October 26, 2005

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000333/2005005

Dear Mr. Sullivan:

On September 30, 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 that were discussed on October 11, 2005, with you and other members of your staff.

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

This report documents one finding of very low safety significance (Green). The finding was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it was entered into your corrective action program, the NRC is treating the finding as a non-cited violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the non-cited violation 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/** Donald Jackson signing for

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

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

Enclosure: Inspection Report 05000333/2005005
w/Attachment: Supplemental Information

cc w/encl
G. Taylor, CEO, Entergy Operations
M. Kansler, President, Entergy
J. Herron, Sr, VP and Chief Operating Officer
C. Schwarz, VP, Operations Support
K. Mulligan, General Manager, Plant Operations
O. Limpias, VP, Engineering
J. McCann, Director, Licensing
C. Faison, Manager, Licensing
M. Colomb, Director of Oversight
D. Wallace, Director, Nuclear Safety Assurance
R. Plasse, Acting Manager, Regulatory Compliance
T. McCullough, Assistant General Counsel
P. Smith, President, New York State Energy Research and Development Authority
P. Eddy, New York State Department of Public Service
S. Lyman, Oswego County Administrator
Supervisor, Town of Scriba
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
J. Sniezek, PWR SRC Consultant
M. Lyster, PWR SRC Consultant
S. Lousteau, Treasury Department
INPO

Distribution w/encl: **(VIA E-MAIL)**
S. Collins, RA
M. Dapas, DRA
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J. Boska, PM NRR
P. Tam, PM (backup)
P. Milano, PM (backup)
L. Cline, DRP - NRC Senior Resident Inspector
D. Dempsey, DRP, Resident Inspector
K. Kolek, Resident OA
Region I Docket Room (with concurrences)
ROPreports@nrc.gov (All IRs)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

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|---|---|
| Docket No.: | 50-333 |
| License No.: | DPR-59 |
| Report No.: | 05000333/2005005 |
| Licensee: | Entergy Nuclear Northeast (Entergy) |
| Facility: | James A. FitzPatrick Nuclear Power Plant |
| Location: | 268 Lake Road<br>Scriba, New York 13093 |
| Dates: | July 1 - September 30, 2005 |
| Inspectors: | L. Cline, Senior Resident Inspector<br>D. Dempsey, Resident Inspector<br>J. Noggle, Senior Health Physicist<br>D. Tifft, Reactor Engineer |
| Approved by: | Brian J. McDermott, Chief<br>Projects Branch 2<br>Division of Reactor Projects |

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## SUMMARY OF FINDINGS

IR 05000333/2005005; 07/01/2005 - 09/30/2005; James A. FitzPatrick Nuclear Power Plant; Event Response, and Cross Cutting Aspects of Findings.

The report covered a three-month period of inspection by resident inspectors, and an announced inspection by a senior health physicist. One Green non-cited violation (NCV) was identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). 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 self-revealing NCV of Technical Specification (TS) 5.4, "Procedures", occurred when Entergy failed to maintain a procedure appropriate to the circumstances. Specifically, abnormal operating procedure (AOP)-21, "Loss of UPS," did not include adequate instructions for restoring automatic feedwater level control following a momentary loss of uninterruptible power supply. This resulted in an automatic reactor scram on September 14, 2005, due to low reactor vessel water level. Entergy revised the procedure as a corrective action for this violation.

  The finding is greater than minor because it affected the procedure adequacy attribute of the initiating event cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations. The inspectors determined the finding to be of very low safety significance using the Phase 1 SDP screening worksheet for at power situations. The finding screened to Green because it does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, and is not potentially risk significant due to external events. (Section 4OA3)

  This finding is associated with the human performance cross-cutting area in that Entergy failed to maintain a procedure appropriate to the circumstances. Specifically, AOP-21 did not include adequate instructions for restoring automatic feedwater level control following a momentary loss of UPS. (Section 4OA4)

B. Licensee-Identified Violations

None.

# REPORT DETAILS

Summary of Plant Status

The reactor began the inspection period shutdown for a forced outage to repair a small torus water leak. Following repairs full power operation was restored on July 15, 2005. On July 28 power was reduced to 55% to replace a B reactor feed water pump seal. Full power operation was restored on July 31. An automatic reactor scram due to low reactor water level occurred on September 14. Full power operation was restored on September 17. FitzPatrick continued to operate at or near rated power for the rest of the inspection period.

## 1. REACTOR SAFETY

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

### 1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors completed one impending adverse weather condition inspection sample. During the week of July 17 when daytime high temperatures were consistently above 90°F, the inspectors verified the status of Entergy's warm weather preparations and the impact of the above average air temperatures on the operability of the plant's ultimate heat sink. The inspectors verified operation of the 600 volts, alternating current (Vac) safety-related switchgear room coolers in accordance with the emergency service water (ESW) and turbine building ventilation system operating procedures. Surveillance testing for the A train 600 Vac switchgear room cooler completed on May 5, 2005, had indicated a degrading trend in ESW flow rates for the cooler. The inspectors reviewed the adequacy of the operability evaluation, interim compensatory measures and Entergy's proposed corrective actions for the condition. The inspectors also reviewed the calibration and operation of the ultimate heat sink temperature monitoring instrumentation used to verify that the average ultimate heat sink temperature was less than 85°F in accordance with the plant's technical specifications (TS), and reviewed plant operating procedures for screenwell intake high temperatures.

b. Findings

No findings of significance were identified.

### 1R04 Equipment Alignment (71111.04 - 4 samples, 71111.04S - 1 sample)

a. Inspection Scope

Partial System Walkdown. The inspectors performed four partial system walkdowns. The selected walkdowns were chosen based on safety significance, scheduled maintenance, and environmental conditions such as high outside air temperatures. The inspectors compared system lineups to system operating procedures (OPs), system drawings, and the applicable chapters in the Updated Final Safety Analysis Report (UFSAR). Other documents reviewed for this inspection are listed in the Attachment. 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:

- Unit coolers 67UC-16B and 67UC-16A, the east and west 600 Vac safety-related switchgear room coolers during high lake temperature conditions with a tube leak on 67UC-16A during the week of August 1;
- Turbine building heating, ventilation, and air conditioning (HVAC) systems during high lake temperature conditions during the week of August 8;
- West diesel-driven and electric fire pumps while the east diesel-driven fire pump was out of service for preventive maintenance (PM) during the week of August 29; and
- Alternate vital 120 vac power supply while motor-generator set 71UPS-1 was out of service for corrective maintenance during the week of September 12.

Complete System Walkdown. The inspectors performed a complete walkdown of reactor building HVAC systems to identify any discrepancies between the existing equipment lineup and the specified lineup. During the walkdown system drawings and OPs were used to verify proper equipment alignment and operational status. The inspectors reviewed the open maintenance work requests on the system for any deficiencies that could affect the ability of the system to perform its function. Documentation associated with unresolved design issues such as temporary modifications, operator workarounds, and items tracked by plant engineering were also reviewed to assess their collective impact on system operation. In addition, the inspectors reviewed the condition report (CR) database to verify that equipment alignment problems were being identified and appropriately resolved. Documents reviewed for this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05AQ - 8 samples)

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," the JAF fire hazards analysis and pre-fire plans in performing the inspection. The areas inspected included:

- Reactor Building (RB) Elevation 326';
- RB Elevation 272' West;
- RB Elevation 272' East;
- RB Elevation 300' East;

- RB Elevation 300' West;
- Turbine Building (TB) Elevation 252' North;
- TB Elevation 272' North; and
- TB Elevation 272' Foam and Miscellaneous Oil Storage Areas.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Annual Sample. The inspectors completed one annual heat sink performance inspection sample. The inspectors reviewed the testing and evaluation of test results for the electric bay and cable tunnel unit coolers conducted during the week of August 1, 2005. ST-8Q, “Testing of the ESW System (IST),” is performed on a quarterly basis to verify safety-related unit cooler thermal performance. The inspectors reviewed performance data to verify that heat exchanger operation was consistent with its design basis. Documents reviewed for this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors completed one licensed operator requalification inspection sample. On August 3, 2005, the inspectors observed licensed operator performance in the simulator during an emergency preparedness exercise to assess operator performance. The exercise scenario involved a security threat that resulted in a loss of all vital AC busses followed by a large break loss of coolant accident (LOCA) inside the primary containment. The inspectors evaluated the performance of risk significant operator actions, including the use of EOPs, EOP-2, “Reactor Pressure Vessel Control” and EOP-4, “Primary Containment Control.” 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. Documents reviewed for this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q - 3 samples)

a. Inspection Scope

The inspectors completed three maintenance effectiveness inspection samples. The inspectors reviewed performance-based problems involving the 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; tracking 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. Other documents reviewed for this inspection are listed in the Attachment. The following samples were reviewed:

- Degraded performance for the reactor feedwater leading edge flow meter (LEFM);
- Performance test failures for the emergency and plant information computer (EPIC) UPS batteries; and
- Maintenance rule (a)(1) action plan for analog transmitter trip system cards and nests.

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 completed five maintenance risk assessments and emergent work evaluation inspection samples. The inspectors reviewed risk assessments associated with five different work weeks. 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. Documents reviewed for this inspection are listed in the Attachment. The following work weeks were reviewed:

- Week of July 11, 2005, that included a plant shutdown and forced outage to repair a torus shell crack and a residual heat removal (RHR) shutdown cooling line crack;
- Week of July 24, 2005, that included PM and full load testing of train A EDGs, standby liquid control system and drywell reactor coolant system leakage detection preventive maintenance, and surveillance on the B RHR/LPCI battery;
- Week of August 1, 2005, that included high lake temperatures, record grid loading and instrument air compressor maintenance;
- Week of August 29, 2005, that included quarterly HPCI system surveillance, corrective maintenance on the C EDG air start compressor, annual PM on a diesel-driven fire pump, and scram discharge instrument volume transmitter calibrations; and
- Week of September 19, 2005, that included A and C emergency diesel generator (EDG) surveillance testing, reactor core isolation system (RCIC) testing, and 120 Vac UPS troubleshooting.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-routine Evolutions and Events (71111.14 - 2 samples)

a. Inspection Scope

The inspectors completed two operator performance during non-routine evolutions and events inspection samples. For the two non-routine events described below the inspectors reviewed plant procedures, operator logs, plant computer data, and strip charts. The inspectors also interviewed operators and plant management to determine what occurred, how the operators responded, and if the response was in accordance with plant procedures and management expectations. Documents reviewed for this inspection are listed in the Attachment. The events reviewed included:

- On July 5, 2005, the inspectors reviewed the site response to an inadvertent trip of both reactor water cleanup pumps (RWCU) while operators were hanging a tagout for seal replacement on the B reactor water recirculation (RWR) pump. While installing the breaker locking device on the supply breaker for the RWR pump vibration monitoring panel the operator inadvertently opened the supply breaker for control rod drive system (CRD) cooling water flow instrumentation. This caused a trip of both RWCU pumps when the loss of power to this instrumentation caused an indicated high reactor building closed loop cooling (RBCLC) water temperature at the outlet of the RWCU pumps. Operators responded in accordance with applicable alarm response and system operating procedures. Due to weaknesses in operator training and the system operating procedure (OP) the restoration of the RWCU system following the trip caused reactor water sulfates to exceed the EPRI BWR Water Chemistry Guidelines prior to startup limit of 20 ppb, but a detailed analysis of the excursion determined that there was no long-term impact on the reliability of the reactor vessel internals and piping.

- On September 14, 2005, the inspectors responded to the control room following a low water level reactor scram. The low reactor vessel water level resulted from operator actions taken to recover from a momentary loss of the 120 Vac UPS bus that occurred while operators were aligning the bus to its alternate power supply in preparation for UPS motor generator set (MG) maintenance. HPCI started on a reactor vessel water level low signal, but did not inject because level recovered before the turbine was up to speed. RCIC started and injected for approximately one minute before tripping on high reactor vessel water level. A primary containment system group two isolation also occurred. All valves closed with the exception of two RWCU PCIS valves that did not close because their breakers were opened in accordance with plant procedures to support the planned UPS transfer. For the flow paths that these valves were designed to provide isolation for, primary containment was maintained by other valves that received isolation signals.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 4 samples)

a. Inspection Scope

The inspectors completed four operability evaluation inspection samples. The inspectors reviewed operability determinations to assess the acceptability of the evaluations; the use and control of compensatory measures, when needed; and compliance with TSs. 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 DBDs. Other documents reviewed for this inspection are listed in the Attachment. The following four evaluations were reviewed:

- CR-2005-03041 concerning more frequent feed and bleed requirements to maintain torus to drywell differential pressure;
- CR-2005-03125 concerning slight scoring of valve plug containment atmosphere dilution system makeup flow control valve 27FCV-103A;
- CR-2005-01901 concerning the need to raise the 67UC-16B maximum operable lake temperature; and

- CR-2005-02951 concerning GE Safety Information Communication SC05-06, "Updated Surveillance Program for Fuel Channel Control Blade Interference Monitoring."

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 and cumulative effects of identified operator workarounds, burdens, and control room deficiencies on the functionality of plant mitigating systems. The items 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 problems were captured in Entergy's corrective action program (CAP.) The inspectors also reviewed Entergy's assessment of the cumulative effects of the identified issues in accordance with ST-99H, "Operations Cumulative Impact Assessment."

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 6 samples)

a. Inspection Scope

The inspectors completed six post maintenance testing inspection samples. The inspectors reviewed post maintenance test procedures and associated testing activities for selected risk significant mitigating systems to assess whether the effect of maintenance on plant systems was adequately addressed by control room and engineering personnel. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design basis documentation; that test instrumentation had current calibrations and the range and accuracy for the application; and that tests were performed, as written, with applicable prerequisites satisfied. Upon completion, the inspectors verified that equipment was returned to the proper alignment necessary to perform its safety function. Documents reviewed for this inspection are listed in the Attachment. The following post maintenance test activities were reviewed:

- WR JAF-04-29449, involving replacement of a control relay 10MOV-25A during the week of July 4. The retest was performed per WO JAF-05-18184 by cycling 10MOV-25A open and closed.
- WR JF- 000216000, involving turning and resloping the sensing lines for torus level transmitter 23LT-201C during the week of August 8. The retest was performed using instrument surveillance procedure (ISP)-29-2, “Torus Water Level (Narrow Ranges) Instrument calibration.”
- WR JAF-05-24248, involving ASME Code repair of a torus leak during the week of July 11. The retest consisted of volumetric, visual, and surface examinations and a 47 psig pneumatic test per TST-97, “Primary Containment Pressurization Test.”
- WR JAF-05-25011, involving ASME Code repair of a shutdown cooling system pipe crack during the week of July 11. The retest consisted of volumetric, visual, and surface examinations and a system leakage test per Section XI of the ASME Code.
- WR JAF-05-23841, involving repair and lubrication of diesel generator switchgear room fire door 76FDR-DG-272-2 during the week of August 8. The retest was performed using ST-76Y, “Fire Door Inspection and Operability Test.”
- WR JAF-05-21373, JAF-05-21437, and JF-030985400, involving annual PM on the east diesel-driven fire pump during the week of August 29. The retest was performed using ST-76AC, ”East Diesel Fire Pump 76P-4 Operational Check.”

b. <u>Findings</u>

No findings of significance were identified.

1R20 <u>Refueling and Other Outage Activities</u> (71111.20 - 2 samples)

a. <u>Inspection Scope</u>

The inspectors completed two outage activities inspection samples. The inspectors observed and reviewed the following activities for two FitzPatrick forced outages. Forced outage 171 from July 1 to July 13, 2005 and Forced outage 172 from September 14 to September 16, 2005.

- Outage Plan: The inspectors reviewed outage schedules and procedures and verified that TSs specified safety system availability was maintained, risk was considered, and that contingency plans existed to restore key safety functions.
- Plant shutdown and cooldown: The inspectors observed portions of the plant shutdown and verified the TS cooldown rate limits were not exceeded.
- The inspectors observed portions of the reactor startup following the outage, and verified through plant walkdowns, control room observations, and surveillance test reviews that safety-related

equipment specified for mode change was operable, and that reactor coolant boundary leakage was within TS limits.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors completed three routine surveillance inspection samples, two inservice testing inspection samples and one containment isolation valve testing inspection sample. The inspectors witnessed the performance of the selected STs and/or reviewed test data for the selected risk-significant SSCs to assess whether the SSCs satisfied TSs, UFSAR, technical requirements manual, and Entergy procedure requirements. Other documents reviewed for this inspection are listed in the Attachment. The inspectors verified that test acceptance criteria were clear, demonstrated operational readiness and were consistent with design basis documentation; that test instrumentation had current calibrations and the range and accuracy for the application; and that tests were performed, as written, with applicable prerequisites satisfied. Upon ST completion, the inspectors verified that equipment was returned to the status specified to perform its safety function. The following STs were reviewed:

- ST-41F, “HVAC Control Valve Fail Position Test(IST);”
- ST-15J, “Torus to Drywell Vacuum Breakers Quarterly Test(IST);”
- ISP-29-2, “Torus Water Level (Narrow Range) Instrument Calibration;”
- ST-8E, “ESW Logic System Functional Test and Simulated Automatic Actuation Test;”
- ST-4N, “HPCI Quick-Start, Inservice, And Transient Monitoring Test (IST);” and
- ST-24R, “RCIC Turbine Slow Roll Test (Mode 1).”

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors completed one drill evaluation inspection sample. The inspectors observed simulator, technical support center and emergency operations facility activities associated with FitzPatrick's emergency planning drill on August 3, 2005. The inspectors verified that emergency classification declarations and notifications were completed in accordance with 10 CFR 50.72, 10 CFR 50, Appendix E, and Entergy emergency plan implementing procedures.

b. Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01 - 12 samples)

a. Inspection Scope

Between August 29 and September 2, 2005, to verify that Entergy properly implemented physical, engineering, and administrative controls for access to radiologically controlled areas; and that workers were adhering to these controls when working in these areas, the inspectors conducted the following activities based on exposure significance. Implementation of the access control program was reviewed against the criteria contained in 10 CFR 20, site TSs, and Entergy's procedures. The following were examined:

- One exposure significant work area associated with a turbine building entry at power to repair a leaking moisture separator stop/control valve was identified and high radiation area (HRA) surveys of the work area were reviewed.
- The work activity listed above was selected for inspection based on the specified HRA controls and the potential radiological risks associated with valve leak repair during full power operation.
- Independent walkdowns and radiation surveys of the above work area and other accessible plant areas were conducted to evaluate whether prescribed radiation work permit (RWP), procedure, and engineering controls were in place and whether surveys and postings were complete and accurate.
- The RWP associated with the work activity listed above, was reviewed with respect to Technical Specification HRA requirements. This review included an evaluation of the adequacy of electronic dosimetry alarm setpoints based on radiation survey information and plant policy. The pre-job briefing with workers was observed to evaluate the adequacy of communication of radiological conditions, actions to take based on electronic dosimetry alarms and other stop work conditions.
- There were no RWPs used to access airborne radioactivity areas for review.

- There were no high radiation work activities during the inspection with significant dose rate gradients to review appropriate dosimetry monitoring of personnel.
- One internal dose assessment that resulted in 29 mrem committed effective dose equivalent was reviewed for adequacy.
- Approximately 15 condition reports (CRs) related to access controls were reviewed with respect to timely and effective corrective actions to address the issues commensurate with their importance to safety and risk.
- There were no safety significant repetitive deficiencies identified in the CRs reviewed during this inspection.
- There were no performance indicator (PI) events for the occupational exposure control effectiveness PI for review during the current reactor oversight assessment cycle.
- Discussion with the Radiation Protection Manager (RPM) and a review of applicable radiation protection procedures verified the effectiveness of controls for very HRA entries.
- Discussions with two RP supervisors and a review of applicable RP standards indicated that areas of the plant that have the potential to become very HRAs are posted and controlled as if they were, requiring RPM approval for entry.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 4 samples)

a. Inspection Scope

Between August 29 and September 2, 2005, the inspectors conducted the following activities to verify that Entergy properly maintained individual and collective radiation exposures as low as is reasonably achievable (ALARA). Implementation of the ALARA program was reviewed against the criteria contained in 10 CFR 20.1101(b) and Entergy's procedures.

- During the repair of a leaking moisture separator stop/control valve, radiation worker and RP technician performance was observed to determine if the workers demonstrated the ALARA philosophy in practice and as planned.
- Site specific collective exposure and source-term trends were reviewed indicating a slight increasing trend in both, currently at median industry performance.
- ALARA exposure estimating methods and procedures were reviewed to determine the accuracy and basis of current dose estimate measures of performance.

- There was one declared pregnant worker during the current reactor oversight program assessment period. The personnel exposure records and procedural controls for the declared pregnant worker were reviewed with respect to 10 CFR 20 requirements.

b. Findings

No findings of significance were identified.

**4. OTHER ACTIVITIES [OA]**

4OA1 Performance Indicator Verification (71151 - 2 samples)

a. Inspection Scope

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

Occupational Radiation Safety

- Occupational Exposure Control Effectiveness

The inspectors reviewed implementation of the Entergy’s Occupational Exposure Control Effectiveness PI Program. 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 HRAs, very HRAs, and unplanned exposures to verify that all occurrences that met the NEI criteria were identified and reported.

Public Radiation Safety

- RETS/ODCM Radiological Effluent Occurrences

The inspectors reviewed relevant effluent release reports for the past four calendar quarters for radiological effluent release occurrences that exceed 1.5 mrem/qtr whole body or 5.0 mrem/qtr organ dose for liquid effluents; 5mrads/qtr gamma air dose, 10 mrad/qtr beta air dose, and 7.5 mrads/qtr for organ dose for gaseous effluents. The inspectors reviewed monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases; quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases; and Entergy’s dose assessment procedures.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution of Problems

Daily Review

a. Inspection Scope

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

In accordance with the baseline inspection modules, the inspectors selected 74 CAP items across the initiating events, mitigating systems, barrier integrity and occupational and public radiation safety cornerstones for additional follow-up and review. The inspectors assessed Entergy’s threshold for problem identification, the adequacy of the cause analyses, extent of condition review, and operability determinations, and the timeliness of the specified corrective actions. The CRs reviewed are listed in the Attachment.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153 - 1 sample)

a. Inspection Scope

On September 14, 2005, the inspectors responded to the control room following a low water level reactor scram. The low reactor vessel water level resulted from a momentary loss of the 120 Vac UPS bus that occurred while operators were aligning the bus to its alternate power supply in preparation for UPS motor generator set (MG) maintenance. HPCI started on a reactor vessel water level low-low signal, but did not inject because level recovered before the turbine was up to speed. RCIC started and injected for approximately one minute before tripping on high reactor vessel water level. A primary containment isolation system (PCIS) group two isolation also occurred. All valves closed with the exception of two RWCU PCIS valves that did not close because their breakers were opened in accordance with plant procedures to support the planned UPS transfer. For the flow paths that these valves are designed to provide isolation for, primary containment was maintained by other valves that received isolation signals. All specified 10 CFR 50.72 reports were completed.

b. Findings

Introduction. A Green self-revealing NCV of TS 5.4, “Procedures”, occurred when Entergy failed to maintain a procedure appropriate to the circumstances. Specifically, AOP-21, “Loss of UPS,” did not include adequate instructions for restoring automatic feedwater level control following a momentary loss of the 120 Vac UPS bus. This resulted in an automatic reactor scram on September 14, 2005, due to low reactor vessel water level.

Description. The 120 Vac UPS bus supplies power to several important instrumentation and control circuits including the feedwater control system (FWCS). The FWCS controls reactor vessel water level by adjusting reactor feed pump (RFP) speed. A loss of power to the UPS bus results in a loss of control signal from the FWCS to the RFP speed control circuit. To provide reactor protection from the consequences of a loss of speed control signal, the RFP motor gear units (MGUs), which adjust RFP turbine throttle position based on the FWCS speed control signal, lock up to maintain the speed level demanded prior to the control system failure.

On September 14, in preparation for maintenance on the UPS MG, operators were in the process of transferring the UPS bus from the MG to the alternate AC source (600 Vac/120 Vac transformer) in accordance with OP-46B, "120 Vac Power System." The procedure directed operators to transfer the UPS MG from the AC to the DC drive motor and then parallel the MG output with the alternate AC power supply. Around 2:12am operators transferred the MG to the DC drive and began adjusting frequency to parallel the UPS with the alternate 120 Vac source. At that point the MG tripped and the plant experienced a momentary loss of the UPS bus, followed by the automatic transfer of the UPS bus to the alternate 120 Vac source. Operators entered AOP-21, "Loss of UPS," and in accordance with the procedure, immediately restored automatic feedwater level control by resetting the A and B FWCS lockout relays, unlocking the feed pump MGUs. Following the reset, reactor vessel water level began to rapidly lower and operators attempted to restore level by taking manual control. Operators were unable to recover level before the low reactor water level scram set point and the reactor scrammed at 2:13am.

In November 1998, FitzPatrick revised AOP-21. Before this revision, following a momentary loss of UPS, AOP-21 directed operators to take manual control of a RFP to maintain reactor vessel level and restore automatic level control using the normal OP-2A, “Feedwater System.” To free up an operator for other specified actions following a momentary loss of UPS event, the 1998 procedure revision directed operators to immediately reset the RFP MGU lockout relays, placing the FWCS back in automatic level control. To avoid a reactor vessel water level transient following an MGU lockout relay reset the procedure specified that the feed pump MGU position matched the FWCS RFP speed demand signal at the time of the reset. Due to the circuit design for the

NUS Instruments PID900-540-01 controller used in the FWCS, when the FWCS loses power the RFP speed demand signal to the feed pump MGUs rapidly decays to zero, and when power is restored the demand signal remains zero until reactor vessel water level deviations from setpoint cause a level error. On September 14, when power was lost to the FWCS, the MGUs locked up at the 100% feed flow position and the FWCS RFP speed demand signal decayed to zero. When power was restored to the controller little or no level error signal existed so the controller demand signal to the RFP speed control circuit remained at zero. When the MGU lockout relays were reset 20 seconds later, the mismatch between MGU position and speed demand signal resulted in rapidly lowering RFP speed that resulted in rapidly lowering reactor vessel level and a reactor scram on September 14, 2005.

Analysis. The performance deficiency associated with this event was the failure to maintain adequate procedures for restoring automatic reactor water level control following a momentary loss of power to the UPS bus. Traditional enforcement does not apply because the issue did not have an actual safety consequence or a potential for impacting the NRC's regulatory function, and it was not the result of any willful violation of NRC requirements. The finding is greater than minor because it affected the procedure adequacy attribute of the initiating event cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during power operations. The inspectors determined the finding to be of very low safety significance using the Phase 1 SDP screening worksheet for at power situations. The finding screened to Green because it does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, and is not potentially risk significant due to external events. The cause of the finding is related to the cross-cutting area of human performance.

Enforcement. TS 5.4.1.a requires written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Appendix A, November 1972. Abnormal operating procedures are recommended in section 5 of RG 1.33, Appendix A. Contrary to the above, AOP-21, "Loss of UPS," was not maintained to address the operating response characteristics of the master reactor level controller resulting in an automatic reactor trip on September 14, 2005. Because this failure to maintain an adequate AOP is of very low risk significance and has been entered into Entergy's CAP as CR 2005-03838, this violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy: **NCV 05000333/2005005-01 Inadequate AOP Resulted in Reactor Trip**.

4OA4 Cross-Cutting Aspects of Findings

Section 4OA3 describes a finding associated with the human performance cross-cutting area. Entergy failed to maintain a procedure appropriate to the circumstances. Specifically, AOP-21, "Loss of UPS," did not include adequate instructions for restoring automatic feedwater level control following a momentary loss of UPS.

4OA5 Other Activities

TI 2515/161 - Transportation of Reactor Control Rod Drives in Type A Packages

a. Inspection Scope

This area was inspected to verify that the Entergy radioactive material transportation program complies with specific requirements of 10 CFR Parts 20, 71, and Department of Transportation (DOT) regulations contained in 49 CFR Part 173. The inspectors interviewed personnel and determined Entergy had undergone refueling/defueling activities between January 1, 2002, and present, and had conducted two shipments of irradiated control rod drives in October 2004, in DOT Specification 7A Type A packages. The shipment records were reviewed with respect to the DOT Specification 7A package documentation requirements.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

The inspectors presented the inspection results to Mr. Theodore Sullivan and other members of Entergy management on October 11, 2005. Entergy acknowledged that no proprietary information was involved.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Entergy Personnel

N. Avrakotos, Manager, Emergency Preparedness
S. Bono, VP Engineering
J. Costedio, manager, Regulatory Compliance
M. Durr, Manager, System Engineering
J. Gerety, Manager, Design Engineering
M. Jacobs, Manager, Training
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
W. Rheaume, Manager, CA&A
B. Sholler, Manager, Plant Maintenance
T. Sullivan, Site Vice President
D. Wallace, Director, Nuclear Safety Assurance

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 05000333/2005005-01 | NCV | Inadequate AOP Resulted In Reactor Trip (Section 4OA3) |

## LIST OF DOCUMENTS REVIEWED

**Section 1R01: Adverse Weather Protection**

OP-4, "Circulating Water System"
OP-21, "Emergency Service Water"
OP-52, "Turbine Building Ventilation"
TST-104, "Testing of ESW Loop A (IST)"

**Section 1R04: Equipment Alignment**

OP-21, "Emergency Service Water"
OP-52, "Turbine Building Ventilation"
OP-46B, "120 VAC Power System"
OP-51A, "Reactor Building Ventilation and Cooling System"
DBD-067, "Turbine Building Ventilation and Cooling Systems"

DBD-066, “Reactor Building HVAC Systems”
DBD-046, “Normal Service Water, Emergency Service Water, RHR Service Water”

Dwg FE-1AB, “120 VAC One Line Diagram”
Dwg FE-1R, “600V One Line Diagram - 71MCC-131, 141, 252 & 253"
WO JAF-05-25945, Repair replace 67UC-16A cooling coils - 2A lower coil leakage
CR-2005-03011, A cooling coil in 67UC-16A has developed a pinhole tube leak

**Section 1RO7: Heat Sink Performance**

JAF-CALC-SWS-00569, “Cooler Performance Methodology For Crescent, Electric Bay And Cable Tunnel Coolers”
JAF-CALC-TBC-02464, “Heat Removal Capability of the Electric Bay Coolers (67UC-16A/B) With Two Fans Operating”
1462.9020-US(N)-003, “Average and Maximum Post-LOCA Temperature In East And West Electric Bays With Unit Coolers Operating With 82EF Of Lake Water”

**Section 1R11: Licensed Operator Requalification Program**

EN-TQ-202, “Simulator Configuration Control”
ENN-PL-163, Operations Expectations and Standards”
AP-12.03, “Conduct of Operations”
ENN-HU-102, “Human Performance Tools”
AP-12.06, “Procedure Use and Adherence”

**Section 1R12: Maintenance Effectiveness**

NUMARC 93-01, “Industry Guideline for Monitoring The Effectiveness of Maintenance at Nuclear Power Plants,”
JAF-RPT-EPIC-02280, “Maintenance Rule Basis Document for System 009-0040, Emergency and Plant Information Computer”
Minor Modification Package M1-93-033, “Replacement of EPIC UPS batteries 71BAT-4A and 71BAT-4B”
Letter from C. Phillips, J&M Schaeffer, Inc., to M. McCormack, New York Power Authority, Regarding Load Testing of 128-ls6-200 Batteries, dated 12/23/93
Dwg FE-3KA, “Meteorological Monitoring and Radiological Assessment Systems Terminal”
Dwg FE-1Y, “600 V One line Diag-Sh.14 71MCC-332, 342, 155 & 165"
Boxes and Miscellaneous Wiring Diagrams for Admin Bldg 71UPS-5
JF-030179500, PM - Performance Discharge Test per IEEE 450
JF-030180400, PM - Performance Discharge Test per IEEE 450
NUREG-0696, “Functional Criteria for Emergency Response Facilities”
NRC Inspection Report IR 50-333/88-07
JAF-SE-97-005, “Nuclear Safety Evaluation for Feedwater Flow Ultrasonic Monitoring System”
RAP-7.3.38, “LEFM Operation and Feedwater Correction Factor Calculation”
IMP-2.4, “Reactor Feedwater Temperature Calibration”

JAF-CALC-FWC-02215, “Total Feedwater flow measurement uncertainty using temperature compensated delta-P instrumentation and the leading edge flow meter”
JAF-APL-05-004, “LEFM Recalibration”
JAF Vendor Manual No. C999-0260 (Binder C09), “LEFM Model 8300 Flow Measurement System”
JAF-RPT-FWC-02277, “Maintenance Rule Basis Document, System 006, Feedwater Control System”
WO JF-010573404, A/C Calibrate 02TT-168B per IMP-2.4
JTS-00-0103, Memorandum from S. Defillippo and R. Post to P. Russell regarding acceptability of continued application of LEFM correction factor, dated 6/12/2000
JENG-APL-05-009, “ATTS Cards and Nests (a)(1) Action Plan”

**Section 1R13: Maintenance Risk Assessments and Emergent Work Control**

AP-12.12, “Protected Equipment Program”
AP-05.13, “Maintenance During LCOs”
AP-10.09, “Outage Risk Assessment”
TOP-353, “Alternate Shutdown Cooling”
TOP-355, “Transition to Mode 3 During RHR SDC Suction Line Repair”
WO JAF-05-25011, PFSK-2285 appears cracked where it attached to the RHR SDC suction line
NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management”
WO JF-030071800, 15MOV-175B out of service for actuator/motor control center PMs
WO JAF-04-16949, PM - open/inspect/repair 46SWS-67A, 67UC-16A service water supply check valve
WO JAF-04-13540, PM - perform periodic compressor maintenance on 39AC-2A
WO JF-021018300, Replace 13TU-2 remote servo
WO JAF-04-12271, Repair oil leak on threaded piping to RCIC turbine lube oil sample valve
ST-24J, “RCIC Flow Rate and Inservice Test”
WO JAF-05-24248, Perform Torus Containment Weld Repairs

**Section 1R14: Operator Performance During Non-routine Evolutions and Events**

ENN-PL-163, “Operations Expectations and Standards”
AP-12.03, “Conduct of Operations”
ENN-HU-102, “Human Performance Tools”
AP-12.06, “Procedure Use and Adherence”

**Section 1R15: Operability Evaluations**

Dwg 11825-6.44-16, 30" Vacuum Breaker Valve
Dwg 11825-6.44-51, 30" Vacuum Breaker Valve
JTS-98-0301, “Surveillance ST-15F for Vacuum breakers 27VB-1 through 5"
JAF-SE-97-039, “Torus/Drywell Vacuum Breaker Alternate Test Method and Review of Primary Containment Inerting and Deinerting Operations”
DBD-016A, “Primary Containment Penetrations and Isolation Devices”

OP-37, “Containment Atmosphere Dilution System”
JAF-CALC-CAD-04450, “Shaft Breakaway Torque Corresponding to 0.5 psid for Vacuum Breakers 27VB-1 thru 5"
ST-15F, “Pressure Suppression Chamber-Drywell Vacuum Breaker Setpoint Test”
ST-15J, “Torus to Drywell Vacuum Breakers Quarterly (IST),” completed 7/19/05
ST-15J, “Torus to Drywell Vacuum Breakers Quarterly (IST),” completed 7/25/05
OP-4, “Circulating Water System”

**Section 1R16: Operator Work-Arounds**

WO JAF-04-40213, A RFP tachometer-generator output low
WO JAF-04-41081, B RFP tachometer-generator output low
WO JAF-05-15730, Torus exhaust inner isolation valve operator air-to-manual actuator adjustment
WO JAF-05-19062, Density compensation for HPCI/RCIC level control
WO JAF-05-25596, Second stage reheat does not track properly in automatic
WO JAF-04-38554, 94PS-4 reset value too high
WO JAF-05-22830, 12-4AOV-42A gross seat leakage
WO JAF-05-26273, A1 waterbox sight glass does not indicate level

**Section 1R19: Post Maintenance Testing**

AP-05.07, “Maintenance Testing and Post-Work Testing”
Dwg FM-25A, “Flow Diagram High Pressure Coolant Injection, System 23"
Dwg FK-4D, “Instrument Piping Level Control and Switches, Sheet 4"
WO JAF-04-12518, EPIC SPDS Torus Water Level Algorithm swaps from narrow to wide range band during system operation
IMP-G42, “Instrument Venting/Filling”
WO JF-000199600, Vent transmitter per IMP-G-42 (Admin WR - Minor Maintenance)
JAF-CALC-HPCI-02102, “Narrow Range Suppression Pool Level Uncertainty Calculation”
IS-S-01, “Tubing and Support Installation”
WO JAF-05-18184, PWT for WO JAF-04-29449
Dwg ESK-6MP, “Elementary diagram 600 V Ckts-MOV RHR Inboard valves 10MOV-25A & B”
Dwg 1.43-227, “AC MCC Elementary and Intern wiring diagram COMPT OJ4"

**Section 1R22: Surveillance Testing**

JTS-94-0378, “Clarification of Surveillance Test Level 2 Acceptance Criteria”
AP-19.01, “Surveillance Testing Program”
Dwg 11825-6.44-16, “30" Vacuum Breaker Valve”
Dwg 11825-6.44-51, “30" Vacuum Breaker Valve”
JTS-98-0301, “Surveillance ST-15F for Vacuum breakers 27VB-1 through 5"
JAF-SE-97-039, “Torus/Drywell Vacuum Breaker Alternate Test Method and Review of Primary Containment Inerting and Deinerting Operations”
DBD-016A, “Primary Containment Penetrations and Isolation Devices”

OP-37, “Containment Atmosphere Dilution System”
JAF-CALC-CAD-04450, “Shaft Breakaway Torque Corresponding to 0.5 psid for Vacuum Breakers 27VB-1 thru 5"
ST-15F, “Pressure Suppression Chamber-Drywell Vacuum Breaker Setpoint Test”
ST-15J, “Torus to Drywell Vacuum Breakers Quarterly (IST),” completed 7/19/05
ST-15J, “Torus to Drywell Vacuum Breakers Quarterly (IST),” completed 7/25/05
DBD-027, “Design basis document for Air Treatment Systems”
WO JAF-04-24492, 70TCV-121A continues to oscillate after troubleshooting
OP-55A, “Control Room Ventilation and Cooling”
JAF-RPT-MULTI-03365, “James A. FitzPatrick Nuclear Power Plant Inservice Testing Program for Pumps and Valves Third Interval Plan,”
JAF-RPT-MULTI-04406, “Inservice Testing Program Basis Document,”
NUREG 1482, “Guidelines for Inservice Testing at Nuclear Power Plants”
IMP-G8, “Temperature Transmitter Calibration”
ISP-85, “Control Room Ventilation Temperature and Differential Pressure Instrument Calibration”
JAF-CALC-HPCI-02102, “Narrow Range Suppression Pool Level Uncertainty Calculation”
Dwg FM-25A, “Flow Diagram High Pressure Coolant Injection, System 23"
Dwg FK-4D, “Instrument Piping Level Control and Switches, Sheet 4"
JAF-CALC-HPCI-02102, “Narrow Range Suppression Pool Level Uncertainty Calculation”

**<u>Section 4OA2: Identification and Resolution of Problems</u>**

<u>Condition Reports</u>

| | | | |
|---|---|---|---|
| 2000-06351 | 2005-03453 | 2005-03099 | 2005-03463 |
| 2001-04649 | 2005-01079 | 2005-03659 | 2005-03272 |
| 2002-04351 | 2005-01579 | 2005-03468 | 2005-03154 |
| 2003-02104 | 2005-03347 | 2005-02967 | 2005-02982 |
| 2003-02269 | 2005-03154 | 2005-03635 | 2005-03818 |
| 2003-01787 | 2005-03452 | 2005-02951 | 2005-01466 |
| 2003-02253 | 2005-03330 | 2005-03962 | 2005-03147 |
| 2003-01581 | 2005-03241 | 2005-03931 | 2005-03199 |
| 2003-02911 | 2005-03041 | 2005-03906 | 2005-03185 |
| 2003-02550 | 2005-03090 | 2005-03896 | 2005-03347 |
| 2003-02943 | 2005-03392 | 2005-03882 | 2005-03374 |
| 2003-02723 | 2005-01290 | 2005-03863 | 2005-03408 |
| 2003-02408 | 2005-03377 | 2005-03860 | 2005-03409 |
| 2003-04173 | 2005-00393 | 2005-03837 | 2005-03422 |
| 2004-00907 | 2005-02341 | 2005-03838 | 2005-03535 |
| 2004-00394 | 2005-03023 | 2005-03844 | 2005-03542 |
| 2004-05462 | 2005-00761 | 2005-02266 | 2005-03562 |
| 2004-00428 | 2005-04039 | 2005-02576 | |
| 2005-01080 | 2005-03314 | 2005-02889 | |

**Section 4OA3: Event Followup**

NUS Technical Bulletin, April 2005, Volume 21, “PID900-540 Controllers - Anti-Windup”
NUS-A021MA, “Operations and Maintenance Manual for PID900 Proportional Integral Derivative Controller”
JAF-MULTI-02107, “James A. FitzPatrick Nuclear Power Station Individual Examination for Severe Accident Vulnerabilities”
Procurement Engineering Technical Evaluation 04-004310, “Scientech-NUS Instruments Controller NUS-A024PA-1"
Item Equivalency Evaluation ENN-04-0404, “Scientech-NUS Instruments Controller NUS-A024PA-1"
Post Transient Evaluation No. 05-003, “September 14, 2005 Automatic Reactor Scram on Low RPV Water Level”
General Electric Dwg No. 187B8307 Sh. 13, “Connection Diagram FVR-3"
Dwg. ESK-11AL, “Elementary Diagram 125Vdc Ckts - MOV HPCI System 23MOV-17 & 19"
Dwg. 1.61-140, “Elementary Diagram HPCI System”
Dwg. 1.61-146, “Elementary Diagram HPCI System”
Dwg. 1.61-142, “Elementary Diagram HPCI System”
NUREG-0800, “U.S. Nuclear Regulatory Commission Standard Review Plan”, Office of Nuclear Reactor Regulation, Section 3.6.1, “Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment”
AP-12.03, “Conduct of Operations”
AP-03.01, “Post Transient Evaluation”
OP-46B, “120 Vac Power System”
AOP-1, “Reactor Scram”
Dwg. ESK-11AQ, “Elementary Diagram - 125 Vdc Ckts - MOV, RCIC System - Outboard Steam Supply Isolation and Steam to Turbine MOVs”
Dwg. ESK-11AR, “Elementary Diagram - 125 Vdc Ckts - MOV, RCIC System - Pump Suction Condensate Storage Tank and Pump Discharge MOVs”
Dwg. 1.61-153, “Elementary Diagram RCIC System”
Dwg. 1.61-152, “Elementary Diagram RCIC System”
SP-01.02, “Reactor Water Sampling and Analysis”
AOP-21, “Loss of UPS”, Revision 17
AOP-21, “Loss of UPS”, Revision 19

## LIST OF ACRONYMS

| | |
|---|---|
| ALARA | as low as is reasonably achievable |
| AOP | abnormal operating procedure |
| CAP | corrective action program |
| CFR | Code of Federal Regulations |
| CR | condition report |
| CRD | control rod drive |
| DBD | design basis document |
| DOT | department of transportation |
| EDG | emergency diesel generator |
| EOP | emergency operating procedure |
| EPIC | emergency and plant information computer |
| ESW | emergency service water |
| FWCS | feedwater control system |
| HRA | high radiation area |
| HVAC | heating, ventilation and air conditioning |
| IMC | inspection manual chapter |
| ISP | instrument surveillance procedure |
| IST | inservice test |
| LEFM | leading edge flow meter |
| LOCA | loss of coolant accident |
| MG | motor generator |
| MGU | motor gear units |
| NCVs | non-cited violations |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| ODCM | offsite dose calculation manual |
| OP | operating procedure |
| PI | performance indicator |
| PCIS | primary containment isolation system |
| PM | preventive maintenance |
| RB | reactor building |
| RBCLC | reactor building closed loop cooling |
| RCA | radiological controlled area |
| RCIC | reactor core isolation cooling |
| RETS | radiological effluents technical specifications |
| RFP | reactor feed pump |
| RG | regulatory guide |
| RHR | residual heat removal |
| RP | radiation protection |
| RPM | radiation protection manager |
| RWCU | reactor water cleanup |
| RWP | radiation work permit |
| RWR | reactor water recirculation |
| SDP | significance determination process |
| SSC | structure, system, and component |

| | |
|---|---|
| TB | turbine building |
| TS | technical specification |
| UFSAR | Updated Final Safety Evaluation Report |
| UPS | uninterruptible power supply |
| Vac | volts, alternating current |
| Vdc | volts, direct current |