UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PA 19406-2713

August 6, 2015

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS TEAM INSPECTION REPORT 05000333/2015007

Dear Mr. Sullivan:

On June 26, 2015, 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 June 26, 2015, with Mr. T. Peter, 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 reviewed selected procedures, calculations and records, observed activities, and interviewed station personnel.

This report documents one NRC-identified finding of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance and because the finding was entered into your corrective action program, the NRC is treating the finding as a non-cited violation (NCV), consistent with Section 2.3.2 of the NRC's Enforcement Policy. If you contest the 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; 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 addition, if you disagree with the cross-cutting aspect assigned to the finding in this report, you should provide a response, within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at the James A. FitzPatrick Nuclear Power Plant.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 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 public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management 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 05000333/2015007
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.:	05000333/2015007
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, New York
Inspection Period:	June 8 through June 26, 2015
Inspectors:	J. Ayala, Reactor Inspector, Division of Reactor Safety (DRS), Team Leader J. Schoppy, Senior Reactor Inspector, DRS B. Haagensen, Millstone Resident Inspector, Division of Reactor Projects (DRP)
Approved By:	Paul G. Krohn, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000333/2015007; 6/8/2015-6/26/2015; James A. FitzPatrick Nuclear Power Plant (FitzPatrick); Permanent Plant Modifications Engineering Team Inspection.

This report covers a 2 week on-site inspection period of the evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three region based engineering inspectors. One finding of very low safety significance (Green) was identified, which was considered to be a non-cited violation. The significance of most findings is indicated by a color (Green, White, Yellow, Red) using 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 5, dated February 2014.

<u>Inspector Identified and Self-Revealing Findings</u>

Cornerstone: Initiating Events

- <u>Green</u>. The inspectors identified a Green, non-cited violation (NCV) of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix B, Criterion III, Design Control, associated with FitzPatrick’s failure to adequately assess and control the acceptance criteria specified in engineering analysis in EC-JAF-56258, “Operability Input for CR-JAF-2015-01271 SRV G Tailpipe Temperature Increase,” which referenced JAF-RPT-03-0056 “Operational Leakage Action Levels for Target Rock Two-Stage Safety/Relief Valves.” Specifically, FitzPatrick concluded that a 2-stage Target Rock Safety Relief Valve (SRV) was operable with pilot valve leakage provided the leak rate was less than 1000 lbm/hr. This conclusion was not adequately supported by the available industry and plant data on setpoint drift and the references provided. As a result, FitzPatrick did not declare 2-stage Target Rock Pilot valves inoperable when the leak rate exceeded 600 lbm/hr in 2007 and 2009. FitzPatrick entered this issue into the corrective action system (CR-JAF-2015-02850) and is reassessing the appropriate operability criteria.

  This performance deficiency is more than minor because it adversely affects the equipment performance attribute of the initiating events cornerstone in IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations by ensuring reactor coolant system (RCS) barrier integrity. This finding screens to Green using IMC 0609, “Significance Determination Process,” Attachment 4, “Initial Characterization of Findings,” and IMC 0609, Appendix A, Exhibit 1, “Initiating Events Screening Questions,” Section A, “LOCA Initiators,” as the finding could not result in leakage exceeding that of a small break loss-of-coolant accident (LOCA) nor could it have resulted in an interfacing system LOCA. The inspectors determined that this performance deficiency had a cross-cutting aspect in human performance, conservative bias, where individuals use decision making-practices that emphasize prudent choices over those that are simply allowable. [H.14] Section 1R17

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (IP 71111.17)

.1 Evaluations of Changes, Tests, or Experiments (20 samples)

a. Inspection Scope

The team reviewed one safety evaluation to evaluate whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance Title 10 of the *Code of Federal Regulations* (10 CFR) 50.59 requirements. In addition, the team evaluated whether Entergy had been required to obtain U.S. Nuclear Regulatory Commission (NRC) approval prior to implementing the changes. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, Technical Specifications (TS), and plant drawings to assess the adequacy of the safety evaluation. The team compared the safety evaluation and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, “Guidelines for 10 CFR 50.59 Evaluations,” Revision 1, as endorsed by NRC Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” to determine the adequacy of the safety evaluation.

The team also reviewed a sample of nineteen 10 CFR 50.59 screenings for which Entergy had concluded that a safety evaluation was not required. These reviews were performed to assess whether Entergy’s threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample included design changes, calculations, and procedure changes.

The team reviewed the safety evaluation and screenings that Entergy had performed and approved during the time period covered by this inspection not previously reviewed by NRC inspectors. All 50.59 safety evaluations completed since the last modifications inspection were reviewed, and the screenings and applicability determinations selected were based on the safety significance, risk significance, and complexity of the change to the facility.

In addition, the team compared Entergy’s administrative procedures used to control the screening, preparation, review, and approval of safety evaluations to the guidance in NEI 96-07 to evaluate whether the procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations and screenings are listed in the Attachment.

b. Findings

No findings were identified.

.2 Permanent Plant Modifications (12 samples)

.2.1 Operability Input for CR-JAF-2015-01271 SRV "G" Tailpipe Temperature Increase

a. Inspection Scope

The team reviewed EC-JAF-56258 that established limits for pilot valve leakage through Target Rock two-stage safety relief valves (SRVs). The SRVs are used to provide protection for reactor vessel pressure transients and allow the control of reactor pressure under various operational and upset conditions. During the last operating cycle, the 02RV-71G SRV showed indications of reactor coolant system (RCS) bypass leakage into the torus in March and May of 2015. SRV tailpipe temperatures had exceeded 212°F for periods of one week. Engineering provided an operability input to the plant operators that recommended 02RV-71G remained operable because the leakage into the torus remained less than the Two-Stage Single Valve Shutdown Limit of 1000 pounds-mass per hour (lbm/hr) as established in report JAF-RPT-03-00056. This report referred to the results and conclusions established in General Electric (GE) BWROG report NEDE 33097 "Guide for Addressing Leaking Safety/Relief Valves in Boiling Water Reactors," Revision 0, dated March 2003 and Target Rock Engineering Test Report Number 3892, dated August 1983.

The team reviewed the modification, results, and conclusions in the associated reports to verify that there was reasonable assurance that the SRVs could perform their safety functions with pilot valve leakage up to a limit of 1000 lbm/hr. The team verified that the design inputs, assumptions and conclusions of the engineering change (EC). The team interviewed engineers, licensed operators, and licensing personnel. The team walked down the SRV controls and monitoring panels in the control room and reviewed CRs and control room logs to verify monitoring of the Two-Stage Single Valve Shutdown leakage limits. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

Introduction. The inspectors identified a finding of very low safety significance (Green) involving a non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for Entergy's failure to conduct an adequate analysis in setting the acceptance criteria in EC-JAF-56258, "Operability Input for CR-JAF-2015-01271 SRV "G" Tailpipe Temperature Increase."

<u>Discussion</u>. In May 2015, the "G" SRV was noted to be leaking by plant operators in CR-JAF-2015-01271, "SRV "G" Tailpipe Temperature Increase." Operations requested an engineering assessment on SRV operability. Engineering recommended that the "G" SRV remained operable with a single valve leak rate of approximately 30 lbm/hr. The conclusion was documented and communicated in Engineering Change EC-JAF-56258, which referred to the prior analysis in FitzPatrick report JAF-RPT-03-0056, "Operational Leakage Action Levels for Target Rock Two-Stage Safety/Relief Valves." The inspectors reviewed this analysis and determined that FitzPatrick's conclusion that the two-stage single pilot valve leakage limit for SRV operability of 1000 lbm/hr was unsubstantiated based on the referenced test data in two vendor reports: 1) General Electric Report NEDE 33097; and 2) Target Rock Engineering Test Report Number (TRETR) 3892.

Regarding the vendor reports, the inspectors observed that in 1983, Target Rock issued TRETR 3892 which advised the industry that pilot valve leakage greater than approximately 400 lbm/hr would cause the SRV setpoint to drift in the low direction. Prior to July 2003, FitzPatrick used the Target Rock analysis and plant specific information to establish a limit of 600 lbm/hr leakage rate for operability. FitzPatrick subsequently performed an engineering evaluation (JAF-RPT-03-0056) in July 2003, that surveyed existing industry operating experience and justified extending the allowable SRV pilot valve leak rates from 600 lbm/hr to 1000 lbm/hr. FitzPatrick determined that the setpoint drift phenomenon described in TRETR 3892 was not supported based on more recent information provided in General Electric (GE) Report NEDE 33097. Specifically, FitzPatrick concluded that extensive testing of in-service SRVs did not support Target Rock's conclusions that the valve setpoint would drift low when leak rates exceeded approximately 400 lbm/hr and that SRV pilot leakage rates below 1000 lbm/hr was not a predictor of setpoint drift. Accordingly, FitzPatrick determined that pilot valve leakage below 1000 lbm/hr would not cause excessive degradation such that SRV setpoint drift would exceed the limiting condition of operation (LCO) of ± 3 percent setpoint drift limits stated in TS 3.4.3.

FitzPatrick subsequently used JAF-RPT-03-0056 to justify operability for measured SRV leakage rates in excess of 600 lbm/hr on at least two occasions. First, in 2004 CR-JAF-2004-01229 reported that total torus in-leakage associated with the SRV "F" was projected to exceed 650 lbm/hr by the end of the operating cycle. Second, in 2007, CR-JAF-2007-02607 reported that SRV "F" was leaking at 750 lbm/hr. The affected SRVs were assessed as operable provided leakage remained below 1000 lbm/hr based on the conclusions in JAF-RPT-03-0056. Furthermore, FitzPatrick more recently validated the conclusions in JAF-RPT-03-0056 in an assessment completed in June 2015.

The inspectors questioned the validity of FitzPatrick's conclusions in JAF-RPT-03-0056 given that the test data in GE NEDE 33097 was limited to only four data points (on two SRVs) that experienced leakage above 600 lbm/hr. Furthermore, three of the four data points associated with SRVs that when removed and tested, reveled as-found lift setpoints that were greater than the LCO upper limit of +3 percent or 1179.3 psig.

Additionally, the inspectors noted that NEDE 33097 Section 5-1 cautioned: "The available data suggest the relationship between [SRV] leakage and set pressure is weaker than indicated by the Target Rock tests. However, there is a need for more as-found data to better assess the relationship, especially for SRV's with leakages >400 lbm/hr. Plants may want to correlate their own as-found set pressure data to as-found leakage rate to justify that this threat/risk is not applicable for their SRVs." Furthermore, Appendix A to this report stated "…the data for leakage rates above 200 lbm/hr is limited (i.e., only six data points above this value)."

The inspectors observed that while the limited data in the GE report may generally support operability for leakage rates to 600 lbm/hr, the four data points that existed for SRV leakage above 600 lbm/hr did not correlate with the TRETR 3892 bench test results which predicted the lift setpoint will drift in the low direction. Furthermore, the inspectors determined that while the 1000 lbm/hr limit may be appropriate for reactor shutdown to prevent spurious SRV actuations, it does not provide reasonable assurance that an individual SRV setpoint will not drift above the TS limit for operability as three of four data points exceeded the LCO limit of +3 percent (1179.3 psig).

<u>Analysis</u>. The inspectors determined that the failure to adequately assess and evaluate the impact of SRV pilot valve leakage on the setpoint drift of the SRVs was a performance deficiency that was within the Entergy's ability to foresee and prevent. This performance deficiency is more than minor because it adversely affects the equipment performance attribute of the initiating events cornerstone in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations by ensuring RCS barrier integrity. In addition, the performance deficiency is similar to example 3.j in IMC 0612 Appendix E, "Examples of Minor Issues," and supports the conclusion that the performance deficiency was more than minor because the engineering analysis error resulted in a condition where there is now a reasonable doubt on the operability of the SRV. Reliable operation of the SRVs ensures the availability and reliability of the RCS barrier to protect the health and safety of the public. This finding screens to Green using IMC 0609, "Significance Determination Process," Attachment 4, "Initial Characterization of Findings," and IMC 0609, Appendix A, Exhibit 1, "Initiating Events Screening Questions," Section A, "LOCA Initiators," as the finding could not result in leakage exceeding that of a small break LOCA nor could it have resulted in an interfacing system LOCA. If the setpoint on a single SRV drifts above 3 percent of the upper limit, prior FitzPatrick accident analyses have shown that the redundant SRVs will have sufficient capacity to limit RCS pressure below safety limits.

The inspectors determined that this performance deficiency had a cross-cutting aspect in human performance, conservative bias, where individuals use decision making-practices that emphasize prudent choices over those that are simply allowable. A proposed action is determined to be safe in order to proceed, rather than unsafe in order to stop. While the original decision to increase the single valve shutdown leak rate limit from 600 to 1000 lbm/hr was made in 2004, a recent (June 2015) analysis of site-specific operating experience "confirm[ed] the visual observation that pilot valve leakage is not a predictor of SRV setpoint drift" and is reflective of current performance. FitzPatrick recently

analyzed the site-specific operating experience and reached the same conclusion that the SRVs were operable as long as the pilot valve leak rate remained below 1000 lbm/hr. This conclusion was based on a data set that did not have any SRV with leakage greater than 600 lbm/hr. Furthermore, lift setpoint test data shows that six of 11 SRV tests (55 percent) showed lift setpoint drift above the +3 percent limit in TS for pilot valve leakage greater than 50 lbm/hr. A reasonable expectation of operability was not established above 50 lbm/hr (where the demonstrated failure probability is greater than 50 percent) and no site-specific data existed for pilot valve leakage above 600 lbm/hr. [H.14]

Enforcement. Title 10 CFR Part 50 Criterion III, "Design Control," states, in part, "Measures shall be established for the … review for suitability of application of … equipment, and processes that are essential to the safety-related functions of the structures, systems and components… The design control measures shall provide for verifying or checking the adequacy of design, such as by … the use of alternate or simplified calculational methods, or by the performance of a suitable testing program… Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing under the most adverse design conditions…Design control measures shall be applied to items such as the following: … delineation of acceptance criteria for inspections and tests."

Contrary to the above, Entergy did not ensure the adequacy of the assessment for the impact of SRV leak rate on setpoint drift that was documented in JAF-RPT-03-0056 and did not delineate adequate acceptance criteria for operability determinations from May 2004 until July 2015. In addition, when provided an opportunity to revise the analysis conclusion during June 2015, FitzPatrick analyzed site-specific operating experience from 2003 to 2015 and reached the same conclusion. Because this issue is of very low safety significance (Green) and Entergy has taken corrective action and entered this issue into their Corrective Action Program (CAP) (CR-JAF-2015-02850), this finding is being treated as an NCV consistent with the NRC Enforcement Policy Section 2.3.2. FitzPatrick is continuing their analysis of this issue. **(NCV 05000333/2015007-01 Failure to Adequately Assess the Impact of SRV Leakage on Operability)**

.2.2 High Pressure Coolant Injection Steam Supply Valve 23MOV-15 Torque Switch Bypass

a. Inspection Scope

The team reviewed EC-JAF-39906 that electrically bypassed the torque switch in motor operated valve (MOV) operator 23MOV-15. Valve 23MOV-15 is the high pressure coolant injection (HPCI) steam supply inboard isolation valve. The valve is normally open and provides a flow path for steam to the HPCI turbine. The valve also has a safety function to automatically close upon initiation of a HPCI steam line break to provide primary containment isolation. Entergy implemented the modification to eliminate the potential for the torque switch to adversely impact the design safety function of 23MOV-15 to close on demand.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the HPCI system had not been degraded by the modification. The team interviewed design engineers, and reviewed evaluations, MOV calculations, surveillance results, and associated maintenance work orders to verify that Entergy appropriately implemented the modification in accordance with design assumptions. The team also performed several walkdowns of the accessible portions of the HPCI system, including control room instrumentation, to independently assess Entergy's configuration control and the material condition of the HPCI system.

The team also reviewed corrective action condition reports (CRs) and the HPCI system health report to determine if there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.3 Emergency Diesel Generator Air Start Compressor Motor Replacement

a. Inspection Scope

The team reviewed EC-JAF-27880 that replaced emergency diesel generator (EDG) air start compressor motors 93AC-A2(M) and 93AC-C2(M). The EDG A2 and C2 air start compressor motors are the prime movers for the EDG air compressors (93AC-A2 and 93AC-C2) and are located in the "A" and "C" EDG rooms, respectively. Each EDG is provided with independent and redundant air start systems with individual air compressors to furnish air for automatic and manual starting. The pre-existing EDG air start compressor motors were obsolete necessitating Entergy to develop and implement this modification to maintain system reliability and support routine preventive maintenance activities.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the EDGs and their support systems had not been degraded by the modification. The team reviewed the associated work order instructions and documentation to verify that maintenance personnel had implemented the modification as designed. The team performed several walkdowns of the EDG air start systems (including a walkdown during an A/C EDG monthly surveillance on June 15) to independently assess Entergy's configuration control, system performance, and the material condition of the associated SSCs. The team reviewed EDG monthly surveillances, equipment operator logs of EDG air bank pressures, and operating procedures to verify that Entergy adequately maintained the EDG air start systems as designed. The team also reviewed corrective action CRs and the EDG system health report to determine if there were reliability or performance issues associated with the EDG air start systems. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.4 High Pressure Coolant Injection Turbine Steam Supply Drain Trap Bypass Valve Replacement

a. Inspection Scope

The team reviewed EC-JAF-11595 that replaced the HPCI turbine steam supply drain trap bypass valve (23AOV-53). During normal operation, the HPCI turbine steam inlet line drain pot level switch, 23LS-90, senses the condensate level in the turbine inlet line drain pot and, if excessively high, initiates a signal to energize the solenoid valve for 23AOV-53, which opens the valve to bypass the steam trap. Valve 23AOV-53 is a normally closed, air-operated valve (AOV) with a safety-related function to maintain the HPCI system pressure boundary. The pre-existing bypass valve had a history of seat leakage after short periods of service resulting in more frequent maintenance and valve replacement, approximately every 2 years. Entergy implemented this modification to evaluate a replacement steam supply drain trap bypass valve and actuator with a robust design more resilient to seat leakage caused by steam cutting.

The team reviewed the modification to verify that the design bases, licensing bases, and pressure boundary of the HPCI system had not been degraded by the modification. The team interviewed design engineers, and reviewed calculations, evaluations, surveillance and post maintenance test results, and associated maintenance work orders to verify that Entergy implemented the modification as designed. The team also performed several walkdowns of the accessible portions of the HPCI system (including 23AOV-53) to ensure that the system configuration was in accordance with design instructions, the HPCI pressure boundary was maintained, and that important-to-safety SSCs in the vicinity were not adversely impacted. The team also reviewed corrective action CRs and the HPCI system health report to determine if there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.5 Standby Liquid Control Pump Motor Space Heater Modification

a. Inspection Scope

The team reviewed EC-JAF-40796 that permanently removed power to the “B” standby liquid control (SLC) pump motor’s internal space heater. Both SLC pump motors were originally designed and equipped with space heaters to prevent condensation buildup in

the motor windings. The “A” SLC pump motor’s space heater remained operational; however, in 2005, operators identified in CR-JAF-2005-01110 that the space heater for the “B” SLC pump motor failed to function properly.

The team reviewed the modification, SLC pump testing results, and associated Environmental Qualification (EQ) scoping evaluations to verify that the design bases, licensing bases, and performance capability of the “B” SLC pump had not been degraded by the modification. The team interviewed design engineers and reviewed the associated maintenance work orders to verify that Entergy appropriately implemented the modification. The team also performed several walkdowns of the SLC system, including associated motor control centers and control room instrumentation, to check for moisture accumulation or evidence of motor winding insulation degradation and to independently assess Entergy’s configuration control and the material condition of the SLC system. The team also reviewed corrective action CRs and the SLC system health report to determine if there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.6 Emergency Service Water/Residual Heat Removal Service Water Pump Room Ventilation with Exhaust Fan out of service (i.e. Extended Natural Ventilation of Room)

a. Inspection Scope

The team reviewed EC-JAF-51412 which was developed to allow Entergy to credit natural circulation buoyant ventilation in the emergency service water (ESW)/Residual Heat Removal (RHR) Pump Rooms when the exhaust fan was out of service. The ESW pump provides cooling water for EDGs during a loss of offsite power. The Residual Heat Removal Service Water (RHRSW) pumps provide a source of cooling water for the RHR heat exchangers during refueling operations and design basis accident conditions. Each emergency pump room is normally cooled by a safety-related thermostatically controlled fan (73FN-3A/B). When the exhaust fan is out of service, the EC provides an analyses that shows the process of natural circulation ventilation will provide adequate cooling of the safety-related components in the rooms for up to 100 days provided that the outside air temperature remains at or below 93 °F.

The team reviewed the modification to verify the design bases, licensing basis, and performance capability of the ESW and RHRSW pumps would be adequately supported by natural circulation ventilation during periods when the room exhaust fan was out of service. The team verified that the modelling and calculations for heat removal capacity were adequate and that, although room temperatures would exceed the specified temperature of 115°F in the UFSAR, they would not exceed the support limits of the

safety-related equipment as long as the outside air temperature was maintained at or below 93°F, and the exposure time at the elevated temperatures was less than 100 days. The team interviewed design engineers and reviewed calculations and associated evaluations to verify that Entergy estimated the effects of naturally buoyant ventilation on the safety-related equipment. The team also reviewed corrective action CRs and control room logs to verify that equipment operation under conditions of natural circulation ventilation was properly monitored and logged. The team walked down the "A" and "B" emergency pump rooms to verify that the design inputs were correct and the configuration of the equipment was as specified in the EC. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.7 Emergency Diesel Generator Fuel Oil Transfer Pump Hydraulic Performance Evaluation

a. Inspection Scope

The team reviewed engineering change EC-JAF-47354 that supported a revision to Entergy calculation 93-6 for the evaluation of EDG fuel oil transfer pump (FOTP) hydraulic performance. Entergy performed the hydraulic evaluation to include the effects of increased pump flow and account for suction strainer pressure drop in response to corrective action CR-JAF-2013- 03196 and CR-JAF-2013-03342. The associated document changes provided limitations on underground fuel oil storage tank (FOST) level to ensure that an adequate supply of fuel oil was available to supply the EDGs for the six and seven day durations required by TS. The changes also accounted for an increase in the minimum tank level required to assure that air-entraining vortices were not formed at the pump suction when the FOTPs were operated at the speed associated with the power supply's maximum frequency.

The team reviewed the modification to verify that the design bases, licensing bases, and performance capability of the EDGs and their support systems had not been degraded by the modification. The team interviewed engineering staff and reviewed technical evaluations associated with the modification to determine if the FOTPs would function in accordance with the design assumptions. The team performed several walkdowns of accessible portions of the EDG fuel oil systems (including a walkdown during an "A" and "C" (A/C) EDG monthly surveillance on June 15) to independently verify calculation assumptions where possible and assess Entergy's configuration control and the material condition of the associated structures, systems, and components (SSCs). The team reviewed EDG monthly surveillances, equipment operator logs for fuel oil inventory, and operating procedures to verify that Entergy adequately maintained and controlled the EDG fuel oil systems as designed. The team also reviewed corrective action CRs and the EDG system health report to determine if there were reliability or performance issues associated with the EDG fuel oil systems. The 10 CFR 50.59 screening determination

associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.8 Replacement of Target Rock 2-Stage Pilot SRVs with Enhanced Target Rock 3-Stage Pilot SRVs

a. Inspection Scope

The team reviewed EC-JAF-48971 to replace the Target Rock 2-stage pilot SRVs with Target Rock 3-stage pilot SRVs because of past operating experience with pilot valve leakage in the 2-stage SRVs. FitzPatrick has 11 SRVs installed in the RCS to provide pressure relief and control. Past operating experience has demonstrated that steam leakage through the second stage of the 2-stage pilot valves and corrosion bonding has caused valve degradation that has led to operability challenges and potential plant shutdowns. FitzPatrick elected to replace the existing Target Rock 2-stage pilot SRVs with 3-stage pilot SRVs to prevent future operability challenges. Currently, three of the 11 SRVs were replaced with 3-stage pilot SRVs with the remaining eight SRVs scheduled for replacement during the next outage. The replacement plan was placed on hold while 3-stage pilot SRV operating experience from another station was being assessed.

The team reviewed the modification to verify that there was reasonable assurance that the replacement SRVs could perform their safety functions and that recent industry operating experience had been considered. The team verified the design inputs, assumptions and conclusions of the EC. The team interviewed engineers, licensed operators, and licensing personnel. The team walked down the SRV controls and monitoring panels in the control room and reviewed CRs and control room logs. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.9 Reserve Service Station Transformer Replacement

a. Inspection Scope

The team reviewed EC-JAF-12703 to replace the two reserve service station transformers (RSSTs) with new transformers of similar impedance. The RSSTs provide 115 kV power from the grid (115kV lines 3 and 4) via the 115 kV switchyard to the safety and non-safety buses during startup and shutdown operations when the normal service

station transformer (NSST) is not available. The RSSTs also provides offsite power to the safeguards equipment for safe shutdown of the plant in the event of an abnormal or accident condition. The new transformers have a higher capacity rating and have an automatic (instead of manual) on-load tap changer (OLTC) capability. FitzPatrick had planned to replace the RSSTs in 2010; however switchyard issues caused Entergy to delay the installation until the fall of 2012.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the alternating current (AC) distribution system had been maintained. The team conducted several walkdowns and visual inspections of the RSSTs and control room interfaces to assess the installed configuration, material condition, and possible adverse impact on safety-related SSCs in the area of the modification. The team reviewed the operating procedures for the new RSSTs including the operation of the automatic OLTC. The team also reviewed corrective actions CRs related to RSST operation to determine if reliability or performance issues have occurred since installation. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.10 Fuel Oil Transfer Motor 93P1-B2 (M) Thermal Overload Replacement

a. Inspection Scope

The team reviewed EC-JAF-26645, Revision 1, which replaced the fuel oil transfer pump motor breaker thermal overload protective devices. The fuel oil transfer pump motor is the prime mover for fuel oil transfer pump, 93P1-B2, and is located in the “B” EDG room. The fuel oil transfer pump will automatically transfer diesel fuel from the fuel oil storage tank to the day tank upon signals from a tank level probe when day tank level reaches 70 percent. Additionally, motor protection, cable protection, and the method of controlling the motor are provided by molded case circuit breaker and overload/relay heaters in the motor control center (MCC) 71MCC-264, cubicle OB2. The fuel oil transfer pump motor was replaced due to obsolescence in 2012. At the same time, the licensee identified that the new motor kilovolt ampere (KVA) rating was not compatible with the thermal overload protective devices. Specifically, the new motor full load current is 0.60 amps with a service factor of 1.15. The original motor full load current was 0.77 amps with a service factor of 1.0. Since, the existing thermal overloads did not provide adequate protection with the lower running currents, replacement thermal overloads were procured and installed under this modification in 2012.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of the fuel oil transfer pump had been maintained. The team conducted several walkdowns and visual inspections of the fuel oil transfer pumps and MCC 71MCC-264 cubicle OB2. The team observed the operation of the “B” EDG from

the control room and EDG cubicle to assess the installed configuration, material condition, and possible adverse impact on safety-related SSCs in the area of the modification. The team reviewed the operating procedures for the fuel oil transfer pumps. The team also reviewed corrective actions CRs related to the fuel oil transfer pumps to determine if reliability or performance issues have occurred since installation. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.11 Evaluate Core Spray Differential Pressure Switches 14DPIS-43A/B Drift Analysis and Setpoint Analysis

a. Inspection Scope

The team reviewed EC-JAF-51400 which calculated a new drift variable for core spray (CS) differential pressure indication switches. The CS differential pressure switches sense the differential pressure between the CS injection line (low pressure) and the above core plate tap (high pressure). The pressure switches provide a control room high differential alarm to indicate a core spray line injection failure in the event of a break in the CS injection line between the vessel nozzle and the shroud. In addition, the team reviewed CR-JAF-2014-00058 which identified issues with the calibration of the differential pressure switch. The pressure switches were found to be outside of administrative limits but within the allowable technical requirements manual (TRM) values and acceptance criteria. Entergy completed a calculation change that analyzed more current data to support the change in drift and setpoint analysis. The change performed affected that way the data was analyzed and did not affect actual TRM required values.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of CS differential pressure indication switches had not been degraded by the modification. The team interviewed plant engineers and reviewed calculation JAF-CALC-CSP-00271, “14DPIS-43A & B Core Spray Diff. Pressure Indicating Switch,” Revision 1, to determine if the changes met design and licensing requirements. Additionally, the team reviewed evaluations to determine if Entergy had properly implemented the modification. The team also reviewed corrective actions CRs related to the CS system to evaluate whether any new performance issues had resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.12 Perform Calculation for Pickup Voltage Available at 70AHU-19A Contactor Coil during the Degraded Bus Voltage Condition

a. Inspection Scope

The team reviewed EC-JAF-42664 which calculated the voltage available across the MCC contactor coil for motor 70AHU-19A during the worst case degraded MCC bus voltage. The MCC contactor design function is to start loads when called upon to mitigate an accident or perform safe shutdown. The 70AHU-19A is one of two air handling units that provide control room cooling. CR-JAF-2013-02200 identified issues during performance of preventive maintenance of the MCC contactor. After maintenance, contactor pickup voltage was 93 volts alternating current (VAC). However, procedure MP-056.01 requires that the contactor pickup voltage be less than or equal to 90 VAC. Entergy replaced the contactor coil and performed an analysis to determine minimum voltage available at the MCC contactor coil for worst case degraded bus voltage. Subsequently, calculation JAF-CAL-05-00117, "Perform 600 volt MCC Control Circuit Voltage Drop Calculation to Verify the Minimum Pickup Voltage for Selected Contactor Circuits," Revision 3 was updated and concluded that the minimum voltage available at the contactor coil was higher at 98 VAC.

The team reviewed the modification to verify that the design basis, licensing basis, and performance capability of air handling unit MCC coils had not been degraded and that minimum voltages were available and that contactor would be capable of performing its function. The team interviewed plant engineers and reviewed calculation JAF-CAL-05-00117 to determine if the changes met design and licensing requirements. Additionally, the team reviewed evaluations to determine if Entergy had properly implemented the change. The team also reviewed CRs to evaluate whether any new performance issues had resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of CRs associated with 10 CFR 50.59 and plant modification issues to evaluate whether Entergy was appropriately identifying, characterizing, and correcting problems associated with these areas, and whether the planned or completed corrective actions were appropriate. In addition, the team reviewed CRs written on issues identified during the inspection to verify Entergy adequately described the problem and incorporated the issue into their corrective action system.

The reviewed CRs are listed in the Attachment.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. T. Peter, Acting General Manager Plant Operations, and other members of Entergy's staff at an exit meeting on June 26, 2015. The team returned the proprietary information reviewed during the inspection.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>
T. Peter, Acting General Manager Plant Operations
C. Molinsky, Shift Manager
D. Arcelus, Senior Reactor Operator
D. Burch, Supervisor, Mechanical Design
G. Foster, Supervisor, Configuration Management
K. Habeyeb, Senior Engineer
K. Myers, MOV Program Engineer
M. Cook, System Engineer (EDG)
M. Hawes, Licensing
R. Giguere, System Engineer (125 Vdc)

ITEMS OPENED, CLOSED AND DISCUSSED

<u>Opened and Closed</u>

05000333/2015007-01	NCV	Failure to Adequately Assess the Impact of SRV Leakage on Operability (Section R17.2.1)

LIST OF DOCUMENTS REVIEWED

<u>10 CFR 50.59 Evaluations</u>
JAF-SE-96-042, Engineering Change (EC) 48365 / Change 32DPS-105 from a Trip/Alarm Function to an Alarm Only Function, Revision 6

<u>Modification Packages</u>
EC-JAF-11595, High Pressure Coolant Injection Turbine Steam Supply Drain Trap Bypass Valve Replacement, Revision 0
EC-JAF-12703, RSST Replacement, Revision 000
EC-JAF-26645, Fuel Oil Transfer Motor 93P1-B2 (M) Thermal Overload Replacement, Revision 0
EC-JAF-27880, Emergency Diesel Generator Air Start Compressor Motor Replacement, Revision 0
EC-JAF-39906, High Pressure Coolant Injection Steam Supply Valve 23MOV-15 Torque Switch Bypass, Revision 0
EC-JAF-40796, Standby Liquid Control Pump Motor Space Heater Modification, Revision 0
EC-JAF-42664, Perform Calculation for Pickup Voltage Available at 70AHU-19A Contactor Coil during the Degraded Bus Voltage Condition, Revision 0
EC-JAF-47354, Emergency Diesel Generator Fuel Oil Transfer Pump Hydraulic Performance Evaluation, Revision 0

EC-JAF-48971, Replacement of Target Rock 2-Stage Pilot SRVs with Enhanced Target Rock 3-Stage Pilot SRVs, Revision 0
EC-JAF-51400, Evaluate Core Spray Differential Pressure Switches 14DPIS-43A/B Drift Analysis and Setpoint Analysis, Revision 000
EC-JAF-51412, ESW/RHRSW Pump Room Ventilation with Exhaust Fan out of service (i.e. Extended Natural Ventilation of Room), Revision 0000
EC-JAF-56258, Operability Input for CR-JAF-2015-01271 SRV G Tailpipe Temperature Increase

<u>10 CFR 50.59 Screened-out Evaluations</u>
ARP 09-8-2-9 R5, EDG A ENG Trouble or Shutdown Process Applicability Determination, dated 7/14/13
EC-JAF-19597, ESW/RHRSW Pump Room Natural Ventilation, Revision 000
EC-JAF-35176, REPLACE 13RCIC-31 Swing Check Valve with an In-Line Check Valve Due to Interferences, Revision 000
EC-JAF-43148, Permanent Modification to Monitor Reactor Vessel Level Indication at the 25RSP Remote Shutdown Panel, Revision 000
EC- JAF-46312, Perform Calculation for Pickup Voltage Available at 70AHU-19A Contactor Coil during the Degraded Bus Voltage Condition, Revision 000
EC-JAF-50301, Resolve the Anomaly Identified On Cr-Jaf-2014-01428 Concerning Ambient Correction Factors for Ambient Compensated Overload Relays, Revision 000
Equipment Support Procedure ESP-22.001, LOCA Bypass of EDG A & C Shutdown Logic Functional Test, 2/6/2013
Equipment Support Procedure, ESP-22.003, EDG A&C Fuel Oil Transfer Pump Operational Check, 9/15/2014
ESP-22.003 R1, EDG A & C Fuel Oil Transfer Pump Operational Check Process Applicability Determination, dated 1/14/11
MST-071.13 R22, 125 VDC Station Battery Quarterly Surveillance Test Process Applicability Determination, dated 3/2/15
OP-15 R61, High Pressure Coolant Injection Process Applicability Determination, dated 5/14/14
OP-22 R60, Diesel Generator Emergency Power Process Applicability Determination, dated 3/23/15
PE Evaluation 107589, Magnetic sensor for HPCI Governor Speed Loop, Terry Turbine, 06/11/2012
PE Evaluation 127941, Target Rock SRV Flexitallic Gasket J03351862, 01/13/2014
PE Evaluation 130027, Target Rock SRV Flexitallic Gasket 309SS or 347SS with Nuclear Grade J0351862, 02/28/2014
PE Evaluation 133193, Fuse, Bussmann LP-CC-15 S21472, One time dedication. (Ref PE Eval # 131029), 04/23/2014
PE Evaluation 136433, Socket, 11 pin octal, Curtis Industries RS11J0770216 Dedication for EDG LCO, 07/24/2014
RAP-7.4.01 R27, Control Rod Scram Time Evaluation Process Applicability Determination, dated 8/20/13
ST-40D R109, Daily Surveillance and Channel Check Process Applicability Determination, dated 11/9/14

Calculations & Analysis

9663-JAF-CALC-0001, Motor Operated Valve Thermal Overload Sizing, Revision 2
JAF-CALC-05-00015, ESW/RHRSW Pump Room Natural Ventilation, Revision 1
JAF-CALC-07-00019, Volume in EDG Underground Fuel Oil Storage Tanks as a Function of Level, Revision 0
JAF-CALC-07-00020, Revised Emergency Diesel Generator (EDG) Fuel Oil Storage Quantities for 7 Day and 6 Day Supplies, Revision 0
JAF-CALC-09-00016, JAF Auxiliary Power Systems Analysis, Revision 2
JAF-CALC-09-00016 EC 27880 Markup, James A. FitzPatrick Auxiliary Power System Analysis, dated 2/28/11
JAF-CALC-12-00022, Determination of Required Stem Thrust for 23MOV-15 using EPRI Performance Prediction Methodology (PPM), Revision 2
JAF-CALC-12-00026, Seismic/Weak Link Analysis of 10"Anchor Darling Gate Valve: 23MOV-15, Revision 0
JAF-CALC-14-00003, ESW/RHRSW Pump Room Cable Life under Natural Ventilation, Revision 0
JAF-CALC-CSP-00271, 14DPIS-43A & B Core Spray Diff. Pressure Indicating Switch, Revision 1
JAF-CALC-CSP-04231, Equivalent Analytical Limit for 14DPIS-43A AND 14DPIS-43B, Revision 3
JAF-CALC-EDG-03358, JAF Single EDG Loading, Revision 0
JAF-CALC-ELEC-02609, 125 VDC Station Battery "A" Sizing & Voltage Drop, Revision 3
JAF-CALC-HPCI-02062, Thrust and Torque Calculation for 23MOV-15, Revision 5
NYPA-23MOV-I5, HPCI Steam Supply Inboard Isolation Valve (23MOV-15) Differential Pressure (DP) Calculations, dated 4/17/95

Design & Licensing Bases

DBD-023, High Pressure Coolant Injection System Design Basis Document, Revision 12
DBD-093, Emergency Diesel Generator (EDG) Design Basis Document, Revision 12
EDP-20 Enclosure 2, Procedure for Establishing if Plant Electrical Equipment is within the Scope of 10CFR50.59 (EQ), Revision 0
JAF-RPT-MISC-04046, Environmental Qualification Service Conditions, Revision 0
TRM Appendix G, Core Operating Limits Report, Revision 28

Drawings

1.12-2, Air Compressor Assembly Emergency Diesel Generator, 93AC-1A, B, C, D & 93AC-2A, B, C, D, Revision 3
1.43-96, MCC 254 Arrangement and Equipment Summary, Revision 10
1.61-140, Elementary Diagram HPCI System, Revision 19
3.52-52, Rack Construction Spent Fuel Storage Racks, Revision A
7.65-424, Edwards Forged Steel Univalve Globe Stop Valve Socket Weld Ends Size 1, Revision 1
16.23-38, High Pressure Coolant Injection System, Revision 9
11825-FV-9A, Spent Fuel Pool Liner Details SH 1, Revision 6
11825-FV-9B, Spent Fuel Pool Liner Details SH 2, Revision 6
11825-FV-9F, Spent Fuel Pool Liner Details SH 6, Revision 4
11825-FV-9H, Spent Fuel Pool Liner Details SH 1, Revision 3

ESK-6MAV, Elementary Diagram 600V CKTS – MOV HPCI Steam Supply Valve 23MOV-15, Revision 15
ESK-6RE, Elementary Diagram 600V CKTS – Nuclear Standby LIQ Control SYS Pumps & Tank HTR, Revision 18
FM-1K, Mach. Loc. Reactor Bldg Sections 4-4 & 5-5, Revision 13
FM-19A, Flow Diagram Fuel Pool Cooling & Clean-up System 19, Revision 43
FM-25A, Flow Diagram High Pressure Coolant Injection System 23, Revision 74
FM-93A, Flow Diagram Fuel Oil Lines Emergency Diesel Generators System 93, Revision 22
FM-94A, Flow Diagram Air Start-up Lines Emergency Diesel Generators System 93, Revision 13
FP-65A, Fuel Oil & Compressed Air Lines Emergency Diesel Generators Sheet 1, Revision 7
FV-6A, Arrangement Reactor Storage Area Liners, Revision 9
JAF-DWG-6.32-101, Pilot Assembly 3-Stage 6x10 Valve, Revision 001
JAF-DWG-6.32-104, Pilot Assembly 304095-1 3-Stage 6x10 Valve, Revision 001
JAF-DWG-6.32-90, Safety Relief Valve Assembly Valve Model 0867F 6x10, Revision 001
JAF-DWG-6.32-96, Safety Relief Valve Assembly Valve Model 0867F 6x10, Revision 001
JAF-DWG-1.43-98, 93P1-B2 (M) EDG B Fuel Oil Transfer Pump B2 Motor, Revision 10
MSK-1282, HPCI Steam Supply Trap/Drain, Revision 9
SE-9JG, 600V Wiring Diagram 11P-2A 11P-2B SYS II, Revision 10
SE-9LL, 600V Wiring Diagram 23MOV-15 System 23, Revision 9
SE-9VV, Distribution Panel RBACD5 Normal Control & Instrument Bus D5, Revision 25

Evaluations
DER-00-2386, Recurring Failure of 23AOV-53 Evaluation, dated 12/28/00
EC 26913, Equivalent Change to Replace EDG Compressor Motor 93AC-D1(M), Revision 0
EC 36155, CALC Markup for EC 11595 to Incorporate the New Actuator Weight and CG (23AOV-53), Revision 0
EC 47354, Calculation 93-6 Revision 1, Computation of Pipe Discharge Loss for Diesel Generator Fuel Oil Transfer Pump, Revision 0
G-STERI No. E-95014, Generic Seismic Technical Evaluation for Replacement Items for AC/DC Motors, Revision 1
JAF-RPT-03-0278, Categorization of James A. FitzPatrick NPP Air Operated Valves, Revision 1
JAF-RPT-12-00013, Assessment of Pressure Retaining Integrity of 23-MOV-15 under a Postulated Motor Stall Scenario, Revision 0
JMD-00-135, Equipment Failure Evaluation for 23AOV-53 from DER 00-02386, ACTS 00-52475, dated 11/15/00
PEEVAL 108004, Globe Valve Procurement Engineering Evaluation (Catalog ID 0032073879), dated 6/20/12
RAL-3822, Seismic & Limiting Component Report (23AOV-53), Revision 0
TCE 15-016, Transient Combustible Evaluation, dated 5/7/15

Miscellaneous
EDG Air Receiver Pressure Logs, dated 5/10/15 – 5/16/15
EDG Fuel Oil Storage Tank Level Indication Logs, dated 5/10/15 – 5/16/15
EN-DC-196 Attachment 9.2, AOV Data Record (23AOV53), dated 5/31/12
EN-MA-118 Attachment 9.6, Foreign Material Exclusion Component Close-Out (WO 326755-08), performed 10/1/12
EN-MA-134, Offline Motor Electrical Testing (11P-2A), performed 3/26/12
EN-MA-134, Offline Motor Electrical Testing (11P-2B), performed 3/15/11

Instrument Calibration Report (23AOV53), dated 5/31/12
IS-E-01 Attachment 4, Installation of Moisture Seals in Cable Entrances to Electrical Equipment, performed 6/8/12
IS-S-04 Attachment 2, Pipe Support Installation Report, dated 6/7/12
MOV Risk vs. Margin Matrix, dated 5/28/15
MP-056.01, AC Motor Control Center Maintenance and Subcomponent Replacement, performed 4/7/11
MP-059.83, Motor Power Monitoring (MPM) Testing and Analysis (11P-2A), performed 5/29/08
MP-059.83, Motor Power Monitoring (MPM) Testing and Analysis (11P-2B), performed 3/17/08
MP-059.87, Viper MOV Diagnostic Testing (23MOV-15), performed 9/3/14
MP-093.11, EDG System Mechanical PM, performed 9/12/13 & 10/23/13
MP-101.20, Periodic Maintenance of Various Air Compressors, performed 4/7/11
RAP-7.4.01, Control Rod Scram Time Evaluation, performed 9/19/14 & 10/8/14
Roark's Formulas for Stress and Strain, 7th Edition
SLC Pump Area Equipment Operator Logs, dated 5/10/15 – 5/16/15

Operating Experience
NRC Information Notice 2007-34: Operating Experience Regarding Electrical Circuit Breakers, dated 10/22/07
NRC Information Notice 2010-09: Importance of Understanding Circuit Breaker Control Panel Indications, dated 4/14/10

Procedures
AOP-31, Loss of Condenser Vacuum, Revision 21
ARP 09-75-1-7, Emergency Pump Room Vent Sys A Trouble, HV-11A Panels, ARP HV-11A-1, ARP HV-11A-02, ARP HV-11A-3, ARP HV-11A-6, ARP HV-11A-07, Revision 2
ARP 09-75-2-7, Emergency Pump Room Vent Sys B Trouble, HV-11B Panel, ARP HV-11B-01, ARP HV-11B-2, ARP HV-11B-03, ARP HV-11B-04, ARP HV-11B-6, ARP HV-11B-7, ARP HV-11B-8, Revision 1
ARP 09-8-2-9, EDG A ENG Trouble or Shutdown, Revision 5
ARP-75-2-15, Screenwell Vent Sys Trouble, HV-11N Panel, ARP HV-11N-01, ARP HV-11N-02, ARP HV-11N-03, ARP-HV-11N-4, ARP HV-11N-05, ARP HV-11N-06, ARP HV-11N-7, ARP HV-11N-8, ARP-HV-11N-9, ARP HV-11N-10, Revision 2
CEP-NDE-0505, Ultrasonic Thickness Examination, Revision 4
EN-DC-115, Engineering Change Process, Rev 017
EN-DC-140, Air Operated Valve Program, Revision 5
EN-DC-210, Environmental Qualification Master List Control, Revision 2
EN-DC-324, Preventive Maintenance Program, Revision 14
EN-DC-331, MOV Program, Revision 4
EN-LI-101, 10 CFR 50.59 Evaluations, Revision 12
EN-LI-100, Process Applicability Determination, Revision 16
EN-MA-134, Offline Motor Electrical Testing, Revision 5
EN-OP-104, Operability Determination Process, Revision 9
ESP-22.003, EDG A & C Fuel Oil Transfer Pump Operational Check, Revision 1
IS-E-07, Installation of Electrical Cable Terminations, Revision 15
IS-S-01, Tubing and Support Installation, Revision 9
MP-101.04, Lubrication of Electric Motor (w/o Disassembly) with Grease Lubricated Bearings, Revision 51

MST-016.01, Primary Containment Air Lock Interlock Test, Revision 1
MST-071.10, LPCI Battery Weekly Surveillance Test, Revision 44
MST-071.11, LPCI Battery Quarterly Surveillance Test, Revision 21
MST-071.13, 125 VDC Station Battery Quarterly Surveillance Test, Revision 22
MST-071.20, 125 VDC Station Battery Service Test, Revision 36
MST-071.24, Station Battery B Modified Performance Test, Revision 18
MST-076.10, Diesel Fire Pump 76P-1 and 76P-4 Battery Quarterly Surveillance Test, Revision 12
MSK-118A1, High Pressure Coolant Injection Piping System 23, Revision 8
MSK-118B1, High Pressure Coolant Injection Piping System 23, Revision 9
MSK-118D1, Piping Diagram System 23, Revision 11
OP-1, Main Steam System, Revision 57
OP-15, High Pressure Coolant Injection, Revision 61
OP-22, Diesel Generator Emergency Power, Revision 60
OP-46A, 4160 V and 600 V Normal AC Power Distribution, Rev. 62
OP-46B, 120 VAC Power System, Revision 33
OP-57, Attachment 2, Screen House Ventilation, Revision 11 (pre and post DRN No. 14-00177)
RAP-7.4.01, Control Rod Scram Time Evaluation, Revision 27
ST-9AA, EDG System A Fuel/Lube Oil Monthly Test, Revision 4
ST-9AB, EDG System B Fuel/Lube Oil Monthly Test, Revision 3
ST-40D, Daily Surveillance and Channel Check, Revision 109

Surveillance and Modifications Acceptance Tests
EN-DC-196 Attachment 9.3, AOV Test Data Evaluation (23AOV53), performed 6/1/12
MST-071.13, 125 VDC Station Battery Quarterly Surveillance Test (71SB-1), performed 3/30/15
MST-071.13, 125 VDC Station Battery Quarterly Surveillance Test (71SB-2), performed 4/10/15
ST-1WA, HPCI and RCIC Primary Containment Inboard Isolation Valve Test (IST), performed 9/8/14
ST-4N, HPCI Quick-Start, Inservice, and Transient Monitoring Test (IST), performed 2/11/15
ST-6HA, Standby Liquid Control A Side Quarterly Operability Test (IST), performed 4/24/15
ST-6HB, Standby Liquid Control B Side Quarterly Operability Test (IST), performed 5/5/15
ST-9AA, EDG System A Fuel/Lube Oil Monthly Test, performed 5/21/15 & 5/22/15
ST-9AB, EDG System B Fuel/Lube Oil Monthly Test, performed 5/4/15
ST-40D R109, Daily Surveillance and Channel Check Process Applicability Determination, performed 5/17/15 – 5/23/15
ST-41K, Remote Valve Position Indication Verification Shutdown (IST), performed 10/13/12 & 10/8/14
WO 51099912-15, Post Modification Functional Testing of Newly Installed 23AOV-53 (HPCI Turbine Steam Supply Drain Trap T-3 Bypass Valve, performed 6/9/12

System Health Reports, Walkdown Reports, & Trending
EDG System Health Report, Q1-2015
HPCI System Health Report, Q4-2014
Standby Liquid Control System Health Report, Q1-2015
Standby Liquid Control System Walkdown Report, dated 4/15/15 & 5/20/15

Vendor Manuals and Specifications

B060-0003, Integral Horsepower AC Induction Motors - Installation & Operation Manual, dated February 2009

G080-0196, Helical Gear-motor Line Instructions, Revision 0

JAF-SPEC-MISC-00334 Class IC-N8, Piping Specification - Stainless Steel Tubing with Tube Fittings, Revision 14

JAF-VMAN T020-001, Target Rock Technical Manual Safety Relief Valve Model 0867F, Revision 1

JAF-VMAN T020-0041, Target Rock Technical Manual Safety Relief Valve Model 7567F, Revision 10

JAF-VMAN T246-0282, Instructions: Installation and Maintenance for 50 and 60 Hertz Motors

L205-C001, Lincoln Motors – AC, Revision 0

R374-C005, Roper Pumps - Series AM, AP and AL Installation & Operating Instructions, Revision 2

Corrective Action Condition Reports (CR-JAF-)

2004-01229	2010-08488	2013-03342	2015-02594
2004-03302	2012-00657	2013-05266	2015-02611*
2005-01110	2012-03244	2015-00179	2015-02662
2007-01609	2012-03291	2015-01271	2015-02728
2007-02108	2012-03331	2015-01980	2015-02731*
2007-02607	2012-03332	2015-02379	2015-02793*
2007-02937	2012-04100	2015-02517*	2015-02805*
2008-00709	2012-07728	2015-02547*	2015-02806*
2009-00118	2013-00132	2015-02548*	
2010-00467	2013-03066	2015-02564	
2010-03468	2013-03196	2015-02567	

(*denotes NRC identified during this inspection)

Work Orders

00122193	00341269	52038568	52262498
00122194	51099912	52038568	52373145
00326755	51103323	52038568	52400085
00339333	52038566	52038568	
00341265	52038568	52185634	
00341268	52038568	52247226	

Miscellaneous

JAF-RPT-03-00056, Operational Leakage Action Levels for Target Rock Two-Stage Safety/Relief Valves, Revision 0 (Proprietary)

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LIST OF ACRONYMS

AC	Alternating Current
A/C	"A" and "C"
AOV	Air-Operated Valve
CFR	Code of Federal Regulations
CR	Condition Report
CS	Core Spray
DP	Differential Pressure
DRS	Division of Reactor Safety
EC	Engineering Change
EDG	Emergency Diesel Generator
Entergy	Entergy Nuclear
EQ	Environmental Qualification
ESW	emergency service water
FOTP	Fuel Oil Transfer Pump
FOST	Fuel Oil Storage Tank
GE	General Electric
HPCI	High Pressure Coolant Injection
IP	Inspection Procedure
IST	In-Service Testing
JAF	James A FitzPatrick
KVA	kilovolt ampere
lbm/hr	pounds-mass per hour
LCO	Limiting Condition of Operation
MCC	Motor Control Center
MOV	Motor-Operated Valve
MPM	Motor Power Monitoring
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NPIRs	Nuclear Plant Interface Requirements
NRC	Nuclear Regulatory Commission
NSST	Normal Service Station Transformer
OLTC	On-Load Tap Changer
PPM	Performance Prediction Methodology
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RSST	Reserve Service Station Transformer
SLC	Standby Liquid Control
SRVs	Safety Relief Valves
SSC	Structure, System, and Component
TRETR	Target Rock Engineering Test Report
TRM	Technical Requirements Manual
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VAC	Volts Alternating Current