UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406-1415

August 5, 2011

EA-11-170

Mr. Kevin Bronson
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 - NRC INTEGRATED INSPECTION REPORT 05000333/2011003 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Bronson:

On June 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant (FitzPatrick). The enclosed inspection report documents the inspection results which were discussed on July 14, 2011, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, this report documents one NRC-identified finding of very low safety significance. This finding was determined to be a violation of NRC requirements that was evaluated under traditional enforcement and categorized as Severity Level IV. However, because of the very low safety significance and because the issue was entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section 2.3.2 of the NRC's Enforcement Policy. If you contest this NCV, you should provide a response within 30 days of the date of the 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 a copy to the Regional Administrator, Region I; Office of Enforcement; U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Senior Resident Inspector at FitzPatrick.

In addition, the inspectors reviewed Licensee Event Report 05000333/2010-004, which described the circumstances associated with local leak rate testing (LLRT) results of the 'C' main steam isolation valves (MSIVs), both inboard and outboard isolation valves, which exceeded the Technical Specification allowable leakage rate. Although this issue constitutes a violation of NRC requirements, in that the combined leakage of the 'C' MSIVs at power constitutes a violation, the NRC concluded that this issue was not within Entergy's ability to foresee and correct, that Entergy staff's actions did not contribute to the degraded condition, and that actions taken were reasonable to identify and address this matter. As a result, the NRC did not identify a performance deficiency. A risk evaluation was performed and the issue was determined to be of very low safety significance. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, to exercise enforcement discretion in accordance with Section 3.5 of the Enforcement Policy, "Violations Involving Special Circumstances."

In accordance with 10 CFR Part 2.390 of the NRC's "Rules of Practice," a copy of this letter and 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 document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

for

Darrell J. Roberts, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000333/2011003
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

foresee and correct, that Entergy staff's actions did not contribute to the degraded condition, and that actions taken were reasonable to identify and address this matter. As a result, the NRC did not identify a performance deficiency. A risk evaluation was performed and the issue was determined to be of very low safety significance. Based on these facts, I have been authorized, after consultation with the Director, Office of Enforcement, to exercise enforcement discretion in accordance with Section 3.5 of the Enforcement Policy, "Violations Involving Special Circumstances."

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Sincerely,

/RA by James W. Clifford Acting For/

Darrell J. Roberts, Director
Division of Reactor Projects

Docket No.: 50-333
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w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	05000333/2011003
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	Scriba, New York
Dates:	April 1, 2011 through June 30, 2011
Inspectors:	E. Knutson, Senior Resident Inspector S. Rutenkroger, PhD, Resident Inspector T. Fish, Senior Operations Engineer J. Noggle, Senior Health Physicist G. Meyer, Senior Reactor Inspector
Approved by:	Darrell J. Roberts, Director Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000333/2011003; 04/01/2011 - 06/30/2011; James A. FitzPatrick Nuclear Power Plant; Adverse Weather Protection.

This report covered a three-month period of inspection by resident inspectors, announced inspections by region-based inspectors, and an in-office review by a region-based inspector. One Severity Level IV finding, which was a non-cited violation (NCV), was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). The cross-cutting aspects for the findings were determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be "Green" or be assigned a severity level after Nuclear Regulatory Commission (NRC) management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

Cornerstone: Mitigating Systems

- Severity Level IV: The inspectors identified a Severity Level IV (SL IV) NCV of Title 10, Code of Federal Regulations (10 CFR) Part 50.71(e) because FitzPatrick personnel did not update the Updated Final Safety Analysis Report (UFSAR) with information consistent with plant conditions. Specifically, FitzPatrick personnel did not remove reference to or correct information in UFSAR Section 8.6.6.c, "Emergency Bus Voltages When Operating From the Reserve Source," to reflect current plant conditions with regard to the listed maximum voltage capable of being produced at the emergency bus from the reserve source during a low load condition. This issue was considered within the traditional enforcement process because it had the potential to impede or impact the NRC's ability to perform its regulatory functions. FitzPatrick issued condition report (CR) CR-JAF-2011-03023 to address the UFSAR discrepancy.

  The inspectors concluded that the violation was more than minor because the longstanding and incorrect information in the UFSAR had a potential impact on safety and licensed activities. Excessive voltage on an emergency bus can result in equipment damage, or loss due to the actuation of protective devices such as overcurrent fuses. Similar to Enforcement Policy Section 6.1, example D.3, the inspectors determined the violation was of SL IV because the erroneous information not updated in the UFSAR was not used to make an unacceptable change to the facility nor did it impact a licensing or safety decision by the Nuclear Regulatory Commission (NRC). (Section 1R01)

REPORT DETAILS

Summary of Plant Status

The James A. FitzPatrick Nuclear Power Plant (FitzPatrick) operated at or near 100 percent reactor power throughout the inspection period with the following exceptions: On April 30 and May 6, short duration power reductions to 55 percent were performed to identify and plug main condenser tube leaks; on June 7 and June 9, short duration power reductions to 75 percent were performed to clean main condenser water boxes; and, on June 27 power was reduced to 65 percent for a control rod sequence exchange, and was restored to full power the following day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 3 samples)

.1 Evaluate Summer Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors reviewed operating procedures to verify continued availability of offsite and alternate alternating current (AC) power systems. The inspectors also reviewed FitzPatrick's agreements and protocols established with the transmission system operator to verify that the appropriate information is exchanged when issues arise that could impact the offsite power system. The documents reviewed are listed in the Attachment. These activities constituted one offsite and alternate AC power systems inspection sample.

b. Findings

Introduction: The inspectors identified a Severity Level IV (SL IV) non-cited violation (NCV) of Title 10, Code of Federal Regulations (10 CFR) Part 50.71(e) because FitzPatrick personnel did not update the Updated Final Safety Analysis Report (UFSAR) with information consistent with plant conditions. Specifically, FitzPatrick personnel did not remove reference to or correct information in UFSAR Section 8.6.6.c, "Emergency Bus Voltages When Operating From the Reserve Source," to reflect current plant conditions with regard to the listed maximum voltage capable of being produced at the emergency bus from the reserve source during a low load condition.

Description: During plant operation, the safety related and non-safety related buses are supplied by the normal station service transformer which is fed directly from the main generator terminals. While the plant is shut down, these buses are supplied by the reserve (offsite) power source. This reserve power source consists of two reserve station service transformers fed from the 115 kilovolt (kV) system. In 2001, FitzPatrick personnel implemented a 115 kV transformer tap change modification from 116 kV tap to 113 kV tap with the voltage values revised in the UFSAR through calculation JAF-

CALC-ELEC-04554, “Emergency Bus Voltage Profile for RSST’s [reserve station service transformer] T2 & T3 Tap Setting @ 113 KV,” Revision 0. The UFSAR was revised to reflect an increase in the maximum emergency bus voltage from 633 volts (V) to 634.5 V. An overly excessive maximum voltage on an emergency bus would be a concern due to the potential to cause equipment damage or the actuation of protective devices such as overcurrent fuses which would prevent further operation of the equipment.

However, FitzPatrick personnel initiated condition report (CR)-JAF-2006-03808, which noted two motor trips during the 2006 refueling outage and a voltage of 642 V on a 600 V bus with the 115 kV system voltage within specification at 120 kV. Corrective actions within CR-JAF-2006-03808 included an engineering request which determined a new acceptable limit of 645 V for the 600 V bus maximum voltages and an associated procedure change request to raise the allowed 600 V bus limit to 645 V during plant outages. FitzPatrick personnel revised OP-46A, “4160 V and 600 V Normal AC Power Distribution,” Revision 49, and ST-9W, “Electrical Lineup and Power Verification,” Revision 8, to change the allowed 600 V bus voltage range from 546 - 635 V to 546 - 645 V. FitzPatrick staff did not initiate action to update the UFSAR.

The inspectors identified that UFSAR Section 8.6.6.c conflicted with OP-46A and ST-9W because the UFSAR stated an acceptable maximum voltage of 634.5 V and the procedures allowed a maximum voltage of 645 V. The inspectors identified that the discrepancy in UFSAR Section 8.6.6.c should have been updated in a timeframe consistent with the standards and expectations delineated in EN-LI-113, “Licensing Basis Document Change Process.” Station procedures require FitzPatrick personnel to periodically update/correct information in the UFSAR within an operating cycle, not to exceed 24 months, to ensure the UFSAR accurately reflects the plant configuration and operation.

FitzPatrick issued CR-JAF-2011-03023 to address the UFSAR discrepancy. FitzPatrick staff identified an additional opportunity to identify and correct the UFSAR discrepancy when calculation JAF-CALC-09-00016, “Electrical Load Flow and Short Circuit Analysis Using Electrical Transient Analysis Program (ETAP),” Revision 0, was issued. The inspectors also noted that CR-JAF-2010-03587 documented an NRC identified issue which described discrepancies with respect to referencing current design basis information from UFSAR Section 8.6.6.c rather than the current calculation of record, JAF-CALC-09-00016. However, the associated corrective actions within CR-JAF-2010-03587 addressed only the specific discrepancy identified and did not recognize the UFSAR discrepancy.

Analysis: This issue was a performance deficiency because FitzPatrick personnel had reasonable opportunity to correct and update the UFSAR to be consistent with current plant conditions. This issue is considered within the traditional enforcement process because it has the potential to impede or impact the NRC's ability to perform its regulatory functions. The inspectors used the Enforcement Policy, Section 6.1, "Reactor Operations," to evaluate the significance of this violation. The inspectors concluded that the violation was more than minor because the longstanding and incorrect information in the UFSAR had a potential impact on safety and licensed activities. Excessive voltage on an emergency bus can result in equipment damage, or loss due to the actuation of protective devices such as overcurrent fuses. Similar to Enforcement Policy Section 6.1, example D.3, the inspectors determined the violation was of SL IV because the erroneous information not updated in the UFSAR was not used to make an unacceptable change to the facility nor did it impact a licensing or safety decision by the NRC.

Enforcement: 10 CFR Part 50.71(e) requires that licensees shall periodically update the UFSAR, originally submitted as part of the application for the operating license, to assure that the information included in the report contains the latest information developed. In part, the submittal shall include the effects of all changes made in the facility or procedures as described in the UFSAR such that the UFSAR as updated remains complete and accurate. Contrary to the above, since 2006 FitzPatrick became aware of contrary information and did not update the UFSAR to accurately reflect the status of the 115 kV and 600 V systems as described within UFSAR Section 8.6.6.c. Not adequately updating the UFSAR as required by 10 CFR Part 50.71(e) is characterized as a Severity Level IV violation. However, because the violation was of very low safety significance and was entered in the corrective action program (CR-JAF-2011-03023), this violation is being treated as an NCV consistent with NRC Enforcement Policy. **(NCV 05000333/2011003-01: UFSAR Emergency Bus Voltage Not Updated, Consistent with Current Plant Conditions)**

.2 Evaluate Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors reviewed and verified completion of the warm weather preparation checklist contained in procedure AP-12.04, "Seasonal Weather Preparations," Revision 18. The inspectors reviewed the operating status of the reactor building and control room ventilation systems, reviewed the procedural limits and actions associated with elevated lake and air temperatures, performed partial walkdowns of the control building chilled water and control building ventilation systems, and walked down accessible areas of the reactor building and control room to assess the effectiveness of the ventilation systems. Discussions with operations and engineering personnel were conducted by the inspectors to ensure that plant personnel were aware of temperature restrictions and required actions. The documents reviewed are listed in the Attachment. These activities constituted one seasonal extreme weather conditions inspection sample.

b. Findings

No findings were identified.

.3 Evaluate Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

On May 26, 2011, the National Weather Service issued a Tornado Watch for portions of New York State, including the area in which FitzPatrick is located. In accordance with plant procedures, the operators entered AOP-13, “High Winds, Hurricanes, and Tornadoes,” Revision 19. The inspectors verified that the actions required by AOP-13 were taken and walked down the site in order to identify any loose material or other concerns. These actions included reviewing work activity in progress related to adverse weather conditions and determining and minimizing plant risk, conducting site walk downs, and ensuring equipment on site is configured appropriately for high wind conditions.

These activities constituted one imminent weather condition inspection sample.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdown (71111.04Q - 3 samples)

a. Inspection Scope

The inspectors performed three partial system walkdowns to verify the operability of redundant or diverse trains and components during periods of system train unavailability or following periods of maintenance. The inspectors referenced system procedures, the UFSAR, and system drawings in order to verify the alignment of the available train was proper to support its required safety functions. The inspectors also reviewed applicable CRs and work orders (WOs) to ensure that FitzPatrick personnel identified and properly addressed equipment discrepancies that could impair the capability of the available equipment train, as required by 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." The documents reviewed are listed in the Attachment. The inspectors performed a partial walkdown of the following systems:

- 'A' emergency service water (ESW) system when the offsite power 115 kV Line 4 was out of service;
- 'B' and 'D' emergency diesel generators (EDGs) while 'C' EDG was inoperable during maintenance; and
- 'A' and 'C' EDGs while 'D' EDG was inoperable during maintenance.

These activities constituted three partial system walkdown inspection samples.

b. Findings

No findings were identified.

.2 Complete System Walkdown (71111.04S - 1 sample)

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the low pressure coolant injection motor operated valve independent power supply system to identify discrepancies between the existing equipment lineup and the required lineup. During the inspection, surveillance procedures and operating procedures were used to verify proper equipment alignment and operational status. The inspectors reviewed the open maintenance WOs associated with the system for deficiencies that could affect the ability of the system to perform its function. Documentation associated with unresolved design issues such as temporary modifications, operator workarounds, and items tracked by plant engineering were also reviewed by the inspectors to assess their collective impact on system operation. In addition, the inspectors reviewed the corrective action program (CAP) database to verify that equipment problems were being identified and appropriately resolved. The documents reviewed are listed in the Attachment.

These activities constituted one complete system walkdown inspection sample.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Review (71111.05Q - 5 samples)

a. Inspection Scope

The inspectors conducted inspections of fire areas to assess the material condition and operational status of fire protection features. The inspectors verified, consistent with applicable administrative procedures, that combustibles and ignition sources were adequately controlled; passive fire barriers, manual fire-fighting equipment, and suppression and detection equipment were appropriately maintained; and compensatory measures for out-of-service, degraded, or inoperable fire protection equipment were implemented in accordance with FitzPatrick's fire protection program. The inspectors evaluated the fire protection program for conformance with the requirements of license condition 2.C(3), "Fire Protection." The documents reviewed are listed in the Attachment.

- Reactor building (RB) 326 foot elevation, fire area/zone IX/RB-1A;
- Administration building 300 foot elevation, fire area/zone IA/AD-6;
- West cable tunnel, fire area/zone IC/CT-1;
- Relay room, fire area/zone VII/RR-1; and
- West diesel fire pump room, fire area/zone IB/FP-1.

These activities constituted five quarterly fire protection inspection samples.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

a. Inspection Scope

The inspectors conducted tours of the EDG rooms and associated switchgear rooms to assess internal flooding protection measures in those areas. The inspectors reviewed selected risk significant plant design features intended to protect the associated safety-related equipment from internal flooding events. The inspectors reviewed flood analysis and design documents, including the Individual Plant Examination and UFSAR.

These activities constituted one internal flood protection measures inspection sample.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Quarterly Review (71111.11Q - 1 sample)

a. Inspection Scope

On June 20, 2011, the inspectors observed licensed operator simulator training to assess operator performance during scenarios to verify that crew performance was adequate and evaluators were identifying and documenting crew performance problems. The inspectors evaluated the performance of risk significant operator actions, including the use of emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, the implementation of appropriate actions in response to alarms, the performance of timely control board operation and manipulation, and the oversight and direction provided by the shift manager. Licensed operator training was evaluated for conformance with the requirements of 10 CFR Part 55, "Operators' Licenses." The documents reviewed are listed in the Attachment.

These activities constituted one quarterly operator simulator training inspection sample.

b. Findings

No findings were identified.

.2 Biennial Review (71111.11B - 1 sample)

a. Inspection Scope

On June 1, 2011, one NRC region-based inspector conducted an in-office review of results of licensee-administered annual operating tests and comprehensive written exams for 2011. The inspection assessed whether pass rates were consistent with the guidance of NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process (SDP)". The inspector verified that:

- Crew pass rate was greater than 80%. (Pass rate was 100%);
- Individual pass rate on the written exam was greater than 80%. (Pass rate was 89.4%);
- Individual pass rate on the job performance measures of the operating exam was greater than 80%. (Pass rate was 97.9%);
- Individual pass rate on the dynamic simulator test was greater than 80%. (Pass rate was 100%); and
- Overall pass rate among individuals for all portions of the exam was greater than or equal to 75%. (Overall pass rate was 87.3%)

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 samples)

a. Inspection Scope

The inspectors reviewed performance-based problems involving selected in-scope structures, systems, or components (SSCs) to assess the effectiveness of the maintenance program. The documents reviewed are listed in the Attachment. The reviews focused on the following aspects when applicable:

- Proper maintenance rule scoping in accordance with 10 CFR Part 50.65;
- Characterization of reliability issues;
- Changing system and component unavailability;
- SSC (a)(1) and (a)(2) determinations;
- Identifying and addressing common cause failures;
- Appropriateness of performance criteria for SSCs classified (a)(2); and
- Adequacy of goals and corrective actions for SSCs classified (a)(1).

The inspectors reviewed system health reports, maintenance backlogs, and maintenance rule basis documents. The follow systems were selected for review:

- Residual heat removal system; and
- Reactor building closed loop cooling system.

These activities constituted two quarterly maintenance effectiveness inspection samples.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 5 samples)

a. Inspection Scope

The inspectors reviewed maintenance activities to verify that the appropriate risk assessments were performed prior to removing equipment for work. The inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The documents reviewed are listed in the Attachment.

- Week of April 4, that included the conclusion of an outage for 115 kV offsite power supply line 4 to support offsite maintenance, a two day maintenance period for the 'A' control room emergency ventilation air supply system, one day maintenance period for the reactor core isolation cooling system, and a one day outage for the 'B' EDG to replace the pre-lubricating oil pump motor;
- Week of April 25, that included 'A' and 'C' EDG monthly surveillance, a one day maintenance period for the 'A' residual heat removal (RHR) system, 'A' RHR service water system quarterly surveillance test, corrective maintenance on the main generator automatic voltage regulator, and emergent activities to troubleshoot the

failure of a drywell cooler, and to reduce power to 55 percent to plug a main condenser tube leak;

- Week of May 9, that included 'B' and 'D' EDG monthly surveillance testing, restoration of reactor power to 100 percent following a power reduction to support main condenser tube plugging, a three day maintenance outage for 115 kV offsite power line 3 and station reserve transformer T-2, and 'B' RHR system quarterly surveillance test;
- Week of May 30, that included functional testing of 'A' reactor protection system (RPS) electronic protection assemblies (EPAs), replacement of electrical components in the 'A' reactor water recirculation pump motor-generator speed control circuit, and a four day maintenance outage for the 'A' emergency service water system; and
- Week of June 6, that included a one day maintenance period for the 'D' EDG, 'B' and 'D' monthly surveillance, 'B' standby liquid control system quarterly surveillance, 'B' core spray quarterly surveillance and initiation logic cycle surveillance, and emergent maintenance to reduce power on two occasions to 75 percent for main condenser water box cleaning.

These activities constituted five maintenance risk assessments and emergent work control inspection samples.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 - 5 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the acceptability of the evaluations; the use and control of applicable compensatory measures; and compliance with technical specifications (TSs). The inspectors' review included verification that the operability determinations were conducted as specified by EN-OP-104, "Operability Determinations." The technical adequacy of the determinations was reviewed and compared to the TSs, UFSAR, and associated design basis documents (DBDs).

- CR-JAF-2011-02043, concerning the 'B' low pressure coolant injection inverter, 71INV-3B, which had a "CAP FUSE BLOWN" alarm lit on its indicator panel;
- CR-JAF-2011-02143, CR-JAF-2011-02352, CR-JAF-2011-02354, CR-JAF-2011-02361, and CR-JAF-2011-02390, concerning periods when the area temperatures for the high pressure coolant injection and reactor core isolation cooling systems were less than the design minimum of 65 degrees Fahrenheit;
- CR-JAF-2011-02592, concerning 'B' core spray check valve, 14CSP-63B, not fully seating;
- CR-JAF-2011-02609, concerning 'A' residual heat removal service water strainer, 10S-5A1, having a through-wall leak on the north side of the strainer outlet; and
- CR-JAF-2011-02918, concerning ultrasonic testing measurements on the 'A' RHRSW strainer housing that indicated less than the minimum allowable wall thickness.

These activities constituted five operability evaluation inspection samples.

b. <u>Findings</u>

No findings were identified.

1R19 <u>Post-Maintenance Testing</u> (71111.19 - 7 samples)

a. <u>Inspection Scope</u>

The inspectors reviewed post-maintenance test (PMT) procedures and associated testing activities for selected risk-significant mitigating systems to assess whether the effect of maintenance on plant systems was adequately addressed by control room and engineering personnel. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness, and were consistent with DBDs; test instrumentation had current calibrations, adequate range, and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon completion, the inspectors verified that equipment was returned to the proper alignment necessary to perform its safety function. PMT was evaluated for conformance with the requirements of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." The documents reviewed are listed in the Attachment.

- WO 52038571, replacement of 93AC-C1(M), the motor of air start compressor 'C1' for the 'C' EDG, as preventive maintenance;
- WO 52038566, replacement of 93AC-C2(M), the motor of air start compressor 'C2' for the 'C' EDG, as preventive maintenance;
- WO 274643, replacement of 93EDG-604A1, the fuel oil filter differential pressure gauge for the electric driven fuel oil pump for the 'A' EDG, due to a permanent plant modification;
- WO 178586; replacement of valves and tubing associated with 93EDG-604A1, the fuel oil filter differential pressure gauge for the electric driven fuel oil pump for the 'A' EDG, due to a cross-threaded connection which was unable to be reused for a permanent plant modification;
- WO 00271495, repair of 10S-5A2, a residual heat removal service water strainer, due to a localized area not meeting minimum wall thickness requirements;
- WO 00272298-01, to replace auxiliary contactor 3/2 in the breaker for control room emergency ventilation air supply system damper 70MOD-106A; and
- WO 00276721 to replace the D-EDG voltage regulator motor operated potentiometer due to reactive electrical load swings.

These activities constituted seven PMT inspection samples.

b. <u>Findings</u>

No findings were identified.

1R22 Surveillance Testing (71111.22 - 5 samples)

a. Inspection Scope

The inspectors witnessed performance of surveillance tests (STs) and/or reviewed test data of selected risk-significant SSCs to assess whether the SSCs satisfied TSs, UFSAR, technical requirements manual, and FitzPatrick procedure requirements. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness, and were consistent with DBDs; test instrumentation had current calibrations, adequate range, and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon ST completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. The following STs were reviewed:

- ST-9BA, "EDG A and C Full Load Test and ESW Pump Operability Test," Revision 12;
- ST-23C, "Jet Pump Operability Test for Two Loop Operation," Revision 26;
- ISP-125A, "HPCI Auto Isolation Instrument Functional Test/Calibration," Revision 29;
- ST-3JB, "Core Spray Initiation Logic System B Functional Test," Revision 2; and
- ISP-71A, "Intermediate Range Monitor Division A Instrument Trip Functional Calibration," Revision 2.

These activities represented five surveillance testing inspection samples.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors observed emergency response organization activities during the emergency preparedness drill that was conducted on May 10, 2011. The inspectors verified that emergency classification declarations, notifications, and protective action recommendations were properly completed. The inspectors evaluated the drill for conformance with the requirements of 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities." The inspectors observed FitzPatrick's critique and compared FitzPatrick's self-identified issues with observations from the inspectors' review to ensure that performance issues were properly identified.

This activity constituted one drill evaluation inspection sample.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06 - 1 sample)

a. Inspection Scope

FitzPatrick's program was evaluated against the requirement to provide adequate protection of the public from effluent releases resulting from normal operations of the plant by maintaining the dose to the maximally exposed member of the public as far below the dose limits in 10 CFR Part 20 and 40 CFR Part 190, as low as is reasonably achievable (ALARA). Criterion 60 in 10 CFR Part 50 Appendix A, requires the control and appropriate mitigation of radioactive materials released as plant effluents. In addition, Paragraph 50.34a (and the associated Appendix I) to 10 CFR Part 50 provide dose based design criteria to ensure the effectiveness of plant effluent processing systems in maintaining effluent releases to the plant environs ALARA.

Event Report and Effluent Report Reviews

The inspectors reviewed the radiological effluent release reports issued since the last inspection. The inspectors determined that the reports were submitted as required by the off-site dose calculation manual (ODCM)/TS. The inspectors identified radioactive effluent monitor operability issues reported by FitzPatrick as provided in effluent release reports, and determined that the issues were entered into the corrective action program (CAP) and adequately resolved.

ODCM and UFSAR Reviews

The inspectors reviewed changes to the ODCM made by FitzPatrick since the last Inspection, against the guidance in NUREG-1301, 1302 and 0133, and Regulatory Guides 1.109, 1.21 and 4.1. The inspectors determined that FitzPatrick had not identified any non-radioactive systems that had become contaminated as disclosed either through an event report or are documented in the ODCM since the last inspection.

Groundwater Protection Initiative (GPI) Program

The inspectors reviewed reported groundwater monitoring results, and changes to FitzPatrick's written program for identifying and controlling contaminated spills/leaks to groundwater.

Procedures, Special Reports & Other Documents

The inspectors reviewed licensee event reports, event reports and/or special reports related to the effluent program issued since the previous inspection. The inspectors identified no additional focus areas for the inspection based on the scope/breadth of problems described in these reports. The inspectors reviewed effluent program implementing procedures, particularly those associated with effluent sampling, effluent monitor set point determinations and dose calculations.

Walkdowns and Observations

The inspectors walked down selected components of the gaseous and liquid discharge systems to verify that equipment configuration and flow paths align with the UFSAR document descriptions, and reviewed and assessed equipment material condition. For equipment or areas associated with the systems selected above, that were not readily accessible due to radiological conditions, the inspectors reviewed FitzPatrick's material condition surveillance records. The inspectors walked down filtered ventilation systems for which test results were reviewed during the inspection. The inspectors verified that there were no conditions, such as degraded high efficiency particulate air (HEPA)/charcoal banks, improper alignment, or system installation issues that would impact the performance, or the effluent monitoring capability, of the effluent system. The inspectors determined that FitzPatrick had not made any significant changes to its effluent release points.

The inspectors observed the routine processing and discharge of effluents (including sample collection and analysis). The inspectors verified that appropriate effluent treatment equipment was being used and that radioactive liquid waste was being processed and discharged in accordance with procedure requirements and aligned with discharge permits.

Sampling and Analyses

The inspectors selected effluent sampling activities and verified that adequate controls had been implemented to ensure representative samples were obtained (e.g. provisions for sample line flushing, vessel recirculation, composite samplers, etc.). The inspectors determined that the facility was not routinely relying on the use of compensatory sampling, in lieu of adequate system maintenance, based on the frequency of compensatory sampling since the last inspection.

The inspectors reviewed the results of the inter-laboratory comparison program to verify the quality of the radioactive effluent sample analyses. The inspectors verified that the inter-laboratory comparison program included hard-to-detect isotopes, as appropriate.

Instrumentation and Equipment

Effluent Flow Measuring Instruments - The inspectors reviewed the methodology used by FitzPatrick personnel to determine the effluent stack and vent flow rates. The inspectors verified that the flow rates were consistent with radiological effluents technical specifications (RETS)/ODCM or UFSAR values, and that differences between assumed and actual stack and vent flow rates did not affect the results of the projected public doses.

Air Cleaning Systems - The inspectors verified that ST results since the previous inspection for TS required ventilation effluent discharge systems (HEPA and charcoal filtration met TS acceptance criteria).

Dose Calculations

The inspectors reviewed one radioactive liquid and three gaseous waste discharge permits. The inspectors verified that the projected doses to members of the public were accurate and based on representative samples of the discharge path. The inspectors evaluated the methods used to determine the isotopes included in the source term to ensure all applicable radionuclides were included, within detectability standards. The inspectors reviewed the current Part 61 analyses to ensure hard-to-detect radionuclides were included in the source term.

The inspectors reviewed changes in FitzPatrick's offsite dose calculations since the last inspection. The inspectors verified that the changes were consistent with the ODCM and Regulatory Guide 1.109. The inspectors reviewed meteorological dispersion and deposition factors used in the ODCM and effluent dose calculations to ensure appropriate factors were being used for public dose calculations. The inspectors reviewed the latest land use census and verified that changes had been factored into the dose calculations.

GPI Implementation

The inspectors verified that FitzPatrick was continuing to implement the voluntary Nuclear Energy Institute (NEI)/Industry GPI since the last inspection. The inspectors reviewed monitoring results of the GPI to determine if FitzPatrick had implemented its program as intended, and to identify any anomalous results. Recent positive tritium results are described below.

The inspectors reviewed leakage and spill events that had been documented by FitzPatrick personnel in accordance with 10 CFR 50.75(g). The inspectors reviewed evaluations of leaks and spills, as well as remediation actions taken to evaluate their effectiveness. The inspectors reviewed onsite contamination events involving contamination of ground water.

The inspectors verified that onsite ground water sample results and a description of any significant onsite leaks/spills into ground water for each calendar year were documented in the annual radiological environmental operating report for radiological environmental monitoring program (REMP) or the annual radiological effluent release report for the RETS.

b. Findings

No findings were identified.

Since September 2009, FitzPatrick has been investigating indications of low level tritium contamination in onsite ground water. During 2010, an additional 29 ground water monitoring wells were installed in order to identify the cause(s) of the subsurface contamination. Recent monitoring well results indicate tritium activity ranging from 500 - 5,000 picocuries per liter (pCi/L) in these onsite wells, with no tritium contamination detected in any offsite environmental samples. To date, the ground water suppression drainage system surrounding the reactor building (reactor building perimeter drain sump) has indicated a maximum ground water tritium concentration of 105,000 pCi/L. The reactor building perimeter drain sump discharge has been collected in holding tanks for monitoring prior to controlled discharge to Lake Ontario since the Fall of 2009. No regulatory limits have been exceeded based on the current condition. FitzPatrick continues its investigation and resolution of this issue through CR-JAF-2009-04166.

4. OTHER ACTIVITIES

4OA2 Problem Identification and Resolution

.1 Review of Items Entered into the Corrective Action Program (71152)

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," to identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all items entered into FitzPatrick's CAP. The review was accomplished by accessing FitzPatrick's computerized database for CRs and attending CR screening meetings. In accordance with the baseline inspection procedures, the inspectors selected items across the Initiating Events, Mitigating Systems, Barrier Integrity, and Public Radiation Safety cornerstones for additional follow-up and review. The inspectors assessed FitzPatrick personnel's threshold for problem identification, the adequacy of the cause analyses, and extent of condition review, operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are listed in the Attachment.

The inspectors reviewed 13 corrective action CRs that were initiated since the last radioactive effluent inspection that were associated with this program area. The inspectors verified that problems identified by these CRs were properly characterized in FitzPatrick's event reporting system and that applicable causes and corrective actions were identified commensurate with the safety significance of the radiological occurrences.

b. Findings and Observations

No findings were identified. The inspectors determined that Entergy staff identified equipment, human performance and program issues at an appropriate threshold and entered them into the CAP.

.2 Semiannual Review to Identify Trends (71152 - 1 sample)

The inspectors performed a semi-annual review of site issues, to identify trends that might indicate the existence of more significant safety issues, as required by Inspection Procedure 71152, "Identification and Resolution of Problems." The inspectors included in this review, repetitive or closely-related issues that may have been documented by FitzPatrick personnel outside of the CAP, such as trend reports, performance indicators, system health reports, and maintenance backlogs. The inspectors also reviewed the FitzPatrick CAP database for the first and second quarters of 2011, to assess CRs written in various subject areas (equipment problems, human performance issues, etc.), as well as individual issues identified during the NRCs daily CR review (Section 4OA2.1). The inspectors reviewed the FitzPatrick quarterly trend report for the first quarter of 2011, conducted under EN-LI-121, "Entergy Trending Process," Revision 10, to verify that FitzPatrick personnel were appropriately evaluating and trending adverse conditions in accordance with applicable procedures.

b. Findings and Observations

No findings were identified. The inspectors determined that Entergy staff identified equipment, human performance, and program issues at an appropriate threshold and entered them into the CAP.

.3 Annual Sample: Review of Safety Relief Valve Circuitry Test Failures (1 sample)

a. Inspection Scope

The inspectors selected CR-JAF-2010-05193 as a problem identification and resolution sample for a detailed follow-up review. This CR documented FitzPatrick's review regarding test failures on the circuitry to actuate safety relief valves (SRVs) from remote shutdown panel 02ADS-071.

The inspectors reviewed the test procedure completed in 2010, maintenance procedures used for the repairs, all three apparent cause evaluations (inspectors had previously reviewed the first two evaluations in Inspection Report 05000333/2009003), corrective actions from the evaluations, and the associated CRs. The documents reviewed are listed in the Attachment. The inspectors assessed FitzPatrick's problem identification threshold, thoroughness and accuracy of apparent cause evaluations, extent of condition reviews, and the appropriateness, priority and timeliness of corrective actions. The inspectors also reviewed an NRC-approved relief request for the American Society of Mechanical Engineers (ASME) code and a TS amendment which enable use of a revised testing method.

b. Findings and Observations

No findings were identified. The apparent cause evaluations and corrective actions were reasonable and appropriate.

The reviewed tests measured the resistance in the actuation circuits of each of the eleven SRVs from the remote shutdown panel to verify the operability of this function, and the tests were performed weeks prior to refueling outages on a 24-month cycle. In each of the three prior testing periods (i.e., 2006, 2008, and 2010), SRV 02RV-71H, SRV 02RV-71J, or both, initially did not meet the acceptance criteria for circuit resistance. Subsequent troubleshooting and repairs determined that the high circuit resistances which caused the test failures had occurred in the multi-pin electrical connectors at the SRV solenoid-operators. Apparent cause evaluations determined that the factors affecting the test outcomes were tightness and torquing of connectors, connector lubricant on the pins, and resistance testing at lower voltages than operating voltage (i.e., 9 volts direct current (Vdc) vs. 125 Vdc). Corrective actions included test procedure revisions to address torquing, pin cleanliness, and testing voltages. Nonetheless, SRV 02RV-71H did not meet resistance acceptance criteria in 2010, despite the two prior evaluations and corrective actions.

The inspectors noted that the longer term resolution of the problem was being accomplished via plans to replace the solenoid-operated SRVs over the next two refueling outages with valves having connectors of an improved design with no need for lubrication, and via revised testing of the actuation circuits such that future tests are planned to be functional tests involving actuating the solenoid operators from the switches at the remote shutdown panel using system operating voltage.

4OA3 <u>Event Follow-up</u> (71153 - 1 sample)

.1 <u>(Closed) LER 05000333/2010-004, "Main Steam Isolation Valve Leak Rate Exceeds Authorized Limit"</u>

a. <u>Inspection Scope</u>

During the 2010 refueling outage, local leak rate testing (LLRT) results indicated that the 'C' main steam line isolation valves (MSIVs), 29AOV-80C and 29AOV-86C, would not hold pressure. Previous at-power operation in this condition constituted a violation of TS 3.6.1.3, "Primary Containment Isolation Valves," because leakage exceeded the acceptance criteria as delineated in TS surveillance requirement (SR) 3.6.1.3.10, which required that the combined main steam line leakage rate be less than or equal to 46 standard cubic feet per hour (scfh). The 'C' MSIVs were disassembled by Entergy personnel to determine the failure mechanisms. For 29AOV-80C, Entergy personnel determined the cause was related to corrosion products on the valve seat. For 29AOV-86C, Entergy personnel determined the failure was due to flow erosion of the valve body and seating surface. Entergy personnel corrected these conditions during the refueling outage and subsequent leak rate tests were completed satisfactorily.

The inspectors reviewed the licensee event report (LER) and CR-JAF-2010-05544 regarding these conditions. The inspectors noted maintenance had been performed on the 29AOV-80C and 29AOV-86C due to excessive seat leakage during the 2000 refueling outage. Subsequent leak rate testing, conducted by Entergy personnel every two years during refueling outages in 2002 through 2008, had been completed with satisfactorily results. The inspectors' review determined that there was no evidence of a trend in the measured leak rates of either valve from 2002 to 2008 that would have indicated to Entergy staff that leakage in excess of the TS-allowed acceptance criteria would develop prior to the 2010 refueling outage. The inspectors verified that all in-service inspection requirements were implemented and there was no previous corrective action information available to station personnel that would have reasonably prevented the excessive valve leakage or indicated valve degradation. The inspectors also determined that station staff identified the leakage at the first reasonable opportunity. The inspectors concluded that, although previous at-power operation with a combined main steam line leakage rate greater than 46 scfh constituted a violation of TS 3.6.1.3, the condition was not reasonably within Entergy staff's ability to have foreseen or prevented, and therefore did not constitute a performance deficiency.

b. <u>Findings</u>

This issue was considered within the traditional enforcement process because there was no performance deficiency identified, and NRC Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening" directs disposition of such issues in accordance with the NRC Enforcement Policy. The inspectors used the Enforcement Policy, Section 6.1, "Reactor Operations," to evaluate the significance of this violation. The inspectors concluded that the violation was more than minor and best characterized as Severity Level IV (very low safety significance) because it is similar to Enforcement Policy Section 6.1, Example d.1. Additionally, the inspectors assessed the risk associated with the issue by using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." The inspectors screened the issue in accordance with Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and determined that a Phase 2 evaluation was required, because the issue constituted an actual open pathway in the physical integrity of the reactor containment, that being the 'C' main steam line with leakage past both MSIVs in excess of the TS limit.

The Phase 2 SDP process for containment barrier issues is conducted in accordance with the guidance provided in IMC 0609, Appendix H. IMC 0609, Appendix H, does not provide specific guidance on establishing the risk associated with penetration leakage for Mark I containments. Therefore, a Region I Senior Reactor Analysis (SRA) conducted a Phase 3 risk evaluation to evaluate the potential increase in the large early release frequency (LERF). The Phase 3 risk assessment uses the best available risk information to make a risk informed decision on the significance of inspection findings.

The Phase 3 risk evaluation determined that the increase in LERF was likely less than the 1E-7 per year threshold and that the finding was, therefore, of very low safety significance. The SRA determined the top fifteen dominant core damage sequences from the NRC's FitzPatrick Standardized Plant Analysis Risk (SPAR) model version 8.16. The SRA used the method as outlined in Section 3.2.8, "Main Steam Isolation Valve Leakage," of NUREG 1785, "Basis Document for Large Early Release Frequency (LERF) Significance Determination Process (SDP)." The result indicated that there was essentially no increase in LERF because the SDP already assumed that all high pressure core damage sequences would cause a large early release, so having a pair of MSIVs leaking would be no worse.

In their LER, the FitzPatrick staff documented an increase in LERF in the low E-8 per year range using their current level 2 PRA model which includes MSIV leakage as a contributor to containment failure. This model goes beyond the SDP assumption that all high pressure core damage sequences result in a large early release, allowing for operator actions to reduce RPV pressure and improving the possibility of getting water to the reactor vessel or the drywell floor before reactor vessel breach.

The SRA considered several qualitative factors in reaching a final risk determination for this issue. Considerations which tended to decrease the risk associated with this finding were that: 1) the leak testing methodology (low rate of pressurization) may not seat an MSIV in the same manner that would occur under actual operating conditions, given the reactor steam pressure tending to close the valves; 2) given the actual mechanisms which allowed the leakage (as described in the LER), it is likely that the two leaking MSIVs with reactor pressure tending to close the valves would have offered considerably more resistance to steam flow than if they were fully open; 3) the downstream turbine bypass and stop/throttle valves would be closed if the condenser was not available, and as such, these valves would also provide some additional (though not specifically quantifiable) isolation function; and 4) there would be significant deposition of radioactive nuclides in the main steam lines and condenser, thus limiting the radiological inventory transported to the site boundary and limiting the potential for a large early release. The SRA also determined that external initiating events such as a seismic event or a fire would not significantly affect these quantitative factors, given the robustness of the main steam piping downstream of the MSIVs and the very low chance that a fire would damage this piping. Therefore, the SRA determined that both the quantitative LERF calculation and qualitative factors supported the conclusion that this finding was of very low risk significance (Green).

Because this issue was of very low safety significance (Green) and it has been determined that the issue was not within Entergy personnel's ability to foresee and correct, that Entergy staff actions did not contribute to the degraded condition, and that actions taken were reasonable to identify and address this matter, and as such no performance deficiency existed, the NRC has decided to exercise enforcement discretion in accordance with Section 3.5 of the NRC Enforcement Policy and refrain from issuing enforcement action for the violation of TSs (EA-11-170). Further, because licensee actions did not contribute to this violation, it will not be considered in the assessment process or the NRC's Action Matrix. This LER is closed.

4OA5 Other Activities

.1 Review of Source Range Monitor Operability Requirements During Core Alterations

a. Inspection Scope

The inspectors reviewed refueling operations during the 2010 refueling outage. At the time that refueling operations began, one of the four installed source range [neutron] monitors (SRMs) was inoperable (SRM 'A'). During core alterations (movement of fuel or control rods within the reactor vessel), TS surveillance requirement (SR) 3.3.1.2.2 requires that an operable SRM is located in the core quadrant where core alterations are being performed. The inspectors questioned how fuel movements were being controlled such that no movements would be performed in the core quadrant that contained SRM 'A'. Entergy staff responded that, in accordance with procedure OSP-66.001, "Management of Refueling Activities," Revision 1, refueling operations could proceed in any core location with any single SRM out of service. Entergy personnel indicated this conclusion was based on a definition of "core quadrant" that was developed and adopted by the station in 2004. The inspectors reviewed the issue of SRM operability requirements during refueling operations to determine if the station's core quadrant definition was consistent with the requirements of TSs. As part of this review, inspectors reviewed the applicable TS and TS bases information regarding SRMs and related design bases documentation.

Findings

Introduction: The inspectors identified an unresolved item (URI) associated with the adequacy of Entergy's bases for a change to the definition of "core quadrant" as applied to refueling operations that was implemented by the Entergy staff at FitzPatrick in 2004.

Description: The inspectors noted that TS Bases 3.3.1.2, "Source Range Monitor (SRM) Instrumentation," documented that the primary indication of refueling, shutdown, and low power operations neutron flux levels is provided by the SRMs or special movable detectors connected to the SRM circuits. The SRMs provide monitoring of reactivity changes during fuel or control rod movement and give the control room operators early indication of unexpected subcritical multiplication. The inspectors further noted that the TS Bases documented that the SRMs have no safety function and are not assumed to function during any UFSAR design basis accident or transient analysis. However, the

TS Bases describe that the SRMs provide the only on-scale monitoring of neutron flux levels during startup and refueling.

The reactor core at FitzPatrick consists of 560 fuel assemblies, arranged symmetrically in an octagonal configuration. Due to this symmetry, the core can be divided into four equal quadrants, using two perpendicular axes (000-180°, and 090-270°) that cross at the geometric center of the core. The reactor core also contains four installed SRMs, with one in each of the quadrants as described above.

The inspectors reviewed station procedures and supporting documentation to evaluate whether the definition of a "core quadrant," as defined in FitzPatrick station procedures, was consistent with the intent of TS 3.3.1.2. The "core quadrant" definition as currently implemented by station personnel indicated that the placement of the axes for a core quadrant was based on the SRM locations. This orientation resulted in quadrant axes that are rotated approximately 18° clockwise from the arrangement that was described above, and resulted in quadrant boundaries that bisect individual fuel assemblies. Entergy personnel determined that such fuel assemblies could be considered to reside in either of the adjacent quadrants. Entergy personnel used this concept to establish two quadrant boundaries, one rotated clockwise by 16° and the other rotated clockwise by 20°, such that the SRM would partially reside in both quadrants using either boundary. This created a set of fuel assemblies along a quadrant boundary that could be considered to be a part of either of the adjacent quadrants. Entergy personnel determined that, by selecting the appropriate boundary in the case that a single SRM is inoperable, this quadrant arrangement supported the requirement of TS SR 3.3.1.2.2, while allowing movement of fuel anywhere in the core.

The inspectors verified that TS 3.3.1.2 and its associated TS Bases do not provide a description or explicit definition of what is considered a "core quadrant." However, the inspectors noted that, as a result of the station's interpretation, an SRM would be expected to detect reactivity changes that occurred at greater distances from the detector than it would using the 000-180°, 090-270° quadrant definition, and therefore, that the approach implemented in 2004 may not be consistent with the current licensing and design bases. The inspectors reviewed the SRM vendor document that was referenced by FitzPatrick personnel as support for the 2004 definition of "core quadrant." The inspectors' review determined that the vendor document did not appear to be analytically based to support the definition of "core quadrant" as implemented by FitzPatrick staff in procedure changes made in 2004.

Based on the above information, the inspectors determined that further NRC evaluation is needed to assess the analyses and licensee regulatory screening reviews to support the rotated core quadrants approach to TS requirements for SRM operability during core alterations, and therefore, to determine whether a performance deficiency exists with regards to Entergy's application of the rotated core quadrant boundary approach. It should be noted that the inspectors' sampling review in the course of inspection during the 2010 refueling outage did not identify a core alteration from a core quadrant with an inoperable SRM using the standard core quadrant definition (symmetrical, four equal quadrants). **(URI 05000333/2011003-02, Source Range Monitor Operability Requirements during Core Alterations)**

.2 (Closed) NRC Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event"

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included: (1) an assessment of the licensee's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines, and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63 and station design bases; (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

Inspection Report 05000333/2011008 (ML111330455) documented detailed results of this inspection activity.

3. (Closed) NRC Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)"

On May 13, 2011, the inspectors completed a review of the licensee's severe accident management guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990's, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee's training of personnel on the use of SAMGs, and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant-specific results for the James A. FitzPatrick Nuclear Power Plant were provided in an Attachment to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated May 27, 2011 (ML111470361).

4OA6 Meetings, Including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. K. Bronson and other members of FitzPatrick's management at the conclusion of the inspection on July 14, 2011. The inspectors asked FitzPatrick management whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified by FitzPatrick personnel.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

K. Bronson, Site Vice President
B. Sullivan, General Manager, Plant Operations
M. Woodby, Director, Engineering
B. Finn, Director, Nuclear Safety Assurance
C. Adner, Manager, Operations
J. LaPlante, Manager, Security
J. Barnes, Manager, Emergency Preparedness
M. Reno, Manager, Maintenance
C. Brown, Manager, Quality Assurance
V. Bacanskas, Manager, Design Engineering
D. Poulin, Manager, System Engineering
P. Scanlon, Manager, Programs and Components Engineering
J. Pechacek, Manager, Licensing
R. Brown, Acting Manager, Radiation Protection

LIST OF ITEMS OPEN, CLOSED, AND DISCUSSED

Opened

05000333/2011003-02	URI	Source Range Monitor Operability Requirements during Core Alterations

Opened and Closed

05000333/2011003-01	NCV	UFSAR Emergency Bus Voltage Not Updated, Consistent with Current Plant Conditions

Closed

05000333/2010004-00	LER	Main Steam Isolation Valve Leak Rate Exceeds Authorized Limit
05000333/2515/183	TI	Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event
05000333/2515/184	TI	Availability and Readiness Inspection of Severe Accident Management Guidelines

Discussed

None

LIST OF DOCUMENTS REVIEWED

Section 1RO1: Adverse Weather Protection

Procedures:
AOP-72, "115 kV Grid Loss, Instability, or Degradation," Revision 9
OP-4, "Circulating Water System," Revision 69
OP-44, "115 kV System," Revision 18
ST-9W, "Electrical Lineup and Power Verification," Revision 10
AOP-13, "High Winds, Hurricanes and Tornadoes," Revision 19
AP-12.04, "Seasonal Weather Preparations," Revision 18, Attachment 3, "Operations Department Warm Weather Preparation Checklist"
AP-12.04, "Seasonal Weather Preparations," Revision 18, Attachment 5, "Maintenance Department Warm Weather Preparation Checklist"
OP-55A, "Control and Relay Room Refrigeration Water Chiller," Revision 23
OP-55B, "Control Room Ventilation and Cooling," Revision 34

Condition Reports:
CR-JAF-2011-02808

Section 1RO4: Equipment Alignment

Procedures:
OP-22, "Diesel Generator Emergency Power," Revision 57
OP-21, "Emergency Service Water (ESW)," Revision 37
MST-071.10, "LPCI Battery Weekly Surveillance Test," Revision 38
OP-43C, "LPCI Independent Power Supply System," Revision 21
ST-16GA, "A LPCI MOV Independent Power Supply Monthly Test," Revision 2

Documents:
System Health Report, "71-DC –DC Distribution System, Q1-2011," Revision 0

Condition Reports:
CR-JAF-2006-03463
CR-JAF-2007-03635
CR-JAF-2007-04225
CR-JAF-2010-08057
CR-JAF-2011-02043
CR-JAF-2011-03265

Section 1RO5: Fire Protection

Procedures:
PFP-PWR08, "Administration Building/Elev. 300', Fire Area/Zone IA/AD-6," Revision 3
PFP-PWR26, "Reactor Building/Elev. 326', Fire Area/Zone IX/RB-1A," Revision 3
PFP-PWR02, "West Cable Tunnel/Elev. 258', Fire Area/Zone IC/CT-1," Revision 4
PFP-PWR12, "Relay Room/Elev. 286', Fire Area/Zone VII/RR-1," Revision 4
PFP-PWR33, "Pump Rooms (Screenwell)/Elev. 255', Fire Area/Zone XII/SP-1, XIII/SP-2, IB/FP-1, FP-3," Revision 1

Section 1R06: Flood Protection Measures

Procedures:
ESP-50.001, "Floor Drain Flow Test," Revision 1

Documents:
SWEC Calculation 14620.9015-US(N)-001-0, "Evaluation of Impact of Flooding Inside Emergency Diesel Generator Rooms on Safety-Related Equipment," Revision 0
SWEC Calculation 14620-8-9017-1, "Potential Flooding Impact for EDG Room Sprinkler Actuation with Floor Drains Plugged and Two Equipment Drains Opened and all Floor Drains Opened," Revision 2

Condition Reports:
CR-JAF-2011-02214

Section 1R11: Licensed Operator Requalification Program

Documents:
Evaluation 2011A, Revision C

Section 1R12: Maintenance Effectiveness

Procedures:
EN-DC-203, "Maintenance Rule Program," Revision 1
EN-DC-204, "Maintenance Scope and Basis," Revision 2
EN-DC-205, "Maintenance Rule Monitoring," Revision 3
EN-DC-206, "Maintenance Rule (a)(1) Process," Revision 1

Documents:
DBD-010, "Design Basis Document for the Residual Heat Removal System," Revision 12
JAF-RPT-RHR-02281, "Maintenance Rule Basis Document System 10 Residual Heat Removal System," Revision 9
System Health Report, 10 RHR & RHRSW, 3rd Quarter 2010
JAF-RPT-RBCLC-02809, "Maintenance Rule Basis Document for System 015 RBCLC," Revision 6
System Health Report for Reactor Building Closed Loop Cooling System, fourth quarter 2010
JENG-APL-10-05, "Maintenance Rule (a)(1) Action Plan, Reactor Building Closed Loop Cooling Containment Isolation Valves," Revision 0

Condition Reports:

CR-JAF-2008-01426	CR-JAF-2008-03539	CR-JAF-2009-01041
CR-JAF-2008-02933	CR-JAF-2008-03544	CR-JAF-2009-01332
CR-JAF-2008-03025	CR-JAF-2008-03750	CR-JAF-2009-02894
CR-JAF-2008-03373	CR-JAF-2008-04163	CR-JAF-2009-02965
CR-JAF-2008-03408	CR-JAF-2008-04242	CR-JAF-2009-03334
CR-JAF-2008-03483	CR-JAF-2009-00094	CR-JAF-2010-00430
CR-JAF-2008-03529	CR-JAF-2009-00624	CR-JAF-2010-00478

CR-JAF-2010-01072
CR-JAF-2010-01382
CR-JAF-2010-03442
CR-JAF-2010-04912
CR-JAF-2010-05081
CR-JAF-2010-05306
CR-JAF-2010-05588
CR-JAF-2010-06255
CR-JAF-2010-06331
CR-JAF-2010-06486
CR-JAF-2010-06503
CR-JAF-2010-05699
CR-JAF-2010-06637
CR-JAF-2010-06638
CR-JAF-2010-07056
CR-JAF-2010-07135
CR-JAF-2010-07253

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures:
AP-05.13, "Maintenance During LCOs," Revision 9
AP-10.10, "On-Line Risk Assessment," Revision 6
EN-WM-104, "On Line Risk Assessment," Revision 4

Section 1R15: Operability Determinations and Functionality Assessments

Procedures:
EN-OP-104, "Operability Determination Process," Revision 5
ST-2AV, "RHR Loop B Keep-Full Check Valve Functional Test (IST)," Revision 0
ST-3U, "Core Spray Hold Pump Min Flow Check Valve Reverse Flow Test (IST)," Revision 8
ST-39J, "Leak Testing of RHR and Core Spray Testable Check Valves (IST)," Revision 17

Documents:
DBD-014, "Design Basis Document for the Core Spray System 014," Revision 10
DBD-066, "Design Basis Document for the Reactor Building HVAC Systems," Revision 11
FM-20A, "Flow Diagram Residual Heat Removal System 10," Revision 72
FM-23A, "Flow Diagram Core Spray System 14," Revision 49

Section 1R19: Post Maintenance Testing

Procedures:
ST-2XA, "RHR Service Water Loop A Quarterly Operability Test (IST)," Revision 13
ST-18, "Main Control Room Emergency Fan and Damper Operability Test," Revision 30
ST-9BB, "EDG B and D Full Load Test and ESW Pump Operability Test," Revision 11

Documents:
DBD-046, "Design Basis Document for the Normal Service Water, Emergency Service Water, RHR Service Water," Revision 18
ST-2XA-110428-52318165

Work Orders:
WO 00272298-01
WO 00276721

Section 1R22: Surveillance Testing

Documents:
ST-23C-110622
ST-23C-110623
ST-23C-110624
ST-23C-110625
ST-23C-110626

Section 1EP6 Drill Evaluation

Documents:
James A. FitzPatrick Nuclear Power Plant Emergency Plan Drill, May 10, 2011

Section 2RS6: Radioactive Gaseous and Liquid Effluent Treatment

Procedures:
SP-01.05, "Waste Water Sampling and Analysis," Revision 12
SP-01.06, "Gaseous Effluent Sampling and Analysis," Revision 15
DVP-01.02, "Offsite Dose Calculation Manual," Revision 11
EN-RP-404, "Operation and Maintenance of HEPA Vacuum Cleaners and HEPA Ventilation Units," Revision 3
EN-RP-402, "DOP Challenge Testing of HEPA Vacuums and Portable Ventilation Units," Revision 4
CHSO-10, "Ground Water Monitoring Program"
PSP-17, "Post-Accident Sampling System Operating Procedure," Revision 24

Documents:
Quality Assurance Audit Report QA-14/15-2009-JAF-1
2008 Annual Radioactive Effluent Release Report
2009 Annual Radioactive Effluent Release Report
Final Safety Analysis Report, Section 11, Radioactive Waste

Condition Reports
CR-JAF-2010-05205
CR-JAF-2010-04686
CR-JAF-2010-00564
CR-JAF-2010-08585
CR-JAF-2009-03564
CR-JAF-2010-06110
CR-JAF-2010-06905
CR-JAF-2009-04166
CR-JAF-2010-08295
CR-JAF-2010-08113
CR-JAF-2010-02236
CR-JAF-2010-00272
CR-JAF-2010-07319

Section 4OA2: Identification and Resolution of Problems

Procedures:
MP-002.04, "Reactor Vessel Safety/Relief Valve (SRV) Maintenance," Revision 33, completed Sept. 13, 2010, under WO 00197909
MP-200.19, “NAMCO Connectors Maintenance,” Revision 7
MP-200.19, “NAMCO Connectors Maintenance,” Revision 6, completed Sept. 13, 2010 under WO 00197909
MST-029.05, “SRV Remote Actuation Maintenance Testing,” Revision 4, completed Sept. 10, 2010, under WO 51680816

Documents:
NRC letter, Relief Request VRR-06, Revision 1, dated October 1, 2009
Safety Evaluation Related to Amendment No. 297, attached to NRC letter dated July 21, 2010

Condition Reports:

CR-JAF-2006-02384	CR-JAF-2010-05518	CR-JAF-2011-01964
CR-JAF-2006-03317	CR-JAF-2010-06852	CR-JAF-2011-02319
CR-JAF-2006-04678	CR-JAF-2010-06862	CR-JAF-2011-02329
CR-JAF-2008-02865	CR-JAF-2010-06877	CR-JAF-2011-02645
CR-JAF-2008-04050	CR-JAF-2010-06924	CR-JAF-2011-02888
CR-JAF-2009-01692	CR-JAF-2011-00057	CR-JAF-2011-03227
CR-JAF-2010-05193	CR-JAF-2011-00152	CR-JAF-2011-03316
CR-JAF-2010-05195	CR-JAF-2011-00179	CR-JAF-2011-03351
CR-JAF-2010-05375	CR-JAF-2011-00696	

LIST OF ACRONYMS

AC	alternating current
ADAMS	Agencywide Documents Access and Management System
ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CFR	Code of Federal Regulations
CR	condition report
DBD	design basis document
EDG	emergency diesel generator
Entergy	Entergy Nuclear Northeast
EPA	electronic protection assembly
ESW	emergency service water
FitzPatrick	James A. FitzPatrick Nuclear Power Plant
GPI	groundwater protection initiative
HEPA	high efficiency particulate air
IMC	inspection manual chapter
JAF	James A. FitzPatrick
kV	kilovolt
LER	licensee event report
LERF	large early release frequency
LLRT	local leak rate test
MSIV	main steam isolation valve
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRR	Nuclear Reactor Regulation
NRC	Nuclear Regulatory Commission
ODCM	off-site dose calculation manual
PARS	Publicly Available Record
pCi/L	picocuries per liter
PMT	post-maintenance testing
RB	reactor building
REMP	radiological environmental monitoring program
RETS	radiological effluents technical specifications
RHR	residual heat removal
RSST	reserve station service transformer
SAMG	severe accident management guideline
scfh	standard cubic feet per hour
SDP	significance determination process
SL	severity level
SPAR	Standardized Plant Analysis Risk
SR	surveillance requirement
SRA	senior risk analyst
SRM	source range monitor
SRV	safety relief valve
SSC	structure, system, or component

ST	surveillance test
TS	technical specification
UFSAR	updated final safety analysis report
URI	unresolved item
V	volts
Vdc	volts direct current
WO	work order