January 27, 2004

Mr. Theodore Sullivan
Vice President - Operations
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 05000333/2003010

Dear Mr. Sullivan:

On December 31, 2003, the United States Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant. The enclosed integrated inspection report documents the inspection findings which were discussed on January 26, 2004, with Mr. Kevin Mulligan and other members of your staff.

The inspections 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 one self-revealing and one NRC-identified finding, both of very low safety significance (Green). One finding was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because the finding was entered into your corrective actions program, the NRC is treating the finding as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, one violation of very low safety significance identified by Entergy is listed in Section 4OA7 of this report. If you deny 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, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement; and the NRC Resident Inspector at the FitzPatrick.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction (TI) 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial power nuclear power plants during calendar year 2002, and the remaining inspection activities for

FitzPatrick are scheduled for completion in calendar year 2003. The NRC will continue to monitor overall safeguards and security controls at FitzPatrick.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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/

Glenn W. Meyer, Chief
Projects Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000333/2003010
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.:	05000333/2003010
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	September 28, 2003 - December 31, 2003
Inspectors:	L. M. Cline, Senior Resident Inspector D. A. Dempsey, Resident Inspector A. J. Blamey, Senior Operations Engineer T. F. Burns, Reactor Inspector P. D. Kaufman, Senior Reactor Inspector T. A. Moslak, Health Physicist
Approved by:	Glenn W. Meyer, Chief Reactor Projects Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000333/2003010; 09/28/2003 - 12/31/2003; James A. FitzPatrick Nuclear Power Plant. Maintenance Implementation, Maintenance Risk Assessments.

The report covered a thirteen-week period of inspection by resident inspectors, a senior health physicist, and regional specialist inspectors. One Green non-cited violation (NCV) and one Green finding were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using 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. Inspector Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. Corrective actions for a 1999 reactor water recirculation (RWR) pump trip were inadequate and resulted in another RWR pump trip and unplanned power reduction on September 25, 2003. This represented a self-revealing finding.

  The finding is considered more than minor, because it is associated with the equipment performance attribute and resulted in an unplanned plant transient that affected the reactor safety initiating events cornerstone objective of limiting the likelihood of events that upset plant stability. The finding is of very low safety significance, because it did not contribute to the likelihood of a primary or secondary system loss of coolant accident initiator, did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, and did not increase the likelihood of a fire or internal/external flood. (Section 1R13)

Cornerstone: Barrier Integrity

- Green. Entergy revised an abnormal operating procedure (OP) such that isolation of the control room envelope following a loss of coolant accident (LOCA) would be initiated later than analyzed in the design basis control room habitability calculation described in the UFSAR. The inspectors identified a non-cited violation of 10CFR 50, Appendix B, Criterion III, "Design Control," that requires the design basis to be correctly translated into procedures.

  The finding is more than minor, because it is associated with the design control attribute and affected the objective of the reactor safety barrier integrity cornerstone to provide reasonable assurance that physical design barriers protect control room operators from radiological releases caused by accidents. The finding was of very low safety significance, because it represented only a degradation of the radiological barrier function provided for the control room, and the increased operator dose would not have exceeded regulatory limits. (Section 1R12)

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by Entergy have been reviewed by the inspectors. Corrective actions taken or planned by Entergy have been entered into Entergy's corrective action program. These violations are listed in Section 4OA7 of this report.

- TS 3.4.3 requires that at least nine SRVs shall be operable in operating modes 1, 2, and 3. On August 19, 2003, Entergy identified that it had operated in these modes during cycle 15 with less than nine operable SRVs. Entergy documented this condition in CR-2003-04321. This finding is of very low safety significance because it did not result in loss of the overpressure relief safety function of the valves.

REPORT DETAILS

Summary of Plant Status

The reactor operated at or near 100 percent power for the entire inspection period. On November 4 power was reduced to 40 percent to perform a control rod pattern adjustment and sequence exchange, and planned maintenance on a reactor water recirculation (RWR) pump motor-generator set. Full power was restored on November 5.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 2 samples)

a. Inspection Scope

The inspectors completed the following two adverse weather protection samples:

- On October 29 the inspectors discussed with plant operators the potential effects of predicted solar magnetic disturbances on the off-site power supplies. The inspectors noted that the independent system operator (ISO) had issued an alert regarding the condition and that operators were familiar with the phenomenon.

- Due to sustained high wind and low temperature conditions on November 13, the inspectors walked down accessible portions of the emergency power and service water systems, and the ultimate heat sink to verify operability. These systems were selected because their safety-related functions could be affected by the adverse weather. Documents reviewed included abnormal operating procedure (AOP)-13, "High Winds, Hurricanes, and Tornadoes," and surveillance test (ST)-8G, "Intake Deicing Heaters Feeder Ammeters Test."

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdown. (71111.04Q - 3 Samples)

The inspectors performed three partial walkdowns of selected systems 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. The inspectors performed three partial walkdowns of the following systems:

- Control room and relay room ventilation systems on October 6
- Trains A and B emergency diesel generators (EDG) and 115 kV line 3 on November 21 while 115 kV line 4 was out of service for corrective maintenance at Nine Mile Point Unit 1
- Residual heat removal (RHR) train B while RHR train A was out of service on October 22 for a planned maintenance outage

The inspectors reviewed the following documentation:

- OP-55A, “Control and Relay Room Refrigeration Water Chiller”
- OP-55B, “Control Room Ventilation and Cooling”
- ST-9W, “Electrical Power and Lineup Verification “
- OP-22, “Diesel Generator Emergency Power”
- OP-13, “RHR”

<u>Complete System Walkdown</u>. (71111.04S - 1 Sample)

The inspectors performed a complete walkdown of the Emergency Service Water (ESW) system to identify any discrepancies between the existing equipment lineup and the specified lineup. During the walkdown 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. The documentation reviewed included:

- Design basis document (DBD)-046, “Service Water Systems”
- DBD-066, “Reactor Building Heating, Ventilation and Air Conditioning Systems”
- DBD-070, “Control Room and Relay Room Ventilation and Cooling Systems”
- DBD-093, “Emergency Diesel Generator Systems”
- DBD-067, “Turbine Building Ventilation and Cooling Systems”
- OP-21, “ESW”

b. <u>Findings</u>

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

Quarterly. (71111.05Q - 12 Samples)

The inspectors toured 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 twelve areas inspected included:

- South safety related pump room, fire area 12/zone SP-1
- North safety related pump room, fire area 13/zone SP-2
- Independent spent fuel storage installation (ISFSI) pad location and hydrogen storage facility
- Relay room, elevation 272 feet, fire area 07/zone RR-1
- West switchgear room, fire area 1C/zones SW-1 and SW-2
- Reactor building, elevation 272 feet, fire areas 9 and 10/zones RB-1A and RB-1B
- Turbine building, elevation 252 feet, fire area 1E/zone TB-1
- West cable tunnel, elevation 258 feet, fire area 1C/zone CT-1
- East cable tunnel, elevation 258 feet, fire area 02/zone CT-2
- Cable spreading room, elevation 272 feet, fire area 07/zone CS-1

Annual. (71111.05A - 1 Sample)

The inspector observed the performance of a fire brigade drill on December 10. The inspector reviewed the post-drill critique and the disposition of issues and deficiencies that were identified.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 Sample)

a. Inspection Scope

The inspectors completed one external flood protection inspection sample. The inspectors reviewed FitzPatrick's Individual Plant Examination (IPE) and the UFSAR concerning external flooding events. The inspectors verified Entergy's actions with respect to industry operating experience regarding the submergence of safety-related cables located in underground vaults. The inspectors walked down the switchyard manholes and verified manhole water levels, the operating condition of the sump pumps, and reviewed the preventive maintenance program for the sump level switches and pumps. The inspectors also reviewed Entergy's program and procedures for monitoring the condition of safety-related underground cables, and in particular, actions taken to verify operability when cables were submerged.

- Drawing FE-41D, "115 kV Switchyard Conduit Plan"
- Drawing 11825-FE-32E, "345 kV Switchyard Cable Trench Arrangement and Manhole Details"
- Drawing 11825-FE-32B-9, "Duct Line Plan and DETS Transformer Area"
- Drawing FE-41L "345 kV Switchyard Conduit Plan"

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

Annual Sample. (71111.07A - 1 Sample)

The inspectors reviewed the testing and evaluation of test results for the crescent area, cable tunnel, and electric bay unit coolers. ST-8Q, "Testing of the Emergency Cooling Water System," is performed on a quarterly basis to verify area cooler performance. Performance data were reviewed to verify that heat exchanger operation was consistent with design.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

Routine Inspection. (71111.11Q - 1 Sample)

The inspectors completed one sample for the licensed operator requalification baseline inspection. On December 9 the inspectors observed licensed operator simulator training to assess operator performance during a scenario involving a reactor coolant leak and an anticipated transient without scram. 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 inspector evaluated the critique provided by the simulator instructors during and following the scenario, and the completeness of the identified areas for improvement.

Biennial Review. (71111.11B - 1 Sample)

The inspector performed an in-office review of the reactor operator 2003 biennial written examinations to evaluate question quality, construction, and difficulty. The examination material reviewed included two biennial written examinations consisting of 40 open reference multiple choice questions. Since these two exams reused more than 50 percent of the questions from one exam to another, the crew mean scores were also reviewed in accordance with Inspection Procedure 71111.11, "Licensed Operator Requalification Program," Appendix D, "Guidance for Excessive Test Item Repetition and Potential for Examination Compromise." Entergy documented the excessive test item repetition in CRs 2004-00145 and 2004-00146.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12)

a. Inspection Scope

Routine Inspection. (71111.12Q - 2 Samples)

The inspectors reviewed performance-based problems involving selected in-scope structures, systems, or components (SSCs) to assess the effectiveness of the maintenance program. Reviews focused on proper maintenance rule scoping, in accordance with 10 CFR 50.65; characterization of reliability issues; charging 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) and the 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 following two maintenance rule samples were reviewed:

- Control room ventilation and cooling
- Core spray

The inspectors reviewed the following documents:

- JAF-RPT-CRC-02299, "Maintenance Rule Basis Document for Control and Relay Room Ventilation Systems"
- JAF-RPT-CSP-02285, "Maintenance Rule Basis Document for Core Spray System"

Biennial Periodic Evaluation. (71111.12B - 6 samples)

The inspector reviewed Entergy's biennial 10 CFR 50.65 (a)(3) evaluation for FitzPatrick. Entergy's most recent periodic evaluation report covered the period of November 1, 2001 to October 31, 2003, and was within the two-year period required by the maintenance rule (MR). To complete this inspection, the inspector performed the following:

- The inspector performed a detailed review of MR documentation for ESW, RHR, automatic depressurization (ADS), standby gas treatment (SGT), reactor building closed loop cooling (RBCLC) and EDG. The inspector verified that goal setting and performance criteria (PC) were appropriate; industry operating experience was considered in the establishment of PC and corrective action plans; corrective action plans were effective, and system performance was effectively monitored.

- The inspector reviewed documentation for SGT, ADS, ESW and RBCLC to verify that these systems were appropriately classified as (a)(1) under the MR and that Entergy adjusted (a)(1) goals as necessary to balance reliability and unavailability. The inspector also reviewed documentation for the reactor protection and 4 kV safety-related breakers, both in (a)(2) status, to verify that appropriate PC were established.

- The inspector reviewed documentation for the feedwater and 4 kV safety-related breakers, because both systems were moved from (a)(1) to (a)(2) during the evaluation period. The inspector verified that the (a)(1) goals for these systems were met before Entergy returned them to (a)(2) status.

- The inspector verified Entergy had established and implemented a preventive maintenance program for systems in (a)(1) and (a)(2) status. The inspector reviewed the performance of condition monitoring and scheduled maintenance on several MR scoped systems.

b. Findings

Introduction. In November 2002 abnormal operating procedure (AOP)-39, "Loss of Coolant Accident," was revised such that isolation of the control room envelope following a loss of coolant accident (LOCA) would be initiated later than analyzed in the design basis control room habitability calculation and described in the UFSAR. The inspector identified a non-cited violation of Criterion III of 10CFR 50, Appendix B, "Design Control," having very low safety significance (Green).

Description. The NRC identified that Entergy revised steps in AOP-39 governing isolation of the control room emergency ventilation air supply system (CREVASS) to be inconsistent with the plant design basis. Calculation JAF-CALC-RAD-00042, "Control Room Radiological Habitability Under Power Uprate Conditions and CREVASS Reconfiguration," shows that control room operator doses will be less than the regulatory limits in 10CFR 50, Appendix A, GDC-19 if the CREVASS is placed in the isolate mode within 30 minutes of the onset of an LOCA. This analysis is described in section 14.8.2.1.1 of the UFSAR.

AOP-39 originally directed operators to isolate the CREVASS within 30 minutes of an LOCA. In November 2002, the procedure was revised to monitor the system radiation monitor and isolated the control within 30 minutes of an alarm. The revision could have delayed the initiation of emergency ventilation and increased operator dose.

Errors in performing the 10 CFR 50.59 screening of the revision per procedure ENN-LI-100, "Process Applicability Determination," contributed to not identifying the conflict between the AOP revision and the plant design basis; i.e., screening questions concerning (1) whether the change involved or changed calculations or assumptions concerning post accident radiological conditions, and (2) whether the revision changed a procedure described in the UFSAR involving methods of performing or controlling design functions were incorrectly answered "no." The erroneous screening was a missed opportunity to have recognized the significance of the revision.

Analysis. The deficiency associated with this finding was an inadequate procedure change that could have resulted in increased post accident operator radiological dose. Incorrect screening of the AOP revision resulted in a violation of Criterion III of 10 CFR 50, Appendix B, "Design Control." The finding is more than minor, because it is associated with the attribute of design control, and affected the objective of the reactor safety barrier integrity cornerstone to provide reasonable assurance that physical design barriers protect the operators from radiological releases caused by accidents. Using phase one of the SDP the inspectors determined that the finding was of very low safety significance, because it represented only a degradation of the radiological barrier function provided for the control room. Also, Entergy's analysis concluded that the delayed control room isolation due to the revision would not have significantly increased operator dose or exceeded GDC-19 limits.

Enforcement. Criterion III of 10CFR 50, Appendix B, requires measures to be established to assure that the design basis is correctly translated into procedures. Contrary to the above, in November 2002, the design basis was not correctly translated into procedures in that procedure AOP-39, "Loss of Coolant Accident," was revised in manner inconsistent with the plant design basis as described in the UFSAR. Because this failure to maintain an adequate procedure is of very low safety significance and has been entered into Entergy's corrective action program (CRs 2003-05501, 2003-05498, and 2003-04889), the violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000333/2003010-01, Failure to Maintain an Adequate Alarm Response Procedure.

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

a. Inspection Scope

The inspectors reviewed the risk assessments for the five WRs listed below. The inspectors verified that risk assessments were performed in accordance with AP-10.10, “On-line Risk Assessment;” risk of scheduled work was managed through the use of compensatory actions and schedule adherence; and applicable contingency plans were properly identified in the integrated work schedule.

- WR 03-10500-00 that replaced the rod control system master clock card and relay 03A-K5 during the week of October 19
- WR 03-00021-00 that replaced the A RWR pump motor generator tachometer generator during the week of November 2
- WR 01-12722-00 that troubleshot and repaired switchgear L44 1000KVA transformer 71T-12 after overheating during the week of November 9
- WR 03-05437-00 that replaced the C RHR service water pump during the week of October 19
- WR JF-020203100 that replaced ESW piping and valves for east crescent area unit cooler 66UC-22D the week of December 7

b. Findings

Introduction. Corrective actions for a 1999 reactor water recirculation (RWR) pump trip were inadequate, and this resulted in another RWR pump trip and unplanned power reduction on September 25, 2003. This represented a self-revealing finding of very low safety significance (Green).

Description. An RWR pump trip occurred in May 1999 and was attributed to excessive motor-generator (MG) set tachometer generator brush wear. Corrective actions to replace the MG tachometer generators with new or overhauled units every two years were not implemented, because a change request was never initiated to schedule the new preventive maintenance tasks. On September 25, 2003, the event recurred, as the B RWR pump tripped due to excessive carbon dust buildup on the MG tachometer generator. The reduction in core flow from the pump trip resulted in a reactivity transient which lowered reactor power to approximately 65% of rated power.

Analysis. The performance deficiency involved in this finding concerned failure to follow the administrative procedure for processing preventive maintenance program changes and inadequate verification that corrective actions had been implemented. The finding is considered more than minor, because it is associated with the equipment performance attribute and resulted in an unplanned plant transient that affected the reactor safety initiating events cornerstone objective of limiting the likelihood of events that upset plant stability. The inspectors evaluated the finding using Phase 1 of the SDP (Initiating Events) and determined it to be of very low safety significance (Green), because the finding did not (1) contribute to the likelihood of a primary or secondary LOCA initiator; (2) contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available; and (3) increase the likelihood of a fire or internal/external flood. Also, the plant responded during the transient as designed and within the bounds of the RWR pump trip safety analysis described in the UFSAR. This finding is in Entergy’s corrective action program as CR

2003-04585. FIN 05000333/2003010-02, Inadequate Corrective Action for Recirculation Pump Trip

Enforcement. No violation of regulatory requirements occurred. The inspectors determined that the finding did not represent a noncompliance, because it occurred on non safety-related primary plant equipment.

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

a. Inspection Scope

On November 10 the inspectors observed operator response to reported smoke coming from 1000 KVA transformer 71T-12. At the time of the incident the transformer was supplying bus L43 and was cross-tied to bus L44. The transformer briefly overloaded when a motor-driven fire pump, powered from bus L44 started during a routine surveillance test. The transformer was found to be undamaged and restored to service later that day. The inspectors reviewed operator logs and plant computer data and interviewed operators to determine what occurred, how the operators responded, and if their response was in accordance with plant procedures.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 3 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to assess the acceptability of the evaluations, the use and control of compensatory measures if needed, and compliance with technical specifications. The inspector's review included a 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 TS, UFSAR, and associated DBD. The following three evaluations were reviewed:

- CR-2003-04689 concerning operation of the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems with elevated torus water temperature
- CR-2003-04707 concerning the basis for operability of the EDGs associated with the extent of condition review for the stationary auxiliary switch failure in the C circulating water pump breaker
- CR-2003-05808 concerning an out of specification core spray pump time delay relay

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16 - 1 sample)

a. Inspection Scope

The inspectors completed one operator workaround inspection sample. The inspectors evaluated individual and cumulative effects of identified operator workarounds on the functionality of the plants mitigating systems. The workarounds were reviewed to determine if the functional capability of the system or human reliability in responding to an initiating event was affected; the effect on the operator's ability to implement abnormal or emergency procedures; and if operator workaround problems were captured in Entergy's corrective action program. The inspectors reviewed Entergy's assessment of the cumulative effects of the identified workarounds in accordance with ST-99H, "Operator Workarounds Assessment." The inspectors also performed a detailed review of operator workarounds associated with circulating water pump discharge valves, and non-density compensated reactor water high-level trip setpoints for HPCI and RCIC.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 5 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 testing on the plant had been adequately addressed by control room and engineering personnel; testing was adequate for the maintenance performed; acceptance criteria were clear and adequately demonstrated operational readiness, consistent with design and licensing basis documents; test instrumentation had current calibrations, range, and accuracy for the application; tests were performed, as written, with applicable prerequisites satisfied; and that equipment was returned to the status required to perform its safety function. The following five post maintenance test activities were reviewed:

- WR 01-02408-00, WR 03-03358-00, and WR 03-08808-00 that involved rebuilding and replacing scram pilot and scram pilot solenoid valves on hydraulic control units 30-03 and 26-51. The retest was performed using reactor analyst procedure(RAP)-7.4.01, "Control Rod Scram Time Evaluation."
- WR JF-010620601 and WR JF-010620501 that involved the inspection and repair of ESW system check valves associated with east electric bay unit cooler 67UC-16B during the week of December 8. The retest was full flow testing performed using ST-8Q, "Testing of the ESW System."

- WR 99-03633-00 that replaced the discharge check valve for the C RHR pump. The retest was performed using ST-2AL, “RHR Loop A Quarterly Operability Test.”
- WR 03-00844-00 and WR 03-00340-43 that repacked HPCI turbine steam supply isolation valve 23MOV-14. The retests included inservice stroke timing and diagnostic testing.
- WR 03-10103-00 that plugged several leaking tubes in east electric bay unit cooler 67UC-16B. Thermal performance of the cooler was verified using ST-8Q, “Testing of the ESW System.”

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 5 samples)

a. Inspection Scope

The inspectors witnessed performance of STs and reviewed test data of selected risk-significant SSCs to assess whether the SSCs satisfied TS, UFSAR, technical requirements manual, and Entergy procedure requirements. The inspectors assessed whether the testing appropriately demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. The following five tests were witnessed:

- ST-18, “Main Control Room Emergency Fan and Damper Operability Test”
- RAP-7.4.01, “Control Rod Scram Time Evaluation”
- Instrument surveillance procedure (ISP)-84, “Standby Gas Treatment System Instrument Calibration”
- ST-3JA, “Core Spray Logic System Functional Test”
- ST-8Q, “Testing of the ESW System”

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

Cornerstone: Initiating Events

4OA1 Performance Indicator Verification (71151 - 5 samples)

a. Inspection Scope

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 7,000 critical hours
- Scrams with a loss of normal heat removal
- Unplanned transients per 7,000 critical hours

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

Occupational Radiation Safety Cornerstone

- Occupational Exposure Control Effectiveness

For the period of October 2002 to December 2003, the inspector reviewed CRs and associated documents to identify all exposure control occurrences that involved locked high radiation areas, very high radiation areas, and unplanned exposures that met the NEI reporting criteria for this PI. The inspector compared that number to the number that Entergy reported for the third quarter.

Public Radiation Safety Cornerstone

- RETS/ODCM Radiological Effluent Occurrences

The inspector reviewed relevant effluent release reports for the period October 2002 through December 2003. Specifically, the inspector reviewed the following documents to verify that Entergy met all NEI reporting requirements for this the PI:

- 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
- Dose assessment procedures

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

1. Annual Sample Review

a. Inspection Scope

The inspectors selected three corrective action issues for detailed review. CR 2002-03684 dealt with Entergy's review of operating experience regarding fatigue failure of two small bore piping lines in the RHR system and a January 2002 service water system pipe fatigue failure at FitzPatrick documented in CR 2002-00333. CR 2002-04695 identified four RBCLC containment isolation valves that repetitively failed to meet local leak rate test limits during refueling outages 14 and 15. CR 2003-04424 dealt with inadequate operator training and procedure revisions following a service air design modification that caused an unnecessary entry into AOP-12, "Loss of Instrument Air after a trip of the C service air compressor. These reports were reviewed to ensure that an appropriate evaluation was performed and appropriate corrective actions were specified. The inspectors evaluated the reports against the requirements of procedure ENN-LI-102, "Corrective Action Process," and 10 CFR 50, Appendix B.

b. Findings and Observations

No significant findings were identified regarding the three samples; however, the following corrective action program insights were observed.

January 2002 Service Water Discharge Pipe Failure. Entergy's apparent cause for this failure determined that the cause of the integral lug support failure and three previous adjacent clevis hanger failures was that the original design of the integral lug pipe support was inadequate based on pipe stress analysis JAF-CALC-SWS-04427. Based on this apparent cause the inspectors determined that previous repairs and failure evaluations for the three previous adjacent clevis hanger failures did not adequately address the cause of those hanger failures, and thus, did not prevent the service water pipe failure in 2002. The inspectors concluded that longstanding, ineffective corrective actions resulted in the hanger and piping fatigue failure in 2002. However, because the pipe that failed was non-safety related, and the corrective actions, cause evaluation, and extent of condition review performed for the 2002 pipe failure were adequate, no violation of regulatory requirements was identified.

Inadequate Operator Training and Procedure Revisions. CR-2003-04424 identified inadequate operator training and procedure revisions following service air compressor modifications as the causes for an unnecessary entry into abnormal operating procedure (AOP)-12, "Loss of Instrument Air," after a trip of the C service air compressor on September 15. As corrective action the CR directed operations to revise the affected procedures, but did not address the broader question of how equipment in the control room was modified and turned over to operations without sufficient procedures and training. The inspectors discussed this concern with management, and Entergy updated the specified corrective actions to include an analysis of the process for procedure revisions and training following system modifications. Entergy completed the analysis and no significant issues were identified. The inspectors verified the adequacy of Entergy's analysis and corrective actions, and because the service air compressors are non-safety related, no actual loss of instrument air occurred, and Entergy's analysis revealed no significant safety issues, no violation of regulatory requirements was identified.

2. Routine PI&R Program Review

a. Inspection Scope

As required 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 corrective action program. 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 43 corrective action program items across the initiating events, mitigating systems, and barrier integrity 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, and operability determinations, and the timeliness of the corrective actions specified. The CRs reviewed are noted in the Attachment.

b. Findings

No findings of significance were identified.

3. Cross-references to PI&R Findings Documented Elsewhere

Section 1R13 describes a finding involving inadequate corrective action that resulted in a reactor water recirculation (RWR) pump trip and unplanned power reduction on September 25, 2003.

4OA3 Event Follow-up (71153)

1. (Closed) LER 05000333/2003001-00, Automatic Reactor Shutdown Due to Grid Instability Associated with the August 14, 2003 Transmission Grid Blackout and Related Plant Mode Change with the A EDG Subsystem Inoperable

On August 29, 2003, during surveillance testing of the A EDG subsystem, Entergy determined that the A subsystem was inoperable due to a no load frequency setting that was out of tolerance high by 0.1 Hz. The root cause analysis determined that the out of tolerance condition was initiated while setting up the A EDG during recovery from the August 14 grid instability event. Consequently, the plant startup following the August 14 automatic shutdown entailed an operating mode change prohibited by Technical Specification 3.0.4 but unrecognized at the time. Follow-up engineering analysis determined that the 0.1 Hz out-of tolerance condition had no impact on the operability of the A EDG. Therefore, this finding constitutes a violation of minor significance not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. Entergy documented the condition in CR 2003-04203. This LER is closed.

2. (Closed) LER 05000333/2003002-00, Safety Relief Valve Setpoints Outside of Allowable Tolerances

On August 19, 2003, Entergy determined that three safety relief valves (SRV) had as-found setpoints outside the tolerance allowed by TS 3.4.3.1; two valves exceeded the high limit and one valve exceeded the low limit. SRV setpoint drift due to oxidic binding of the pilot valves has been a generic industry issue that has continued to be addressed by the NRC and the GE Boiling Water Reactors Owners Group. FitzPatrick installed a diverse SRV pressure switch actuation system that is not affected by oxidic binding. The finding is more than minor, because it had a credible impact on safety in that reduced reliability and functionality of the safety system designed to mitigate a reactor overpressure event could adversely affect fuel cladding and reactor coolant system pressure boundary integrity. The condition was mitigated by two considerations: (1) while two SRVs did not lift within the TS-prescribed high limit, they did actuate at higher pressures; and (2) a diverse SRV pressure switch actuation system was available. Since the plant continued to operate within the bounds of the design basis safety analyses, there was no loss of safety function. Using Phase 1 of the SDP the inspectors determined that the condition was a design or qualification deficiency confirmed not to result in a loss of function per Generic Letter 91-18, Revision 1. Therefore, the risk associated with this condition was of very low significance (Green). This licensee-identified finding involved a violation of TS 3.4.3, "Safety/Relief Valves." The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

4OA6 Meetings, Including Exit

The inspectors presented the inspection results to Entergy management at the conclusion of the inspection on January 26, 2004. Entergy acknowledged that no proprietary information was involved.

4OA7 Licensee-identified Violations

The following violation of very low safety significance (Green) was identified by Entergy and is a violation of NRC requirements that meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- TS 3.4.3 requires that at least nine SRVs shall be operable in operating modes 1, 2, and 3. On August 19, 2003, Entergy identified that it had operated in these modes during cycle 15 with less than 9 operable SRVs. Entergy documented this condition in CR-2003-04321. This finding is of very low safety significance because it did not result in loss of the overpressure relief safety function of the valves.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy personnel

N. Avrakatos, Emergency Preparedness Coordinator
P. Berry, Manager, Training
V. Bhardwaj, Manager, Programs and Components Engineering
S. Bono, Manager, System Engineering
C. Boucher, Chemistry Superintendent
B. Drain, Manager, Project Management
J. Haley, Manager, Security
A. Halliday, Manager, Regulatory Compliance
D. Johnson, Manager, Operations
A. Khanifar, Manager, Design Engineering
O. Limpias, Director, Engineering
G. Lozier, Maintenance Superintendent
B. Maguire, Director, Nuclear Safety
K. Mulligan, General Manager, Plant Operations
K. Pushee, Manager, Radiation Protection
D. Ruddy, Manager, CA&A
T. Spencer, Manager, Plant Maintenance
T. Sullivan, Vice President, Operations
D. Wallace, Quality Assurance Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000333/2003010-01	NCV	Inadequate procedure for isolation of control room ventilation during an LOCA. (Section 1R12)
05000333/2003010-02	FIN	Inadequate corrective action resulted in RWR pump trip and unplanned power reduction. (Section 1R13)

Closed

05000333/2003001-00	LER	Automatic Reactor Shutdown Due to Grid Instability Associated with the August 14, 2003 Transmission Grid Blackout and Related Plant Mode Change with the A EDG Subsystem Inoperable (Section 4OA3)

05000333/2003002-00	LER	Safety Relief Valve Setpoints Outside of Allowable Tolerances (Section 4OA3)

Discussed

NONE

LIST OF DOCUMENTS REVIEWED

Section 4OA1: Performance Indicator Verification

Procedures:

SP-01.05, “Wastewater sampling and Analysis”
SP-01.06, “Gaseous Effluent Sampling and Analysis”
AP-20.08, “NRC Performance Indicators”
CHSO-02, “Chemistry Notifications”

Effluent and Dose Assessment Reports:

2002 Annual Radioactive Effluent Release Report
Quarterly and Monthly Liquid Release and Dose Summary Reports for 10/01/2002 to 12/09/2003
Quarterly and Monthly Gaseous Release and Dose Summary Reports for 10/01/2002 to 12/09/2003

Section 4OA2: Identification and Resolution of Problems

Condition Reports:

CR-2002-00333	CR-2002-03684	CR-2002-04695	CR-2003-02714
CR-2002-04019	CR-2003-04742	CR-2002-01263	CR-2002-02335
CR-2003-05520	CR-2003-05514	CR-2003-06369	CR-2003-05371
CR-2003-05966	CR-2003-05957	CR-2003-05949	CR-2003-05950
CR-2003-05951	CR-2003-05948	CR-2003-05934	CR-2003-02771
CR-2003-05060	CR-2003-05064	CR-2003-05074	CR-2003-05048
CR-2003-04689	CR-2003-05872	CR-2003-04889	CR-2003-05056
CR-2003-05763	CR-2003-05801	CR-2003-05808	CR-2003-05356
CR-2003-04933	CR-2003-05467	CR-2003-05583	CR-2003-05195
CR-2003-04890	CR-2003-05498	CR-2003-05501	CR-2003-03930
CR-2003-04203	CR-2003-04321	CR-2003-04887	

Procedures:

ENN-LI-102, “Corrective Action Process”
ENN-OP-104, “Operability Determinations”

Miscellaneous:

JAF-CALC-MISC-02875, “Suppression Pool Temperature Following a Small Break Accident with HPCI Operation”
JAF-CALC-SWS-04427, “Pipe Stress Calculation”
Periodic Assessment of Maintenance Rule Program, November 2001 through October 2003
Maintenance Rule Performance Indicators and Unavailability Time for ESW, RHR, ADS, SGT and RBCLC
System Health Reports for ESW, RHR, Reactor Protection System (RPS) and Emergency Diesel Generator (EDG) for second and third quarters of 2003
JENG-APL-03-014, “Maintenance Rule Action Plan, ‘A’ Standby Gas Treatment System”
JTS-APL-97-011, “Maintenance Rule Action Plan, Main Steam Safety Relief Valves”
JTS-APL-00-011, “Maintenance Rule Action Plan, ESW”
JENG-APL-03-016, “Maintenance Rule Action Plan, RHR ‘B’ System”
JENG-APL-01-004, “Maintenance Rule Action Plan, Feedwater System”
JTS-APL-99-013, “Maintenance Rule Action Plan, Air Operated Valves, Extraction Steam and Feedwater Heater Vents and Drains”
JMD-APL-97-010, “Maintenance Rule Action Plan, 4KV Circuit Breakers”
JTS-APL-97-018, “Maintenance Rule Action Plan, Containment Isolation Valves, (Reactor Water Clean-up Supply, Reactor Core Isolation Cooling Steam Supply, Reactor Feedwater, Main Steam Isolation Valves, Incore Probe Ball valves and RBCLC Containment Isolation Valves)”
Performance Indicators, ESW, RHR, ADS, SGT and RBCLC (Systems 001, 002, 010, 015, 046)
Preventive Maintenance Overview Report and Interval Data Report, dated December 12, 2003
Expert Panel Meeting Minutes, dated November 21, 2002

LIST OF ACRONYMS

ADS	automatic depressurization system
AOP	abnormal operating procedure
CR	condition report
DBD	design basis document
EDG	emergency diesel generator
EOP	emergency operating procedure
ESW	emergency service water
HPCI	high pressure coolant injection
IPE	individual plant examination
ISFSI	independent spent fuel storage installation
ISO	independent system operator
ISP	instrument surveillance procedure
kV	kilovolt
LER	licensee event report

MR	maintenance rule
NCVs	non-cited violations
NRC	Nuclear Regulatory Commission
OP	operating procedure
PC	performance criteria
PI	performance indicator
RAP	reactor analyst procedure
RBCLC	reactor building closed loop cooling system
RCIC	reactor core isolation cooling
RHR	residual heat removal
RWR	reactor water recirculation
SDP	significance determination process
SGT	standby gas treatment system
SRV	safety relief valve
SSC	systems, structures and components
ST	surveillance test procedure
TI	temporary instruction
TS	technical specification
UFSAR	updated final safety evaluation report
WR	work request