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

November 7, 2008

Mr. Peter T. Dietrich
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 05000333/2008004

Dear Mr. Dietrich:

On September 30, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant. The enclosed inspection report documents the inspection results, which were discussed on October 16, 2008, with you and other members of your staff.

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

Based on the results of this inspection, one finding of very low safety significance (Green) was identified. This finding was determined to be a violation of NRC requirements. A reporting violation was also identified that was evaluated under traditional enforcement and categorized at Severity Level IV. Additionally, a licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of the very low safety significance, and because the violations were entered into your corrective action program, the NRC is treating these violations as a non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a written response within 30 days of the date of this inspection report with the basis for your denial, 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 Senior Resident Inspector at the James A. FitzPatrick Nuclear Power Plant.

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

NRC's 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/

Mel Gray, Chief
Projects Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000333/2008004
w/Attachment: Supplemental Information

cc w/encl:
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Vice President, Oversight, Entergy Nuclear Operations
Senior Manager, Nuclear Safety and Licensing, Entergy Nuclear Operations
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P. Church, Oswego County Administrator
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C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
S. Lousteau, Treasury Department, Entergy Services
J. Spath, Program Director, New York State Energy Research and Development Authority

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-333
License No.:	DPR-59
Report No.:	05000333/2008004
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	July 1, 2008 through September 30, 2008
Inspectors:	G. Hunegs, Senior Resident Inspector S. Rutenkroger, PhD, Resident Inspector H. Gray, Senior Reactor Inspector P. Kaufman, Senior Reactor Inspector J. Noggle, Senior Health Physicist R. Cureton, Emergency Preparedness Inspector C. Hott, Resident Inspector S. McCarver, Project Engineer
Approved by:	Mel Gray, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000333/2008-004; 07/01/2008 - 09/30/2008; James A. FitzPatrick Nuclear Power Plant; Refueling and Other Outage Activities; and Identification and Resolution of Problems.

The report covered a three-month period of inspection by resident inspectors and announced inspections by region based inspectors. A Green finding and Severity Level (SL) IV violation, both of which were 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 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. A self-revealing NCV of 10 CFR Part 50.65 (a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified when Entergy did not manage the increase in risk that resulted from removal of the 'B' reactor protection system from service in preparation for conducting maintenance. The removal of the 'B' reactor protection system from service resulted in an unanticipated loss of shutdown cooling (SDC). Entergy took prompt action to communicate the error to station personnel; provide additional oversight for equipment tagouts affecting required safety systems during the remainder of the refueling outage; and entered the issue into the corrective action program.

This finding is more than minor because it is related to maintenance risk assessment and management. In this instance, Entergy did not implement prescribed significant compensatory measures and effectively manage those measures. Specifically, this finding reflects inadequate risk management that contributed to a short duration loss of shutdown decay heat removal capability resulting from the inadvertent interruption of flow through the operating train of shutdown cooling during cavity flood-up, in preparation for refueling. In accordance with IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and Appendix G, "Shutdown Operations Significance Determination Process," the inspectors determined this finding was of very low safety significance (Green). In accordance with IMC 0609, Appendix G, this finding did not require quantification and did not constitute a significant loss of thermal margin, based upon the slow reactor coolant system heat-up rate and minimal time of interruption in shutdown cooling system operation.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not plan and coordinate work activities properly to manage operational impact of work activities. Specifically, the impact on shutdown cooling of deenergizing the 'B' reactor protection system was not recognized or assessed. Additionally, a number of processes and barriers, such as the outage risk assessment and protective equipment program, were not used effectively. (H.3(b)) (Section 1R20)

Cornerstone: Public Radiation Safety

Severity Level (SL) IV. A self-revealing NCV of 10 CFR Part 71.95 was identified because Entergy did not provide a written report to the NRC as required by 10 CFR Part 71.95 relative to a non-conforming condition involving the shipment of a NRC-approved package. Entergy was informed that a package it shipped to Energy*Solutions*™ Barnwell Low Level Radioactive Waste Disposal Facility was found to be in non-conformance with the applicable Certificate of Compliance for the package upon receipt, Entergy did not report the condition to the NRC within 60 days of the occurrence, as required. Failure of Entergy to report the condition, as required by 10 CFR Part 71.95, constitutes a performance deficiency in that the issue is the result of Entergy not meeting a regulatory requirement that was reasonably within Entergy's ability to foresee and correct, and should have been prevented. Entergy entered this issue into the corrective action program as condition report (CR)-2008-02772.

This violation involved a failure to make a required report to the NRC and is considered to impact the regulatory process. Such violations are dispositioned using traditional enforcement process instead of the Significance Determination Process. Using the Enforcement Policy Supplement IV "Transportation," example D4 which states, "a noncompliance with shipping papers, marking, labeling, placarding, packaging or loading not amounting to a Severity Level I, II, or III violation;" the NRC determined this violation is categorized as a SL IV Violation. The Enforcement Policy Supplement I "Reactor Operations" examples D3, D4, and D5 are similar to this issue, in that they discuss examples of failures to make required reports for more than minor events, which are also categorized at Severity Level IV.

This finding has a cross-cutting aspect in the area of problem identification and resolution related - corrective action program, because Entergy performed an insufficient evaluation of a non-conforming condition associated with an NRC-approved package to assure the matter was properly classified, prioritized and evaluated relative to reportability. (P.1(c)) (Section 4OA2)

B. Licensee-Identified Violations

A violation of very low safety significance identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The James A. FitzPatrick Nuclear Power Plant (FitzPatrick) began the inspection period operating at 100 percent reactor power. On July 14, 2008, Entergy reduced reactor power to 55 percent to remove the 'B' feedwater pump from service to facilitate replacement of the inboard seal. Following replacement of the 'B' feedwater pump inboard seal, reactor power was restored to 100 percent on July 19, 2008. On September 14, 2008, the reactor was shutdown to conduct a refueling outage and remained shutdown for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors reviewed and verified completion of the warm weather preparation checklist contained in procedure AP-12.04, "Seasonal Weather Preparations." The inspectors reviewed the operating status of the reactor and turbine building cooling systems, reviewed the procedural limits and actions associated with elevated lake temperature, and walked down accessible areas of the reactor and turbine buildings to assess the effectiveness of the ventilation systems. Walkdowns were also conducted in the emergency diesel generator (EDG), emergency service water, and switchgear rooms. Discussions with operations and engineering personnel were conducted to ensure that they were aware of temperature restrictions and required actions. The documents reviewed are listed in the Attachment. The inspection satisfied one inspection sample for seasonal weather conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdown (71111.04Q – 3 samples)

a. Inspection Scope

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

- Instrument battery system while the reactor vessel pressure and level pen recorder and associated control room annunciators were out of service;
- Reactor core isolation cooling system while the high pressure coolant injection system was out of service; and
- 'B' reactor protection system when 'A' reactor protection system was on a backup power supply during the installation of a modification.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the residual heat removal system to identify any discrepancies between the existing equipment lineup and the required lineup. During the inspection, system drawings and operating procedures were used to verify proper equipment alignment and operational status. The inspectors reviewed the open maintenance work orders associated with 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 by the inspectors to assess their collective impact on system operation. In addition, the inspectors reviewed the condition report database to verify that equipment problems were being identified and appropriately resolved. The documents reviewed are listed in the Attachment. The inspection represented one inspection sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Quarterly Inspection (711111.05Q – 6 samples)

a. Inspection Scope

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

This inspection represented six inspection samples for fire protection tours and was conducted in the following plant areas:

- Fire Area/Zone IX/RB-1A, elevation 369 foot;
- Fire Area/Zone IE/TB-1 North, elevation 252 foot;
- Fire Area/Zone IE/TB-1 South, elevation 252 foot;
- Fire Area/Zone ISFSI Yard, elevation 272 foot;
- Fire Area/Zone II/CT-2, elevation 258 foot; and
- Fire Area/Zone IC/CT-1, elevation 258 foot.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 – 2 samples)

a. Inspection Scope

The inspectors conducted tours of the East and West crescent rooms to assess internal flooding protection measures in those areas. The inspectors reviewed selected risk-significant plant design features and Entergy's procedures intended to protect the associated safety-related equipment from internal flooding events. The inspectors reviewed flood analysis and design documents, including the Individual Plant Examination and the UFSAR, engineering calculations, and abnormal operating procedures. The documents reviewed are listed in the Attachment. These inspection activities represented two inspection samples.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08 – 1 sample)

a. Inspection Scope

The inspectors reviewed non-destructive examination activities during the refueling outage 18 (RO18) that included observations of in-progress ultrasonic testing (UT) and analysis of test results from the phased array UT technique and the General Electric smart computer based UT. These ultrasonic test systems were used to examine alloy 82/182 dissimilar metal welds in the recirculation system, including welds N1A (28" diameter), N2B and N2C (both 12" diameter). The inspectors reviewed the applicable UT procedures, qualification certification for the personnel and procedures, observed UT data analysis review and verified that relevant indications were properly documented and presented to Entergy for disposition.

The inspectors compared Entergy's dissimilar metal weld program with the Electric Power Research Institute guidance, "Boiling Water Reactor Vessel and Internals Project;" letter 2007-367 (BWRVIP-2007-367), "Recommendations Regarding Dissimilar Metal Weld Examinations;" and BWRVIP-75A, "Technical Basis for Revisions to NRC Generic Letter 88-01 Inspection Schedules." The inspectors reviewed the data of previous and current automated ultrasonic examination of the safe-end to nozzle welds N2B and N2C.

The inspectors reviewed a UT of the alloy 82/182 N2C safe-end to nozzle weld that identified an inside diameter surface-breaking, transverse (axial to the weld circumference) indication which required a mitigating weld overlay. The inspectors reviewed the weld overlay procedure; essential weld overlay variables; controls on the weld process and preparations for welding; and the procedure for examination of the completed overlay weld.

The inspectors reviewed the UT procedure ENN-NDE-904, Revision 2 and observed the equipment calibration for the UT examination of main steam pipe welds 24-29-584 and 589.

For in-vessel visual inspection, the inspectors sampled the remote enhanced visual examination of reactor vessel internals including core spray piping, core shroud and steam dryer. The in-vessel examination included re-examination of previously identified indications. The inspection included a review of the applicable in-vessel visual inspection procedure, observation of a sample of digital video records, the analysis process for the observations, and documentation of indications.

The inspectors reviewed portions of the radiography procedure ENN-NDE-10.05, Revision 1 and reviewed the in-process radiographs for the replacement welds on the reactor core isolation cooling weld FW-3 for comparison with the ASME Code and site radiography testing procedural requirements. The sensitivity of the radiographic method as shown by the penetrameter and densitometer measurement, the identification of the radiographer, and provision for acceptance by the data reviewers were observed.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On August 18, 2008, the inspectors observed licensed operator simulator training to assess operator performance during several scenarios to verify that operator performance was adequate and evaluators were identifying and documenting crew performance problems. The inspectors evaluated the performance of risk significant operator actions, including the use of emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, the implementation of appropriate actions in response to alarms, the performance of timely control board operation and manipulation, and the oversight and direction provided by the shift manager. The inspectors also reviewed simulator fidelity to evaluate the degree of similarity to the actual control room. Licensed operator training was evaluated against the requirements of 10 CFR Part 55, "Operators' Licenses." The documents reviewed are listed in the Attachment. This observation of operator simulator training represented one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 samples)

a. Inspection Scope

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

- Proper Maintenance Rule scoping in accordance with 10 CFR Part 50.65;
- Characterization of reliability issues;
- Changing system and component unavailability;
- 10 CFR Part 50.65 (a)(1) and (a)(2) classifications;
- Identifying and addressing common cause failures;
- Trending of system flow and temperature values;
- Appropriateness of performance criteria for SSCs classified (a)(2); and
- Adequacy of goals and corrective actions for SSCs classified (a)(1).

The inspectors reviewed system health reports, maintenance backlogs, and Maintenance Rule basis documents. The inspectors evaluated the maintenance program against the requirements of 10 CFR Part 50.65. The documents reviewed are listed in the Attachment. The following maintenance effectiveness samples were reviewed and represented three inspection samples:

- Residual heat removal system;
- Offgas system; and
- Stack and stack equipment including dilution fans.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- The week of June 30, 2008, which included fire pump 17P-4B repair, inoperable fire hose stations, and instrument surveillances that affect the reactor core isolation cooling system;
- The week of July 14, 2008, which included reduced power operation during repair work on the 'B' reactor feed pump and high pressure coolant injection system testing;
- The week of July 21, 2008, which included adverse weather, planned testing and calibrations impacting reactor core isolation cooling and low pressure injection systems, and failure of the 'A' service water strainer;
- The week of August 11, 2008, which included schedule changes due to an unplanned outage of the 'D' EDG subsystem, and planned testing and calibrations of primary containment isolation system and the reactor protection system instruments; and
- The week of August 25, 2008, which included a planned outage of the 115 kV offsite power source line number four, planned surveillance testing of condensate storage tank low level switches, and scaffold erection activities in preparation for the upcoming outage.

b. Findings

No findings of significance were identified.

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 compliance with Technical Specifications (TS). The inspectors' 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 TSs, UFSAR, and associated design basis documents. The documents reviewed are listed in the Attachment. The following evaluations were reviewed and represented five inspection samples:

- CR 2008-02075, impact of external water leak into the west crescent area and associated conduit, cables, piping and junction boxes;

- CR 2008-02384, abraded 'D' EDG fuel oil hose;
- CR 2008-02601, steam leak on 'B' reactor water cleanup pipe;
- CR 2008-00781, 'B' EDG jacket water cooler degradation; and
- CR 2008-03213, bent and cracked steam separator standpipe tie strap.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18 – 4 samples)

a. Inspection Scope

The inspectors reviewed the following plant modifications to verify the design bases, licensing bases, and performance capability of the systems were not degraded by the modifications. The inspectors reviewed the modifications against the requirements of 10 CFR 50.59. The documents reviewed are listed in the Attachment. The following modifications were reviewed and represented four inspection samples:

- The inspectors reviewed temporary modification EC-10523, which was implemented to remove a broken steam separator tie bar identified during remote visual inspection of separator gussets. Extent of condition inspections of remaining tie rods around the 0° steam separator did not disclose any additional damaged tie rods. The inspectors reviewed information provided by GE-Hitachi, and operating experience information from other nuclear power plants to determine industry experience with this type of issue and acceptability of removal of the damaged tie rod without immediate replacement. The inspectors also reviewed documentation to determine if sufficient steam separator support was provided by supporting tie bars from adjacent separators.

- The inspectors reviewed permanent plant modification EC-6660, which was implemented to replace the 'B' and 'C' traveling water screens. The modification included replacement of the existing traveling water screens, motor drives, screen differential level instrumentation and modification of the screen wash piping and debris removal sections. The inspectors also verified 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 reviewed temporary modification EC-8402, which provided temporary cooling for the steam tunnel. The modification provided incremental cooling to partially compensate for lost cooling from a degraded unit cooler. A rental chiller was placed outside the turbine building and had a flexible hose routed from the rental chiller into the condenser bay. The inspectors also verified 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 reviewed permanent plant modification EC-4572, which provided a variable speed drive for the 'A' traveling water screen. The change was designed to allow the speed of the traveling water screen to be adjusted to optimize debris removal. The inspectors also verified 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.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 8 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; test instrumentation had current calibrations, adequate range, and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon completion, the inspectors verified that equipment was returned to the proper alignment necessary to perform its safety function. Post-maintenance testing was evaluated against the requirements of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." The documents reviewed are listed in the Attachment. The following post-maintenance test activities were reviewed and represented eight inspection samples:

- Work order 00161355, troubleshooting of load swings on the 'D' EDG and replacement of the governor actuator;
- Work order 00134115, installation of the 'A' traveling water screen variable speed drive (engineering change 4572);
- Work order 00125926, 'B' EDG system governor modification;
- Work order 51192253, maintenance on the 'B' standby gas treatment system;
- Work order 00166677, repair of 27 AOV-115, torus purge supply valve;
- Work order 00165300, repair of 29 AOV-80D, main steam inboard isolation valve;
- Work order 00155433, 'B' and 'C' traveling water screen modification; and
- Work order 00125926, 'D' EDG system governor modification.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

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. The outage was in progress at the end of the inspection period, therefore this sample will be completed during the next inspection period.

- The inspectors reviewed outage schedules and procedures; and verified that TS required safety system availability was maintained, shutdown risk was considered, and contingency plans existed for restoring key safety functions such as electrical power and primary coolant system makeup.

- The inspectors observed portions of the plant shutdown and cooldown on September 13 and 14, 2008, and verified that the TS cooldown rate limits were satisfied.

- During the course of the refueling outage, the inspectors observed selected reactor disassembly activities and walked down clearances to verify that tagouts were properly implemented 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 alternate decay heat removal 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 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.

b. Findings

Introduction: A Green, self-revealing NCV of 10 CFR Part 50.65 (a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” was identified when Entergy failed to manage the increase in risk that resulted from removal of the ‘B’ reactor protection system from service in preparation for conducting maintenance. Removal of the ‘B’ reactor protection system from service resulted in an unanticipated loss of shutdown cooling (SDC).

Description: On September 16, 2008, shutdown cooling flow was isolated while hanging a tagout on the ‘B’ reactor protection system. When removing fuses as directed by the tagout, isolation logic for shutdown cooling suction valve 10MOV-18, was actuated. This suction valve isolates the common shutdown cooling suction line to all residual heat removal pumps. At the time of the isolation, the reactor was in the refueling mode, (Mode 5) reactor cavity flood-up was in progress and the time to boil was greater than 5.5 hours. Operator’s entered abnormal operating procedure (AOP)-30, “Loss of Shutdown Cooling,” and restored shutdown cooling flow in approximately 53 minutes. All equipment performed as expected based on the conditions caused by the removal of the fuse.

Entergy determined that several failed barriers contributed to or did not prevent this event from occurring. Entergy’s evaluation of the event identified a combination of human performance and programmatic/organizational errors that occurred when station personnel did not adequately prepare and install an adequate tagout that considered the work scope/schedule and applicable reactor operational conditions (high decay heat load). Specifically, the tagout cover page did not contain pertinent system status information or amplifying notes as required by station procedures. This lack of information contributed to a mistaken assumption by operators that the installation of the tagout was appropriate during shutdown cooling operation with logic bypassed using procedurally allowed jumpers to prevent the SDC suction isolation. Operations also expressed a concern of potentially affecting SDC during the RPS work and requested a second technical review of the tagout. However, due to ineffective communication between outage scheduling and tagging groups, that review was not performed. Additionally, the scheduling group did not effectively communicate updates to the tagging group when subsequent outage schedule changes were made, thereby, missing an opportunity to assure that the tagging group was aware of critical schedule changes. For example, the outage scheduling group was aware that the RPS work should not be performed when SDC was required to be in-service as evidenced by outage schedule status entry dated June 5, 2008 that stated, “ensure work in window is reviewed for impact on SDC and scheduled appropriately…PCIS [primary containment isolation system] relays are in the respective RPS windows and the windows are tied to SDC out of service.”

The inspectors determined that the impact of scheduling the removal of the ‘B’ reactor protection system and primary containment isolation system at a time with high decay heat load was not fully understood or managed by station personnel. The potential risk of interrupting SDC flow to the reactor was not adequately managed as evidenced by the communication and procedural barriers that were not effective. Specifically, Entergy personnel did not adequately manage the increased risk with this evolution to properly implement procedure AP-10.09, “Outage Risk Assessment,” that states, “When shutdown cooling is required, there shall be NO work in progress that affects the availability of the common shutdown cooling suction flow path or either of the common RHR [residual heat removal] shutdown cooling suction valves.” The inspectors also noted that protected equipment signs and barriers were not adequately applied by the station.

The inspectors determined that the failure to manage the risk associated with the ‘B’ RPS work is a performance deficiency.

Analysis: This finding is more than minor because it is related to maintenance risk assessment and management. In this instance, Entergy did not implement prescribed significant compensatory measures and effectively manage those measures. Specifically, this finding reflects inadequate risk management that contributed to a short duration loss of shutdown decay heat removal capability resulting from the inadvertent interruption of flow through the operating train of shutdown cooling during cavity flood-up, in preparation for refueling. This was reasonably within Entergy's ability to foresee and prevent because there were opportunities to recognize and manage the potential risk of losing shutdown cooling and schedule the maintenance activity at a more appropriate maintenance window or take actions to prevent the loss of shutdown cooling.

In accordance with IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and Appendix G, "Shutdown Operations Significance Determination Process," the inspectors determined this finding was of very low safety significance (Green). The basis for this determination is that in accordance with IMC 0609, Appendix G, Table 1, "Losses of Control," and Checklist 8, "BWR Cold Shutdown or Refueling Operation Time to Boil > 2 Hours: RCS Level < 23' Above Top of Flange," this finding did not require quantification and did not constitute a significant loss of thermal margin, based upon the slow reactor coolant system heat-up rate and minimal time of interruption in shutdown cooling system operation. The problem was entered into Entergy's corrective action program as CR-JAF-2008-02997. Entergy implemented corrective actions that included communicating the error to personnel and providing additional oversight for tagouts affecting required safety systems.

The inspectors determined that this finding had a cross-cutting aspect in the area of human performance because Entergy did not plan and coordinate work activities properly to manage operational impact of work activities. Specifically, the impact on shutdown cooling of deenergizing the 'B' reactor protection system was not properly assessed. (H.3(b))

Enforcement: 10 CFR Part 50.65 (a)(4), requires, in part, that before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, on September 16, 2008, Entergy did not manage the increase in risk prior to removal of the 'B' reactor protection system. Removal of the 'B' reactor protection system resulted in a loss of shutdown cooling. Because this finding was of very low safety significance and was entered into Entergy's corrective action system as CR-JAF-2008-02997, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000333/2008004-01, Failure to Manage Risk During Maintenance Activity Resulted in Loss of Shutdown Cooling.**

1R22 Surveillance Testing (71111.22 - 8 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 documents; test instrumentation had current calibrations, adequate range, and accuracy for the application; and tests were performed, as written, with applicable prerequisites satisfied. Upon ST completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. The following STs were reviewed and represented eight inspection samples:

- ST-9BB, "B and D Full Load Test and ESW Pump Operability Test, " Revision 9;
- ST-2AL, "RHR Loop A Quarterly Operability Test (IST)," Revision 27;
- ST-9LA, "EDG A & C Fuel Oil Transfer Pump Operability Test," Revision 6;
- ST-4E, "HPCI and SGT Logic System Functional and Simulated Automatic Actuation Test," Revision 52;
- ST-3F, "Core Spray Full Flow Test (IST)," Revision 3;
- ST-39B-X7D, "Type C Leak Test Main Steam Line D MSIVs (IST)," Revision 9;
- ST-39B-X205, "Type C Leak Test of Torus Purge Exhaust Line Valves (IST)," Revision 6; and
- ST-39B-X202, "Type B Leak Test of Drywell to Torus Vacuum Breakers," Revision 5.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing (71114.02 - 1 sample)

a. Inspection Scope

An onsite review was conducted to assess the maintenance and testing of Entergy's alert and notification system (ANS). During this inspection, the inspectors interviewed emergency preparedness staff responsible for implementation of the ANS testing and maintenance. The inspectors reviewed CRs pertaining to the ANS for causes, trends, and Entergy's corrective actions. The inspectors further discussed the ANS with the assigned technical specialist, reviewing system performance from January 2007 through June 2008. The inspectors reviewed the ANS procedures and the ANS design report to ensure Entergy's compliance with those commitments for system maintenance and testing. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 2. 10 CFR Part 50.47(b)(5) and the related requirements of 10 CFR Part 50, Appendix E, were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation (71114.03 - 1 sample)

a. Inspection Scope

A review of Entergy's emergency response organization (ERO) augmentation staffing requirements and the process for notifying the ERO was conducted. This was performed to ensure the readiness of key staff for responding to an event and to ensure timely facility activation. The inspectors reviewed procedures and CRs associated with the ERO notification system and drills, and reviewed records from call-in drills. The inspectors interviewed personnel responsible for testing the ERO augmentation process, and reviewed the training records for a sampling of ERO to ensure training and qualifications were up to date. The inspectors reviewed procedures for ERO administration and training, and verified a sampling of ERO participation in drills and exercises conducted in 2007 and 2008. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 3. 10 CFR Part 50.47(b)(2) and related requirements of 10 CFR Part 50 Appendix E were used as reference criteria.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Prior to this inspection, the NRC had received and acknowledged changes made to Entergy's emergency plan and implementing procedures. Entergy developed these changes in accordance with 10 CFR Part 50.54(q), and determined that the changes did not result in a decrease in effectiveness to the emergency plan. Entergy also determined that the emergency plan continued to meet the requirements of 10 CFR Part 50.47(b) and Appendix E to 10 CFR Part 50. During this inspection, the inspectors conducted a review of Entergy's 10 CFR Part 50.54(q) screenings for all the changes made to the emergency action level and all of the changes made to the emergency plan from April 2007 through July 2008 that could potentially result in a decrease in effectiveness. This review of the emergency action level and emergency plan changes did not constitute NRC approval of the changes and, as such, the changes remain subject to future NRC inspection. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 4. The requirements in 10 CFR Part 50.54(q) were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05 – 1 sample)

a. Inspection Scope

The inspectors reviewed a sampling of self-assessment procedures and reports to assess Entergy's ability to evaluate their performance and programs. The inspectors reviewed a sampling of CRs from January 2007 through July 2008 initiated by Entergy at FitzPatrick from drills, self-assessments and audits. Other drill reports reviewed include: medical/health physics, fire, integrated and call-in. Additionally, the inspectors reviewed audits for 2007 and 2008 required by 10 CFR Part 50.54(t). This inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 5. Planning Standard, 10 CFR Part 50.47(b)(14) and the related requirements of 10 CFR Part 50 Appendix E were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06 – 1 sample)

a. Inspection Scope

The inspectors observed simulator activities associated with licensed operator requalification training on August 18, 2008. The inspectors verified that emergency classification declarations and notification activities were properly completed. The inspectors evaluated the drill against the requirements of 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities." The inspectors observed Entergy's critique and compared Entergy's self-identified issues with observations from the inspectors' review to ensure that performance issues were properly identified. This evaluation represented one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01 – 21 samples)

a. Inspection Scope

During August 25 through 29, 2008, the inspectors conducted the following activities to verify that Entergy was properly implementing physical, engineering, and administrative controls for access to high radiation areas, and other radiologically controlled areas, and that workers were adhering to these controls when working in these areas. Implementation of the access control program was reviewed against the criteria contained in 10 CFR Part 20, Technical Specifications, and Entergy's procedures. Activities (15) through (21) were conducted September 22 through 26, 2008 during the refueling outage.

(1) There were no occupational exposure cornerstone performance indicator (PI) incidents during the current assessment period.

(2) The inspectors walked down accessible exposure significant work areas of the plant and reviewed licensee controls and surveys to determine if licensee surveys, postings, and barricades were acceptable and in accordance with regulatory requirements.

(3) The inspectors walked down accessible exposure significant work areas of the plant and conducted independent surveys to determine whether prescribed radiation work permits and procedural controls were in place and whether licensee surveys and postings were complete and accurate.

(4) During 2008, there were no internal dose assessments >10 mrem committed effective dose equivalent (CEDE) and therefore, no assessment of internal exposure calculations was performed.

(5) Entergy's physical and programmatic controls for highly activated materials stored underwater in the spent fuel pool was reviewed and evaluated through observation and a review of the applicable access control procedure.

(6) A review of Entergy's radiation protection program self-assessments and audits during 2008 was conducted to determine if identified problems were entered into the corrective action program for resolution.

(7) Ten condition reports associated with the radiation protection access control and 'as low as is reasonably achievable (ALARA)' areas between April 2008 and August 2008, were reviewed and discussed with licensee staff to determine if the follow-up activities were being conducted in an effective and timely manner commensurate with their safety significance.

(8) Based on the condition reports reviewed, repetitive deficiencies were screened to determine if Entergy's self-assessment activities were identifying and addressing these deficiencies.

(9) There were no occupational exposure PI incidents reported during the current assessment period to evaluate utilizing the SDP.

(10) Changes to the high radiation area and very high radiation area procedures since the last inspection in this area were reviewed and management of these changes were discussed with the Radiation Protection Manager.

(11) Controls associated with potential changing plant conditions to anticipate timely posting and controls of radiation hazards was discussed with a radiation protection supervisor.

(12) All accessible locked high radiation area entrances in the plant were verified to be locked through challenging the locks or doors. All locked and very high radiation area keys were inventoried and controls reviewed.

(13) Several radiological condition reports were reviewed to evaluate if the incidents were caused by radiation worker errors and determine if there were any trends or patterns and if Entergy's corrective actions were adequately addressing these trends.

(14) Several radiological condition reports were reviewed to evaluate if the incidents were caused by radiation protection technician errors and determine if there were any trends or patterns and if Entergy's corrective actions were adequately addressing these trends.

(15) Radiation work permits (RWPs) that provide access to exposure significant areas of the plant, including high radiation areas were reviewed. Specified electronic personal dosimeter alarm set points were reviewed with respect to current radiological condition applicability and workers were queried to verify their understanding of plant procedures governing alarm response and knowledge of radiological conditions in their work area.

(16) There was no radiation work performed in airborne radioactivity areas with the potential for individual worker internal exposures of >50 mrem CEDE.

(17) During September 22 through 26, 2008, the following radiologically significant work activities were selected; the radiological work activity job requirements were reviewed; and work activity job performance was reviewed with respect to the radiological work requirements.

In-service inspection: UT of various reactor vessel nozzle penetration welds (N2E, N2C, N8A);

- In-vessel visual inspection and shroud cleaning activities;
- Control rod drive replacement;
- Safety relief valve replacement; and
- Radiation protection support in the drywell.

(18) During observation of the work activities listed in (17) above, the adequacy of surveys, job coverage and contamination controls were reviewed.

(19) The adequacy of effectively monitoring occupational dose in work areas of significant dose gradients requiring relocation of dosimetry was reviewed for control rod drive replacement activities.

(20) During observation of the work activities listed in (17) above, radiation worker performance was evaluated with respect to the specific radiation protection work requirements and their knowledge of the radiological conditions in their work areas.

(21) During observation of the work activities listed in (17) above, radiation protection technician work performance was evaluated with respect to their knowledge of the radiological conditions, the specific radiation protection work requirements and radiation protection procedures.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

During August 25 through 29, 2008, the inspectors conducted the following activities to verify that Entergy was properly maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors reviewed implementation of the ALARA program against the criteria contained in 10 CFR Part 20.1101(b) and Entergy's procedures. Activities (5), (6), and (7) were conducted September 22 through 26, 2008, during the refueling outage.

(1) The highest exposure significant outage ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements were reviewed for the upcoming September 2008 refueling outage.

(2) The assumptions and bases for the September 2008 refueling outage collective exposure estimates were reviewed. This review involved the detailed preparation of exposure estimates based on dose rate and man-hour estimates for the highest exposure significant outage work activities.

(3) There were no declared pregnant workers during 2008. Therefore Entergy performance in this area was not observed.

(4) Radiation protection related condition reports were reviewed between April 2008 and August 2008 for repetitive deficiencies in ALARA to determine if Entergy's self-assessment activities were identifying and addressing these deficiencies.

(5) The following highest exposure work activities for the September 2008 refueling outage were selected for review:

- In-service inspection: ultrasonic examination of various reactor vessel nozzle penetration welds (N2E, N2C, N8A);
- In-vessel visual inspection and shroud cleaning activities;

- Control rod drive replacement;
- Safety relief valve replacement; and
- Radiation protection support in the drywell.

(6) With respect to the work activities listed in (5) above, these job sites were observed to evaluate if surveys and ALARA controls were implemented as planned.

(7) With respect to the work activities listed in (5) above, radiation worker and radiation protection technician performance was observed during the performance of these work activities to demonstrate the ALARA principles.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification (71151- 10 samples)

a. Inspection Scope

The inspectors reviewed PI data for the cornerstones listed below and used Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, to verify individual PI accuracy and completeness.

Cornerstone: Initiating Events

- Unplanned Scrams;
- Unplanned Power Changes; and
- Unplanned Scrams with Complications.

The inspectors reviewed Entergy’s event reports, operator logs, and PI data sheets to determine whether Entergy adequately identified the number of reactor scrams and unplanned power changes greater than 20 percent that occurred between July 2007 and June 2008. This number was compared to the number reported for the PI during the applicable quarter. The inspectors also verified the accuracy of the number of critical hours reported.

Cornerstone: Barrier Integrity

- Reactor Coolant System Leak Rate; and
- Reactor Coolant System Specific Activity.

The inspectors reviewed operator logs, plant computer data, chemistry records, and procedure ST-40D, “Daily Surveillance and Channel Check,” to verify the accuracy of

Entergy's reported maximum reactor coolant system identified leakage and specific activity between July 2007 and July 2008.

Cornerstone: Emergency Preparedness

- Drill/Exercise Performance;
- Emergency Response Organization Drill Participation; and
- Alert and Notification System Reliability.

The inspectors reviewed supporting documentation from drills and tests for April 2007 through March 2008, to verify the accuracy of the reported data. Additional acceptance criteria used for the review was included in 10 CFR Part 50.9.

Cornerstone: Occupational Radiation Safety

- Occupational Exposure Control Effectiveness

The inspectors reviewed implementation of Entergy's Occupational Exposure Control Effectiveness PI Program. Specifically, the inspectors reviewed CRs, and radiological controlled area dosimeter exit logs for the past four calendar quarters. These records were reviewed for occurrences involving locked high radiation areas, very high radiation areas, and unplanned exposures against the criteria specified in NEI 99-02, Revision 5.

Cornerstone: Public Radiation Safety

- RETS/ODCM Radiological Effluent Occurrence

The inspectors reviewed a listing of relevant effluent release reports for the past four calendar quarters, for issues related to the public radiation safety PI, which measures radiological effluent release occurrences per site 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 review was conducted against applicable criteria specified in NEI 99-02, Revision 5. The inspectors reviewed the following documents to ensure the licensee met all requirements of 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; and
- dose assessment procedures.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, “Identification and Resolution of Problems,” to identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of all items entered into 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 procedures, the inspectors selected items across the initiating events, mitigating systems, and barrier integrity cornerstones for additional follow-up and review. Additionally, an NRC Senior Health Physicist reviewed 22 CRs that were initiated between April 2008 and August 2008 and were associated with the radiation protection program. The inspectors assessed Entergy’s threshold for problem identification, the adequacy of the cause analyses, and extent of condition review, operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are listed in the Attachment.

b. Assessment and Observations

No findings of significance were identified. The inspectors determined that Entergy appropriately identified equipment, human performance and program issues at an appropriate threshold and entered them into the corrective action program.

.2 Annual Sample: Category ‘D’ Dissimilar Metal Welds (71152 -1 sample)

a. Inspection Scope

The inspectors reviewed Entergy’s plans for completing examinations of Category ‘D’ dissimilar metal welds that were not previously examined in accordance with the requirements of ASME Section XI, Appendix VIII, Supplement 10, Performance Demonstration Initiative (PDI) program. These welds contain Inconel 182 weld metal and are susceptible to intergranular stress corrosion cracking. The inspectors verified that the examination data for these Category ‘D’ dissimilar metal welds was reviewed, evaluated and corrective actions developed in conformance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition with 2003 Addenda; licensing commitments; Fitzpatrick risk-informed inservice inspection program; and Boiling Water Reactor Vessel Internal Program (BWRVIP)-75-A guidance recommendations.

The inspectors conducted a review of condition reports (CR) 2006-00004 and CR 2007-00417 in response to industry operating experience related to dissimilar metal weld data reviews and flaws found in dissimilar metal welds at several facilities. The inspectors reviewed Entergy’s Intergranular Stress Corrosion Cracking Program (Refueling Outage R18 Selection/Scope) and commitment tracking COM-2008-00008, including other documents which are listed in the attachment to this report, to verify that the corrective actions to resolve the dissimilar metal welds examinations were adequate.

b. Assessment and Observations

No findings of significance were identified. The inspectors concluded that Entergy had appropriately reviewed the previous non-destructive examination data records of the Category 'D' dissimilar metal welds and the corrective actions to examine the Category 'D' dissimilar metal welds through the PDI program were considered adequate. The inspectors also determined that Entergy accelerated examination of reactor pressure vessel nozzle-to-safe-end dissimilar metal welds to be PDI examined during refueling outage (RO18) to 12 welds, with the four remaining to be examined in the next refueling outage.

.3 Annual Sample: Review of February 11, 2008 Transportation Incident (71152 - 1 sample)

a. Inspection Scope

One problem identification and resolution (PI&R) sample associated with the radioactive material transportation program was selected and reviewed by a regional inspector during August 25-29, 2008. This PI&R sample involved a February 11, 2008 NRC-licensed cask shipment to the Barnwell Disposal Facility that arrived with one of twelve cask closure bolts loose. Condition report (CR) 2008-1539 documented Entergy's investigation including an apparent cause evaluation and corrective actions to prevent recurrence of this issue.

b. Findings

Introduction: The inspectors identified a violation for failure to file a 60-day report with the NRC as required by 10 CFR Part 71.95 based on shipment receipt notification on February 15, 2008, that one of twelve cask closure bolts was found loose on a NRC-licensed cask shipment upon arrival at the shipping destination.

Description: On February 11, 2008, Entergy made a shipment of irradiated reactor hardware in an NRC-licensed cask (Model CNS 3-55-1) to the Energy*Solutions*™ Barnwell Low Level Radioactive Waste Disposal Facility in Barnwell, South Carolina. Upon receipt, the disposal site operator observed that one of the twelve cask lid closure bolts was loose. Energy*Solutions*™ informed station personnel at Fitzpatrick of the as-found condition by telephone on February 15, 2008; and indicated its intent to initiate a condition report to investigate this instance.

Entergy subsequently reviewed applicable procedures, records, and independent verification activities that were applied at Fitzpatrick, and the actions of the Energy*Solutions*™ contractors who actually prepared the package for shipment. Entergy determined that the package was prepared and shipped in accordance with the applicable procedure (DVP-TR-OP-019, Revision 24) and the Certificate of Compliance (No. 5805); and confirmed that all bolts were torqued to 75 ft-lbs (+/- 7 ft-lbs.), as required, upon shipment.

On April 10, 2008, Entergy received a written notification of the Energy*Solutions*™ as-found observation from the State of South Carolina Department of Health and Environmental Control, which informed Entergy of the State's concern in this matter, and requested that Entergy provide a description of corrective measures to prevent recurrence.

The inspectors determined that Entergy had not taken appropriate action to document these external communications for resolution, such as initiating a Condition Report, or reporting these observations to the NRC as specified by 10 CFR Part 71.95, "Reporting." The regulatory requirement specified reporting to be accomplished within 60 days of the occurrence, i.e., April 15, 2008, in this case.

Analysis: Failure of Entergy to report the condition, as required by 10 CFR Part 71.95, constitutes a performance deficiency in that the issue is the result of Entergy not meeting a regulatory requirement that was reasonably within Entergy's ability to foresee and correct, and should have been prevented. This violation involves a failure to make a required report to the NRC and is considered an impact to the regulatory process. Such violations are dispositioned using traditional enforcement process instead of the Significance Determination Process. Using the Enforcement Policy Supplement IV "Transportation", example D4 which states, "a noncompliance with shipping papers, marking, labeling, placarding, packaging or loading not amounting to a Severity Level I, II, or III violation;" the NRC determined this violation is categorized as a SL IV Violation. The Enforcement Policy Supplement I "Reactor Operations" examples D3, D4, and D5 are similar to this issue, in that they discuss examples of failures to make required reports for more than minor events, which are also categorized at Severity Level IV.

The finding was also reviewed using the Significance Determination Process to assess the insights afforded by this process. It was determined that this finding is more than minor because Entergy's failure to report a deficiency in the package shipped is associated with the Public Radiation Safety cornerstone attribute of transportation packaging, and affected the cornerstone objective of ensuring adequate protection of public health and safety from exposure to radioactive material as a result of the offsite transport of radioactive materials and wastes. Applying IMC 0609, Appendix D, "Public Radiation Safety SDP," the matter constitutes a finding in Radioactive Material Control relative to Transportation. In this case, radiation limits were not exceeded, and there was no package breach. However, the Certificate of Compliance was affected in that the matter did constitute a design documentation deficiency related to the use of an NRC-approved package. Specifically, the finding involved 10 CFR Part 71.95, "Reports." Accordingly, this finding is considered as having very low safety significance. Comparing this item to the examples in the Enforcement Policy Supplement I, this finding is similar to Item D.5 "Violations of 10 CFR 50.59 that result in conditions evaluated as having very low safety significance (i.e., green) by the SDP," which is an example of a Severity Level IV violation.

This finding has a cross-cutting aspect in the area of problem identification and resolution related - corrective action program, because Entergy performed an insufficient evaluation of a non-conforming condition associated with an NRC-approved package to assure the matter was properly classified, prioritized and evaluated relative to reportability. (P.1(c))

Enforcement: 10 CFR Part 71.95 requires licensee's to submit a written report, within 60 days, of instances in which the conditions of approval in the Certificate of Compliance were not followed during a shipment. Certificate of Compliance No. 5805, Section 9, was not followed in that one of the closure bolts was found loose, a non-conforming condition, upon receipt of the shipping package at Energy*Solutions*™ Barnwell Disposal Facility on February 14, 2008, since all twelve closure bolts were to be closed with 75+/-7 ft-lbs torque. Contrary to 10 CFR Part 71.95, Entergy failed to report the occurrence to the NRC within 60 days. While it is was determined that Entergy properly prepared the package for shipment and did not cause the non-conforming condition that was observed upon receipt by Energy*Solutions*™, Entergy's failure to report the condition to the NRC constitutes a finding of very low safety significance. This matter has been entered in to Entergy's corrective action program as condition report (CR)-2008-02772. Accordingly, this matter is being treated as a NCV of 10 CFR Part 71.95 consistent with Section VI.A of the NRC Enforcement Policy: **NCV 05000333/2008004-02, Failure to Make a Written Report of a Non-Conforming Condition Relative to an NRC-Approved Package.**

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period the inspectors conducted observations of security force personnel and activities to ensure that the activities were consistent with Entergy security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours. These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather they were considered an integral part of the inspectors' normal plant status reviews and inspection activities.

b. Findings

No findings of significance were identified.

.2 Institute of Nuclear Power Operations Plant Assessment Report Review

a. Inspection Scope

The inspectors reviewed the final report for the Institute of Nuclear Power Operations Plant Assessment of Entergy conducted in December 2007. The inspectors reviewed the report to ensure that issues identified were consistent with the NRC perspectives of licensee performance and to verify if any significant safety issues were identified that required further NRC follow-up.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

Exit Meeting Summary

On October 16, 2008, the inspectors presented the inspection results to Mr. Peter T. Dietrich and other members of his staff. The inspectors asked Entergy whether any of the material examined during the inspection should be considered proprietary. Entergy did not identify any material as proprietary information.

4OA7 Licensee-Identified Violations

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

- A violation of TS 5.4.1, "Procedures," occurred when Entergy did not provide an adequate maintenance procedure for work on a safety-related component. Maintenance procedure MP-93.06, "EDG Woodward Governor Actuator Maintenance," Revision 15, did not include adequate instructions to perform filling and venting of the hydraulic actuator of the 'D' EDG governor. The issue was entered into Entergy's corrective action program as CR-JAF-2008-02562. The issue was of very low safety significance because it was identified during the post-maintenance test and corrected prior to operability restoration.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

P. Dietrich, Site Vice President
C. Adner, Manager Operations
P. Cullinan, Manager, Emergency Preparedness
E. Dorman, Acting Licensing Manager
B. Finn, Director Nuclear Safety Assurance
D. Johnson, Manager, Training
J. LaPlante, Manager, Security
A. Mitchell, Manager, System Engineering
K. Mulligan, General Manager, Plant Operations
J. Solowski, Radiation Protection
M. Woodby, Director Engineering

LIST OF ITEMS OPEN, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed

05000333/2008004-01	NCV	Failure to Manage Risk During Maintenance Activity Resulted in Loss of Shutdown Cooling (Section 1R20)
05000333/2008004-02	NCV	Failure to Make a Written Report of a Non-Conforming Condition Relative to an NRC-Approved Package (Section 4OA2)

Closed

None

Discussed

None

LIST OF DOCUMENTS REVIEWED

<u>Section 1RO1: Adverse Weather Protection</u>
OP-51A, "Reactor Building Ventilation and Cooling System," Revision 47
OP-52, "Turbine Building Ventilation," Revision 16
DBD-066, "Design Basis Document for the Reactor Building Heating, Ventilation and Air Condition (HVAC) Systems"
DBD-067, "Design Basis Document for the Turbine Building HVAC Systems"

<u>Section 1RO4: Equipment Alignment</u>
<u>Procedures</u>
OP-13, "Residual Heat Removal System," Revision 93
OP-19, "Reactor Core Isolation Cooling System," Revision 46

<u>Drawings</u>
FM-20B, "Residual Heat Removal System 10," Revision 64

<u>Work Orders</u>
00141037
00120991

<u>Miscellaneous</u>
Quarterly system health reports Residual heat Removal system
DBD-010, "Design Basis Document for the Residual heat Removal System," Revision 12

<u>Section 1RO5: Fire Protection</u>
Fire Area/Zone IX/RB-1A, elevation 369 foot – PFP-PWR 28
Fire Area/Zone IE/TB-1 North, elevation 252 foot – PFP-PWR 42
Fire Area/Zone IE/TB-1 South, elevation 252 foot – PFP-PWR 43
Fire Area/Zone ISFSI Yard, elevation 272 foot – PFP-OUT 39
Fire Area/Zone II/CT-2, elevation 258 foot PFP- PWR 01
Fire Area/Zone IC/CT-1, elevation 258 foot PFP- PWR 02

<u>Section 1RO6: Flood Protection Measures</u>
JAF-RPT-MULTI-02107, "IPE Update, Appendix H, Internal Flooding Analysis," Revision 2

<u>Section 1RO8: Inservice Inspection Activities</u>
<u>Procedures</u>

VT-FPK-204V12, In-Vessel Visual Examination procedure, Revision 0
GEH-ADM-1061, AP for in-vessel inspection, Revision 1
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2007-01820	2008-02704	2008-0930
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LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ALARA	as low as is reasonably achievable
ANS	alert and notification system
AOV	air-operated valve
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CEDE	committed effective dose equivalent
CFR	Code of Federal Regulations
CR	condition report
EDG	emergency diesel generator
Entergy	Entergy Nuclear Northeast
ERO	emergency response organization
FitzPatrick	James A. FitzPatrick Nuclear Power Plant
IMC	inspection manual chapter
IST	inservice test
kV	kilovolt
MOV	motor-operated valve
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OA	other activities
PARS	Publicly Available Records
PDI	performance demonstration initiative
PI	performance indicator
PI&R	problem identification and resolution
RO18	refueling outage 18
RWP	radiation work permit
SDC	shutdown cooling
SDP	significance determination process
SL	Severity Level
SSC	structures, systems, or components
ST	surveillance test
TS	technical specification
UFSAR	updated final safety analysis report
UT	ultrasonic testing