February 2, 2005

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000333/2004005

Dear Mr. Sullivan:

On December 31, 2004, the US Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant. The enclosed integrated inspection report documents the inspection findings that were discussed on January 10, 2005, 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.

This report documents three findings of very low safety significance (Green). All three findings were determined to involve a violation of NRC requirements. However, because of their very low safety significance and because all three were entered into your corrective action program, the NRC is treating the findings as non-cited violations, in accordance with Section VI.A. of the NRC's Enforcement Policy. If you deny any of the non-cited violations noted in this report, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at FitzPatrick.

In accordance with 10 CFR 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 Web Site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Brian J. McDermott, Chief
Projects Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000333/2004005
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.:	05000333/2004005
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	October 1 - December 31, 2004
Inspectors:	L. M. Cline, Senior Resident Inspector D. A. Dempsey, Resident Inspector B.A. Bickett, Reactor Inspector J. D. Noggle, Senior Health Physicist D. M. Silk, Senior Emergency Preparedness Inspector
Approved by:	Brian J. McDermott, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000333/2004005; 10/01/2004 -12/31/2004; James A. FitzPatrick Nuclear Power Plant; Post Maintenance Testing, Problem Identification and Resolution, Event Follow-up.

The report covered a three month period of inspection by resident inspectors, and announced inspections by a senior health physicist, and regional specialist inspectors. Three Green non-cited violations (NCVs) were 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). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a non-cited violation of Technical Specification (TS) limiting condition for operation (LCO) 3.8.1, "Electrical Power Systems - AC Sources - Operating," for Entergy's failure to comply with the LCO required actions for one offsite power circuit inoperable within the specified time requirements.

  This issue is more than minor because it is associated with the Initiating Events cornerstone attribute of configuration control and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the finding was determined to be of very low risk significance (Green) because as a transient initiator it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. (Section 4OA2.1)

- Green. A self-revealing violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," occurred when Entergy failed to provide a procedure appropriate to the circumstances. Specifically, surveillance procedure ST-39H, "RPV System Leakage Test and CRD Class-2 Piping Inservice Test," did not include adequate precautions for reactor vessel level control. This resulted in operators draining 120 inches from the reactor vessel with the only on-scale level indicator out of service for testing.

  This finding is more than minor because it is associated with the procedure quality and configuration control attributes of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions while shutdown. In accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," dated May 25, 2004, a Region I Senior

Reactor Analyst (SRA) determined the finding to be of very low risk significance using a Phase 2 SDP evaluation.

The finding is associated with the cross-cutting area of human performance because in addition to the inadequate procedure, it involved the operators' failure to maintain adequate control of equipment status during operations in accordance with Entergy administrative procedure (AP)-19.01, "Conduct of Operations." (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for Entergy's failure to correct a condition adverse to quality involving high pressure coolant injection (HPCI) turbine steam supply isolation valve 23MOV-14 seat leakage. In November 2004 this resulted in 53 hours of unplanned HPCI system unavailability due to emergent corrective maintenance to address degradation of the valve disc and seat.

  This issue is more than minor because it is associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability of systems that respond to initiating events. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," a Region I SRA determined the finding to be of very low risk significance using a Phase 2 SDP evaluation. (Section 1R19)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The inspection period began with the plant shutdown for a refueling outage. Operators commenced a reactor start up on October 23 and placed the main generator on line on October 24. Except for planned power reductions for control rod pattern adjustments, FitzPatrick operated at or near rated power for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity [REACTOR - R]

1R01 Adverse Weather Protection (71111.01 - 2 samples)

a. Inspection Scope

The inspectors completed the following two samples (representing one actual adverse weather condition review and one preparation review). Documents reviewed in completing these inspections are listed in the attachment.

- On December 21 the inspectors reviewed Entergy's actions regarding the potential for frazil ice intrusion into the intake structure. The inspectors verified the condition and monitoring of the intake structure bar rack heaters, a system designed to mitigate the potential for ice intrusion. The inspectors reviewed the operating requirements for the heaters as specified in TS 3.7.2, and verified satisfactory completion of all surveillance requirements. The inspectors also verified that the heaters were operated in accordance with procedures and that test acceptance criteria ensured compliance with the design basis requirements.

- The inspectors reviewed and verified completion of the operations department cold weather preparation checklist contained in procedure AP-12.04, "Seasonal Weather Preparations." The inspectors reviewed the operating status of outdoor facilities and the reactor and turbine building heating and ventilation systems, reviewed the procedural limits and actions associated with low lake temperature, and walked down accessible areas of the buildings to assess the effectiveness of the ventilation systems. The walkdowns included discussions with operations and engineering personnel to ensure that they were aware of temperature restrictions and specified actions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04 - 3 samples, 71111.04S - 1 sample)

a. Inspection Scope

Partial System Walkdown (71111.04 - 3 samples)

The inspectors performed three partial system walkdowns to evaluate the operability of one train while the opposite train was inoperable or out of service for maintenance and testing. The inspectors compared system lineups to system operating procedures (OPs), system drawings, and the applicable chapters in the Updated Final Safety Analysis Report (UFSAR). The inspectors also verified the operability of critical system components by observing component material condition during the system walkdown and reviewing the maintenance history for each component. Other documents reviewed for this inspection are listed in the attachment. The inspectors performed partial walkdowns of the following systems:

- Train B residual heat removal (RHR) inspected on November 16 while train A RHR was out of service for planned maintenance;
- The alternate decay heat removal system (DHR) was inspected on October 1 while both trains of shutdown cooling were out of service for planned maintenance during week one of the refueling outage; and
- Train B emergency diesel generators (EDGs) during maintenance on train A EDGs on October 14.

Complete System Walkdown (71111.04S - 1 sample)

The inspectors performed a complete walkdown of the turbine building closed loop cooling (TBCLC) system to identify any discrepancies between the existing equipment lineup and the required lineup. During the walkdown the UFSAR, system drawings, and OPs were used to verify proper equipment alignment and operational status. The inspectors reviewed the open maintenance work requests (WRs) on the system for any 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 to assess their collective impact on system operation. In addition, the inspectors reviewed the condition report (CR) database to verify that equipment alignment problems were being identified and appropriately resolved. Other documents reviewed for this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 8 samples)

a. Inspection Scope

The inspectors toured eight areas important to reactor safety to evaluate conditions related to Entergy's control of transient combustibles and ignition sources; the material condition, operational status, and operational lineup of fire protection systems, equipment and features; and the fire barriers used to prevent fire damage or fire propagation. The inspectors used procedure ENN-DC-161, "Transient Combustible Program," in performing the inspection. The areas inspected included:

- Fire Area 1C/Zone CT-1;
- Fire Area 02/Zone CT-2;
- Fire Area 09/Zone SG-1;
- Fire Area 1A/Zone MG-1;
- Fire Area 07/Zone CS-1;
- Fire Area 07/Zone CR-1;
- Fire Area 12/Zone SP-1;
- Fire Area 13/Zone SP-2; and
- Fire Area 1B/Zone SH-1

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A - 1 sample)

a. Inspection Scope

The inspectors reviewed the testing and evaluation of test results for the control room ventilation air handling units (AHUs) performed on December 21-22. Surveillance procedure ST-18C, "Control Room Ventilation Air Handling Unit Performance Test," is performed every two years to verify safety-related unit cooler thermal performance. The inspectors reviewed performance data to verify that AHU operation was consistent with design. Other documents that were reviewed for this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08G - 1 sample)

a. Inspection Scope

The inspector observed non-destructive examination (NDE) activities and reviewed documentation of NDE and repair activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation could result in a significant increase in risk of core damage. The direct observations and documentation reviews were performed to verify that activities were performed in accordance with Sections IX and XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The inspector reviewed a sample of Entergy's inspection reports initiated to document the performance and record results of in-service inspection (ISI) examinations completed during the current refueling outage as well as those since the last refueling outage. The inspector also evaluated Entergy's effectiveness in resolving relevant indications identified during ISI activities. Documents reviewed for this inspection are listed in the attachment.

The inspector observed in-process data analysis of ultrasonic testing (UT) of the axial reactor pressure vessel (RPV) shell welds and reviewed selected documentation of the UT on axial RPV shell weld VV-2B. The inspector reviewed several NDE examinations including magnetic particle, visual, and UT examination documentation to verify the effectiveness of Entergy's program for monitoring degradation of risk significant piping structures, systems, and components (SSCs). The inspector evaluated the activities for compliance with the requirements of Section XI of the ASME Code. The inspector observed the procedure qualification for examination of RHR weld 24-10-131 to confirm that the qualification was performed per ASME Section XI, Appendix VIII requirements. The inspector examined Entergy's evaluation and disposition for continued operation without repair or rework of nonconforming conditions identified during ISI activities by review of CRs CR 2004-04225, CR 2004-04366, and JAF-CALC-04-00516. These reports documented various relevant indications observed during the visual examinations (VT-1) of the steam dryer.

The inspector reviewed one ASME Section XI code repair and its associated NDE from this refueling cycle. Specifically, the inspector reviewed welding activities and documentation performed on component H10-42A, RHR support. This review was performed to verify that the activities associated with welding on ASME Class I or II components were in accordance with applicable ASME code requirements.

The inspector observed the videotape portion of the in-vessel visual examinations performed on the steam dryer, including equipment calibration and final record review. The review was performed to evaluate examiner skills; to test equipment performance, examination technique, and inspection environment (water clarity); to assess Entergy's contractor oversight activities; and also to verify Entergy and the contractor's ability to identify and characterize observed indications.

b. <u>Findings</u>

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 1 sample)

a. Inspection Scope

On November 10 the inspectors observed licensed operator simulator training to assess operator performance during an emergency planning exercise involving failure of control rods to insert and a loss of coolant accident (LOCA) at low reactor vessel pressure. The inspectors evaluated the performance of risk significant operator actions, including the use of emergency operating procedures (EOPs), EOP-2, "Reactor Pressure Vessel Control" and EOP-3, "Failure to Scram." 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. The inspectors also reviewed simulator fidelity to evaluate the degree of similarity to the actual control room.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12Q - 2 samples)

a. Inspection Scope

The inspectors reviewed performance-based problems involving the primary containment atmosphere dilution and fuel pool cooling systems to assess the effectiveness of their maintenance programs. Reviews focused on: proper Maintenance Rule (MR) scoping in accordance with 10 CFR 50.65; characterization of reliability issues; changing system and component unavailability; 10 CFR 50.65 (a)(1) and (a)(2) classifications; identifying and addressing common cause failures; trending key parameters; and the appropriateness of performance criteria for SSCs classified (a)(2), as well as the adequacy of goals and corrective actions for SSCs classified (a)(1). The inspectors reviewed system health reports, maintenance backlogs, and MR basis documents. Documents reviewed for this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed risk assessments associated with five different work weeks during the inspection period. The inspectors verified that risk assessments were performed in accordance with AP-10.10, "On-line Risk Assessment;" that risk of scheduled work was managed through the use of compensatory actions and schedule adherence; and that applicable contingency plans were properly identified in the integrated work schedule.

The following work weeks were reviewed:

- Week of October 31 that included planned B and D RHR service water strainer basket repairs, a planned C service air compressor maintenance outage, emergent B reactor water recirculation system motor generator set control system troubleshooting, and emergent feedwater control system electronic component replacements;

- Week of November 7 that included planned maintenance on the A TBCLC pump, and emergent repairs on the HPCI steam admission valve, 23MOV-14;

- Week of November 14 that included an A RHR maintenance outage, feedwater control system transient response testing, and A and C EDG surveillance testing;

- Week of November 28 that included quarterly inspection and individual cell replacements in the B low pressure coolant injection inverter battery and planned maintenance on the B EDG; and

- Week of December 12 that included planned maintenance on the A emergency service water (ESW) system and the A EDG.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14 - 2 samples)

a. Inspection Scope

For the two non-routine events described below, the inspectors reviewed operator logs, plant computer data, and strip charts. The inspectors also interviewed operators and plant management to determine what occurred, how the operators responded, and if the response was in accordance with plant procedures and management expectations.

- On October 7, a short duration, invalid, low pressure emergency core cooling system (ECCS) actuation injected approximately 30,000 gallons of water into the reactor vessel and flooded the refuel cavity. The actuation was caused by an instrumentation and controls (I&C) technician returning a remote shutdown reactor vessel pressure indicator to service after calibration. The ECCS injection caused a large crud burst that elevated radiation levels on the bridges, and overflowed water from the refuel pool into the reactor building floor drains. The overflow resulted in 2000 square feet of high level, floor contamination on all elevations of the north side of the reactor building due to floor drain backups.

- On October 20, during refueling outage 16 operators drained approximately 120 inches of reactor water level without adequate reactor vessel RPV level indication. Operators were completing the RPV system leakage test as part of refueling outage closeout, and in order to depressurize the RPV following completion of leakage inspections, operators lowered reactor water level and pressure unaware that the refueling zone water level indicator was out of service.

b. Findings

Section 4OA3 discusses one finding associated with the October 20 inadvertent RPV level change.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

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

- CR-2004-05242 concerning 400 kW load oscillations experienced while at rated load on C EDG during monthly surveillance testing;
- CR-2004-5301 concerning an oil leak from the upper motor bearing for the A core spray pump;
- CR-2004-04893 concerning reactor vessel flange seal leakage;
- JAF-RPT-04-00467 concerning analysis of loose parts identified during in-vessel visual inspections; and
- CR-2004-05462 concerning calibration of the feedwater leading edge flow meter.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16 - 2 samples)

a. Inspection Scope

The inspectors completed the following two operator workaround inspection samples. Documents reviewed for this inspection are listed in the attachment.

- The inspectors evaluated individual and cumulative effects of identified main control board deficiencies on the functionality of the plant's mitigating systems. The equipment deficiencies were reviewed to determine the effect on the functional capability of the systems, or human reliability in responding to an initiating event; and to assess the potential effects on the operators' ability to implement abnormal or emergency procedures.

- The inspectors performed a detailed review of the Entergy-identified operator workarounds associated with operation of HPCI and reactor core isolation cooling (RCIC) for reactor water level control at less than normal operating pressure. During implementation of EOPs, at less than normal operating pressure, the allowable reactor water level operating band can be significantly reduced because HPCI and RCIC high reactor water level instrumentation is not density compensated. This causes additional challenges to plant operators when controlling reactor pressure and level using HPCI or RCIC and safety relief valves (SRVs).

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A - 2 samples)

a. Inspection Scope

The inspectors completed the following two permanent plant modification reviews:

- The inspectors reviewed modification documents and observed the testing of agastat relay replacements; specifically, the B core spray pump start interlock relay in accordance with modification No. JE-03-024. This modification replaced the existing agastat relays with electronic timers due to excessive drift and accuracy problems. The modification was completed under WR JF-0030597700. The post-modification testing, completed under WR JAF-04-28962, included performance of ST-9BB, "EDG B and D Full Load Test and ESW Pump Operability Test," TST-106, "Core Spray Pump Start Timer Testing," followed by satisfactory performance of ST-9C, "Emergency AC Power Load Sequencing Test and 4 KV Emergency Power System Voltage Relays Instrument Functional Test."

- The inspectors reviewed modification No. JAF-04-18673 and walked down the completed installation. This modification upgraded a plant access defense position for a plant vital area.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 6 samples)

a. Inspection Scope

The inspectors reviewed post maintenance test 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 design basis documentation; that test instrumentation had current calibrations and the range and accuracy for the application; and that 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. Documents reviewed during this inspection are listed in the attachment. The following six post maintenance test activities were reviewed.

- WR JF-003059770, involving installation of a new pump start interlock relay for the B core spray pump. The retest was performed using ST-9BB, "EDG B and D Full Load Test and ESW Pump Operability Test" and temporary surveillance test (TST)-106, "Core Spray Pump Start Timer Testing."
- WR JF-01099490, involving replacement of the C EDG woodward governor actuator. The retest was performed using ST-9BA, "EDG A and C Full Load Test and ESW Pump Operability Test," and ST-9C, "Emergency AC Power Load Sequencing Test and 4 KV Emergency Power System Voltage Relays Instrument Functional Test."
- WR JAF-04-39659, involving the repair of HPCI steam isolation valve 23MOV-14 seat leakage. The retest included stroke time testing, position indication verification, and seat leakage testing, in accordance with WR JAF-04-39856 and ST-4N, "HPCI Quick Start, Inservice, and Transient Monitoring Test."
- WR JAF-03-26816, involving installation of a new disc holder for C RHR pump discharge check valve 10RHR-42C. The retest included a forward flow test and leak check using OP-13B,"RHR - Containment Control," and a forward flow/reverse flow test using ST-2AL, "RHR Loop A Quarterly Operability Test (IST)."
- WR JAF-04-40183, involving replacing a spent fuel pool cooling return line vacuum breaker. The retest was performed in accordance with the work order, and consisted of measuring the opening force of the vacuum breaker to confirm its conformance with design requirements.
- WR JF-020239000, involving preventive maintenance (PM) replacement of rod control relays. The retest was performed using ST-20C, "Control Rod Operability For Fully Withdrawn Control Rods and HCU Cooling Water Supply Check Valve Reverse Flow Check (IST)."

b. Findings

Introduction. The inspectors identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for Entergy's failure to correct a condition adverse to quality involving HPCI turbine steam supply isolation valve 23MOV-14 seat leakage. In November 2004, this resulted in 53 hours of unplanned HPCI system unavailability due to emergent corrective maintenance to address degradation of the valve disc and seat.

Description. Normally closed valve 23MOV-14 opens to supply steam to the HPCI turbine. Leakage through the valve, if severe enough, can challenge HPCI turbine operability by degrading the rotor, gland seals, bearings, and lubricating oil. Leakage through 23MOV-14 has been a perennial problem. Since 1998, approximately 9 instances of steam leakage have been documented in 11 CRs. Elevated turbine thrust bearing temperatures were identified in August 1998. In October 2000, corrosion, shaft pitting, and flaking of carbon seal ring chrome plating were documented. In March 2001, the HPCI system was declared inoperable due to a large amount of water in the lubricating oil sump. As a result, in May 2001, the NRC issued NCV 00050333/2001003-01 for failure to implement adequate corrective actions to prevent repetitive challenges to the operability of HPCI in accordance with 10 CFR 50, Appendix B, Criterion XVI.

In 1998, Entergy recognized that the problem involved valve design and orientation in the system. Entergy's long term corrective action was to replace the valve with one of superior design and to reorient the valve. The valve replacement was removed from the scope of three refueling outages for various reasons including reclassifying the corrective action as an enhancement in December 2003 due to the belief that repairs in February 2003 had been successful, and the ability to work on the valve on-line. At that time, Entergy also determined that reorienting the valve would not be required. Since 1998, Entergy has either repaired the valve internals or repeatedly re-stroked the valve to re-position the valve disc and seat to control seat leakage such that leakage was lowered to an acceptable level. Most recently, in November 2004, Entergy incurred approximately 53 hours of HPCI unavailability to overhaul the valve due to increasing leakage past the seat.

Analysis. The performance deficiency was the failure to correct a long-standing condition adverse to quality involving steam leakage through HPCI steam supply isolation valve 23MOV-14, which required Entergy to take HPCI out of service from November 11 to November 13, 2004, to perform emergent corrective maintenance on 23MOV-14. This resulted in 53 hours of unplanned HPCI system unavailability. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements. The issue was of more than minor significance because it was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability of systems that respond to initiating events.

In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the Phase 1 screen required a Phase 2 analysis because the removal of the HPCI system from service to perform corrective maintenance represented an actual loss of the system’s safety function. The Region I senior reactor analyst (SRA) conducted a modified Phase 2 SDP analysis and made the following assumptions:

- Based upon the 53 hours of unavailability due to corrective maintenance, the exposure time for the condition was less than 3 days; and
- A revised mitigation credit was assigned to the high pressure injection (HPI) function based upon more recent industry data (SPAR). Industry data supports a revised HPI failure probability of approximately 4E-4. This value was rounded up to 1E-3 consistent with SDP usage rules.

Based on the modified Phase 2 worksheets, the SRA determined that this finding was of very low risk significance. The most dominant core damage sequences involve the loss of HPI and failure of operators to depressurize the reactor vessel. However, these sequences are mitigated by the availability of the RCIC system. Consistent with IMC 0609, Appendix A and Appendix H, “Containment Integrity SDP,” this finding did not require external events or large early release frequency related evaluations because the delta-CDF was less than 1E-7 per year.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires measures to be established to assure that conditions adverse to quality, such as malfunctions and deficiencies, are promptly identified and corrected. Contrary to the above, since May 2001 Entergy did not promptly correct a condition adverse to quality involving HPCI steam supply isolation valve 23MOV-14 seat leakage, which resulted in 53 hours of HPCI unplanned unavailability between November 13 and 15, 2004. Because the violation is of very low risk significance and Entergy entered the deficiency into its corrective action program (CAP) as CR-2005-00088, this finding is being treated as an NCV consistent with Section VI.A of the Enforcement Policy **(NCV 05000333/2004005-01, Inadequate corrective action for 23MOV-14 seat leakage)**.

1R20 Refueling and Outage Activities (71111.20 - 1 sample)

a. Inspection Scope

The inspectors observed and reviewed selected refueling outage activities to verify that operability requirements were met and that risk, industry experience, and previous site specific problems were considered. Documents reviewed during this inspection are listed in the attachment.

- During the course of the refueling outage, the inspectors observed selected reactor disassembly activities and walked down clearances to verify that tagouts were properly hung and that equipment was properly configured. Through plant tours, the inspectors verified that Entergy maintained and adequately protected electrical power supplies to safety-related equipment and that TS requirements were met.

- The inspectors periodically verified proper alignment and operation of the shutdown cooling and decay heat removal (DHR) systems. The verification also included reactor cavity and fuel pool makeup paths and water sources, and administrative control of drain down paths.

- The inspectors reviewed reactor analyst procedures RAP-7.1.04B, “Refueling Procedure,” and RAP-7.1.04C, “Neutron Instrument Monitoring During In-Core Fuel Handling” and the results of refueling platform interlock functional tests to ensure that the TS requirements for fuel movement were met. The inspectors also verified that fuel assemblies were loaded in the reactor core locations specified by the design and that containment requirements for refueling activities were met.

- The inspectors observed portions of the reactor startup following the outage, and verified through plant walkdowns, control room observations, and surveillance test reviews that the safety-related equipment specified for mode changes was operable, that containment integrity was set, and that reactor coolant boundary leakage was within TS limits.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 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 Entergy procedure requirements. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design basis documentation; that test instrumentation had current calibrations and the range and accuracy for the application; and that 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. Documents reviewed during this inspection are listed in the attachment. Six STs were reviewed:

- ST-9C, “Emergency AC Power Load Sequencing Test And 4 KV Emergency Power System Voltage Relays Instrument Functional Test;”
- ST-4N, “HPCI Quick Start, Inservice, and Transient Monitoring Test (IST);”
- ST-2F. “LPCI And LPCI Power Supply Simulated Automatic Actuation Test;”
- ST-39H, “RPV System Leakage Test And CRD Class 2 Piping Inservice Test (ISI);”

- ST-9BA, “EDG A And C Full Load Test And ESW Pump Operability Test;” and
- ST-24J, “RCIC Flow Rate And Inservice Test (IST).”

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 2 samples)

a. Inspection Scope

The inspectors reviewed the temporary modifications (TMs) listed below. The inspectors assessed the adequacy of the 10 CFR 50.59 evaluations for these temporary modifications; that the installation was consistent with the modification documentation; that the drawings and procedures were updated as applicable; and that the post-installation testing was adequate. The inspectors also reviewed the results of ST-1X, “Protective Tags and Temporary Alterations Audit.”

- TM 04-043 involving level control of second stage reheater drain tank 31TK-4B
- Temporary shielding package (TSP) No. 97-024 - temporary shielding in the RCIC room

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04 - 1 sample)

a. Inspection Scope

On October 20, 2004, the inspectors performed an in-office inspection that reviewed recent changes to the emergency plan and implementing procedures. These changes were made by Entergy in September 2004. A thorough review was performed for documents related to the risk significant planning standards (RSPS) and a general review was completed for non-RSPS documents. The review verified that the changes satisfied the standards of 10 CFR 50.54(q), 10 CFR 50.47(b), the requirements of 10 CFR 50 Appendix E, and the intent of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants” and that the changes did not decrease the effectiveness of the plan. These changes are subject to future NRC inspections to ensure that as a result of changes the emergency plan continues to meet NRC regulations. Documents reviewed for this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 10 samples)

a. Inspection Scope

During October 4 - 8, 2004, the inspector conducted the following activities that were selected based on their exposure significance, to verify that Entergy properly implemented physical, engineering, and administrative controls for access to high radiation areas (HRAs), and other radiologically controlled areas, and that workers adhered to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10CFR20, site TSs, and Entergy's procedures.

On October 4 - 7, 2004, the inspector observed the following work activities during the Fall 2004 refueling outage:

- Control rod drive (CRD) replacements
- "G" SRV replacement
- 53A recirculation discharge isolation valve repair
- "D" inboard main steam isolation valve actuator repair
- In-vessel visual inspection of core spray spargers
- CRD rebuild room equipment removal
- Refuel floor general work activities
- Turbine operating deck general work activities
- Drywell general work activities

With respect to these work activities, the following inspection activities were conducted:

(1) Based on scheduled work activities the above jobs were selected for review based on their estimated exposure significance.

(2) Walkdowns of the work areas were performed with a survey instrument to determine whether prescribed radiation work permit (RWP), procedure, and engineering controls were in place, to confirm the accuracy of Entergy's surveys and postings, and whether air samplers were properly located.

(3) The inspector reviewed the applicable RWPs used to access these work activities, confirmed the implementation of the specified radiation work controls, and evaluated the applicability of the specific electronic dosimetry alarm setpoints.

(4) The inspector observed radiological briefings to the workers and conduct of these work activities with respect to established radiological controls.

(5) The inspector verified the adequacy of the required surveys to include system breach radiation, contamination, and airborne surveys and the oversight provided by radiation protection (RP) staff (to include remote monitoring and control technologies).

(6) Observation of radiation worker performance was performed to determine if they adhered to the RWP and as low as is reasonably achievable (ALARA) work requirements and demonstrated the use of established low dose waiting areas and proper contamination procedural controls.

(7) Observation of RP technician performance with respect to the applicable radiological work requirements was conducted consistent with their job responsibilities/qualifications and with respect to the radiological hazards associated with the work activities.

The following inspection activities were conducted associated with the CAP:

(8) Six CRs were reviewed (see Section 4OA2.3), dated between September and October 2004, that were related to the access control program. The inspector reviewed applicable documents and interviewed the staff to ensure that Entergy conducted effective and timely followup activities commensurate with the safety significance.

(9) Of the CRs reviewed in (8) above, one was the result of radiation worker error and was reviewed for appropriate and timely corrective actions.

(10) Of the CRs reviewed in (8) above, there were no incidents reported associated with RP technician performance errors.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 7 samples)

.1 ALARA Planning for the Five Highest Exposure Outage Tasks (71121.02 - 2 samples)

a. Inspection Scope

The inspector reviewed Entergy's ALARA performance in accordance with 10 CFR 20.1101(b). Areas reviewed included implementation of the exposure reduction requirements based on ALARA planning for the five highest exposure outage tasks as listed below.

- CRD replacement
- In-service inspection
- Motor-operated valve (MOV) testing
- Reactor disassembly/reassembly
- SRV replacement

With respect to these five outage work activities, the following ALARA inspection activities were conducted:

(1) The inspector evaluated the use of engineering controls to achieve dose reductions. These included the use of remote monitoring technologies and drywell shielding dose benefit evaluations.

(2) The inspector observed radiation worker and RP technician performance during work activities to determine if ALARA principles were practiced with respect to the applicable ALARA review requirements and radiological briefings for low dose area use.

b. Findings

No findings of significance were identified.

.2 Individual and Collective Radiation Exposure ALARA Review (71121.02 - 5 samples)

a. Inspection Scope

The inspector performed the following activities to verify that Entergy was properly maintaining individual and collective radiation exposures ALARA. Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and Entergy's procedures. Other documents reviewed for this inspection are listed in the attachment. The five samples are as follows:

- The plant collective exposure history trend and current 3-year rolling average collective exposure data were reviewed. Based on the 2001-2003 exposure performance of 115 person-rem, FitzPatrick ranks in the second quartile of boiling water reactors.

- The following seven highest exposure work activities for the Fall 2004 refueling outage were selected for review: reactor disassembly/reassembly; CRD replacement; MOV PM; inservice inspection; refueling activities; SRV maintenance, and snubber inspection and replacement.

- The ALARA reviews for the outage work activities listed above were evaluated with respect to initial exposure estimates and any subsequent credits due to emergent work or increased dose rates, and then compared to the actual exposure results obtained. Any causes for exposure overruns were identified and quantified when appropriate.

- With respect to the ALARA reviews that were evaluated, the methods for adjusting exposure estimates were reviewed relative to changes in work scope or increased dose rates in order to preserve the original work activity exposure performance measurement of the work activities.

- The inspector reviewed the site specific source term trend (an increase from 97.5 mrem/hr average on the recirculation piping in 2002 up to 155 mrem/hr as measured during the Fall 2004 refueling outage).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification

a. Inspection Scope (71151 - 5 samples)

The inspectors reviewed performance indicator (PI) data for the below listed cornerstones and used NEI 99-02, “Regulatory Assessment Performance Indicator Guidance,” to verify individual PI accuracy and completeness.

Initiating Events Cornerstone

- Unplanned Scrams per 7000 Critical Hours
- Scrams with a Loss of Normal Heat Removal
- Unplanned Transients per 7000 Critical Hours

The inspectors reviewed a selection of licensee event reports, operator log entries, the monthly operating reports, and PI data sheets to determine whether Entergy adequately identified the number of scrams and unplanned power changes greater than 20 percent that occurred from July 2002 to June 2003. This number was compared to the number reported for the PI during the current quarter. The inspectors also verified the accuracy of the number of critical hours reported and Entergy’s basis for crediting normal heat removal capability for each of the reported reactor scrams. In addition, the inspectors also interviewed Entergy personnel associated with the PI data collection, evaluation, and distribution.

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

The inspector reviewed CRs, and radiological controlled area dosimeter exit logs from October 2003 to September 2004. These records were reviewed for occurrences involving locked HRAs, very high radiation areas, and unplanned exposures against the criteria specified in NEI 99-02 to verify that all occurrences that met the NEI criteria were identified and reported as PIs.

Public Radiation Safety Cornerstone

- Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent

The inspector reviewed a listing of relevant effluent release reports issued between October 2003 and September 2004 for issues related to this PI. The indicator measures radiological effluent release occurrences that exceed 1.5 mrem/qtr whole body or 5.0 mrem/qtr organ dose for liquid effluents; 5mrads/qtr gamma air dose, 10 mrad/qtr beta air dose, and 7.5 mrads/qtr for organ dose for gaseous effluents.

The inspector reviewed monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases; quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and dose assessment procedures to verify Entergy met the requirements of the indicator.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Routine Problem Identification and Resolution Program Review

a. Inspection Scope (71152)

As specified by Inspection Procedure 71152, “Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all items entered into Entergy’s CAP. The review was accomplished by accessing Entergy’s computerized database for CRs and attending CR screening meetings.

In accordance with the baseline inspection modules, the inspectors selected 69 CAP items across the Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Planning, and Occupational and Public Radiation Safety cornerstones for additional follow-up and review. The inspectors assessed Entergy’s threshold for problem identification, the adequacy of the cause analyses, extent of condition review, operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are noted in the attachment.

The inspectors also performed a semiannual review of Entergy's CAP to access trends that may indicate the existence of more significant safety issues. This semiannual review included a review of Entergy's system health reports, maintenance backlogs, engineering requests, self assessment reports, and the CR data base.

b. Findings

Introduction. The inspectors identified a Green non-cited violation of TS LCO 3.8.1, “Electrical Power Systems - AC Sources - Operating,” for Entergy’s failure to comply with the LCO required actions for one offsite power circuit inoperable within the specified time requirements.

Description. The FitzPatrick 115 kV emergency offsite power system consists of two independent offsite circuits, one supplied by the Lighthouse Hill hydroelectric generating station (Line 3) and the other by the Nine Mile Point (NMP) Unit 1 switchyard (Line 4). Each line normally supplies power to two reserve station service transformers (RSSTs) through a normally-closed 115 kV bus disconnect. The system was designed such that either line alone will supply both RSSTs that supply both safeguards buses under normal, shutdown, and design basis accident (LOCA) loads. On November 13, 2003, NMP isolated Line 4 due to arcing in their switchyard. This deenergized Line 4 to the FitzPatrick switchyard and rendered one of FitzPatrick's qualified offsite power circuits inoperable. TS LCO 3.8.1 requires, in part, that two qualified offsite circuits between the offsite transmission network and the plant class 1E AC electrical power distribution system are operable during power operation, startup and hot shutdown. Because Entergy misinterpreted the TS requirements related to the operability of offsite circuits, they did not declare the applicable offsite power source inoperable in accordance with TS LCO 3.8.1, or take the appropriate required action. TS LCO 3.8.1 required action A, in part, states that when one offsite circuit is inoperable it must be restored to operable status in seven days. If required action A is not completed in the required completion time, required action F states that the plant must be placed in hot shutdown in 12 hours and cold shutdown in 36 hours. Line 4 was restored on November 22, resulting in an outage time of approximately 9 days and 8 hours. This exceeded the allowed outage time of TS LCO 3.8.1.

Analysis. The performance deficiency was the failure to correctly interpret TS LCO 3.8.1 requirements concerning the operability of offsite power circuits. This resulted in exceeding the TS LCO 3.8.1 allowed outage time. Traditional enforcement does not apply because the issue did not have an actual safety consequence or potential for impacting the NRC’s regulatory function, and it was not the result of any willful violation of NRC requirements. The issue was of more than minor significance because it was associated with the Initiating Events cornerstone attribute of configuration control and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

In accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” the finding was determined to be of very low risk significance (Green) because as a transient initiator it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available.

Enforcement. TS LCO 3.8.1. requires, in part, that two qualified offsite power circuits be operable. TS LCO 3.8.1 required action A states that an inoperable offsite power circuit must be restored to operable status within seven days. If the required action and associated completion time of condition A are not met, TS LCO 3.8.1. required action F states that the plant must be placed in hot shutdown in 12 hours and cold shutdown in 36 hours. Contrary to the above, from November 13 to 22, 2003, one qualified offsite power circuit was inoperable for greater than 7 days 36 hours and the plant was not placed in the cold shutdown condition. Because the violation is of very low risk significance and Entergy entered the deficiency into its CAP as CR-2005-00089, this finding is being treated as a NCV consistent with Section VI.A of the Enforcement Policy **(NCV 05000333/2004005-02, Failure to comply with TS 3.8.1 required actions for one offsite circuit inoperable)**.

.2 Annual Sample Review

a. Inspection Scope (71152 - 2 samples)

The inspectors performed detailed reviews of Entergy's evaluations and corrective actions for CR-2003-04441 and CR-2003-04739. The CRs were reviewed to ensure that the correct level of investigation was assigned, that appropriate causal evaluations were performed, and that required corrective actions were specified. The inspectors evaluated the CRs against the requirements of 10 CFR 50, Appendix B, and procedure ENN-LI-102, "Corrective Action Process."

b. Findings and Observations

There were no findings identified. CR-2003-04441 involved the apparent ineffectiveness of preventive and corrective maintenance activities to prevent safety-related check valve failures. Entergy determined that existing PM frequencies were inadequate. To correct the problem, Entergy evaluated approximately 160 check valves and revised their PM intervals. CR-2003-04739 involved industry operating experience concerning CRD scram time degradation due to premature aging of Buna-N™ scram pilot solenoid valve exhaust diaphragms. Although the condition did not result in slow scram times at FitzPatrick, Entergy ultimately replaced its valves with a different design that eliminated suspect elastomers.

.3 Identification and Resolution of Problems - Occupational Radiation Safety

a. Inspection Scope (71121.01)

The inspector reviewed the following corrective action CRs associated with the RP program: CR-JAF-2004-3634, CR-JAF-2004-3843, CTR-JAF-2004-3912, CR-JAF-2004-3917, CR-JAF-2004-4152 and CR-JAF-2004-4220. The inspector verified that problems identified by these CRs were properly characterized in Entergy's event reporting system, and that applicable causes and corrective actions were identified commensurate with the safety significance of the radiological occurrences.

b. Findings

No findings of significance were identified.

.4 Cross-reference to Problem Identification and Resolution Findings

Section 1R19 describes a finding involving failure to correct a condition adverse to quality involving HPCI turbine steam supply isolation valve 23MOV-14 seat leakage.

4OA3 Event Follow-up (71153 - 1 sample)

a. Inspection Scope

On October 20, during refueling outage 16, operators drained approximately 120 inches of reactor water level without adequate RPV level indication. Before the start of the drain down, level was at 485 inches above top of active fuel (TAF); and after the drain down was terminated, level was at 365 inches above TAF. To assess operator response to the event, the inspectors interviewed control room operators, reviewed operating logs and Entergy's post-event evaluation. The inspectors also reviewed Entergy's immediate corrective actions following the event, and the final results of its root cause analysis.

b. Findings

Introduction. The inspectors identified a self-revealing violation of 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for Entergy's failure to provide a procedure appropriate to the circumstances. Specifically, surveillance procedure ST-39H, "RPV System Leakage Test and CRD Class-2 Piping Inservice Test," did not include adequate precautions for reactor vessel level control. This resulted in operators draining 120 inches from the reactor vessel with the only on-scale level indication out of service for testing.

Description. On October 19 and 20, operators completed RPV system leakage inspections, RPV instrumentation line excess flow check valve testing, and control rod scram time testing in accordance with ST-39H, ISP-1, "Instrument Line Excess Flow Check Valve Operability Test,"and RAP-7.4.1, "Scram Time Testing."

At 5:50 p.m. on October 19, in accordance with ST-39H, operators raised reactor water level to greater than 410 inches (above TAF) and reactor pressure to between 1050 and 1080 psig using the CRD system. At 9:00 p.m., after initial conditions were established for ISP-1, I&C technicians requested and received permission from operations to perform excess flow check valve testing. To perform this testing the procedure directed removing instrument racks from service one at a time. In accordance with the procedure, once permission was given to perform the test, the I&C technicians removed the racks from service at their discretion. As a result, during testing the operations department shift was not always aware what instruments were in or out of service.

At 1:59 a.m. on October 20, with pressure and temperature in the required band, operators commenced leak inspections in accordance with ST-39H. At 7:20 a.m., in accordance ISP-1, an I&C technician removed the refueling zone water level transmitter from service for excess flow check valve testing. At 11:17 a.m., following completion of RPV leak inspections and excess flow check valve testing, but before completion of the system restoration from check valve testing, operators reduced reactor water level and pressure in accordance with ST-39H section 8.6. ST-39H section 8.3 contained a note that stated "...when excess flow check valve testing is complete, RPV depressurization may be conducted per subsection 8.6 at the discretion of the test coordinator." In previous outages depressurization was not commenced until full completion of ISP-1 including system restoration. In this instance operators and the test coordinator believed that the intent of the ST-39H section 8.3 note was fulfilled based on satisfactory excess flow check valve testing results and commenced depressurization. The procedure did not provide adequate direction and precautions regarding the availability and use of level instrumentation during depressurization. As a result, because the operators had not maintained control of the status of available level indication during the testing, they were not aware that the refueling zone water level indicator, the only on-scale level indication at this water level (>410 inches), was still out of service. After approximately 30 minutes of draining through the main steam line (MSL) drains to the main condenser, and through reactor water cleanup (RWCU) system to radwaste, operators noted a 10-inch increase in condenser hotwell level without a concurrent change in indicated reactor vessel level. Due to this discrepancy, at 11:45 a.m. operators terminated the drain down. They confirmed that an I&C technician had removed the refueling zone level indication from service for excess flow check valve testing and had not yet returned it to service.

Before the start of the drain down, level was at 485 inches above TAF, and after the drain down was terminated, level was at 365 inches above TAF. Had the drain down continued, drainage from the MSL drains would have stopped at 285 inches above TAF, the level of the MSL penetrations. The B train of wide range and narrow range level was available, and would have alarmed as level continued to decrease. Drainage from the RWCU system would have terminated when RWCU was automatically isolated by the primary containment isolation system at 177 inches above TAF. One loop of RHR and both loops of core spray were available for automatic injection. The redundant train of RHR was lined up for shutdown cooling and would not have automatically realigned to its low pressure coolant injection lineup.

<u>Analysis</u>. The performance deficiency associated with this finding was the failure to provide adequate procedures to control reactor vessel water level during the performance of RPV pressure boundary leak inspections in accordance with 10 CFR 50 Appendix B Criterion V. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements. The finding was of more than minor significance because it was associated with the procedure quality and configuration control attributes of the Initiating Events cornerstone and adversely affected the objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions while shutdown. To assess the risk

significance of this event, the inspector used IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” dated May 25, 2004. The initial review of this issue using the Phase 1 screen criteria in Table 1, “Losses of Control,” and Appendix G, Attachment 1, identified that the inadvertent loss of reactor coolant system (RCS) inventory exceeded the two feet threshold. The actual RCS inventory loss was 120 inches of level change without an operable channel of vessel level indication. The Region I SRA completed a Phase 2 analysis using Worksheet 1, “Loss of Inventory in POS 1.” Based upon the availability of multiple injection/cooling systems, automatic level alarms and isolation capability, and alternate feed and bleed capability, the Phase 2 analysis provided for nearly maximum mitigation system credit. Accordingly, this self-revealing performance deficiency was determined to be of very low (approximately 1E-9) risk significance. The results of the Phase 2 analysis were independently reviewed by the NRR shutdown risk specialist who concurred in the risk assessment methodology and results. The dominant core damage sequences for this condition involve the failure to isolate the draindown pathway and an inadequate vent path for pressure control during subsequent reflood of the reactor vessel.

The finding was associated with the cross-cutting area of human performance because in addition to the inadequate procedure it involved operators failure to maintain adequate control equipment status during operations in accordance with AP-19.01, “Conduct of Operations.”

Enforcement. 10 CFR 50 Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to this requirement, surveillance procedure ST-39H, which governed the performance of RPV pressure boundary leak inspections, was not appropriate to the circumstances in that it did not provide adequate precautions regarding the availability of level instrumentation prior to system depressurization. On October 20, 2004, this resulted in the operating crew lowering reactor vessel water level over two feet without adequate indication. Because this violation was of very low risk significance and has been entered into Entergy’s CAP as CR-2004-04791, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy **(NCV 05000333/2004005-03, Inadequate procedure for RPV leak testing resulted in an inadvertent two foot level change)**.

4OA4 Cross Cutting Aspects of Findings

Section 4OA3 describes a finding associated with the cross-cutting aspect of human performance. On October 20, during refueling outage 16, operators drained approximately 120 inches of reactor water level without adequate RPV level indication. A contributing cause to the event was operators’ failure to maintain adequate control of equipment status during operations in accordance with AP-19.01, “Conduct of Operations.”

4OA6 Meetings, Including Exit

The inspectors presented the inspection results to Mr. Ted Sullivan and other members of Entergy management on January 10, 2005. Entergy acknowledged that some of the material reviewed by the inspectors during this period was proprietary, but the content of this report contains no proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

N. Avrakatos, Emergency Preparedness Coordinator
P. Berry, Manager, Training
S. Bono, Director, Nuclear Safety
M. Durr, Manager, System Engineering
J. Gerety, Manager, Programs and Components Engineering
A. Halliday, Manager, Regulatory Compliance
D. Johnson, Manager, Operations
J. LaPlante, Manager, Security
O. Limpias, Director, Engineering
K. Mulligan, General Manager, Plant Operations
J. Pechacek, Manager, Design Engineering
K. Pushee, Manager, Radiation Protection
W. Rheaume, Manager, CA&A
B. Sholler, Manager, Plant Maintenance
T. Sullivan, Vice President, Operations
D. Wallace, Quality Assurance Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000333/2004005-01	NCV	Inadequate corrective action for 23MOV-14 seat leakage (Section 1R19)
05000333/2004005-02	NCV	Failure to comply with TS 3.8.1 required actions for one offsite circuit inoperable (Section 4OA2.1)
05000333/2004005-03	NCV	Inadequate procedure for RPV leak testing resulted in inadvertent reactor vessel level decrease (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

NYPA letter, Ralph E. Beedle to USNRC, dated February 18, 1992, “Response to Request for Additional Information Regarding Certain Diagnostic Team Findings at the James A. FitzPatrick Nuclear Power Plant”

Technical Specification Interpretation (TSI) No. 28, Minimum Number of Operable Intake De-icing Heaters, January 27, 1993
JAF-CALC-CWS-00384, “Determine Operability of Intake De-icing Heater Elements”
JAF-CALC-CWS-02604, “Intake De-icing Heater Minimum Circuit Ground Resistance and Minimum Total Required Feeder Amps”
JAF-SE-93-036, “Verification of Intake De-icing Heaters QA Category II/III”
JAF-RAS-92-003, “Circulating Water Intake De-icing Heaters Reasonable Assurance of Safety”
SWEC Calculation No. 36-8, “Ice Formation on Intake Bar Racks”
JAF-RPT-3971, “A Study on the Potential for Blockage by Frazil and Grease Ice”
MST-071.17, “Intake Deicing Heaters Rated Power Surveillance Test”
MST-071.06, “Intake Deicing Heaters Insulation Resistance Surveillance Test”
NYPA Memorandum, H. Kodali to T. Savory, dated December 6, 1996, “Bases for Acceptance Criteria in MST's 71.6 and 71.7 and ST-8G which Monitor the Technical Specification Requirements 4.11.E.1, 4.11.E.2, and 4.11.E.3 for Intake De-icing Heaters”
ST-8G, “Intake Deicing Heaters Feeder Ammeters Test,” Revision 13
OP-51A, “Reactor Building Ventilation and Cooling System”;
OP-52, “Turbine Building Ventilation”;
OP-60, “Diesel Generator Room Ventilation”;
DBD-066, “Design Basis Document (DBD) for Reactor Building Heating, Ventilation and Air Conditioning (HVAC) Systems”; and
DBD-067, “DBD for Turbine Building HVAC Systems”

<u>Section 1R04: Equipment Alignment</u>

DBD-046, “Design basis document (DBD) for the Normal, Emergency and Residual Heat Removal Service Water Systems”
OP-13, “Residual Heat Removal System”
DBD-010, “DBD for the Residual Heat Removal System”
DBD-014, “DBD for the Core Spray System”
OP-14, “Core Spray System”
OP-30B, “Decay Heat Removal System”
OP-41, “Turbine Building Closed Loop Cooling”
DBD-067, “DBD for the Turbine Building Ventilation and Cooling System”

<u>Section 1RO7: Heat Sink Performance</u>

JAF-CALC-CRC-04276, “Maximum Allowable Tube Plugging Limit For Control Room & Relay Room Air Handling Units”;
JAF-CALC-04-00478, “Mathcad Worksheet Validation For ST-18C Air Handling Unit Calculation”; and
DBD-070, “DBD for Control Room and Relay Room Ventilation and Cooling Systems”

<u>Section 1RO8: In-service Inspection</u>

<u>NDT Examination Reports</u>

VC-BH-2B/E, RPV Bottom Head, UT, Work Order JF-011199400

04UT073, 24-10-131 RHR Discharge Line, UT, Work Order JF-030711300
04S082, Main Steam ‘C’ Pipe Support, MT, Work Order JF-030633000
04S071, RHR Pressure Boundary Weld 20-10-636A, MT, Work Order JF-030633300

In-Vessel Remote Visual Examination

VT-1, Steam Dryer Structural Welds, Vibration Block Welds

Repair-Replacement

H10-42A, 16" RHR Pipe Support Weld, Work Order JF-020209700

Flaw Evaluation

CR 2004-04366, ”Steam Dryer Vibration Block Attachment Welds”
CR 2004-04225, “Steam Dryer Drain Channel #8 Indication”
JAF-CALC-04-00516, “Evaluation of Steam Dryer Vibration Block No.1”

Miscellaneous

AP-19.06, “Piping Support and Pressure Retaining Component In-service Inspection”
JAF-RPT-MISC-03409, “JAF Risk-informed ISI Program”
NDE, “Personnel Certifications for UT - Level II and III”
GE SIL No. 644, “Supplement 1, BWR Steam Dryer Integrity,” September 2003
Drawing ISI-FM-20A, “RHR System 10"
Drawing MSK-3038, “ISI-Scram Tank Weld Identification”

Procedures

ENN-NDE-9.31, “Magnetic Particle Examination”
ENN-NDE-10.01, “VT-1 Examination”
ENN-NDE-9.04, “Ultrasonic Examination of Ferritic Piping Welds”

Section 1R12: Maintenance Rule Implementation

ENN-DC-171, “Maintenance Rule Monitoring”
JAF-RPT-CAD-02312, “Maintenance Basis Document for the Primary Containment Atmosphere Control and Dilution System”
OP-37, “Containment Atmosphere Dilution System”
System Health Report - Primary Containment Atmosphere Control and Dilution System, 1st Quarter 2004
DBD-027, “DBD for the Air Treatment Systems”
JENG-03-006, “PCAD Maintenance Rule (a)(1) Action Plan”
JAF-RPT-FPC-02288, “Maintenance Rule Basis Document for Fuel Pool Cooling”
TOP-282, “Draining of the Fuel Pool Cooling System”
02268-—08-14, “Calculation of Maximum Differential Opening Pressure Acceptable For Siphon Breaker Valves 19VB-1A and 19VB-1B For Surveillance Tests”

Section 1R15: Operability Evaluations

Woodward Governor Company Manual 37708J, “EG-B10C Governor/Actuator”
Alarm Response Procedure (ARP) 09-4-2-23, “Reactor Vessel Flange Seal Leakage”
JMD-APL-04-002, “Core Spray Motor (14P-1A) Oil Leak Action Plan”
JAF-RPT-FWC-02479, “Design Engineering Evaluation and Summary Of LEFM Data Collection and Trending (1996)”
JAF-CALC-FWC-02215, “Total Mass Feedwater Flow Uncertainty Calculation Using Delta-P and LEFM Instrumentation”

Section 1R16: Operator Workarounds

JAF-RPT-04-00398, “HPCI and RCIC Level Instrumentation Density Compensation Position Paper”
ER JF-03-01858, “HPCI and RCIC needs density compensation”

Section 1R17: Permanent Plant Modifications

AP-16.10, “Engineering Change Process”
AP-16.13, “Equivalent Change Process”
AP-16.11, “Design Change Process”
DBD-014, “DBD for Core Spray System”
DBD-093, “DBD for Emergency Diesel Generator System”

Section 1R19: Post Maintenance Testing

AP-05.07, ”Maintenance Testing and Post-Work Testing (ISI)”
DBD-023, “DBD for High Pressure Coolant Injection System”
JAF-RPT-MULTI-03155, “Evaluation of Thrust Requirements Relative to Leakage Criteria for GL 89-10 Anchor/Darling Double Disc Gate Valve”
DBD-093, “DBD for the Emergency Diesel Generator System”
DBD-010, “DBD for Residual Heat Removal System”
Woodward Governor Company Manual 37708J, “EG-B10C Governor/Actuator”

Section 1R20: Refueling and Outage Activities

ST-39Q, “Drywell Inspection”
RAP-7.4.9, “Shutdown Margin Check”
RAP-7.3.3, “Core Thermal Power Evaluation”
RAP-7.3.5, “Core Power Symmetry Analysis”
RAP-7.4.1, “Control Rod Scram Time Evaluation”
RAP-7.3.38, “LEFM Operation and Feedwater Correction Factor Calculation”
RAP-7.4.3, “LPRM Calibration”
RAP-7.3.7, “Core Flow Evaluation and Indication Calculation”
Post Transient Evaluation No. 04-002, October 7, 2004, unplanned ECCS injection
Engineering Request (ER) JF-03-0155, Cycle 17 Reload

AOP-30, “Loss of Shutdown Cooling”
OP-65B, “Shutdown Operation”
AP-05.06, “Foreign Material Exclusion (FME)”
Entergy Memorandum, G. Rorke to W. Drews, dated July 22, 2004, “JAF Rotated SRM Quadrant Definition”
RAP-7.1.04C, “Neutron Instrumentation Monitoring During In-Core Fuel Handling”
NRC letter, G. Vissing to M. Kansler, dated September 12, 2002, “James A. FitzPatrick Nuclear Power Plant - Amendment RE: Technical Specification Change to the Requirements for High Irradiated Fuel Assemblies (TAC No. MB5328)”
RAP-7.1.03G, “Control Blade Replacement (with Fuel in Core)”
MP-004.03, “CRD Removal and Replacement (ISI)”
AP-10.09, “Outage Risk Assessment”
NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” dated December 1991
ST-39H, “RPV System Leakage Test and CRD Class 2 Piping Inservice Test (ISI)”
ISP-1. “Instrument Line Excess Flow Check Valve Operability Test (IST)”

Section 1R22: Surveillance Testing

DBD-010, “DBD for Residual Heat Removal System”

Section 1R23: Temporary Plant Modifications

RP-ALARA-01.04, “Shielding”
JAF-CALC-MISC-02900, “Load capacity for standard fasteners and hardware used in lead shielding evaluations”
DESO-19, “Procedure for the Evaluation of Radiation Shielding Installations”

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Section 1, “Definitions/Acronyms”
Section 9, “Recovery”
Appendix K, “Evacuation Travel Time Estimates and Population Distribution for the JAF/Nine Mile Point Emergency Planning Zone”
EAP-2, “Personnel Injury”
EAP-4.1, “Release Rate Determination”
SAP-3, “Emergency Communications Testing”
SAP-7, “Monthly Surveillance Procedure for On-Call Employees”

Section 2OS2: ALARA Planning and Controls

ALARA Review Nos.: 04-30, 04-37, 04-29, 04-28, 04-36, 04-38, 04-32
Gaseous and Liquid release dose summary reports for 4th quarter 2003 through 3rd quarter 2004

Section 4OA2: Identification and Resolution of Problems

Condition Reports

2002-05380	2004-03717	2004-04940	2004-05267
2003-01003	2004-03755	2004-04981	2004-05271
2003-01068	2004-03826	2004-04982	2004-05301
2003-02846	2004-03839	2004-05025	2004-05312
2004-00685	2004-03905	2004-05033	2004-05323
2004-01123	2004-04269	2004-05034	2004-05391
2004-01666	2004-04360	2004-05063	2004-05412
2004-02190	2004-04448	2004-05077	2004-05462
2004-02643	2004-04461	2004-05080	2004-05472
2004-02651	2004-04506	2004-05139	2004-05474
2004-03003	2004-04507	2004-05152	2004-05479
2004-03201	2004-04644	2004-05215	2004-05480
2004-03245	2004-04657	2004-05219	2004-05516
2004-03328	2004-04752	2004-05239	2004-05568
2004-03556	2004-04786	2004-05242	2004-05572
2004-03596	2004-04791	2004-05243	2004–01415
2004-03647	2004-04893	2004-05252	

LIST OF ACRONYMS

AHU	air handling unit
ALARA	as low as reasonably achievable
AP	administrative procedure
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CDF	core damage frequency
CR	condition report
CRD	control rod drive
DBD	design basis document
DHR	decay heat removal
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOP	emergency operating procedure
ESW	emergency service water
HPCI	high pressure coolant injection
HPI	high pressure injection
HRA	high radiation area
HVAC	heating, ventilation and air conditioning
I&C	instrumentation and controls
IMC	Inspection Manual Chapter
ISI	in-service inspection
ISP	instrument surveillance procedure

IST	in-service testing
kV	kilovolt
LCO	limiting condition for operation
LOCA	loss of coolant accident
MOV	motor-operated valve
MR	Maintenance Rule
MSL	main steam line
NCV	non-cited violation
NDE	non-destructive examination
NEI	Nuclear Energy Institute
NMP	Nine Mile Point
NRC	Nuclear Regulatory Commission
OP	operating procedure
PI	performance indicator
PI&R	problem identification and resolution
PM	preventive maintenance
RAP	reactor analyst procedure
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RHR	residual heat removal
RP	radiation protection
RPV	reactor pressure vessel
RSPS	risk significant planning standard
RSST	reserve station service transformer
RWCU	reactor water cleanup
RWP	radiation work permit
SDP	significance determination process
SRA	Senior Reactor Analyst
SRV	safety relief valve
SSC	structure, system, and component
ST	surveillance test procedure
TAF	top of active fuel
TBCLC	turbine building closed loop cooling
TS	technical specification
TSP	temporary shielding package
TST	temporary surveillance test procedure
TM	temporary modification
UFSAR	Updated Final Safety Evaluation Report
UT	ultrasonic testing
WR	work request