the associated voltage drop calculations to verify that adequate voltage was available to components supplied by the bus under the worst loading and degraded voltage conditions. The team reviewed the bus supply and feeder breaker ratings and trip settings to verify that protection coordination was provided for the loads and for the feeder conductors. The team reviewed vendor specifications, nameplate data, and calculations related to the 600V bus supply 4160V/600V transformer to verify its rating was adequate to provide the capacity and capability required by 600V bus 11500. The team interviewed system and design engineers to answer questions that arose during document reviews to evaluate the adequacy of maintenance and configuration control. The team performed several walkdowns of the bus, its associated transformer, and adjacent areas to assess the installed configuration, material condition, the operating environment, and potential vulnerability to hazards.

b. Findings

No findings were identified.

.2.1.11 'B' Emergency Service Water Pump (46P-2B)

a. Inspection Scope

The team inspected the 'B' ESW pump to verify that it was capable of meeting its design basis requirements. The ESW system is designed to provide cooling water to essential components when the service water system or the reactor building closed cooling water system cannot provide sufficient cooling to those essential components under transient and accident conditions. The team reviewed the UFSAR, associated design basis documents (including the ESW DBD), the vendor manual, plant drawings, and procedures to identify the most limiting requirements for the pump. The team reviewed a sample of ST results to verify that pump performance met the acceptance criteria and that the criteria were consistent with the design basis. The team also reviewed calculations for NPSH to ensure that the pump could successfully operate under the most limiting conditions. The team discussed the design, operation, and corrective maintenance of the pump with the engineering staff to gain an understanding of the performance history and overall component health. The team performed several walkdowns of the ESW pump and accessible ESW piping to assess the material condition, operating environment, and configuration control. The team also reviewed the maintenance and operating history of the B ESW pump, associated corrective action

CRs, ESW system health reports, and applicable test results to determine if there were any adverse operating trends and to ensure that Entergy adequately identified and addressed any adverse conditions.

b. Findings

No findings were identified.

.2.1.12 Train A 125 VDC Station Battery (71SB-1)

a. Inspection Scope

The team reviewed the design, testing, and operation of the ‘A’ 125 VDC station battery (71SB-1) to verify that it was capable of performing its design function of providing a reliable source of direct current (DC) power to connected loads under operating, transient, and accident conditions. The team reviewed design calculations to assess the adequacy of the battery’s sizing to ensure that it could power the required equipment for a sufficient duration, and at a voltage above the minimum required for equipment operation. The team reviewed short circuit and breaker coordination calculations to ensure that breakers were adequately sized and were capable of interrupting short circuit faults. The team verified that proper coordination existed to provide adequate isolation of the affected portion of the circuit. The team reviewed battery test results to ensure that the testing was in accordance with design calculations, the FitzPatrick TSs, and industry standards; and that the results confirmed acceptable performance of the battery. The team interviewed design engineers regarding design margin, operation, and testing of the DC system. The team performed a walkdown of the batteries, DC buses, battery chargers, and associated distribution panels to assess the material condition, configuration control, and the operating environment. Finally, the team reviewed a sample of corrective action CRs to ensure Entergy was identifying and properly correcting issues associated with the ‘A’ 125 VDC station battery.

b. Findings

No findings were identified.

.2.1.13 ‘A’ Residual Heat Removal Pump (10P-3A)

a. Inspection Scope

The team reviewed applicable portions of FitzPatrick’s TSs, the UFSAR, and the residual heat removal (RHR) system DBD to identify design basis requirements for ‘A’ RHR pump. The team reviewed design calculations and site procedures to verify that the design bases and design assumptions were appropriately translated into these documents. The team also reviewed design and operational requirements with respect to pump flow rate, developed head, achieved system flow rate, and NPSH. The team verified that minimum flow requirements as described in NRC Bulletin 88-04 were satisfied to avoid pump damage. The team verified the adequacy of pump discharge check valve testing, the associated acceptance criteria, and the measures taken to avoid flow diversion through the idle pump located on the same train. The team reviewed the adequacy of assumptions, limiting parameters, the pump's protection from the formation of air vortices, and the adequacy of its suction from the torus. The team verified that test procedures and test results were supported by design basis documents, acceptance criteria for tested parameters were supported by calculations or other engineering documents, and that tests and analyses served to validate component operation under accident and transients. The team also verified the adequacy of the pump baseline for inservice testing. The team reviewed operating and emergency operating procedures (EOP) to verify that selected operator actions could be accomplished. The team reviewed the pump motor's voltage and current requirements as well as the overcurrent relay setting and verified that the design was adequate to avoid spurious tripping. Additionally, the team reviewed control schematics to verify that the system operation complied with the system design requirements. The team performed several walkdowns of the pump, accessible RHR piping, and the adjacent area to assess the material condition, operating environment, and configuration control. The team also reviewed a sample of RHR system corrective action CRs, the RHR system health report, and applicable test results to determine if there were any adverse operating trends and to ensure that Entergy adequately identified and addressed any adverse conditions.

b. Findings

No findings were identified.

.2.1.14 Reserve Station Service Transformer T-2

a. Inspection Scope

The team inspected reserve station service transformer (RSST) T-2 to verify that it was capable of meeting its design basis requirements. The RSST is designed to provide the preferred power source to safety-related 4160V bus 10600. The team reviewed load flow, short circuit current calculations, and nameplate ratings to verify that the T-2 design basis for maximum load and breaker duty requirements were within the 10600 switchgear bus equipment vendor ratings. The team also reviewed the T-2 RSST protective relay trip settings and nameplate ratings to verify the adequacy of protection coordination under the worst case loading condition. The team also reviewed a sample of completed routine PMs and modifications associated with the T-2 RSST to ensure that test results and modifications were in accordance with design requirements. The team interviewed system and design engineers to answer questions that arose during document reviews and walkdowns to evaluate the adequacy of maintenance and configuration control. The team performed a walkdown of the T-2 transformer, surrounding transformer yard, and associated control room panels to assess the installed configuration, material condition, operating environment, and the potential vulnerability to hazards. Finally, the team reviewed corrective action documents and system health reports to verify that Entergy appropriately identified and resolved deficiencies and properly maintained the T-2 RSST.

b. Findings

No findings were identified.

2.1.15 Low Pressure Coolant Injection Valve (10MOV25A)

a. Inspection Scope

The team inspected the low pressure coolant injection (LPCI) inboard injection valve (10MOV25A) to verify that it was capable of performing its design function. The normally closed 10MOV25A valve is required to open for the LPCI system to perform its ECCS function and required to close on a shutdown cooling (SDC) isolation signal to isolate a potential reactor drain path created during SDC operations. The team reviewed applicable portions of FitzPatrick's TSs, the UFSAR, and the RHR system DBD to identify design basis requirements for 10MOV25A.

The team reviewed design calculations, including the seismic and weak link analysis, to verify that the valve met its design basis requirements. The team reviewed a sample of ST results to verify that valve performance met the acceptance criteria and that the criteria were consistent with the design basis. The team reviewed applicable RHR system operating procedures to verify technical details and adequacy, and to ensure that the procedures adequately addressed the requirements of NRC Bulletin 93-02, Supplement 1, regarding ECCS suction strainer debris plugging. The team interviewed the MOV program engineer and reviewed valve diagnostic test results and trending to assess valve performance capability, design margin, and susceptibility to recent industry OE. The team reviewed a sample of RHR system corrective action CRs, the RHR system health report, and applicable test results to determine if there were any adverse operating trends and to ensure that Entergy adequately identified and addressed any adverse conditions. The team also performed a walkdown of the valve, adjacent area, accessible portions of the RHR system, and associated control room instrumentation to assess the material condition, operating environment, and configuration control.

b. Findings

No findings were identified.

.2.1.16 419 VDC Battery (71-BAT-3A) and Inverter (71INV-3A)

a. Inspection Scope

The team reviewed the design, testing, and operation of the 419 VDC station battery (71-BAT-3A) and inverter (71INV-3A) to verify that they could perform their design function of providing a reliable source of DC power to connected MOVs under operating, transient, and accident conditions. The team reviewed design calculations to assess the adequacy of the battery's sizing to ensure it could power the required equipment for a sufficient duration, and at a voltage above the minimum required for equipment operation. The team reviewed battery test results to ensure that the testing was in accordance with design calculations, the FitzPatrick TSs, and industry standards; and that the results confirmed acceptable performance of the battery. The team interviewed design engineers regarding design margin, operation, and testing of the DC system. The team performed a walkdown of the battery and the inverter to assess the material condition, configuration control, and the operating environment. Finally, the team reviewed a sample of corrective action CRs to ensure Entergy was identifying and properly correcting issues associated with the 419 VDC battery and inverter.

b. Findings

No findings were identified.

.2.2 Review of Industry Operating Experience and Generic Issues (4 samples)

The team reviewed selected OE issues for applicability at FitzPatrick. The team performed a detailed review of the OE issues listed below to verify that Entergy had appropriately assessed potential applicability to site equipment and initiated corrective actions when necessary.

.2.2.1 NRC Information Notice 2012-14: Motor-Operated Valve Inoperable Due to Stem-Disc Separation

a. Inspection Scope

The team selected Information Notice (IN) 2012-14 for a detailed review. The team evaluated Entergy's applicability review and disposition of NRC IN 2012-14. The NRC issued this IN to inform licensees about an event where an MOV failed at the connection between the valve stem and disc. The team reviewed Entergy's actions relative to the conditions described within the IN to ensure that Entergy had performed appropriate evaluations for FitzPatrick. The team interviewed the MOV program engineer, reviewed a sample of valve diagnostic test results and trending, and reviewed a sample of Entergy's remote valve position verifications to independently assess Fitzpatrick's susceptibility to this failure mechanism.

b. Findings

No findings were identified.

.2.2.2 NRC Information Notice 2012-25: Performance Issues with Seismic Instrumentation and Associated Systems for Operating Reactors

a. Inspection Scope

The team evaluated Entergy's applicability review and disposition of NRC IN 2012-25. The NRC issued the IN to inform licensees of an occurrence where seismic instrumentation and associated monitoring and alarm systems did not provide reliable indications or alarms. In response to IN 2012-25, Entergy initiated CR-JAF-2013-00735 to review the IN and determine the applicability to FitzPatrick. Entergy had previously evaluated industry OE associated with seismic instrumentation and monitoring concerns under CR-JAF-2012-00963. In response, Entergy initiated corrective actions for procedure reviews and equipment upgrades (CR-JAF-2012-00963, CA-4). Specifically, Entergy initiated Engineering Change (EC) 37682 to upgrade and replace their existing seismic monitoring system. The team reviewed EC 37682 to verify that the design bases, licensing bases and performance capability of the seismic instrumentation system would not be adversely impacted by the planned modification. Entergy designed the replacement system to maintain battery backup and alarm functions to the plant process computer. Also, Entergy designed the replacement system to provide operations with prompt (< 4 hours) indication of the seismic event magnitude to assist in their determination if an operating basis earthquake (OBE) or a design basis earthquake (DBE) had occurred. In accordance with the guidance of IN 2012-25, Entergy planned to install three ground acceleration sensor packages: one on top of the reactor building structure, a second one near the reactor within the containment, and a third, free field sensor, located in the torus room. Entergy designed the detection capability of the replacement system to meet the original detection levels specified in FitzPatrick UFSAR Section 2.6. At the time of the inspection, the team noted that Entergy was actively engaged in the procurement, installation, and testing of the new system and planned to have the system operational by November 2013.

b. Findings

No findings were identified.

.2.2.3 NRC Part 21 Reports Regarding Failure of Scram Solenoid Pilot Valves

a. Inspection Scope

The team evaluated Entergy's applicability review and disposition of NRC Part 21 Report 1997-36-0 and NRC Part 21 Report 1997-34-2. The NRC issued these Part 21 reports to inform licensees of scram solenoid pilot valve (SSPV) issues that had the potential to adversely impact control rod scram times. The team reviewed the Entergy modification package that implemented the design change for AVCO SSPVs to replace the ASCO SSPVs to confirm that the NRC Part 21 report regarding pre-mature hardening of Buna-N diaphragms was adequately addressed. The team reviewed recent control rod scram time test results and trending data to verify that Entergy maintained control rods operable in accordance with FitzPatrick TSs and adequately addressed any adverse trends. In addition, the team reviewed corrective action CRs and system health reports to determine if there were any adverse trends associated with SSPVs and to ensure that Entergy was identifying and properly correcting issues with SSPVs. Finally, the team performed several walkdowns of the hydraulic control units (HCU) to assess the SSPV material condition and operating environment.

a. Findings

No findings were identified.

.2.2.4 NRC Information Notice 2012-17: Inappropriate Use of Certified Material Test Report Yield Stress and Age-Hardened Concrete Compressive Strength in Design Calculations

a. Inspection Scope

The team evaluated Entergy's applicability review and disposition of NRC IN 2012-17. The NRC issued the IN to inform licensees of issues identified during recent NRC inspections regarding the design of seismic Category I or safety-related structures. The team verified that Entergy entered the OE into their CAP for review (CR-JAF-2012-05339) and reviewed Entergy's evaluation and follow-up actions. The team discussed a LPCI battery room structural modification with a structural engineer and reviewed a sample of recent seismic calculations to independently assess the applicability of the industry OE to FitzPatrick.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

The team reviewed a sample of problems that Entergy had previously identified and entered into the CAP. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions. In addition, the team reviewed corrective action CRs written on issues identified during the inspection to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that the team sampled and reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA6 Meetings, Including Exit

On July 11, 2013, the team presented the inspection results to Mr. Dave Poulin, Acting General Manager, Plant Operations, and other members of Entergy management. The team verified that no proprietary information was documented in the report.

On July 31, 2013, Entergy completed an apparent cause evaluation and past-operability review for the EDG ventilation configuration control issue (see Section 1R21.2.1.9.b.1). Based on a review of these documents, the team revised the NRC's previous preliminary assessment provided at the Exit on July 11. On August 8, 2013, the team presented the updated inspection results to Mr. Dave Poulin, Operations Manager, and Mr. Chris Adner, Licensing Manager, via a conference call.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel
C. Adner, Manager, Licensing
M. Annett, EFIN Team Supervisor
A. Barton, Electrical Design Engineer
T. Britt, MOV Program Manager
D. Burch, Supervisor, Mechanical Design
B. Burnham, System Engineer
M. Cook, System Engineer
J. Cooney, PSA Engineer
B. Finn, Director, Nuclear Safety Assurance
G. Foster, Supervisor, Configuration Management
S. Glover, Electrical Design Engineer
R. Kester, Engineering Projects
B. Marks, Electrical Design Engineer
D. Poulin, Manager, Operations
M. Sansonne, System Engineer
S. Scott, Design Engineer
A. Storm, System Engineer
T. Yost, Electrical Design Engineer

NRC Personnel
E. Knutson, Senior Resident Inspector
W. Schmidt, Senior Reactor Analyst
B. Sienel, Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Open and Closed

05000333/2013007-01	NCV	Failure to correctly position EDG room ventilation temperature controllers in automatic (Section 1R21.2.1.9.b.1)
05000333/2013007-02	NCV	Failure to verify adequacy of the FOTP NPSH (Section 1R21.2.1.9.b.2)

LIST OF DOCUMENTS REVIEWED

Calculations

93-6, EC 5827, Computation of Pipe Discharge Loss for Diesel Generator Fuel Oil Transfer Pump, Revision 0

93-6, EC 22069, Pipe Discharge Loss for EDG Fuel Oil Transfer Pump, Revision 0

5016-4, Verification of ESW System Hydraulic Balance for New EDG Cooler and Bypass Piping and Restriction Orifices, Revision 2

11825, RHR Min Flow Orifice, dated 1/30/73

CALC 1.20-4, Station Battery Rack Seismic Calculation, Revision 0

E-43, Motor Feeder Cable Sizing Calculation for 10P-3A (RHR Pump Motor), dated 12/20/06

E-47, Normal and Reserve Station Service Transformer Differential Protection CT Ratios and Relay Taps, dated 12/15/09

E-51, Sizing of Neutral Grounding Equipment for Emergency Diesel Generator, Revision 0

E-76, Load Duty Cycles for LPCI MOV Power Supplies, Revision 0

ELEC-01539, RHR Motor Operation during Degraded Voltage Conditions, dated 6/2/94

JAF-CALC-01-07, ESW Flow Response to Variations in Lake Level, Strainer Differential Pressure and Pump Speed, file 12-485, Revision A

JAF-CALC-07-00020, Revised Emergency Diesel Generator Fuel Oil Storage Quantities for 7 Day and 6 Day Supplies, Revision 0

JAF-CALC-09-00016 Attachment 32.04-1, JAF Auxiliary Power System Analysis, 3-Winding Transformer Input Data, dated 12/1/11

JAF-CALC-09-00016 Attachment 36, JAF Auxiliary Power System Analysis, dated 12/20/11

JAF-CALC-12-00004, Seismic Validation for Walkway and Operator Platform, Refueling Platform FitzPatrick Nuclear Station, Revision 0

JAF-CALC-12-00026, Seismic/Weak Link Analysis of 10” Anchor Darling Gate Valve 23MOV-15, Revision 0

JAF-CALC-DGV-02026, 092/EDG Building Heating and Ventilation Sys EDG SWGR Rooms - Temp Following HELB in the Turbine Building, Revision 2

JAF-CALC-ELEC-00523, Testing Duty Cycle – LPCI UPS System, Revision 3

JAF-CALC-ELEC-00562, LPCI UPS Battery Testing Duty Cycle, Revision 0

JAF-CALC-ELEC-01488, 4KV Emergency Bus Loss of Voltage, Degraded Voltage and Time Delay Relay Uncertainty and Set-point Calculation, Revision 6

JAF-CALC-ELEC-01860, LPCI Battery Performance Test and End Volt Cutoff Due to Inverter Jumper Voltage Drop, Revision 1

JAF-CALC-ELEC-02016, 125V DC System Short-Circuit Calculation and Coordination Evaluation, Revision 0

JAF-CALC-ELEC-02213, LPCI UPS System Testing Load Bank Characteristic and LPCI Battery and Inverter On-Line Testing Condition and/or Limitation, Revision 0

JAF-CALC-ELEC-02307, Station Battery Charger Performance Test Amp-Hour Calculation, Revision 0

JAF-CALC-ELEC-02551, Determination of Float and Equalize Voltage for Station Batteries 71SB-1 & 71SB-2 for a Battery of 60 Cells, Revision 1

JAF-CALC-ELEC-02609, 125VDC Station Battery “A” Sizing and Voltage Drop, Revision 2

JAF-CALC-MISC-03340, Evaluation of HELB Barriers Including Penetration Seals, Revision 2

JAF-CALC-RHR-00633 (with DRN-05-03929), POT-10T Data Reduction, Revision 0

JAF-CALC-SWS-0326, Minimum ESW Flow Requirements for the EDG Jacket Water Coolers with Elevated Lake Temperature Up to and Including 85ºF, Revision 0

JAF-CRVE-ELEC-CC-HO3 & HO4 – Bus Coordination, Revision 0
JAF-CRVE-ELEC-CC-HO5 & HO6 – Bus Coordination, Revision 0
JAF-CRVE-ELEC-CC-HO5 & HO6 – EDG AC & BD, Revision 0
JAF-ECAF-4O5&HO6-10P-3AB&CD, 4160V Electrical Distribution System Coordination Adequacy Form, dated 2/6/95
JAF-ECAF-HO3-Bus Coordination, 4160V Electrical Distribution System Coordination Adequacy Form, dated 2/10/95
JAF-ECAF-HO5&HO6-Bus Coordination, 4160V Electrical Distribution System Coordination Adequacy Form, dated 2/10/95
JAF-ECAF-HO5&HO6-EDG-AC&BD, 4160V Electrical Distribution System Coordination Adequacy Form, dated 2/10/95
JAF-ECAF-Station Grounding, Revision 0
MDE 88-5068-C-1, Miniflow Capacity, Revision 0

Corrective Action Condition Reports (CR)

1998-3163	2010-6293	2012-1911	2012-7309	2013-3138
2007-2506	2010-6902	2012-2488	2012-8384	2013-3157
2009-2429	2010-7756	2012-2984	2012-8690	2013-3196*
2009-2514	2010-7759	2012-3015	2012-8840	2013-3240
2009-2646	2010-8288	2012-3684	2013-0202	2013-3241
2009-2881	2011-0934	2012-3891	2013-0735	2013-3247*
2009-4018	2011-2079	2012-4131	2013-1512	2013-3272
2010-2593	2011-2422	2012-4248	2013-1835	2013-3325*
2010-2984	2011-2478	2012-4266	2013-1854	2013-3334*
2010-3100	2011-2592	2012-4963	2013-2237	2013-3342*
2010-3468	2011-2593	2012-5052	2013-2271	2013-3352*
2010-3587	2011-4642	2012-5067	2013-2556	2013-3354*
2010-3609	2012-0057	2012-5339	2013-2696	2013-3403*
2010-3662	2012-0265	2012-5955	2013-2881	2013-3417*
2010-3686	2012-0491	2012-6103	2013-3083	2013-3428*
2010-3933	2012-0584	2012-6130	2013-3093*	2013-3493*
2010-3936	2012-1320	2012-6616	2013-3106*	2013-3503*
2010-4408	2012-1482	2012-6868	2013-3109*	2013-3576*
2010-5695	2012-1614	2012-7069	2013-3110*	2013-3578*
2010-6235	2012-1835	2012-7201	2013-3114*	2013-3599*

* CR written as a result of this inspection

Design and Licensing Bases

ASME OM Code-2001, Code for Operation and Maintenance of Nuclear Power Plants, ASME OMb Code-2003 Addenda
DBD-010, Residual Heat Removal System Design Basis Document, Revision 13
DBD-023, High Pressure Coolant Injection System Design Basis Document, Revision 12
DBD-046, Normal Service Water, Emergency Water, and RHR Service Water Design Basis Document, Revision 18
DBD-071 Tab I, Electrical Distribution Systems 4160V and 600V AC Power Systems Design Basis Document, Revision 7

DBD-071 Tab III, Electrical Distribution System 125V and 24V DC Power System Design Basis Document, Revision 3
DBD-092, EDG Building Heating and Ventilation System Design Basis Document, Revision 3
DBD-093, Emergency Diesel Generator (EDG) Design Basis Document, Revision 12
JAFP-05-0082, Docket No. 50-333, Technical Specification Amendment to DC Electrical System Requirements, dated 4/27/05

Drawings
1.12-41, Static Exciter and Voltage Regulator Emergency Diesel Generator Schematic Diagram, Revision 7
1.41-132, 4KV Switchgear Bus 10300 Elementary Power Circuits, Revision 0
1.41-134, Elementary Diagram-4KV Switchgear Bus 10500, Revision G
1.41-172, 4KV Switchgear Breaker 10560 and 10660 Wiring Diagram, Revision J
1.41-198, 4KV Switchgear Breaker 10502 or 10602 Wiring Diagram, Revision 20
1.41-200, 4KV Switchgear Breaker 10512 and 10612 Wiring Diagram, Revision 15
1.65-120, Outline Residual Heat Removal Pump Motor, Revision F
6.37-49, Motor Operated Valve (10MOV-25A & B), Revision 14
11825-FM-1G Sh. 7, Section 1-1, Machinery Location Reactor Building, Revision 11
11825-FV-IA, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 19
11825-FV-IC Sh. 3, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 6
11825-FV-ID Sh. 4, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 9
11825-FV-IE Sh. 5, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 6
11825-FV-IF Sh. 6, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 0
11825-FV-IG Sh. 7, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 6
11825-FV-IH, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 13
11825-FV-IK Sh. 10, JAF Drywell & Suppression Chamber Penetrations Locations & Details, dated 4/22/90
11825-FV-IL Sh. 2, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 13
11825-FV-IM Sh. 12, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 5
11825-FV-IN Sh. 13, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 7
11825-FV-IP Sh. 14, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 7
11825-FV-IQ Sh. 15, JAF Drywell & Suppression Chamber Penetrations Locations & Details, Revision 15
11825-FV-IS Sh. 17, JAF Drywell & Suppression Chamber Penetrations Locations & Details, dated 11/13/70
ESK-5BB, 4160V Circuit Emergency Generator 93 EDGA Output ACB DC Elementary Diagram, Revision 28

ESK-5BC, 4160V Circuit Emergency Generator 93 EDGC Output ACB DC Elementary Diagram, Revision 32
ESK-5BE, 600V Emergency Switchgear to Transformers T13 & T15 DC Elementary Diagram, Revision 13
ESK-5BU, 4160V Circuit Emergency Residual Heat Removal Pump 10-p-3A DC Elementary Diagram, Revision 24
ESK-5BZ, 4160V Circuit Emergency Bus 10600 Undervoltage DC Elementary Diagram, Revision 7
ESK-8C, Current Circuits Normal Station Service Transformer T4 AC Elementary Diagram, Revision 12
ESK-8E, Current Circuits Res. Station Service Transformer-T3 AC Three Line Diagram, Revision 11
ESK-8HE, Emergency Diesel Generator Excitation Elementary Diagram, Revision 21
ESK-11BA Sh. 1, Emer. Diesel Generator EDGA Control Elementary Diagram, Revision 25
ESK-11BB Sh. 2, Emer. Diesel Generator EDGA Control Elementary Diagram, Revision 18
FB-4A, JAF Reactor Bldg. Floor Drainage El. 277'-6" & 256'-6", Revision 17
FB-10H, Reactor Building Service Water Cooling System 66 JAF Flow Diagram, Revision 43
FB-35E, Control Room Area Service & Chilled Water System 70 Flow Diagram, Revision 38
FE-1AH Sh. 1, 125V DC One Line Diagram, Revision 31
FE-1AJ Sh. 2, 125V DC One Line Diagram, Revision 21
FE-1AK Sh. 3, 125V DC One Line Diagram, Revision 14
FE-1AL Sh. 4, 125V DC One Line Diagram, Revision 28
FE-1AM Sh. 5, 125V DC One Line Diagram, Revision 9
FE-1AN Sh. 6, 125V DC One Line Diagram, Revision 19
FE-1AX Sh. 7, 125V DC One Line Diagram, Revision 20
FE-1B Sh. 2, Station Service Transformers Main One Line Diagram, Revision 14
FE-1D Sh. 4, 115KV Switchyard Main One Line Diagram, Revision 10
FE-1F Sh.2, Bus 10300 4160V One Line Diagram, Revision 8
FE-1H Sh. 4, Emergency Bus 10500 4160V One Line Diagram, Revision 14
FE-1L Sh. 2, Swgr 71L15 & 71L16, 71MCC-153 & 71MCC-163 600V One Line Diagram, Revision 36
FE-1Y Sh. 14, 600V One Line Diagram (71MCC-332, 342, 155, & 165), Revision 38
FM-17A, Radwaste System 20 Flow Diagram, Revision 43
FM-46A, Service Water System 46 JAF Flow Diagram, Revision 91
FM-46B, Emergency Service Water System 46 & 15 JAF Flow Diagram, Revision 56
FPSSK-178, Fire Barrier Penetration Arrangement Fire Area/Zone IE/TB-1 El. 272', Revision 4
MSK-1899, JAF Drywell-Torus East Pipe Tunnel Piping Isometric Line N32WL-151-2, System #20 Reactor Bldg. Floor Drain Mechanical, Revision 5

Engineering Evaluations

CR-JAF-2009-02514, HELB Door 76FDR-DG-272-11 was Open and Unattended when Required to be Closed Apparent Cause Evaluation Report, dated 9/4/09
CR-JAF-2013-02700, Diesel Generator Ventilation Supply Fan Trouble Annunciator during ST-09QA Apparent Cause Evaluation Report, dated 5/29/13
CR-JAF-2013-03354 CA-3, 'A' and 'C' EDG Ventilation Temperature Controllers found out of Position Apparent Cause Evaluation Report, dated 7/31/13
CR-JAF-2013-03354 CA-7, Past Operability Evaluation for CR-JAF-2013-03354, dated 7/24/13

EC 7400, Evaluate Seismic Adequacy for a Hanes Supply Load Rail Ceiling-Hung Bridge Crane to Remain Installed in the B LPCI Battery Room with 71BAT-3b Operable, Revision 0
EC 37682, Replace Seismic Monitoring System, dated 5/24/12
EC 40367, Degraded Areas Identified from IWE General Visual Exams - Torus Interior, Revision 0
JAF-RPT-05-00100, Laboratory Investigation of Torus Wall Cracking, Revision 0
JAF-SE-83-003, Mark I Torus Modification – Phase III, Revision 0
JAF-SE-88-185, Additional Torus and Drywell Pressure Indications, dated 9/27/88
JAF-SE-91-038, Modifications to the Torus Temperature Monitoring System, dated 4/11/91
JE-02-127, EDG Fuel Oil Transfer Pump Motor Replacement, Revision 0

<u>Functional, Surveillance and Modification Acceptance Testing</u>
IMP-60, Seismic Recording System Functional Test, performed 5/9/13
IMP-71.2, Overcurrent Relays - Type IAC66K Calibration, performed 04/26/11
IMP-71.3, General Electric Type PJC11AV Overcurrent Relay Calibration, performed 03/4/08
IMP-71.4, General Electric Type IAC51A and IAC53A Overcurrent Relay Calibration, performed 9/12/10
ISP-29, Suppression Chamber Water Level HPCI Instrument Functional Test/Calibration, performed 2/16/13
ISP-90-2, 4KV Residual Bus Transfer Functional Test, performed 10/7/12
ISP-100A-PCIS, PCIS Instrument Functional Test/Calibration (ATTS), performed 3/18/13
ISP-100B-PCIS, PCIS Instrument Functional Test/Calibration (ATTS), performed 3/19/13
ISP-100C-PCIS, PCIS Instrument Functional Test/Calibration (ATTS), performed 3/18/13
ISP-100D-PCIS, PCIS Instrument Functional Test/Calibration (ATTS), performed 3/19/13
ISP-103A, PCIS Group 1 Isolation Reactor Level Instrument Response Time Test (ATTS), performed 12/12/12
ISP-103B, PCIS Group 1 Isolation Reactor Level Instrument Response Time Test (ATTS), performed 5/18/12
ISP-105A, MSIV Closure Low Main Steam Line Pressure Response Time Test (ATTS), performed 1/14/13
ISP-105B, MSIV Closure Low Main Steam Line Pressure Response Time Test (ATTS), performed 5/18/12
ISP-106A, MSIV Closure High Steam Line Flow Response Time Test (ATTS), performed 1/14/13
ISP-106B, MSIV Closure High Steam Line Flow Response Time Test (ATTS), performed 5/18/12
ISP-125A, HPCI Auto Isolation Instrument Functional Test/Calibration, performed 2/12/13
ISP-125B, HPCI Auto Isolation Instrument Functional Test/Calibration, performed 2/12/13
MST-071.10, LPCI Battery Weekly Surveillance Test, performed 5/13/13
MST-071.11, LPCI Battery Quarterly Surveillance Test, performed 4/1/13
MST-071.12, 125 VDC Station Battery and Charger Weekly Surveillance Test, performed 5/13/13
MST-071.13, 125 VDC Station Battery Quarterly Surveillance Test, performed 4/29/13 & 10/5/12
MST-071.25, LPCI Battery Modified Performance Test, performed 2/6/13
MST-071.26, Station Battery A Modified Performance Test, performed 10/4/12
MST-071.29, LPCI Charger-Inverter Performance Surveillance Test, performed 2/7/13
RAP-7.4.01, Control Rod Scram Time Evaluation, performed 10/11/12 and 3/21/13
ST-1B, MSIV Fast Closure Test (IST), performed 3/3/13
ST-1F, Condenser Air Removal Pump and Valve Isolation Logic System Functional Test, performed 10/9/10, 10/14/12 and 10/15/12

ST-2CA, RHR Loop A LPCI Inboard Injection Valve Operability Test (IST), performed 3/3/13
ST-2JA, 10MOV-25A Leakage Test (IST), performed 6/21/12
ST-4F, HPCI Automatic Isolation Logic System Functional and Simulated Automatic Actuation Test, performed 2/12/13
ST-4N, HPCI Quick-Start, Inservice, and Transient Monitoring Test (IST), performed 5/13/13
ST-8Q, Testing of the Emergency Service Water System (IST), performed 2/3/12 & 2/13/13
ST-9AA, EDG System A Fuel/Lube Oil Monthly Test, performed 4/1/13
ST-9CA, EDG A and C Load Sequencing Test and 4KV Emergency Power Voltage Relays Instrument Functional Test, performed 10/3/12
ST-9NA, EDG Subsystem A Logic System Functional Test, performed 10/4/12
ST-9PA, Functional Test of Breaker 10514 EDG Logic Auxiliary Stationary Contacts, performed 10/3/12
ST-9QA, EDG A and C Full Load Test (8 Hour Run), performed 5/23/11
ST-9Y, Manual Transfer Test of 10300 and 10400 Bus from Normal to Reserve, performed 9/15/12
ST-15B, Suppression Chamber and Drywell Deterioration Inspection, performed 6/8/09
ST-15G, Pressure Suppression Chamber - Reactor Building Vacuum Breaker Operability and Setpoint Test (IST), performed 12/8/09, 12/15/09, and 2/24/12
ST-16GA, A LPCI MOV Independent Power Supply Monthly Test, performed 4/29/13
ST-39E, Pressure Suppression Chamber - Drywell Vacuum Breaker Leak Test (IST), performed 12/9/07, 2/4/08, and 9/9/12
ST-41D, Remote Valve Position Indication Verification Online (IST), performed 9/21/12
ST-41K, Remote Valve Position Indication Verification Shutdown (IST), performed 10/6/12

Maintenance Work Orders

00147788	01086700	52216405	52288862	52415872
00293179	51204114	52233547	52347826	
00345375	51690168	52250873	52347827	
01077200	52185637	52288704	52404062	

Miscellaneous
DRN No. 13-00312, ST-9AA EDG System A Fuel/Lube Oil Monthly Test Temporary Change Form, dated 6/24/12
DRN No. 13-00313, ST-9AB EDG System B Fuel/Lube Oil Monthly Test Temporary Change Form, dated 6/25/12
EN-DC-149, Test Report of General Electric 4KV Switchgear and Breaker Momentary, Close & Latch and Interrupting Ratings, dated 12/20/11
GDCD-E15, Protection and Coordination of Plant Electrical Systems, Revision 0
IEEE 450-2010, IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Application, dated 12/9/10
IEEE 485-2010, IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications, dated 11/8/10
IST-02-03, Basis for the Exclusion of EDG Support Systems from IST Testing, dated 2/19/02
JAF-ICD-ELEC 04347, Tolerances for G.E. Electro-Mechanical Overcurrent Relay Calibration, Revision 2
JAF-RPT-MULTI-00746, Generic Letter 89-10/96-05 Motor Operated Valve Program Plan, Revision 18

Letter from General Electric Company to Carolina Power and Light Company, RHR and Core Spray Minimum Flow Limitation for Byron Jackson DVDS Pumps, dated 11/22/89
Modification No. JF-03-01758, Scram Solenoid Valve Replacement, Revision 0
ODSO-17 Control Room Tour, dated 6/22/13 - 6/25/13
ODSO-17 Float Tour, dated 6/22/13 - 6/25/13
ODSO-17 Reactor Building Tour, dated 6/22/13 - 6/25/13
Operations Department Equipment Status and Temporary Label Control Log, dated 6/12/13
PM Evaluation Report, 23INV-79 INVERT, dated 4/17/12
SEP-AP-J-007, Primary Containment Leakage Rate Testing (Appendix J) Program, Revision 5
SEP-ISI-007, Inservice Testing for Pumps and Valves, Fourth Ten Year Interval Program Section, dated 8/9/12

Normal and Special (Abnormal) Operations Procedures
AOP-14, Earthquake, Revision 13
AOP-18, Loss of 10500 Bus, Revision 15
AOP-19, Loss of 10600 Bus, Revision 14
AOP-56, High Traveling Screen or Trash Rack Differential Level, Revision 10
AOP-64, Loss of Intake Water Level, Revision 8
ARP 09-75-1-4, EDG Vent Sys A Trouble, Revision 2
ARP 93ECP-A-6, Low Day Tank Level, Revision 4
ARP HV-9A-02, Room A Temp High, Revision 3
EOP-4, Primary Containment Control, Revision 8
ODSO-17, Operator Plant Tour and Operating Logs, Revision 81
OP-13, Residual Heat Removal System, Revision 96
OP-21, Emergency Service Water (ESW), Revision 38
OP-22, Diesel Generator Emergency Power, Revision 58
OP-46A, 4160V and 600V Normal AC Power Distribution, Revision 59
OP-60, Diesel Generator Room Ventilation, Revision 8
OP-65A, Normal Operation, Revision 11

Operating Experience
GE SIL No. 622, Noisy Operation of Dual-type Scram Solenoid Pilot Valves, dated 11/10/98
Letter PEM-88-026, Final Response to NRC Bulletin 88-04, dated 1/25/89
Letter PEM-88-331, Technical Evaluation of NRC Bulletin 88-04, dated 7/7/88
LO-JAFLO-2012-00004, OE34934-20120107 - Nuclear Regulatory Commission Red Finding Root Cause Analysis Results OE Review, dated 2/29/12
NRC Bulletin 88-04, Potential Safety-Related Pump Loss, dated 5/5/88
NRC Bulletin 93-02 Supplement 1, Debris Plugging of Emergency Core Cooling Suction Strainers, dated 2/18/94
NRC Generic Letter 2003-01, Control Room Habitability, dated 6/12/03
NRC Information Notice 2007-27, Recurring Events Involving Emergency Diesel Generator Operability, dated 8/6/07
NRC Information Notice 2011-12, Reactor Trips Resulting from Water Intrusion into Electrical Equipment, dated 6/16/11
NRC Information Notice 2012-03, Design Vulnerability in Electric Power System, dated 3/1/12
NRC Information Notice 2012-11, Age-Related Capacitor Degradation, dated 7/23/12
NRC Information Notice 2012-25, Performance Issues with Seismic Instrumentation and Associated Systems for Operating Reactors, dated 2/1/13

NRC Part 21 1997-34-2: Potential Safety-Related Problem with ASCO HV 266000-007J Scram Solenoid Pilot Valves, dated 5/27/97
NRC Part 21 1997-36-0: Failure of Scram Solenoid Pilot Valves, dated 5/1/97

Preventive Maintenance, Calibrations and Inspections
ISP-29, Suppression Chamber Water Level HPCI Instrument Functional Test/Calibration, performed 9/14/11
ISP-64-3A, Main Steam Radiation Monitor Channel Calibration, performed 3/14/13
ISP-64-3B, Main Steam Radiation Monitor Channel Calibration, performed 3/19/13
ISP-200A, RPS-PCIS (A1 Channel) Pressure Transmitter Calibration (ATTS), performed 10/6/12
ISP-200C, RPS-PCIS (A2 Channel) Pressure Transmitter Calibration (ATTS), performed 9/21/12
ISP-201C, RPS-PCIS Reactor Level Transmitter Calibration (ATTS), performed 10/7/12
ISP-201D, RPS-PCIS Reactor Level Transmitter Calibration (ATTS), performed 9/22/12
ISP-202, Main Steam Line High Flow Transmitter RTD Calibration and Channel Functional Test (ATTS), performed 9/19/12
ISP-203, PCIS Main Steam Line High Temperature Instrument RTD Calibration, performed 9/18/12
MP-071.42, Station Service Transformer Maintenance 71T-2, 71T-3 and 71T-4, performed 9/11/11
MP-076.20, Fire Damper Maintenance (92FD-1), performed 5/21/13
MP-093.04, EDG Electrical Preventive Maintenance, performed 9/13/11 & 4/24/13
MP-093.11, EDG System Mechanical PM, performed 4/23/13
PM-71EDC-10025, South 115KV Bus Reserve Station Transformer T-2 Disconnect Switch, performed 6/25/10
PMQR 50057049-01, LR-PM Perform Visual Exam; Torus and ADS System Tailpipes, performed 10/2/12
SP-03.01, Main Steam Line and SJAE Radiation Monitor Calibration, performed 10/7/12

Procedures
EN-DC-132, Control of Engineering Documents, Revision 5
EN-DC-311, MOV Periodic Verification, Revision 2
EN-DC-312, Motor Operated Valve (MOV) Test Data Review, Revision 2
IMP-093.6, Emergency Diesel Generator FODT Level Functional Test, Revision 14
MP-054.01, 4.16KV Magne Blast Breakers, Revision 31
MP-059.45, Piston Check Valves, Revision 13
MP-093.11, EDG System Mechanical PM, Revision 43
MP-093.12, HCU Scram Inlet & Outlet Valve Maintenance (03AOV-126 & 127), Revision 17
MST-059.45, Piston Check Valves Inspection (IST), Revision 3
ST-1B, MSIV Fast Closure Test, Revision 25
ST-2AL, RHR Loop A Quarterly Operability Test, Revision 33
ST-9BA, EDG A and C Full Load Test and ESW Pump Operability Test, Revision 14
ST-39B, Type B and C LLRT of Containment Penetrations, Revision 33
ST-39B-X7C, Type C Leak Test Main Steam Line C MSIVs, Revision 14
ST-39B-X9A/B, Type C Leak Test of FWS Line A and B Valves, Revision 9
ST-39B-X13A, Type C Leak Test of 1-MOV-25A, Revision 10
ST-39J, Leak Testing of RHR and Core Spray Testable Check Valves, Revision 17
ST-41J, Inservice Check Valve Testing, Revision 10
TST-1, EDG Air Starting Reservoir Capacity Test, dated 9/29/88

Risk and Margin Management

NUREG/CR-5485, INEEL/EXT-97-01327, Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment, dated November 1998

System Health, System Walkdowns, and Trending

4 KV Distribution System Health Report, Q1-2013
71-345 and 115KV Distribution System Health Report, Q1-2013
600V AC Distribution System Health Report, Q1-2013
CRD Hydraulic System Health Report, Q1-2013
DC Distribution System Health Report, Q1-2013
EDG System Health Report, Q1-2013
Emergency Service Water System Health Report, Q3-2012, Q4-2012, & Q1-2013
Feedwater System Health Report, Q2-2013
HPCI System Health Report, Q1-2013
Main Steam System Health Report, Q1-2013
RHR & RHRSW System Health Report, Q1-2013

Vendor Technical Manuals and Specifications

A180-0052(A08), ESW Pump Vendor Manual, dated 1/27/02
E095-0193 (E04), Edward Valve Main Steam Isolation Valve, Revision 3
K238-C002, Kraissl Single Separator Strainers and Filters Class 72 Series, Revision 0
M494-0208, MKW Power Systems Model # 645-S20E4GW Emergency Diesel Generator, Revision 7
N989-0002, Instruction Manual for Inverter Assembly P/N 250-125-117F, Revision 0
PDW 3400-400/600, Operating Instruction Uninterruptible Power Supply, Revision 4
R374-C005, Roper Pumps Bulletin 1-83-9 Series AM, AP and AL Installation and Operation Instructions, dated 10/11/95
V080-0013, Velan Forged Steel Valves Bolted Bonnet, Revision 12

LIST OF ACRONYMS

ADAMS	Agency-Wide Documents Access and Management System
ACE	Apparent Cause Evaluation
AOP	Abnormal Operating Procedure
AOV	Air-Operated Valve
ARP	Alarm Response Procedure
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CAP	Corrective Action Program
CCF	Common Cause Failure
CDBI	Component Design Bases Inspection
CFR	Code of Federal Regulations
CR	Condition Report
DBD	Design Basis Document
DBE	Design Basis Earthquake
DC	Direct Current
DRS	Division of Reactor Safety
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFIN	Electrical Fix-It-Now
EOP	Emergency Operating Procedure
EQ	Environmental Qualification
ESW	Emergency Service Water
FOTP	Fuel Oil Transfer Pump
FOST	Fuel Oil Storage Tank
FSS	Field Support Supervisor
HCU	Hydraulic Control Unit
HPER	Human Performance Error Review
HPCI	High Pressure Coolant Injection
Hz	Hertz
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IN	Information Notice
IP	Inspection Procedure
IST	In-Service Test
JAF	James A FitzPatrick
KV	Kilovolt
LCO	Limiting Condition for Operation
LERF	Large Early Release Fraction
LOOP	Loss-of-Offsite Power
LPCI	Low Pressure Coolant Injection
MOV	Motor-Operated Valve
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NPO	Nuclear Plant Operator
NPSH	Net Positive Suction Head

NRC	Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OE	Operating Experience
P&ID	Piping and Instrument Diagram
PM	Preventive Maintenance
PMT	Post Maintenance Test
PRIB	Plant Risk Information e-Book
PSA	Probabilistic Safety Assessment
RAW	Risk Achievement Worth
RHR	Residual Heat Removal
RPM	Revolutions Per Minute
RRW	Risk Reduction Worth
RSST	Reserve Station Service Transformer
SDC	Shutdown Cooling
SDP	Significance Determination Process
SPAR	Standardized Plant Analysis Risk
SSPV	Scram Solenoid Pilot Valve
ST	Surveillance Test
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
V	Volt