February 1, 2006

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000333/2005006

Dear Mr. Sullivan:

On December 31, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your James A. FitzPatrick Nuclear Power Plant. The enclosed integrated inspection report documents the inspection findings which were discussed on January 13, 2006, with you and other members of your staff.

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

This report documents three findings of very low safety significance (Green). All three findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating the findings as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. 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, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at FitzPatrick.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) will be available electronically for public inspection in the

NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web Site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000333/2005006
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.:	05000333/2005006
Licensee:	Entergy Nuclear Northeast (Entergy)
Facility:	James A. FitzPatrick Nuclear Power Plant
Location:	268 Lake Road Scriba, New York 13093
Dates:	October 1 - December 31, 2005
Inspectors:	G. Hunegs, Senior Resident Inspector D. Dempsey, Resident Inspector K. Diederich, Reactor Inspector J. Lilliendahl, Reactor Inspector J. Noggle, Senior Health Physicist
Approved by:	Brian J. McDermott, Chief Projects Branch 2 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000333/2005-006; 10/01/2005 - 12/31/2005; James A. FitzPatrick Nuclear Power Plant; Evaluation of Changes, Tests or Experiments, Maintenance Effectiveness, and Maintenance Risk Assessment and Emergent Work Control.

This report covers a three-month period of inspection by resident inspectors, and announced inspections by three regional specialist inspectors. Three Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A Green self-revealing non-cited violation (NCV) of Technical Specification (TS) limiting condition for operation (LCO) 3.8.1, "Electrical Power Systems - AC Sources - Operating," occurred for Entergy's failure to comply with the LCO required actions for one inoperable offsite power circuit. The performance deficiency is that the condition of Line 4 was not effectively monitored such that the degraded phase A bus bar was not identified. This resulted in exceeding the TS 3.8.1 allowed outage time. This issue was entered into the corrective action program. The bus bar was repaired and a process to monitor bus voltage was implemented. Long term corrective actions are under development.

  The finding is greater than minor significance because it is associated with the Initiating Events Cornerstone attribute of configuration control and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the finding is determined to be of very low risk significance because as a transient initiator it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Because this finding is of very low safety significance and has been entered in Entergy's corrective action program, the violation is being treated as a non-cited violation. (Section 1R13)

Cornerstone: Mitigating Systems

- Green. A Green (Severity Level IV) non-cited violation of 10 CFR 50.59 was identified for failure to perform an adequate safety evaluation (SE) of a change to the facility. Specifically, Entergy's SE did not adequately evaluate the

potential for a malfunction with a different result associated with the elimination of safety relief valve (SRV) accumulator check valve leakage testing. The issue was entered into the corrective action program. An operability evaluation concluded that the equipment was operable and additional corrective actions are under review.

Entergy's less than adequate 10 CFR 50.59 safety evaluation constitutes a performance deficiency. This finding has been addressed using traditional enforcement since it potentially impacted or impeded the regulatory process in that a required 10 CFR 50.59 evaluation was not adequate. This is contrary to the regulatory process that allows licensees to make changes without a license amendment provided that licensees comply with the 10 CFR 50.59 process. The finding is greater than minor, because there was a reasonable likelihood that the change would have required NRC review and approval prior to implementation. This finding was evaluated using the SDP for the mitigating systems cornerstone and was determined to be a finding of very low safety significance (Green), because it did not impact operability of the SRVs, and was not potentially risk-significant due to possible external events. Because this finding is of very low safety significance and has been entered in Entergy's corrective action program, the violation is being treated as a non-cited violation. (Section 1R02)

- Green. A Green NRC-identified non-cited violation of 10 CFR 50.65(a)(2) was identified for a failure to demonstrate that the performance of the back-up diesel-driven fire pump 76P-4 was being effectively controlled through the performance of appropriate preventive maintenance. Specifically, the pump did not complete its surveillance runs on at least four occasions between October 2003 and December 2005 due to fouling of the diesel engine cooling water strainer. To address this, maintenance was performed in each case to clean the strainer. However, this maintenance did not prevent recurrence and did not ensure the pump remained capable of performing its intended function. The issue was entered into the corrective action program and corrective actions are under review. The finding is associated with the cross cutting area of problem identification and resolution since there were repetitive failures of the back-up diesel driven fire pump.

  The finding is more than minor, because the performance of the component was degraded, and that the degraded performance affected the objectives of the Mitigating Systems Cornerstone. Specifically, the continued reliability of the pump was affected. The inspectors evaluated this finding using the site-specific Phase 2 SDP worksheets. This analysis showed the safety significance to be very low based on alternate sources remaining available. Because this finding is of very low safety significance and has been entered in Entergy's corrective action program, the violation is being treated as a non-cited violation. (Section 1R12)

B. Licensee-Identified Violations

None

REPORT DETAILS

Summary of Plant Status

The reactor operated at or near 100 percent power for the entire inspection period.

1. **REACTOR SAFETY**

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors reviewed and verified completion of the operations department cold weather preparation checklist contained in procedure AP-12.04, “Seasonal Weather Preparations.” The inspectors reviewed the operating status of outdoor facilities and the reactor and turbine building ventilation systems, reviewed the procedural limits and actions associated with low lake temperature, and walked down accessible areas of the buildings to assess the effectiveness of the ventilation systems. The walkdowns included discussions with operations and engineering personnel to ensure that they were aware of temperature restrictions and required actions, and constituted one inspection program sample. The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R02 Evaluation of Changes, Tests or Experiments (71111.02 - 18 samples)

a. Inspection Scope

The inspectors reviewed five safety evaluations (SE), which constituted inspection program samples, among the Initiating Events, Barrier Integrity and Mitigating Systems cornerstones to verify that changes and tests were reviewed and documented in accordance with 10 CFR 50.59, “Changes, Tests, and Experiments,” and that NRC approval was obtained prior to implementation if required. The inspectors assessed SE adequacy through interviews with plant staff and review of supporting information such as calculations and analyses, design change documentation, procedures, the Updated Final Safety Analysis Report (UFSAR), technical specifications (TSs), and plant drawings. In order to evaluate JAF-SE-03-003, the inspectors accompanied a reactor operator in a simulated swap of the instrument nitrogen supply trains at the containment atmosphere dilution (CAD) control panel and walked down portions of the CAD system, including the nitrogen storage tanks.

The inspectors also reviewed 13 changes, which constituted inspection program samples, that Entergy had screened and determined to be outside of the scope of 10 CFR 50.59. The inspectors performed this review to assess if Entergy’s conclusions with respect to 10 CFR 50.59 applicability were appropriate. Issues that were screened out included design, procedure, and setpoint changes, and temporary alterations.

In addition, the inspectors reviewed the administrative procedures for screening, preparation, and issuance of SEs to ensure that the procedures adequately implemented the requirements of 10 CFR 50.59.

The SE's, 50.59 screens, and other documents that were reviewed are listed in the Attachment.

b. Findings

Introduction. The team identified a Green (Severity Level IV) non-cited violation (NCV) of 10 CFR 50.59 for failure to perform an adequate safety evaluation of a change to the facility. Specifically, Entergy's SE did not adequately evaluate the potential for a malfunction with a different result associated with the elimination of safety relief valve (SRV) accumulator check valve leakage testing.

Description. Following the Three Mile Island accident, the NRC required licensees to demonstrate (1) that their SRV accumulators were adequate to meet design requirements, and (2) that the design requirements were adequate to support a 100-day post-loss of coolant accident (LOCA) mission time involving the use of SRVs for long-term alternate core cooling. In response to this requirement, FitzPatrick committed to leak test the SRV accumulators each refueling outage and to upgrade the drywell pneumatic supply (DPS) subsystem to provide redundant safety-related supply trains. In June 2003, Entergy approved JAF-SE-03-003 which eliminated SRV accumulator check valve leakage testing. By not testing the SRV accumulator check valves, Entergy relied on DPS system capacity to assure short and long-term SRV functionality. The short-term SRV function enables operators to rapidly depressurize the reactor coolant system for low pressure emergency core cooling system operation.

The inspectors questioned the adequacy of JAF-SE-03-003 regarding the creation of a possibility for a malfunction of a system important to safety with a different result. Specifically, assuming a single active failure of the in-service DPS train, operator action would be required to align the redundant standby train. Given this potential nitrogen supply interruption, Entergy did not fully evaluate whether the combined drywell DPS ring header nitrogen inventory (one common header that supplies the SRV accumulators and four inboard main steam isolation valve (MSIV) accumulators) was adequate to ensure continued operability of the seven automatic depressurization system (ADS) valves for one hour post-LOCA. In particular, Entergy did not fully evaluate the potential depressurization of the ring header from either closing the inboard MSIVs or system leakage within existing acceptance criteria.

Entergy entered this issue into its corrective action program as CR 2005-04711. A prompt operability evaluation concluded that the DPS was fully capable of maintaining an adequate pneumatic supply to the SRVs. Entergy's determination was based on: (1) assumed minimal accumulator check valve leakage based on historical leakage test results; (2) the upgraded, reliable, seismically qualified design of the ADS pneumatic supply; (3) historical trending of quarterly DPS leakage monitoring per ST-22D, “Nitrogen Instrument Header Integrity Test,” and daily DPS nitrogen usage monitoring per ST-40D, “Daily Surveillance and Channel Check;” (4) the seismic qualification and preventive maintenance of active DPS components; and (5) the ability to restore DPS following a single active failure (control room annunciated low pressure alarm, controlled procedure to realign trains, and post-LOCA accessibility). Entergy also initiated an engineering request to provide a leakage acceptance criterion for daily DPS usage monitoring per ST-40D. Entergy’s long-term corrective actions include the re-evaluation of JAF-SE-03-003 SE either to adequately support the change or to reinstate accumulator check valve testing.

Analysis. The team determined that Entergy’s less than adequate 10 CFR 50.59 SE constituted a performance deficiency. This finding was addressed using traditional enforcement since it potentially impacted or impeded the regulatory process in that a required 10 CFR 50.59 evaluation was not adequate. This is contrary to the regulatory process that allows licensees to make changes without a license amendment provided that licensees comply with the 10 CFR 50.59 process. The finding is greater than minor, because there was a likelihood that the change would have required NRC review and approval prior to implementation. This finding was evaluated using the Phase I significance determination process (SDP) for the mitigating systems cornerstone and was determined to be a finding of very low safety significance (Green), because it did not result in a loss of function per existing operability determination guidance (NRC Inspection Manual Part 9900 Technical Guidance, dated September 26, 2005) and was not potentially risk-significant due to possible external events. Since the finding is being addressed under traditional enforcement and was determined to have very low safety significance it is categorized as Severity Level IV, consistent with Supplement I.D of the NRC Enforcement Policy.

Enforcement. 10 CFR 50.59 defines changes to the facility that require detailed evaluations to determine whether the changes can be implemented without obtaining prior NRC approval. Contrary to the above, Entergy implemented a change to the facility that required a detailed evaluation without performing an adequate 10 CFR 50.59 analysis that addressed all of the criteria in the regulation. Specifically, on June 19, 2003, Entergy approved a change to the facility as described in the UFSAR without an adequate evaluation to determine that the elimination of SRV accumulator check valve leakage testing did not create the potential for a malfunction with a different result. Because the failure to provide an adequate written evaluation of the potential to depressurize the DPS ring header is of very low safety significance, and has been entered into Entergy’s corrective action program as CR 2005-04711, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. **NCV 05000333/2005006-01 Inadequate 10 CFR 50.59 Safety Evaluation Associated with Safety Relief Valves.**

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

a. Inspection Scope

Partial System Walkdown. The inspectors performed three partial system walkdowns, each constituting inspection program samples, to verify equipment alignment and to identify any discrepancies that could potentially increase risk, cause initiating events, or impact the system operability. 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 partial walkdowns of the following systems:

- Train A and B emergency diesel generators (EDG) inspected on October 5 while the "D" EDG was out of service for preventive maintenance;
- "B" and "C" service water pumps and strainers on November 9 while the "A" service water pump and strainer were out of service for preventive maintenance and modification; and
- "A" residual heat removal (RHR) system following maintenance on October 19.

Complete System Walkdown. The inspectors performed a complete walkdown of the ADS to identify any discrepancies between the existing equipment lineup and the required lineup. This walk down constituted one inspection sample. 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 (TM), operator work-arounds, 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 documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q - 8 samples, 71111.05A - 1 sample)

a. Inspection Scope

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

- Motor Generator Set Room;
- Reactor Building, Elevation 369';
- East and West Cable Tunnels;
- A EDG and Emergency Switchgear Rooms;
- Control Room Heating, Ventilation, and Air Conditioning Room;
- Diesel Fire Pump Rooms;
- North and South Cable Tunnels; and
- Relay Room

Annual. The inspectors observed a fire brigade drill on December 15, 2005, including performance of the drill and the post-drill critique, and reviewed the disposition of issues and deficiencies that were identified. This fire brigade drill observation constituted one inspection program sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On November 29, 2005, the inspectors observed licensed operator simulator training to assess operator performance during several scenarios. The inspectors evaluated the performance of risk significant operator actions, including the use of emergency operating procedures (EOPs). 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. This observation of operator simulator training constituted one inspection program sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 2 samples, 71111.12B - 4 samples)

.1 Quarterly Review

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. Reviews focused on: proper Maintenance Rule (MR) scoping in accordance with 10 CFR 50.65; characterization of reliability issues; changing system and component unavailability; 10 CFR 50.65 (a)(1) and (a)(2) classifications; identifying and addressing common cause failures, trending key parameters, and the appropriateness of performance criteria for SSCs classified (a)(2) as well as the adequacy of goals and corrective actions for SSCs classified (a)(1). The inspectors reviewed system health reports, maintenance backlogs, and MR basis documents. The following two maintenance rule samples were reviewed:

- Primary containment atmosphere control and dilution system;
- Emergency service water system.

The documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.2 Biennial Review

a. Inspection Scope

The inspectors assessed the effectiveness of the Entergy's 10 CFR 50.65 (a)(3) periodic evaluation, and the resulting adjustments or corrective actions performed since the last inspection. The periodic evaluation covered the period from November 2003 to October 2005, and the inspectors confirmed that it met the periodicity requirements, and that it adequately evaluated performance monitoring activities, associated goals, and preventive maintenance activities.

To aid in determining the effectiveness of the Entergy's (a)(3) activities, four maintenance rule in-scope SSCs that had suffered degraded performance or condition were reviewed, based on SSC performance or condition, plant specific risk assessment, past inspection results, and operating experience. Reviews of each of the following systems were considered inspection program samples:

- Recirculation Flow Control
- Main Steam
- Feedwater
- Fire Protection

The inspectors conducted the review to verify that: performance of SSCs was being effectively monitored against licensee-established goals which took into account industry operating experience where practical; that goals and performance criteria were appropriate; that balancing of reliability and availability was given adequate consideration; that corrective action plans were adjusted appropriately when performance of SSCs did not meet established goals; that the monitoring was sufficient to provide reasonable assurance that SSCs are capable of fulfilling their intended functions; that monitoring plans were appropriately closed; that performance of SSCs was being effectively controlled through the performance of appropriate preventive maintenance; and that problem identification and resolution of maintenance rule-related issues were addressed.

The inspectors walked down accessible portions of the selected SSCs, interviewed the maintenance rule coordinator and system engineers, and reviewed documentation for applicable systems. The documents that were reviewed are listed in the attachment.

The inspectors reviewed a sample of condition reports related to maintenance effectiveness and to selected SSCs to ensure that problems were identified at an appropriate threshold, characterized, and that adequate corrective actions were implemented.

b. Findings

Introduction. A Green NRC-identified NCV of 10 CFR 50.65(a)(2) was identified for a failure to demonstrate that the performance of the back-up diesel-driven fire pump 76P-4 was being effectively controlled through the performance of appropriate preventive maintenance. Specifically, the pump did not complete its surveillance runs on at least four occasions from 10/16/03 to 12/20/05 due to fouling of the diesel engine cooling water strainer. To address this, maintenance was performed in each case to clean the strainer. However, this maintenance did not prevent recurrence and did not ensure the pump remained capable of performing its intended function.

The finding is more than minor, because the performance of the component was degraded, and that the degraded performance affected the objectives of the Mitigating Systems Cornerstone. Specifically, the continued reliability of the pump was affected. The inspectors evaluated this finding using the site-specific Phase 2 SDP worksheets. This analysis showed the safety significance to be very low based on alternate sources remaining available. The finding is associated with the cross cutting area of problem identification and resolution since there were repetitive failures of the back-up diesel driven fire pump.

Description. Fire protection pump 76P-4 is a diesel-driven fire pump which is used in the Emergency Operating Procedures (EOPs) during certain loss of AC bus scenarios. It is monitored under the Maintenance Rule and is designated risk significant in the FitzPatrick scoping basis document.

To ensure that the fire pump and its engine can meet operational demands, monthly surveillance runs are conducted. These runs are at least 20 minutes in length. On the following dates: October 16, 2003, January 7, 2004, November 22, 2005,, and December 20, 2005, CRs documented fouling of the diesel engine cooling water strainer resulting in either loss of flow or impending loss of flow. In each of these cases, the engine was shut down and maintenance similar to a section of the annual preventive maintenance to clean the cooling water strainer was performed to address the issue.

The backup diesel-driven fire pump suction is susceptible to some types of debris due to its location. Certain environmental conditions appear to cause mollusk shell debris in the pump intake. In each of the above cases, this led to blockage or impending blockage of cooling water flow through the strainer in 20 minutes or less. The cooling water supplies the engine heat exchanger and oil cooler, and its loss would rapidly lead to engine failure.

Although an alternate strainer is available and its use approved in current procedures, the inspector noted that the alternate strainer is similar to the primary strainer and it could not be concluded the mission time could be fulfilled. Further, current operating procedures do not contain a coping strategy to address fouled strainers.

Analysis. The performance deficiency is that the backup diesel-driven fire protection pump was not being effectively controlled through the performance of appropriate preventive maintenance such that it remained capable of performing its intended function. It is reasonable that Entergy could have recognized and prevented this problem. The inspectors determined that this finding is more than minor because the performance of the component was degraded, and that the degraded performance affected the objectives of the Reactor Safety Strategic Performance Area of the Mitigating Systems Cornerstone as discussed in NRC IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening." Specifically, the continued reliability of 76P-4 was affected and the finding is therefore more than minor. The inspectors evaluated this finding in accordance with NRC IMC 0609, "Significance Determination Process." The SDP Phase 1 Worksheet called for a Phase 2 analysis since the finding represented an actual loss of safety function of non-technical specification equipment designated as risk significant for a period greater than 24 hours. The site-specific SDP Phase 2 dominating events are loss of offsite power (LOOP) and loss of safeguard AC bus 10600 in which the pump provides one of multiple sources for late injection and containment heat removal. These analyses showed the safety significance to be very low based on alternate sources for late injection and containment heat removal remaining available.

The finding is associated with the cross cutting area of problem identification and resolution since there were repetitive failures of the back up diesel driven fire pump.

Enforcement. 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," section (a)(2) states, "Monitoring as specified in paragraph (a)(1) of this section is not required where it has been demonstrated that the performance or condition of a structure, system, or component is being effectively controlled through the performance of appropriate preventive maintenance, such that the structure, system, or component remains capable of performing its intended function." Section (a)(1) states, in part, that the licensee shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components...are capable of fulfilling their intended functions. Contrary to the above, prior to December 20, 2005, Entergy failed to demonstrate that the performance of pump 76P-4 was being effectively controlled through the performance of appropriate preventive maintenance such that the component remained capable of performing its intended function. Between January 7, 2004, and December 20, 2005, Entergy failed to establish goals and monitor 76P-4 under paragraph (a)(1) or demonstrate that monitoring under (a)(1) was not required. Because the finding is of very low safety significance and has been entered into Entergy's corrective actions program as CR 2005-05276, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **NCV 05000333/2005006-002, Failure to Maintain Diesel-Driven Fire Pump Performance**.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 4 samples)

a. Inspection Scope

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

The following work weeks were reviewed:

- Week of October 24, that included emergent repairs to the refuel mast during cask loading operations.
- Week of November 7, that included refurbishment of the "A" service water pump and preventive maintenance on the "A" turbine building component cooling water pump;
- Week of November 14, that included planned maintenance on 115kV line #3, the "A" containment atmosphere dilution system, and "B" turbine building component cooling water pump; and
- Week of December 19, that included emergent repairs to the 115kV line #4 breaker 71BKR-10012 input bus bar.

b. Findings

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

Description. The FitzPatrick 115 kV emergency offsite power system consists of two independent offsite circuits, one supply by the Lighthouse Hill hydroelectric generating station (Line 3) and the other by the Nine Mile Point (NMP) Unit 1 switchyard (Line 4). Each line normally supplies power to two reserve station service transformers (RSSTs) through a normally-closed bus disconnect. The system was designed such that either line alone will supply both RSSTs that supply both safeguards buses under normal, shutdown, and design basis accident (LOCA) loads.

During weekly checks on December 19, 2005, National Grid (the local grid operator) identified a difference in Line 4 phase amperes and contacted the NMP Unit 1 control room. Using control room ammeters, NMP operators confirmed that Line 4 phase A was reading zero amperes and notified Entergy. Entergy investigated and identified that the phase A bus bar to 115 kV breaker 71BKR-10012 was disconnected and hanging down in the FitzPatrick switchyard. Entergy declared Line 4 inoperable on December 19, repaired the bus bar, and returned the line to service on December 20. Through review of computer data Constellation (the NMP operator) determined that Line 4 phase A had failed at 9:57 a.m. on November 29. The delay in identifying the degraded

condition of Line 4 until December 19 occurred because Entergy did not have an effective means to monitor the line and did not physically identify the broken bus bar during operator rounds in the switchyard. TS LCO 3.8.1 required action A, in part, states that when one offsite circuit is inoperable it must be restored to operable status in seven days. If required action A is not completed in the required time, action F states that the plant must be placed in hot shutdown in 12 hours and cold shutdown in 36 hours. Line 4 was restored at 3:35 p.m. on December 20, resulting in an outage time of approximately 20 days and 5 -1/2 hours. This exceeded the allowed outage time of TS LCO 3.8.1.
The bus bar was repaired and a process to monitor bus voltage was implemented. Long term corrective actions are under development.

Analysis. The performance deficiency was that the condition of Line 4 was not effectively monitored such that the degraded phase A bus bar was not identified, and it was reasonable that Entergy could have provided a method to verify Line 4 phase continuity. This resulted in exceeding the TS 3.8.1 allowed outage time. Traditional enforcement does not apply because the issue did not have an actual safety consequence or potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements. The issue was of greater than minor significance because it was associated with the Initiating Events cornerstone attribute of configuration control and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

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

Enforcement. TS LCO 3.8.1 requires in part that two qualified offsite power circuits be operable. TS LCO 3.8.1 required action A states that an inoperable offsite power circuit must be restored to operable status within seven days. If the required action and associated completion time of condition A are not met, TS LCO 3.8.1 required action F states that the plant must be placed in hot shutdown in 12 hours and cold shutdown in 36 hours. Contrary to the above, from November 29 to December 20, 2005, one qualified offsite power circuit was inoperable for greater than seven days and 36 hours and the plant was not placed in the cold shutdown condition. Because the violation is of very low risk significance and Entergy entered the deficiency into its corrective action program as CR-2005-05180, this finding is being treated as an NCV consistent with Section VI.A of the Enforcement Policy. **NCV 05000333/2005006-03, Failure to Comply with TS 3.8.1 Required Actions for One Offsite Power Circuit Inoperable.**

1R15 Operability Evaluations (71111.15 - 3 samples)

a. Inspection Scope

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

- CR-2003-02968, concerning residual heat removal service water (RHRSW) check valve back leakage;
- JENG-REO-03-0011, concerning RHRSW and emergency service water (ESW) pump operability under high ambient temperature conditions; and
- CR-2005-04711, concerning the potential for a loss of safety relief valve pneumatic supply and the loss of the ADS.

b. Findings

No findings of significance were identified.

1R16 Operator Work-Arounds (71111.16 - 1 sample)

a. Inspection Scope

The inspectors completed one operator workaround inspection sample. The inspectors evaluated individual cumulative effects of identified operator work-arounds on the functionality of the plants mitigating systems. The work-arounds were reviewed to determine the effect on the functional capability of the systems, or human reliability in responding to an initiating event; and to assess the potential effects on the operators' ability to implement abnormal or emergency procedures; and if operator workaround problems were captured in Entergy's corrective action program. The inspectors also reviewed Entergy's assessment of the cumulative effects of the identified work-arounds in accordance with ST-99H, "Operator Work Arounds Assessment."

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B - 8 samples)

a. Inspection Scope

The inspectors reviewed eight risk-significant plant modifications completed within the past two years, each constituting one inspection program sample. The review was performed to verify that: (1) the design bases, licensing bases, and performance capability of risk significant SSCs had not been degraded through the modifications; and, (2) the modifications performed during increased risk configurations did not place the plant in an unsafe condition. The listing of the modifications reviewed is provided in the Attachment.

The plant modifications were distributed among the Initiating Event, Mitigating Systems, and Barrier Integrity cornerstones. The inspectors interviewed plant staff and reviewed the design inputs, assumptions, and design calculations to assess design adequacy. The inspectors also reviewed field change notices that were issued during installation to confirm that the problems associated with the installation were adequately resolved. In addition, the inspectors reviewed post-modification testing, functional testing, and instrument and relay calibration records to determine readiness for operation. Finally, the inspectors reviewed the affected procedures, drawings, design basis documents, and UFSAR sections to verify that the affected documents were appropriately updated.

For the accessible components associated with the modifications, the inspectors also walked down the systems to detect possible abnormal installation conditions. The documents reviewed during this inspection are listed in the Attachment.

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 maintenance on plant systems was adequately addressed by control room and engineering personnel. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design basis documentation; that test instrumentation had current calibrations and the appropriate range and accuracy for the application; and that tests were performed, as written, with applicable prerequisites satisfied. Upon completion, the inspectors verified that equipment was returned to the proper alignment necessary to perform its safety function. The following five post maintenance test activities were reviewed, and constitute inspection program samples:

- WR JAF-04-13819, involving electrical preventive maintenance on “D” emergency diesel generator during the week of October 3. The retest was performed using TST-128D, “D EDG Governor Control Operability Test,” and ST-9BB, EDG “B” & “D” Full Load Test and ESW Pump Operability Test.”
- WR JF- 030155200, involving refurbishment of “A” service water pump during the week of November 7. The retest consisted of pump functional and vibration testing and motor monitoring using MP 059-.83, “Motor Power Monitoring, Testing, and Analysis.”
- WR JAF-05-33207, involving repair of containment atmosphere dilution nitrogen makeup isolation valve 27AOV-131A during the week of November 7. The retest was performed using ST-39B-X25-71, “Type C Leak Test of DW Purge Supply and Atmosphere Dilution Line Valves (IST),” and ST-25B, “CAD Nitrogen Injection and Valve Exercise Test (IST).”
- WR JAF-05-34383, involving cleaning of east diesel fire pump 76P-4 cooling lines and strainer during the week of November 21. The retest was performed using ST-76AC, “East Diesel Fire Pump 76P-4 Operational Check.”

- WR JAF-05-36156, involving repair of the bus bar lug on the input side of 115kV breaker 71BKR-10012 during the week of December 19. The retest was performed per the WR by measuring resistance across the lug connections using a digital low resistance ohmmeter.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 3 samples)

a. Inspection Scope

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

- ST-20C, “Control Rod Operability and HCU Cooling Water Supply Check Valve Reverse Flow Check (IST);”
- ST-76AC, “East Diesel Fire Pump 76P-4 Operational Check;” and
- ST-41F, “HVAC Control Valve Fail Position Test (IST).”

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152 - 1 sample)

.1 Annual Sample Review

a. Inspection Scope

The inspectors reviewed approximately 74 corrective action condition reports affecting the Occupational Radiation Safety Cornerstone that were initiated between September 2004 and October 2005. Several repetitive condition reports had been categorized as steam affected entries with annual collective dose increases from 5 person-rem during 2004 up to 14 person-rem from January 2005 through October in 2005. Another category of repetitive condition reports reported weekly departmental exposure goals being exceeded. These condition reports as well as others were screened to review that the issues were properly identified, appropriately evaluated, and appropriate corrective actions were specified to prevent recurrence. This review was performed with respect to licensee's procedures and the requirements of 10 CFR 50, Appendix B.

This annual PI&R review constitutes one inspection program sample.

b. Findings and Observations

No findings of significance were identified. Significant issues were appropriately identified, evaluated and corrective actions were assigned commensurate with their safety significance. Corrective actions specified to address the repetitive steam affected entry condition reports have been addressed by the Engineering Department in implementing a multi-year turbine building equipment reliability improvement plan to reduce steam leaks. Corrective actions specified to address the weekly department exposure goals being exceeded had been addressed in the initiation of a Radiation Field Control Plan to address long-term plant source term concerns.

.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 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 specified corrective actions. The CRs reviewed are noted in the Attachment.

The inspectors also performed a semi-annual review of Entergy's corrective action program to assess trends that might indicate the existence of more significant safety issues. This semi-annual review included a review of system health reports, maintenance backlogs, engineering requests, self assessment reports and the condition report database.

b. Findings

No findings of significance were identified.

.3 Biennial Review of Condition Reports Associated with 10 CFR 50.59

Inspection Scope

The inspectors reviewed condition reports (CRs) associated with 10 CFR 50.59 issues and plant modification issues to ensure that Entergy was identifying, evaluating, and correcting problems associated with these areas and that the planned or completed corrective actions for the issues were appropriate. The inspectors also reviewed self-assessments related to 10 CFR 50.59 SEs and plant modification activities at FitzPatrick. The listing of the condition reports and self assessments reviewed is provided in the attachment.

b. Findings

No findings of significance were identified.

.4 Cross-References to PI&R Findings Documented Elsewhere

Inspection Scope

Section 1R12 describes a finding that is associated with the cross cutting area of problem identification and resolution since there were repetitive failures of the back up diesel driven fire pump.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153 - 4 samples)

.1 (Closed) LER 05000333/2005003-00, Plant Shutdown Due to Through-Wall Crack in Torus

On June 30, 2005, the plant was shutdown per TS 3.6.1.1 when Entergy concluded that the operability of the primary containment due to small torus vessel through-wall crack could not be assured. This event and its NRC enforcement aspects are documented in section 4OA3.1 of inspection report 05000333/2005009. Entergy entered the event into its corrective action program as CR-2005-02593. No new findings were identified during the inspector's review of the LER. This LER is closed.

.2 (Closed) LER 05000333/2005003-01, Plant Shutdown Due to Through-Wall Crack in Torus

This supplemental LER documented completion of a formal risk evaluation and the results of a revised torus leakage calculation. The inspectors reviewed the LER and identified no significant new findings. This event was documented in Entergy's corrective action program as CR-2005-02593. This LER revision is closed.

.3 (Closed) LER 05000333/2005004-00, Residual Heat Removal (RHR) Shutdown Cooling Line Through-Wall Crack

This LER documents Entergy's identification of a through-wall crack in the common suction pipe to both RHR shutdown cooling trains. The crack was determined to have been caused by low-stress, high cycle fatigue of a weld due to inadequate engagement of an adjacent pipe support. The violation of Criterion V, "Instructions, Procedures, and

Drawings," of 10 CFR 50, Appendix B, occurred because Entergy's inservice inspection in April 1985 did not identify a configuration error in the construction of the pipe support. This event and NRC enforcement aspects of this violation are documented in section 4OA3.2 of inspection report 05000333/2005009. Entergy entered the event into its corrective action program as CR-2005-02749. No new findings were identified during the inspector's review of the LER. This LER is closed.

.4 (Closed) LER 05000333/2005005-00, Automatic Reactor Scram on Low Reactor Vessel Water Level During Reactor Feed Pump Control Reset

This LER documents an automatic scram that occurred on September 14, 2005 due to a momentary loss of the uninterruptible power supply system. During the power loss, the reactor feedwater pump controls locked up as designed. When the controls were reset, a level transient occurred causing the scram. The violation of TS 5.4 occurred because abnormal operating procedure AOP-21, "Loss of UPS," did not have adequate instructions for restoring automatic feedwater control following a momentary loss of the UPS bus. This event and NRC enforcement aspects of this violation are documented in section 4OA3 of inspection report 05000333/2005005. Entergy entered the event into its corrective action program as CR-2005-03818. No new findings were identified during the inspector's review of this LER. This LER is closed.

4OA6 Meetings, Including Exit

On January 13, 2006, the inspectors presented the inspection results to Mr. Theodore Sullivan and other members of Entergy management. Entergy acknowledged that no proprietary information was involved.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Entergy Personnel

T. Sullivan, Vice President, Operations
S. Bono, VP Engineering
D. Wallace, Director, Nuclear Safety Assurance
K. Mulligan, General Manager, Plant Operations
N. Avrakotos, Manager, Emergency Preparedness
J. Costedio, Manager, Regulatory Compliance
M. Durr, Manager, System Engineering
J. Gerety, Manager, Design Engineering
D. Johnson, Manager, Operations
J. LaPlante, Manager, Security
A. McKeen, Manager, Radiation Protection
K. Mulligan, General Manager, Plant Operations
J. Pechacek, Manager, Programs and Components Engineering
S. Reininghaus, Manager, Training
W. Rheaume, Manager, CA&A
B. Sholler, Manager, Plant Maintenance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000333/2005006-01	NCV	Inadequate 10 CFR 50.59 Safety Evaluation Associated with Safety Relief Valves
05000333/2005006-02	NCV	Failure to Maintain Diesel-Driven Fire Pump Performance
05000333/2005006-03	NCV	Failure to Comply with TS 3.8.1 Required Actions for One Offsite Power Circuit Inoperable

Closed

05000333/2005003-00	LER	Plant Shutdown Due to Through-Wall Crack in Torus
05000333/2005003-01	LER	Plant Shutdown Due to Through-Wall Crack in Torus
05000333/2005004-00	LER	Residual Heat Removal (RHR) Shutdown Cooling Line Through-Wall Crack

05000333/2005005-00	LER	Automatic Reactor Scram on Low Reactor Vessel Water Level During Reactor Feed Pump Control Reset

LIST OF DOCUMENTS REVIEWED

Section 1RO1: Adverse Weather Protection

OP-51A, "Reactor Building Ventilation and Cooling System"
OP-52, "Turbine Building Ventilation"
DBD-066, "Design Basis Document for the Reactor Building Heating, Ventilation and Air Conditioning System"
DBD-067, "Design Basis Document for the Turbine Building Heating, Ventilation, and Air Conditioning System"

Section 1R02: Evaluation of Changes, Tests, or Experiments
Section 1R17: Permanent Plant Modifications

10 CFR 50.59 Safety Evaluations
JAF-SE-03-002, "Updated Reactor Pressure Vessel Fatigue Analysis"
JAF-SE-03-003, "Elimination of SRV Accumulator Check Valve Leakage Testing Per ST-39M"
JAF-SE-03-004, "Reduction of the sample number of welds requiring inspection under the Main Steam and Feedwater Augmented Inspection Program"
JAF-SE-03-005, "EHC Scram Frequency Reduction"
JAF-SE-04-001, "Compensatory Measures for a Possible Control Room Envelope Tracer Gas Test Failure"

10 CFR 50.59 Safety Evaluation Screens
DRN 05-00337, "HPCI Operation Procedure Revision for warming and pressurizing HPCI Steam Line following 23MOV-60 closure"
DRN 05-02431, "RHR Loop "A" Quarterly Operability Test Procedure Revision"
ER-JAF-03-01758, "Scram Solenoid Pilot Valve (03SOV-117 & 118) Replacement"
ER-JAF-04-14834, "Changing the Setpoints for SLC Tank Heater Controller 11TIC-48"
JAF-04-13240, "Qualify 345kV Backfeed to satisfy TS requirement to remove 115kV"
JD-04-003, "ESW Check Valve Change"
TA-05-023, "Rewire Reactor Vessel Flange Seal Leakage Pressure Switch"
TA-05-025, "Operate B Control Rod Drive Pump with Larger Diameter Impellers"
TA-05-033, "Install Leakage Clamp for Off-Gas A Condenser 01-107E-7A"
TA-05-039, "Removing RPS channel trip fuses in lieu of MSIV slow closure"
TA-04-017, "Maintain 10MOV-39A operable during 10MOV-15A overhaul"

Permanent Plant Modifications
DRN 05-00815, "Revise EOP & Operating Procedures to eliminate steam condensing mode of RHR"
JAF-05-16780, "Relocate Relay Room Temperature Sensor, 70TS-100, and Remove Relay Room Temperature Sensor, 70-TIC-101"
JAF-CALC-CAD-01767, "Torque Limits Calc. for Containment Isolation Valve 27MOV-121"

JD-03-060, “ATWS-RPT test circuit modification”
JD-03-069, “Reactor Feed Pump Seal Replacement”
JD-01-020, “23MOV-14 Replacement”
JF-03-01758, “Scram Solenoid Pilot Valves Modification”
ST-43C, “Remote Shutdown Panel Component Operation and Isolation Verification”

Calculations
JAF-CALC-CAS-02771, “ADS Accumulator Pressure Determination”
JAF-CALC-HPCI-02962, “Thrust and Torque Limits Calculation for 23MOV-14"
JAF-CALC-RAD-00023, “Power Uprate Program - Technical Support Center Post-Accident Radiological Habitability Study”
JAF-CALC-04-00455, “Structural Evaluation of Replacement Scram Solenoid Pilot Valves (SSPV)”
14620-EM-75-01, “HPCI Steam Supply Piping to Pump Turbine - Pipe Stress and Pipe Support Analysis”

10 CFR 50.59 Applicability Determinations
FSAR Change Request No. 04-018

Audits and Self-Assessments
LO # JAFLO-2005-00066, “Engineering Quality Focused Assessment”
QA-4-2004-JAF-1, “Design Control (LO-JAF-2004-00005)”

Completed Surveillances
RAP 7.4.1, “Control Rod Scram Time Evaluation (IST)”
ST-4N, “HPCI Quick-Start, Inservice, and Transient Monitoring Test (IST)”
ST-22D, “Nitrogen Instrument Header Integrity Test”

Evaluations
“Analysis of MOV Diagnostic Testing Using Liberty Technologies "Votes" System (for 23MOV14)”
CR-JAF-2005-04711 Operability Evaluations
JAF-04-38799, “SE Evaluation of RAP 7.4.1(Scram Time Test Results)”
JAF-05-25852, “System Engineer Evaluation of 7/13/05 Scram Time Test RAP 7.4.1"
JAF-SE-86-166, “ADS Pneumatic Supply System Upgrade”
JAF-SE-89-034, “Emergency Diesel Generator Air Start System Air Start System Capacity and Original Design Bases”

Drawings
DSK-23J, “Support Stand for HPCI Overspeed Test Motor”
ESK-5F, “B MG Drive Motor Elementary”
ESK-7FH, “ARI/RPT Systems Elementary”
FM-16B, “Flow Diagram Off Gas”
FM-39C, “Flow Diagram Instrument Air Reactor Bldg.”

FSAR Figure No. 7.4-3, “High Pressure Coolant Injection System (FCD)”, Sheet 2
MSK-3031, “Main Steam System”
PFSK-5631, “Pipe Support Vertical”

Miscellaneous
DBD-005, “Reactor Protection System”
DBD-23, “High Pressure Coolant Injection System”
DBD-46, “Normal Service Water Emergency Service Water RHR Service Water”
DBD-93, “Emergency Diesel Generator (EDG) Systems”
GE-NE-0000-0002-4213-01, “HPCI System Steam Supply Valve Replacement Report”
JAFP-05-0069, “Summary of Plant Changes, Tests, and Experiments for 2003 and 2004"
JAF-RPT-03-00289, “Augmented Main Steam and Feedwater High Stressed Weld Inspection Program”
JAF-RPT-04-00352, “Nuclear Environmental Test Report for Automatic Valve SSPV Model B7122-145"
JAF-SPEC-04-00021, “Technical Procurement Specification for Scram Solenoid Pilot Valves”
NEDE-24956, “BWR ADS Pneumatic System Comparison to NUREG-0737 Requirement II.K.3.28"
NEI 96-07, “Guidelines For 10 CFR 50.59 Implementation”
NRC Letter, D. B. Vassallo (NRR) to J.C. Brons, “TMI Item II.K.3.28"
NYPA Letter JPN-84-13, J.P. Bayne to D. B. Vassallo (NRR), “Qualification of ADS Accumulators NUREG-0737 Item II.K.3.28"
NYPA Letter JPN-84-58, J.P. Bayne to D. B. Vassallo (NRR), “Qualification of ADS Accumulators NUREG-0737 Item II.K.3.28"
NYPA Letter JPN-85-24, J.P. Bayne to D. B. Vassallo (NRR), “NUREG-0737 Item II.K.3.28 Qualification of ADS Accumulators on Automatic Depressurization System Valves”
PASNY Letter JPN-82-35, J.P. Bayne to D. B. Vassallo (NRR), “Qualification of ADS Accumulators NUREG-0737 Item II.K.3.28"
PASNY Letter JPN-82-7, J.P. Bayne to T. A. Ippolito (NRR), “NUREG-0737 Post-TMI Requirements Submittals and Modifications Required by January 1, 1982"
PCR for AOP-40, Main Steam Line Break, dated September 10, 2004
Preliminary Tracer Gas Test Results, dated June 28, 2004
Relief Request, RR-20, “Risk-Informed Inspection Program Plan”
TST-1, “EDG Air Starting Reservoir Capacity Test”

Operating Experience
NRC Generic Letter 95-07, “Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves”
NRC IE Bulletin No. 80-01, “Operability of ADS Valve Pneumatic Supply”
NRC Information Notice 2003-17, “Reduced Service Life of Automatic Switch Company (ASCO) Solenoid Valves With Buna-N Material”
NRC Information Notice 80-40, “Excessive Nitrogen Supply Pressure Actuates Safety-Related Valve Operation to Cause Reactor Depressurization”
NRC Information Notice 86-51, “Excessive Leakage in the Automatic Depressurization System”
NRC Information Notice 94-06, “Potential Degradation of Long-Term Emergency Nitrogen Supply for the Automatic Depressurization System Valves”

NRC Information Notice 94-71, “Degradation of Scram Solenoid Pilot Valve Pressure and Exhaust Diaphragms”
NRC Information Notice 96-07, “Slow Five Percent Scram Insertion Times Caused by Viton Diaphragms in Scram Solenoid Pilot Valves”
NRC Information Notice 96-08, “Thermally Induced Pressure Locking of a High Pressure Coolant Injection Gate Valve”
NRC Part 21 Report 1997-34-2, “Potential Safety-Related Problem with ASCO HV 266000-007J Scram Solenoid Pilot Valves”
NRC Part 21 Report 1997-36-2, “Failure of Scram Solenoid Pilot Valves Issued by Automatic Valves Corporation”
NRC Regulatory Issue Summary 01-015, “Performance of DC-Powered Motor-Operated Valve Actuators”

Safety Review Committee (SRC) and On Site Review Committee (OSRC)
SRC OSRC Subcommittee Meeting Minutes dated June 4, 2004, December 2, 2004, and May 20, 2005
OSRC Meeting Minutes dated September 15, 2005, October 13, 2005, and November 3, 2005

Work Orders
JF-000682001
JAF-04-12393
JAF-04-21163
JAF-04-29214
JAF-04-36543
JAF-04-37763
JAF-04-40282

Procedures
AOP-27, “Control Rod Drift”
AOP-32, “Unexplained/Unanticipated Reactivity Change”
AOP-40, “Main Steam Line Break”
AP-19.01, “Surveillance Testing Program”
ARP 09-4-2-33, “DW N2 Supp Press Hi or Lo”
ENN-DC-115, “ER Response Development”
ENN-DC-136, “Temporary Alterations”
ENN-LI-100, “Process Applicability Determination”
ENN-LI-101,”10CFR50.59 Review Process”
JAF-RAP 7.4.1, “Control Rod Scram Time Evaluation (IST)”
JAF-RAP-7.4.10, “Component Cyclic or Transient Limit Program”
OP-13, “Residual Heat Removal System”
OP-15, “High Pressure Coolant Injection”
OP-17, “Standby Liquid Control System”
OP-21, “Emergency Service Water (ESW)”
OP-37, “Containment Atmosphere Dilution System”
ST-2AL, “RHR Loop A Quarterly Operability Test (IST)”
ST-39M, “Leak Rate Test of ADS Pneumatic Supply Check Valves”
ST-40D, “Daily Surveillance and Channel Check”

ST-8Q, “Testing of the Emergency Service Water System (IST)”
ST-9BB, “EDG B and D Full Load Test and ESW Pump Operability Test”

Condition Reports

2003-01356	2004-00488	2004-04093	2005-02056
2003-02771	2004-01412	2004-04315	2005-03160
2003-03968	2004-01624	2004-04443	2005-03195
2003-04691	2004-01675	2004-04461	2005-03408
2003-04701	2004-02262	2004-04476	2005-03419
2003-04702	2004-02567	2004-04698	2005-04633*
2003-04739	2004-02726	2004-04844	2005-04634*
2003-05511	2004-02838	2004-05120	2005-04672*
2004-00190	2004-03355	2004-05373	2005-04705*
2004-00301	2004-03459	2005-00367	2005-04711*
2004-00385	2004-03850	2005-00636	2005-04720*

* Initiated as a result of this inspection

Engineering Requests
ER-03-0208, “Rewire ATWS/ARI test switches”
JAF-04-27499, “Update turbine missile analysis”

Section 1RO4: Equipment Alignment

OP-22, “Diesel Generator Emergency Power”
OP-42, “Service Water System”
OP-68, “Automatic Depressurization System”

Section 1R12: Maintenance Effectiveness

LO-JAFLO-2005-00076, JAF NPP 10 CRF 50.65 (a)(3) Periodic Assessment Nov 03 to Oct 05, Revision 0, 11/10/05
JAF-RPT-MULTI-02107, James A. FitzPatrick Nuclear Power Station IPE Update, Revision 2, 10/27/04
FM-29A, Flow Diagram Main Steam System 29, Revision 53, 10/6/04
FM-29B, Flow Diagram Main Steam System 29, Revision 49, 11/5/02
FM-29C, Flow Diagram Turbine Steam Supply Exhaust and Drains System 29, Revision 14, 9/2/99
FM-34A, Flow Diagram Feedwater System 24, Revision 59, 12/17/04
ENN-DC-121, Maintenance Rule, Revision 2, 5/20/04
ENN-DC-171, Maintenance Rule Monitoring, Revision 2, 7/8/04
ENN-DC-172, Maintenance Rule (a)(3) Periodic Assessment, Revision 0, ½4/05
JTS-APL-97-002, Maintenance Rule Action Plan (a)(1) Systems / Components 76P-4 East Diesel Fire Pump, 3/13/97
JTS-APL-97-002, Maintenance Rule Action Plan (a)(1) Systems / Components 76P-4 East Diesel Fire Pump, Revision 2, 7/27/97

JTS-APL-97-002, Maintenance Rule Action Plan (a)(1) Systems / Components 76P-4 East Diesel Fire Pump, Revision 3, 11/10/98
JTS-APL-01-011, Feedwater System Maintenance Rule (a)(1) Action Plan, Revision 3, 6/10/02
JTS-APL-02-005, Maintenance Rule Action Plan (a)(1) Systems / Components Recirculation Flow Control System, Revision 2, 6/21/04
JTS-APL-02-005, Maintenance Rule Action Plan (a)(1) Systems / Components Recirculation Flow Control System, Revision 3, 3/30/05
JTS-APL-03-011, Maintenance Rule Action Plan (a)(1) Systems / Components Inboard MSIV 80D, Revision 0, 10/15/03
JTS-APL-03-011, Maintenance Rule Action Plan (a)(1) Systems / Components Inboard MSIV 80D, Revision 0 status update, 4/6/05
JTS-APL-03-011, Maintenance Rule Action Plan (a)(1) Systems / Components Inboard MSIV 80D, Revision 1, 3/17/04
JAF-RPT-FPS-02496, Maintenance Rule Basis Document for System 076 Fire Protection System, Revision 6, 4/22/05
JAF-RPT-FPS-02496, Maintenance Rule Basis Document for System 076 Fire Protection System, Revision 5, 4/20/04
JAF-RPT-FPS-02496, Maintenance Rule Basis Document for System 076 Fire Protection System, Revision 4, 12/10/01
JAF-RPT-FPS-02496, Maintenance Rule Basis Document for System 076 Fire Protection System, Revision 3, 6/4/99
JAF-RPT-FPS-02496, Maintenance Rule Basis Document for System 076 Fire Protection System, Revision 2, ½2/98
JAF-RPT-FPS-02496, Maintenance Rule Basis Document for System 076 Fire Protection System, Revision 1, 11/6/97
JAF-RPT-FPS-02496, Maintenance Rule Basis Document for System 076 Fire Protection System, Revision 0, 10/26/95
JAF-RPT-FWS-03079, Maintenance Rule Basis Document for System 034 Feedwater System, Revision 0, ½6/99
JAF-RPT-MST-02480, Maintenance Rule Basis Document for System 029 Main Steam System, Revision 6, 3/16/05
JAF-RPT-RFC-02315, Maintenance Rule Basis Document for System 002-0184 Reactor Water Recirculation Flow Control System, Revision 4, 9/15/03
Memo RET-97-233, JAF Nuclear Power Plant Close Out of Acts Item 25812, 5/16/97
System 034 - Feedwater, System Health Report, Q3 2005
Maintenance Rule, Quarterly System Health Report, Q3 2005
Fire Protection-076, System Health Report, Q3 2005
Recirculation Flow Control System - 002-0184, System Health Report, Q3 2005
Main Steam (29), System Health Report, Q2 2005
600V and Under Systems - 71AC, System Health Report, Q3 2005
HPCI System-023, System Health Report, Q3 2005
Radwaste - 020, System Health Report, 1st half 2005

Condition Reports

2002-05528	2003-00495	2003-01354	2003-01418
2003-02116	2003-04001	2003-04888	2004-00095

2004-00289	2004-00308	2004-00769	2004-02308
2004-02930	2004-04233	2004-04531	2004-05018
2004-05034	2005-00589	2005-00998	2005-01995
2005-02715	2005-04407	2005-04767	2005-04806
2005-04837	2005-04838	2005-04842	2005-05187
2005-05224	2005-05243		

Section 4OA2: Identification and Resolution of Problems

.1 Condition Reports

2004-03959	2004-04116	2004-04152	2004-04153
2004-04220	2004-04439	2004-04440	2004-04786
2004-04867	2004-04981	2004-04982	2004-05036
2004-05077	2004-05284	2004-05286	2004-05290
2004-05292	2004-05363	2004-05554	2004-05558
2004-05625	2004-05626	2004-05628	2005-00058
2004-05530	2005-00157	2005-00059	2005-00073
2005-00081	2005-00093	2005-00097	2005-00103
2005-00261	2005-00291	2005-00382	2005-00383
2005-00411	2005-00549	2005-00460	2005-00809
2005-00834	2005-00879	2005-01432	2005-01650
2005-01864	2005-02248	2005-03535	2005-03503
2005-00887	2005-02186	2005-02036	2005-01884
2005-02150	2005-02419	2005-02420	2005-02833
2005-02838	2005-02917	2005-02965	2005-03433
2005-03444	2005-04436	2005-03542	2005-03496
2005-03540	2005-03541	2005-04485	2005-03983
2005-03984	2005-03982	2005-04261	2005-04262
2005-04317	2005-03748		

.2 Condition Reports

2004-02490	2005-01396	2005-04478	2005-05195
2005-05180	2005-05243	2005-05195	2005-05139
2005-05119	2005-04859	2005-04838	2005-04842
2005-05041	2005-04752	2005-04687	2005-04639
2005-03931	2005-05054	2005-00998	2005-04472
2005-04463	2005-04407	2005-04125	

LIST OF ACRONYMS

ADS	automatic depressurization system
AOP	abnormal operating procedure
CAD	containment atmosphere dilution
CFR	Code of Federal Regulations
CR	condition report
DBD	design basis document

DPS	drywell pneumatic supply
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOP	emergency operating procedure
ER	engineering request
ESW	emergency service water
HCU	hydraulic control unit
HPCI	high pressure coolant injection
HVAC	heating, ventilation and air conditioning
IMC	inspection manual chapter
IP	inspection procedure
IST	inservice test
kV	kilovolt
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss of coolant accident
LOOP	loss of offsite power
MOV	motor-operated valve
MR	maintenance rule
MSIV	main steam isolation valve
NCV	non-cited violation
NMP	Nine Mile Point
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NYPA	New York Power Authority
OP	operating procedure
OSRC	on site review committee
PASNY	Power Authority of the State of New York
PCR	procedure change request
RHR	residual heat removal
RHRSW	residual heat removal service water
RSST	reserve station service transformer
SDP	significance determination process
SE	safety evaluation
SLC	standby liquid control
SRC	safety review committee
SRV	safety relief valve
SSC	structure, system, and component
SSPV	scram solenoid pilot valve
ST	surveillance test procedure
TM	temporary modification
TS	technical specification
TST	temporary surveillance test procedure
UFSAR	Updated Final Safety Evaluation Report
WR	work request