UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

May 8, 2008

Mr. Mark Bezilla
Site Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P. O. Box 97, 10 Center Road, A-PY-290
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT NRC INTEGRATED INSPECTION REPORT 05000440/2008002

Dear Mr. Bezilla:

On March 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings of the baseline inspection program which were discussed on April 15, 2008, with you and other members of your staff. Included in this inspection report is the inspection conducted under Inspection Procedure (IP) 92702, to review results of the independent assessment of your corrective action program.

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. In addition, it has been determined that Perry is in the Regulatory Response column of the Action Matrix (as outlined in our letter of May 8, 2007). In March of 2008, the NRC reviewed Perry operational performance, inspection findings, and performance indicators for the fourth quarter of 2007. Your staff updated the performance indicator for unplanned scrams and this caused the performance indicator to cross the Green to White threshold for the second quarter of 2007. Based on this review, we concluded that Perry was operating safely.

Based on the results of this inspection, five NRC-identified and four self-revealed findings of very low safety significance were identified and also involved violations of NRC requirements. In addition, two NRC-identified non-cited violations of NRC requirements, without an associated finding, were identified. However, because findings associated with these violations were of very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, five licensee-identified violations are listed in Section 4OA7 of this report.

If you contest the subject or severity of any Non-Cited Violation in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspectors' Office at the Perry Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

Docket No. 50-440
License No. NPF-58

Enclosure: Inspection Report 05000440/2008002
w/Attachment: Supplemental Information

cc w/encl: J. Hagan, President and Chief Nuclear Officer - FENOC
J. Lash, Senior Vice President of Operations and Chief Operating Officer - FENOC
D. Pace, Senior Vice President, Fleet Engineering - FENOC
J. Rinckel, Vice President, Fleet Oversight - FENOC
Director, Fleet Regulatory Affairs - FENOC
Manager, Fleet Licensing - FENOC
Manager, Site Regulatory Compliance - FENOC
D. Jenkins, Attorney, FirstEnergy Corp.
Public Utilities Commission of Ohio
C. O'Claire, Chief, Ohio Emergency Management Agency
R. Owen, Ohio Department of Health

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R. Owen, Ohio Department of Health

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Letter to M. Bezilla from J. Cameron dated May 8, 2008

SUBJECT: PERRY NUCLEAR POWER PLANT NRC INTEGRATED INSPECTION REPORT 05000440/2008002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	050000440/2008002
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	Perry, Ohio
Dates:	January 1, 2008, through March 31, 2008
Inspectors:	M. Franke, Senior Resident Inspector M. Wilk, Resident Inspector G. Wright, Project Engineer A. Dahbur, Senior Reactor Engineer
Approved by:	Jamnes L. Cameron, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000440/2008002; 01/01/2008 – 03/31/2008; Maintenance Effectiveness; Maintenance Risk Assessments and Emergent Work Control; Evaluations of Changes, Test, or Experiments and Permanent Plant Modifications; Surveillance Testing; Event Follow-up

The inspection was conducted by resident and regional inspectors. The report covers a three month period of resident inspection, in addition to the IP 92702 inspection. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be "Green," or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated July 2006.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Event

- Green A finding of very low safety significance and a non-cited violation of Technical Specification (TS) 5.4, “Procedures,” was self-revealed on January 4, 2008, when reactor steam was observed coming from the from the 'A' reactor water cleanup (RWCU) system as operators opened the pump suction shutoff valve. A system isolation valve that was danger-tagged as shut to provide double-boundary protection from the reactor coolant system was found in the open position. At the time of the event, licensee personnel were in the process of restoring the 'A' RWCU pump to service following maintenance and the reactor was at rated power and pressure. As part of their immediate corrective actions, licensee personnel isolated the leak, performed a system alignment, and entered this issue into their corrective action program.

  The finding was considered more than minor because it was associated with the Human Performance attribute of the Initiating Events cornerstone 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 functions. Specifically, the finding resulted in a reactor coolant leak to the safety-related auxiliary building. The finding was determined to be of very low safety significance because the reactor water leak was readily isolable. The primary cause of this finding was related to the cross-cutting area of Human Performance as defined by IMC 0305 H.1(b) because licensee personnel failed to use conservative assumptions in decision making associated with the valve tagging procedure. Section (1R13)

- Severity Level IV The inspectors identified a non-cited violation of 10 CFR Part 50.72(b)(2)(iv)(B), "Four Hour Reports." The inspectors determined that the licensee failed to report a manual actuation of the reactor protection system when it was not part of a preplanned sequence. Specifically, on June 22, 2007, the 'B' reactor recirculation pump failed during a plant shutdown sequence and the licensee inserted a manual scram above preplanned power levels and not in accordance with the preplanned sequence. Licensee operators decided to insert the manual scram earlier than planned due to the unexpected loss of flow in the 'B' reactor recirculation system loop. (Section 4OA1.b.1)

- Green The inspectors identified a finding associated with the licensee's reporting of Unplanned Scram Performance Indicator (PI) data for the second quarter 2007. On July 23, 2007, Perry plant personnel submitted PI data to the NRC that included one unplanned scram for the second quarter of 2007. In August 2007, the inspectors informed the licensee that the NRC disagreed with the reported number of unplanned scrams. The inspectors determined that the licensee failed to pursue resolution of the discrepancy in a timely manner in accordance with established industry standards.

  The finding was considered more than minor because it was related to a PI and would have caused the PI to exceed a threshold. Had all three unplanned scrams been reported in July 2007, the Unplanned Scram PI would have crossed the Green to White threshold. The finding was determined to be of very low safety significance after management review. (Section 4OA1.b.2)

Cornerstone: Mitigating Systems

- Green A finding of very low safety significance and a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," was self-revealed when the 'B' emergency service water (ESW) system pump discharge strainer failed on December 27, 2007. A strainer inspection cover, about 6 inches wide and 9 inches tall, became dislodged due to a loose fastener, and water discharged into the ESW pump house when the 'B' ESW pump was started. The strainer was last worked during a refueling outage in April 2007. The maintenance procedures associated with the strainer were determined to be inappropriate because they resulted in the unexpected failure of the strainer cover. As part of their immediate corrective actions, licensee personnel revised strainer cover installation procedures, repaired the strainer, and restored availability of the 'B' ESW system.

  The finding was considered more than minor because it was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events in order to prevent undesirable consequences. Specifically, the finding resulted in the unavailability of the 'B' ESW system train. The finding was determined to be of very low safety significance because it did not represent an actual loss of safety function of a single train for greater than the TS-allowed outage time. (Section 1R12)

- Green The inspectors identified a finding having very low safety significance and an associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the licensee's failure to adequately evaluate and take appropriate corrective actions for a condition adverse to quality affecting the Emergency Service Water (ESW) Pump 'A' and its associated discharge valve. Specifically, the licensee did not implement adequate actions to ensure that the ESW Pump 'A' discharge valve (1P45F0130A) would remain open and would not be damaged during the loss of direct current (DC) Bus ED-1-A while the pump was in operation. In addition, the licensee did not identify and evaluate the impact of this condition on the plant's safe shutdown equipment in the event of an Appendix R fire in the control room. The licensee entered the issue into their corrective action program.

  This finding was more than minor because the failure to assure that the ESW Pump 'A' discharge valve would remain open and would not be damaged affected the mitigating system corner stone objective of ensuring the availability, reliability and capability of the safety-related components to respond to initiating events to prevent undesirable consequences. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because the specific condition/scenario only affected the ESW Pump 'A' and its associated discharge valve and it did not exist for the redundant ESW Pump 'B'. In addition, safe shutdown components for the Division 2 and/or Division 3 systems would remain available, free of fire damage, to safely shut down the plant in the event of a fire in the control room. The finding has a cross-cutting aspect in the area of problem identification and resolution as defined in Inspection Manual Chapter 0305 P.1(c), because the licensee failed to thoroughly evaluate the problem when it was first identified in 2006. (Section 1R17)

- Green A finding of very low safety significance and a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," was self-revealed during the reactor scram and plant response on November 28, 2007, when reactor core isolation cooling (RCIC) failed to perform its design function. The RCIC system started automatically on low reactor water level, began to inject into the reactor pressure vessel, and then tripped on low suction pressure. The RCIC pump flow controller was found to have been incorrectly tuned in January 2006. As part of their immediate corrective actions, licensee personnel tuned the RCIC controller prior to the December 6, 2007, plant startup.

  The finding was considered more than minor because it was associated with Equipment Reliability attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events in order to prevent undesirable consequences. The finding was determined, through Phase 3 analysis, to be of very low safety significance. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution as defined in IMC 0305 P.2(b) because the licensee failed to institutionalize operating experience through changes to procedures regarding flow controller settings. (Section 1R22.b.1)

- NCV. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," associated with testing of the reactor core isolation cooling (RCIC) system between January 20, 2006, and November 28, 2007, a period when RCIC was determined to have been inoperable. Specifically, the program failed to incorporate the requirements and acceptance limits contained in applicable design documents to assure that RCIC flow controller configuration and performance met design requirements during testing. (Section 1R22.b.2)

- Green The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," while observing a periodic test associated with the reactor core isolation cooling (RCIC) system on February 14, 2008. The inspectors determined that the licensee's procedure was inappropriate for the circumstances of the test. Specifically, the purpose of the test was to detect and quantify gas formation in RCIC system piping and the procedure did not provide an adequate method to determine whether acceptance criteria were met. The repeated performance of the test resulted in the unnecessary inoperability of the RCIC system.

This finding was greater than minor because it adversely affected the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events in order to prevent undesirable consequences. Specifically, the performance of the test affected the capability of the RCIC system to respond to events. The finding was of very low safety significance because the time RCIC was inoperable was less than TS-allowed inoperability time. The primary cause of this finding was related to the cross-cutting area of Human Performance as defined by Inspection Manual Chapter 0305 H.2(c), because the licensee failed to provide complete and accurate procedures related to nuclear safety. As part of their immediate corrective action, the licensee revised the test procedure. (Section 1R22.b.3)

- Green The inspectors identified a finding of very low safety significance and a non-cited violation of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Action," when the reactor core isolation cooling (RCIC) system was declared inoperable on December 12, 2007, due to improper flow controller settings. The inspectors noted that the cause of RCIC inoperability on December 12, 2007, was the same cause of RCIC inoperability from January 21, 2006, to November 28, 2007. The licensee failed to perform adequate corrective actions to preclude repetition of a significant condition adverse to quality. As part of their immediate corrective actions, the licensee entered the issue into the corrective action program and adjusted flow controller settings to 1987 pre-startup settings when RCIC successfully injected into the reactor pressure vessel.

  The finding was more than minor because it was associated with the Equipment Performance attribute of the reactor safety Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the controller settings affected the capability of the RCIC system to respond to initiating events as designed. The finding was determined to be of very low safety significance because it was determined that the period of inoperability was less than the TS-allowed outage time. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution as defined in Inspection Manual Chapter 0305 P.2(a) because the licensee failed to communicate relevant external operating experience in a timely manner. (Section 4AO3.3)

- Green The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," during a review of the licensee's treatment of the safety-related reactor core isolation cooling (RCIC) system's failure to perform its safety function when called upon during an event. On November 28, 2007, the licensee experienced an unplanned scram with complications that included a failure of the feedwater system affecting all feed pumps. During the event, RCIC failed to function as designed when aligned to the suppression pool and when re-aligned to the condensate storage tank. Licensee personnel failed to identify the RCIC failures as a significant condition adverse to quality within their corrective action program. As part of their immediate corrective actions, licensee personnel reclassified the condition as a significant condition adverse to quality.

  The finding was considered more than minor because the failure to identify significant conditions adverse to quality would become a more significant safety concern if left uncorrected. The finding was determined to be of very low safety significance after management review. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution as defined in Inspection Manual Chapter

IMC 0305 P.1(a), because the licensee failed to identify the issue completely, accurately, and in a timely manner commensurate with its safety significance. (Section 4OA3.5)

Cornerstone: Barrier Integrity

- Green A finding of very low safety significance and a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," was self-revealed when a loss of the annulus exhaust gas treatment system (AEGTS) safety function occurred on December 21, 2007. Maintenance procedures failed to include adequate instructions and acceptance criteria related for a hydramotor assembly and this resulted in the inoperability of the 'B' AEGTS train while the 'A' train was inoperable for charcoal sampling. As part of their immediate corrective actions, licensee personnel restored 'A' train to operable status and entered the issue into the corrective action program.

  The finding was more than minor because it was associated with the Procedure Quality attribute related to maintenance of containment function of the Barrier Integrity cornerstone and affected the cornerstone objective of reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the finding was determined to have resulted in a degraded condition of secondary containment. The finding was of very low safety significance because the finding only represented a degradation of the radiological barrier function. The primary cause of this finding was related to the cross-cutting area of Human Performance per Inspection Manual Chapter 0305 H.2(c), because the licensee failed to provide complete and accurate procedures related to nuclear safety. (Section 4OA3.6)

B. Licensee-Identified Violations

Five violations of very low safety significance that were identified by the licensee have been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee’s corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period at 100 percent power. On March 1, 2008, operators reduced reactor power to 90 percent to confirm the existence of a small fuel defect. Operators returned reactor power to 100 percent on March 3, 2008. On March 14, 2008, operators commenced a series of power suppression tests in order to determine the location of the fuel defect. Power was reduced to as low as 59 percent during testing and operators were unable to determine the location of the fuel defect. The reactor was returned to 100 percent power on March 22, 2008.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors conducted a partial walkdown of the following systems to determine whether the system was correctly aligned to perform its designed safety function. The inspectors used valve lineup instruction (VLIs) and system drawings during the walkdown. The walkdown included selected switch and valve position checks, and verification of electrical power to critical components. Finally, the inspectors evaluated other elements, such as material condition, housekeeping, and component labeling. The documents used for the walkdown are listed in the attached List of Documents Reviewed.

The following partial system walkdowns represent three quarterly inspection samples:

- emergency closed cooling water 'A' following maintenance during the week of February 4, 2008;
- safety air system following maintenance during the week of February 4, 2008; and
- 13.8KV and 4.16KV plant electrical system during maintenance affecting the Unit 1 LH1A power transformer during the week of March 10, 2008.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (Quarterly/Annual) (71111.05AQ)

a. Inspection Scope

The inspectors walked down the following areas to assess the overall readiness of fire protection equipments and barriers:

- Fire Zone 1AB-1C; Unit 1 RCIC System Pump Room, elevation 574’10”;
- Fire Zone 1AB-1A; Unit 1 LPCS System, elevation 574’10”;
- Fire Zone 0CC-1A,1B, and 1C; Control Complex, elevation 599’;
- Fire Zone 0IB-2; Intermediate Building, elevation 599’;
- Fire Zone 0IB-5; Intermediate Building, elevation 682’;
- Fire Zone 1RB-1C-1B; Containment to Drywell Space;
- Fire Zone 0IB-1; Intermediate Building, elevation 574’; and
- Fire Drill-1125 0328 08 "Fire In Radwaste Truck Bay" conducted March 28, 2008.

Emphasis was placed on evaluating the licensee's control of transient combustibles and ignition sources, the material condition of fire protection equipment, the material condition and operational status of fire barriers used to prevent fire damage or propagation. The inspectors utilized the general guidelines established in Fire Protection Instruction (FPI)-A-A02, "Periodic Fire Inspections," Revision 5; Perry Administrative Procedure (PAP)-1910, "Fire Protection Program," Revision 15; and PAP-0204, "Housekeeping/Cleanliness Control Program," Revision 20; as well as basic National Fire Protection Association Codes, to perform the inspection and to determine whether the observed conditions were consistent with procedures and codes.

The inspectors observed fire hoses, sprinklers, and portable fire extinguishers to determine whether they were installed at their designated locations, were in satisfactory physical condition, and were unobstructed. The inspectors also evaluated the physical location and condition of fire detection devices. Additionally, passive features such as fire doors, fire dampers, and mechanical and electrical penetration seals were inspected to determine whether they were in good physical condition. The documents listed in the List of Documents Reviewed at the end of this report were used by the inspectors during the inspection of this area.

These reviews represent seven quarterly inspection samples and one annual inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope

The resident inspectors reviewed licensed operator training conducted in the plant simulator on February 18, 2008. The inspectors evaluated crew performance in the areas of:

- clarity and formality of communication;
- ability to take timely action in the safe direction;
- prioritizing, interpreting, and verifying alarms;
- correct use and implementation of procedures, including alarm response procedures;

- timely control board operation and manipulation, including high-risk operator actions; and
- group dynamics.

This review represents one quarterly inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule requirements to determine whether component and equipment failures were identified and scoped within the maintenance rule and that select structures, systems, and components were properly categorized and classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors reviewed station logs, maintenance work orders (WOs), selected surveillance test procedures, and a sample of condition reports (CRs) to determine whether the licensee was identifying issues related to the maintenance rule at an appropriate threshold and that corrective actions were appropriate. Additionally, the inspectors reviewed the licensee's performance criteria to determine whether the criteria adequately monitored equipment performance and to determine whether changes to performance criteria were reflected in the licensee's probabilistic risk assessment. During the inspection period the inspectors reviewed the following systems:

- emergency service water (ESW); and
- emergency diesel generators (EDGs).

These maintenance effectiveness reviews constitute two inspection samples.

b. Findings

Introduction: A finding of very low safety significance and a non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," was self-revealed when a 'B' ESW system pump discharge strainer failed on December 27, 2007. A strainer inspection cover, about 6 inches wide and 9 inches tall, separated from the strainer body, and water discharged into the ESW pump house when the 'B' ESW pump was started.

Description: On December 27, 2007, licensee personnel were performing routine surveillance testing of the 'B' ESW system train. After obtaining satisfactory results, operators stopped the pump. When the pump was stopped, operators observed that system keepfill pressure was about 2.5 psig, which was below a minimum of 14.5 psig required for system operability.

About four minutes after the pump was stopped, in response to the low keepfill pressure, operators restarted the 'B' ESW pump. After the pump was restarted, control room operators observed that alarms associated with low heat exchanger flows did not reset as expected, and keepfill pressure remained lower than expected at less than 3 psig.

The operators observed that indicated heat exchanger flows were significantly less than expected and a plant operator was sent to the ESW pump house to investigate.

About three minutes after the pump was started, a plant operator arrived at the ESW pump house. The plant operator reported that the 'B' ESW pump discharge strainer had a sizable leak, water was spraying into the pump house, and water level on the pump house floor was about 1 foot deep. Operators in the control room stopped the 'B' ESW pump. The 'B' ESW system was declared inoperable and unavailable. Operators estimated that the pump had been running for a total of about five minutes before it was stopped.

The 'B' ESW strainer was designed with upper and lower inspection ports. These ports were normally closed with a dish-shaped metal cover and gasket that fit against the inside of the strainer body. A strongback assembly, including a single bolt and fastener, was used to hold the cover in place. The design was such that the bolt head fit into a U-shaped bracket on the cover. A nut on the strongback end of the bolt was used to draw the cover and strongback together to press the cover up against the inside of the strainer body wall.

After the event, the upper portal was found open and the upper dish-shaped cover was missing. The strongback assembly was found on the pump house floor near the strainer. The strainer upper cover and gasket were later found inside the strainer. The bolt nut on the strongback assembly was found loose in that it was flush with the end of the bolt threads. The licensee determined that, for a normally fastened cover, the bolt should have protruded from the nut by about two to three threads.

Licensee personnel performed an investigation to determine whether the water spray from the event had caused damage to equipment in the pump house. Water spray had wetted the motor fire pump and safety-related 'C' ESW system components, including electrical switchgear cabinets. The non-safety-related motor fire pump was running at the time of the event and remained running after the event without any indications of degradation. Licensee personnel inspected the 'C' ESW pump and affected electrical cabinets, and determined that no equipment damage had occurred and that the systems remained operable. A temperature instrument associated with the 'A' ESW system was found to be degraded, but this was not determined to have affected the 'A' ESW system operability.

The licensee determined that the strainer cover was last worked in April 2007 during a refueling outage and that the nut had most likely come loose following this maintenance. The licensee determined that, when the strongback assembly became loose enough, the cover most likely slid off the bolt head. The cover was determined to have fully dislodged during shutdown of the 'B' ESW pump following the surveillance test run that preceded the event.

A review of internal and external operating experience (OpE) found that similar events had occurred in the past. In particular, an internal OpE review found that the 'A' ESW pump strainer lower cover had fallen off in the year 1990. The licensee had implemented corrective actions to periodically check the tightness of the covers and to install a design change to preclude the event from recurring. The corrective actions of periodic tightness checks were eventually stopped because the licensee determined that no adjustments were required following the checks. The corrective action to install a design change was not implemented.

As part of their immediate corrective actions following the December 2007 event, a procedure and design change was implemented to stake the nuts to the bolt by scoring the bolt threads after installation. The licensee performed an extent-of-condition review and entered the issue into their corrective action program (CAP) as CR 07-31994.

Because the April 2007 implementation of maintenance procedures associated with the ESW strainer cover resulted in the failure of the cover, the inspectors determined that the procedures were inappropriate.

Analysis: The inspectors determined that the failure to implement appropriate maintenance procedures affecting the 'B' ESW system was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," dated September 20, 2007. The finding was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events in order to prevent undesirable consequences. Specifically, the finding resulted in the unavailability of the 'B' ESW train.

The inspectors performed a significance determination of this issue using IMC 0609.04, "Phase 1- Initial Screening and Characterization of Findings," dated January 10, 2008. Though the finding was determined to also affect the Initiating Events cornerstone, the Mitigating Systems reactor safety cornerstone was determined to be the cornerstone that best reflected the dominant risk of the finding. The finding was determined to be of very low safety significance because it did not represent an actual loss of safety function of a single train for greater than its Technical Specification (TS)-allowed outage time.

Enforcement: Criterion V, "Procedures," of 10 CFR Part 50, Appendix B, required in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to this requirement, on April 7, 2007, licensee personnel failed to implement procedures appropriate to the circumstances affecting the safety-related 'B' ESW system. Specifically, the maintenance procedure, WO 200188470, did not ensure that the 'B' ESW pump strainer cover was adequately fastened and this led to strainer failure on December 27, 2007. However, because of the very low safety significance of the issue and because the issue has been entered into the licensee's CAP (CR 07-31994); the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. This issue (NCV 05000440/2008002-01) is related to Unresolved Item (URI) 05000440/2007005-09 and closes this URI.

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

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk, scheduling, configuration control, and performance of maintenance associated with planned and emergent work activities to determine whether scheduled and emergent work activities were adequately managed in accordance with 10 CFR 50.56(a)(4). In particular, the inspectors reviewed the licensee's program for conducting maintenance risk assessments to determine whether the licensee's planning, risk management tools, and the assessment and management of on-line risk were adequate. The inspectors also reviewed licensee actions to address increased on-line risk when equipment was out of service for maintenance, such as establishing compensatory actions, minimizing the duration of the activity, obtaining appropriate management approval, and informing appropriate plant staff, to determine whether the actions were accomplished when on-line risk was increased due to maintenance on risk-significant structures, systems, and components. The following assessments and/or activities were reviewed and represent a total of five samples:

- Division 2 electrical outage during the week of January 28, 2008;
- reactor water cleanup (RWCU) system repairs during the week of January 2, 2008;
- RCIC system flow controller repairs during the week of January 14, 2008;
- RCIC outage during the week of February 19, 2008; and
- reactor power suppression testing during the week of March 10, 2008.

b. Findings

Introduction: A finding of very low safety significance and an NCV of TS 5.4, “Procedures,” was self-revealed on January 4, 2008, when steam unexpectedly came out of the 'A' RWCU system as operators opened the pump suction shutoff valve. A system isolation valve that had been danger-tagged shut, to provide double-boundary protection from the reactor coolant system, was found in the open position.

Description: On January 4, 2008, maintenance personnel had completed work to replace the 'A' RWCU pump and plant operators were in the process of restoring the pump to service. The clearance control requirements for the maintenance included the danger-tagging of a valve in the pump suction path. The tagged suction valve was required to be closed, in conjunction with a second closed suction valve, to ensure double-boundary protection from the reactor coolant system during the maintenance activity. The maintenance was performed while the reactor plant was operating at full rated power and pressure and the 'B' RWCU pump remained in service.

Following the maintenance activity, operators began to place the 'A' RWCU pump back in service. The system restoration procedure required operators to fill and vent the portion of the system associated with the 'A' RWCU pump. The system fill piping connection was located between the two suction valves that were providing isolation for the maintenance. The pump shutoff isolation valve, which was the valve nearest to the pump, was in the closed position. Operators opened vent valves on the pump casing to check for stored system energy. Operators then began to open the suction shutoff valve to provide a path between the fill line connection and the pump. As they opened the valve, operators observed steam coming into the room from the area of the pump casing and vents. Operators immediately closed the pump shutoff valve and the leak stopped.

On investigation, operators noted that the danger-tagged isolation valve was in the open position, contrary to the clearance requirement. Clearance procedure PY1-G33-0004; RWCU Pump A; dated January 4, 2008, had required personnel to place valve, 1G33F0034A, in the closed position. Licensee personnel documented placing this valve in the closed position on December 24, 2007. On January 4, 2008, the administratively controlled shutoff valve was opened during the fill and vent attempt, a path was opened for high pressure reactor water to exit out the pump vents. No personnel injuries occurred due to the event and the licensee determined no equipment damage occurred.

As part of their immediate corrective actions, licensee personnel performed an equipment alignment for the RWCU system and entered the issue into their CAP. The licensed operator and the plant operators associated with the initial valve line-up were removed from duties pending investigation of the event.

Analysis: The inspectors determined that the licensee's failure to adequately implement clearance control procedures associated with the 'A' RWCU pump maintenance was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," dated September 20, 2007. The finding was associated with the Human Performance attribute of the Initiating Events cornerstone and 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. Specifically, the finding resulted in a leak of reactor coolant that was at rated pressure into the safety-related auxiliary building.

The inspectors performed a significance determination of this issue using IMC 0609.04, "Phase 1- Initial Screening and Characterization of Findings," dated January 10, 2008. The Initiating Events reactor safety cornerstone was determined to be the cornerstone that best reflected the dominant risk of the finding. The inspectors considered that the finding resulted in a reactor water leak into the safety-related auxiliary building. The leak was isolated promptly, the fill line was not opened to expose it to reactor pressure, and the leak was downstream of automatic isolation valves that could have been used to isolate it. The inspectors determined that it was not reasonable to assume that a "worst case" scenario would result in exceeding a TS limit or likely affect the mitigating systems in the building before isolation. Therefore, the inspectors determined that the finding was of very low safety significance. The primary cause of this finding was related to the cross-cutting area of Human Performance as defined by IMC 0305 H.1(b) because licensee personnel failed to use conservative assumptions in decision making associated with the RWCU alignment and tagging clearance procedure. Specifically, personnel verified the position of a valve using a reach rod that was known to stick in position. When personnel were unable to turn the reach rod wheel past a certain point, they assumed that the valve was closed when it was actually open. The licensee determined that the known condition of the reach rod should have warranted added efforts to verify the actual position of the valve.

Enforcement: Technical Specification 5.4, "Procedures," required the implementation of the applicable procedures recommended in Regulatory Guide 1.33, "Quality Assurance Program Requirements," Revision 2, dated February 1978. Regulatory Guide 1.33, Appendix A, recommended procedures for the reactor cleanup system. Clearance procedure PY1-G33-0004 for the reactor water cleanup system instructed operators to close valve 1G33F0043A. Contrary to this requirement, on December 24, 2007, licensee personnel failed to close valve 1G33F0043A and this resulted in a reactor water leak into the auxiliary building on January 4, 2008. However, because of the very low safety significance of the issue and because the issue has been entered into the licensee's CAP (CR 08-32531); the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2008002-02)

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected CRs related to potential operability for risk-significant components and systems. These CRs were evaluated to determine whether the operability of the components and systems was justified. The inspectors compared the operability design criteria in the appropriate sections of the TS and Updated Safety Analysis Report (USAR) to the licensee's evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures were in place, would function as intended, and were properly controlled.

Additionally, the inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. The inspectors reviewed the follow issues:

- Division 3 ESW during the week of January 7, 2008;
- RCIC during the week of January 21, 2008;
- Residual Heat Removal (RHR) 'A' during the week of March 10, 2008; and
- Division 1 EDG during the week of March 17, 2008.

These reviews represent four inspection samples.

b. Findings

No findings of significance were identified.

1R17 Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors followed up on URIs 05000440/2007006-01 and 05000440/2007006-02 that were opened during the Modification/50.59 Inspection in September 2007. Specific documents reviewed are listed in the attachment.

These follow-ups do not represent an inspection sample.

b. Findings

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” having very low safety significance (Green) involving the licensee’s failure to adequately evaluate and take appropriate corrective actions to address a condition that affected ESW Pump 'A' and its associated discharge valve. Specifically, the licensee did not implement appropriate actions to ensure that the ESW Pump 'A' discharge valve (1P45F0130A) would remain open when the pump was in operation during the loss of direct current (DC) Bus ED-1-A. In addition, the licensee did not identify and evaluate the impact of this condition on the plant’s safe shutdown equipment in the event of an Appendix R fire in the control room.

<u>Description</u>: The licensee initiated CR 06-03087 in July 2006, which identified a condition that affected ESW Pump 'A' and its associated discharge valve (1P45F0130A). Specifically, the licensee identified that if ESW Pump 'A' was in operation when a loss of DC Bus ED-1-A occurred, then the discharge valve 1P45F0130A would automatically close while the pump continued to run. The pump breaker would not trip because of the loss of its DC control power. In addition, remote operation of the breaker from the control room would also be lost. Therefore, the ESW Pump 'A' would continue to run at shutoff head until operator action was taken to open the breaker locally. At the time of discovery, there was no procedural guidance in Off-Normal Instruction (ONI)-R42-1 "Loss of DC Bus ED-1-A," to direct the operators to shut down the pump.

The licensee reviewed the ESW pump and discharge valve electrical drawings B-208-176, Sheets 1 and 4, which indicated that a loss of DC control circuit for ESW Pump 'A' would result in de-energizing control relay 1P45-K8 that would consequently initiate a close signal to the discharge valve. The CR evaluation also indicated that if an open signal was present as it would be in the case of the Division 1 EDG running or a Division 1 loss-of-coolant-accident (LOCA) signal, the discharge valve would cycle open and closed until the operator took manual control of the valve. In the case of no open signal being present, the discharge valve would close and remain closed until manual control was taken to reopen the valve. However, due to the loss of DC power, once the valve was fully opened it would automatically cycle closed. As a result of the CR evaluation, the licensee revised ONI-42-1 and added steps that directed the operators to repeatedly open the ESW Pump 'A' discharge valve until an operator removed control relay 1P45-K8 at panel 1H13-P872. New instructions were also added which directed the operators to trip the ESW Pump 'A' locally, if the pump was not required to support equipment operation.

During the inspectors' review of Screening 06-03964 for the revision to Procedure ONI-R42-1, the inspectors questioned if there were any limitations on the number of times that the discharge valve could be cycled/stroked. Based on information specified in FTI-F0016, "Motor Operated Valve Diagnostic Testing," the licensee indicated that there was a limit of five times per every five minutes. Since test results indicated the valve strokes in approximately 30 seconds, the inspectors were concerned that the discharge valve motor could be damaged, while the valve was in the closed position because of the repeated cycling either by the operator action or automatically due to an EDG start or LOCA signal.

During the Modification/50.59 Inspection, the inspectors were also concerned that the licensee did not evaluate this condition/scenario impact on the Appendix R safe shutdown equipment and analysis. Specifically, the inspectors were concerned regarding the operability and reliability of the Division 1 EDG and its associated ESW pump in the event of a fire in, and evacuation of, the control room. Assuming a loss of offsite power (LOOP) that would result in starting the Division 1 EDG and its associated ESW Pump 'A', a subsequent loss of DC control power for the ESW pump due to fire-induced failures (i.e., loss of fuse) that could occur prior to control room isolation at the alternate shutdown panel, would cause the ESW discharge valve to close. Operating under no flow conditions could result in damage to the ESW Pump 'A' and/or damage to the Division 1 EDG due to the loss of cooling provided by ESW. Following identification of this issue, the licensee entered this issue into their corrective action program as CR 07-26412. This issue was left unresolved at the conclusion of the 50.59 inspection pending further NRC review of Perry's Appendix R evaluation for a control room fire and the affect of the above scenario on safe shutdown equipment.

The licensee completed a cause analysis/evaluation for CR 07-26412 and concluded that the postulated series of fire induced conditions, as described above, that would potentially lead to the equipment (i.e., EDG 1) being unavailable was not considered to be credible under the Perry Licensing Basis and current regulatory guidance. The licensee's conclusion was based on the premise that the postulated conditions would require two independent spurious operations. These spurious operations were the startup of the EDG due to fault in the diesel starting circuit (i.e., a fault that could be caused by disconnection of the offsite power or other fire induced fault that could cause a diesel initiation signal) and the loss of DC power for the Pump 'A' start control logic due to open circuit in the DC power or a short that causes the fuse on the power supply to open. The evaluation also indicated that the original plant design for the Unit 1 Control Room fire was consistent with the current state of regulatory guidance which did not require the plant to postulate multiple spurious operations for fire areas defined by Appendix R, Paragraph III.G.3 (i.e., Control Room Fire Area).

The inspectors reviewed the licensee evaluation, Regulatory Guide 1.189, and Branch Technical Position CMEB 9.5-1 "Guidelines for Fire Protection for Nuclear Power Plants," and determined that the control room fire scenario described above was not a result of multiple/two independent spurious operations as stated in the licensee evaluation. Specifically, the inspectors did not consider the start of the EDG-1 upon LOOP a spurious operation, but per plant design.

As a result of the inspectors' finding, the licensee initiated a procedure change and provided operator guidance to prevent any damage to the Division 1 EDG in the event of a fire in the control room. Also, based on the investigation of CR 07-26073, the licensee concluded that the actions stated in ONI-R42-1 to protect ESW Pump 'A' thru cycling its discharge valve 1P54F0130A to the open position were undesirable. Accordingly, Corrective Action 07-26073-01 was assigned to present a proposed Engineering Change to the Plant Health Committee for assigned priority and schedule with respect to removing the DC control circuit vulnerability in the discharge valve circuit design.

Analysis: The inspectors determined that the failure to adequately evaluate and take appropriate corrective actions for a condition adverse to quality associated with the ESW Pump 'A' and its discharge valve was a performance deficiency warranting a significance evaluation.

The inspectors determined that the performance deficiency was more than minor in accordance with IMC 0612, Appendix B, "Issue Disposition Screening," because the finding was associated with the design control attribute of the Mitigating System cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of the ESW Pump 'A' and its associated discharge valve to respond to initiating events to prevent undesirable consequences. Specifically, the procedural guidance provided in ONI-R42-1 was not adequate to ensure that ESW Pump 'A' associated discharge valve (1P45F0130A) would have remained open and would not have been damaged in the event of loss of DC control power to ESW Pump 'A' supply breaker, DC Bus ED-1-A. In addition, there was no procedural guidance to ensure that the Division 1 EDG would not be damaged due to loss of cooling water in the event of a fire in the main control room.

The inspectors evaluated the finding using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening. The inspectors answered “No” to all the screening questions under the Mitigating System cornerstone because the specific condition/scenario only affected the ESW Pump 'A' and its associated discharge valve and it did not exist for the redundant ESW Pump 'B'. In addition, because IMC 0609 Appendix F does not currently include explicit treatment of fires leading to main control room abandonment, the Region III Senior Reactor Analyst (SRA) performed a Phase 3 SDP analysis using data and information from the draft NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities.” The overall control room fire frequency was determined to be 4.8E-3. However, the control room fire frequency was adjusted by considering that a fire in only one of the 29 main control board cabinets would result in a scenario that would have affected all three divisions, including Division 1, 2 and high pressure core spray (HPCS). Based on Appendix “S” of NUREG/CR-6850, “Fire Propagation to Adjacent Cabinets,” the non-suppression probability for a control room fire lasting 30 minutes was estimated to be 1E-3. The result for an unsuppressed main control room fire that could result in the failure of all three divisions was estimated 1.7E-7. The SRA determined that this result was bounding and additional analysis would remove conservative assumptions. Therefore, the finding was determined to be best characterized as having very low safety significance (Green).

The finding has a cross-cutting aspect in the area of Problem Identification and Resolution because the licensee did not thoroughly evaluate and correct the issue when it was identified in July 2006 (IMC 0305 P.1(c)).

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Contrary to the above, in July 2006, the licensee did not adequately evaluate or implement appropriate corrective actions for a condition adverse to quality (CR 06-03087) associated with ESW Pump 'A' and its discharge valve 1P45F0130A. Specifically, the licensee did not consider the limitation on the number of times that the discharge valve could be cycled/stroked when they revised procedure ONI-R42-1 including adding guidance to cycle the discharge valve 1P45F0130A in the event of loss of DC Bus ED-1-A while the pump was running. In addition, the licensee failed to evaluate the effect of this condition on the Division 1 EDG in the event of a fire in the main control room. However, because this violation was of very low safety significance and because the issue was entered into the licensee’s CAP (CR 07-26073), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2008002-03).

1R19 <u>Post-Maintenance Testing</u> (71111.19)

a. <u>Inspection Scope</u>

The inspectors selected the following post-maintenance testing activities for risk-significant systems to ensure the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written; and equipment was returned to its operational status following testing. The inspectors evaluated the activities against TS, the USAR, 10 CFR Part 50 requirements, licensee procedure and various NRC generic communications. In addition, the inspectors reviewed CRs associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP. The specific procedures and CRs reviewed are listed in the attached List of Documents Reviewed. The following post-maintenance activities were reviewed:

- emergency closed cooling pump 'A' during the week of January 7, 2008;
- RCIC repairs during the week of January 21, 2008;
- RCIC during the week of February 25, 2008;
- hydraulic control unit for Rod 06-47 during the week of March 17, 2008; and
- remote shutdown panel fuse during the week of March 17, 2008.

The inspectors' reviews of these post-maintenance testing activities represent five inspection samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing or reviewed test data for risk-significant systems or components to assess compliance with TS, 10 CFR 50, Appendix B, and licensee procedure requirements. The testing was also evaluated for consistency with the USAR. The inspectors verified that the testing demonstrated that the systems were ready to perform their intended safety functions. The inspectors determined whether test control was properly coordinated with the control room and performed in the sequence specified in the surveillance instruction (SVI), and if test equipment was properly calibrated and installed to support the surveillance tests. The procedures reviewed are listed in the attached List of Documents Reviewed.

The inspectors selected the following surveillance testing activities for review:

- RCIC tuning and routine testing during the week of January 21, 2008;
- Division 2 EDG routine testing during the week of January 28, 2008;
- RCIC system in-service testing during the week of February 11, 2008;
- RCIC leak detection system testing during the week of February 18, 2008; and
- control rod scram time routine testing during the week of March 17, 2008.

These reviews constitute one in-service; one reactor cooling system leak detection; and three routine inspection samples.

b. Findings

b.1 Introduction: A finding of very low safety significance and an NCV of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," was self-revealed during the reactor scram and subsequent RCIC actuation on November 28, 2007. The RCIC pump started in response to the actuation signal, but then tripped on a low suction pressure signal. The cause was determined to be an improperly tuned flow controller.

Description: On November 28, 2007, the reactor scrammed due to a failure in the digital feedwater control system (DFWCS) and the reactor vessel water level reached Level 2. At Level 2, RCIC received an actuation signal and commenced injecting into the reactor vessel. Shortly thereafter, RCIC tripped on low suction pressure. It was identified that the RCIC flow controller was tuned to an over reactive state, rather than an over dampened condition necessary for proper control of the system. The over reactive condition resulted in flow oscillations that eventually caused a low suction pressure condition and a pump trip. During this event there was a loss of all feedwater due to degraded power supplies associated with the DFWCS and reactor pressure vessel (RPV) water level was maintained by the HPCS system.

The licensee's procedure, ICI-C-E51-3, "RCIC Control System Tuning," Revision 5, did not contain appropriate guidance or reference to the vendor manual to assure adequate system response. The RCIC flow controller was a Bailey Type 701 Controller, and the licensee's procedure for this controller type, ICI-B16-15, "Plant Instrument Calibration Instruction – Bailey Type 701 Controller," Revision 3, did not contain adequate guidance from the vendor manual. Specifically, the vendor manual, Bailey Controls 4570K11-300G, "Type 701 Basic Controller," Section VI, "Operation," contained guidance on setting limitations. The procedure did not contain guidance, referenced in the Bailey vendor manual, that the product of rate and reset settings should be $\leq$ 1.0. Without this guidance, licensee personnel could improperly tune the system. As a result, inappropriate changes were made to the RCIC flow controller in January 2006. The failure to include the vendor manual precaution associated with rate and reset settings resulted in the inappropriate tuning of the flow controller in January 2006 from a required over damped to an over reactive response setting for flow changes.

The licensee's investigation of the RCIC tuning process determined that in 1998 changes were made to ICI-C-E51-3. The reference to the Bailey vendor manual was removed, and pre-startup settings and procedural references to base flow response curves were deleted. This allowed individual instrumentation and control technicians to improperly adjust RCIC flow controller settings. The inspectors determined the settings for the November 28, 2007, event were established during January 2006.

Based on the observations, the licensee entered this issue into their CAP as CR 07-31441 and pursued establishing the correct settings for RCIC operability.

Analysis: The inspectors determined that the failure to provide adequate tuning procedures affecting the RCIC system was a performance deficiency warranting a significance determination review. The inspectors concluded that the finding was greater than minor in accordance with Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," dated September 20, 2007. The finding was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, improper tuning of the RCIC controller impacted operability of the RCIC pump.

The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Attachment 4, "Initial Screening and Characterization of Findings." In accordance with Table 3b, "SDP Phase 1 Screening Worksheet for Initiating Events, Mitigating Systems, and Barriers Cornerstones," the finding affected the safety of an operating reactor; specifically, the Mitigating Systems cornerstone. In accordance with Table 4a, "Characterization Worksheet for IE, MS, and BI Cornerstones," the finding represented a loss of system safety function. Therefore, the inspectors contacted a Region III SRA who used IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," to perform an SDP Phase 2 review of the finding.

The analyst determined that the Phase 2 pre-solved table/spreadsheet for Perry did not adequately address recovery credit for the RCIC pump. The SRA used the Perry Risk-Informed Inspection Notebook, Revision 2.1a to perform the Phase 2 analysis. The following assumptions were made:

A. The identified performance deficiency occurred beginning in January 2006. The maximum exposure time used in the SDP is limited to one year. Therefore the SRA assumed an exposure time of one year.

B. An operator could recover the RCIC system in time to mitigate the assumed initiating event. Therefore operator recovery credit of "1" was assumed.

Using the counting rule worksheet, the result from this Phase 2 estimation indicated that the finding was of very low safety significance (Green). The dominant sequence involved a LOOP initiating event, failure of onsite emergency alternating current (AC) power, and failure to recover offsite power. The ΔCDF was 2E-7. In order to further refine the result, the analyst performed a Phase 3 analysis in accordance with IMC 0609, Appendix A, and the Risk Assessment Standardization Project (RASP) Handbook.

The analyst performed the Phase 3 analysis using the Standardized Plant Analysis Risk (SPAR) Model for Perry, Revision 3.31, dated June 2006, to simulate the failed RCIC pump. The analyst assumed an exposure time of one year, RCIC failure to start, and a non-recovery probability. To determine an appropriate non-recovery probability, the analyst conducted a human reliability analysis using the SPAR-H method. The analyst assumed that emergency response personnel would be under high stress during the diagnosis and recovery. The complexity of the diagnosis of the RCIC pump failure to start was considered to be obvious. All other performance shaping factors were considered to be nominal. The analyst calculated a non-recovery probability of 4E-3 using SPAR-H and used a more conservative value of 1E-2 in the SPAR Model calculation.

The SPAR baseline core damage frequency (CDF_{BASE}) was 3.523E-6/yr. The evaluation case was run assuming the RCIC pump failure to start and non-recovery probability from the failure to start of 1.0E-2. The evaluation case resulted in a conditional core damage frequency ($CCDF_{SPAR}$) of 3.560E-6/yr. The ΔCDF was calculated to be 3.693E-8/yr, or about 3.7E-8, since an exposure of one year was assumed. The dominant core damage sequence involved a LOOP/station blackout, failure of HPCS, failure of RCIC, and failure to recover AC power within 30 minutes. Therefore the total SDP result was 3.7E-8, representing a Green finding. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution per IMC 0305 P.2(b) because the licensee failed to institutionalize OpE through changes to procedures regarding flow controller settings.

Enforcement: Criterion V, "Procedures," of 10 CFR Part 50, Appendix B, required in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to this requirement, on January 21, 2006, licensee personnel performed activities to tune the RCIC flow controller in accordance with procedures that were not appropriate to the circumstances in that they resulted in an inoperable RCIC system. The tuning procedure was changed in 1998 removing pertinent guidance that would have precluded the improper tuning. This issue is related to URIs 05000440/2007005-11 and 05000440/2007010-02 and both are closed. However, because of the very low safety significance of the issue and because the issue has been entered into the licensee's CAP (CR 07-31441), the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2008002-04)

b.2 Failure to Adequately Control Testing of the RCIC System

Introduction: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," when the licensee test program failed to ensure RCIC system operability between January 2006 and November 2007. The inspectors noted on two occasions during the November 28, 2007, loss of feedwater event that RCIC failed to inject into the RPV and thus failed to perform its safety function. The licensee's test program was required to ensure safety systems perform satisfactorily when required.

Description: On November 28, 2007, the reactor scrammed due to a loss of feedwater. When RPV level reached Level 2, RCIC received an initiation signal, RCIC started and commenced injecting water into the RVP. Within seconds, RCIC tripped due to a low suction pressure. RCIC was initially lined up to the suppression pool before the November 28, 2007, event. Licensee personnel realigned suction to the condensate storage tank and finally placed the RCIC flow controller in manual control to allow use of the RCIC system to maintain reactor water level.

Section 5.4.6 of the Updated Final Safety Analysis Report, Revision 12, stated the design basis of RCIC. RCIC was considered safety-related and was designed to assure that sufficient reactor water inventory was maintained in the reactor vessel to permit adequate core cooling to take place. The RCIC system was designed to respond in the event of plant shutdown with loss of normal feed by operating until the reactor is depressurized and shutdown cooling can be initiated. The RCIC system was to meet these requirements with suction lined up to the condensate storage tank or the suppression pool.

Investigation into the cause of the RCIC trip determined that the flow controller was improperly tuned during January 2006. The licensee's test program and surveillance procedures did not identify this deficiency. As noted in the licensee's investigation, the surveillance procedure for operability, SVI-E51-T2001, "RCIC Pump and Valve Operability Test," Revision 27, did not identify any issues related to the flow controller or system operability. This test would likely not trigger responses in system pressure or flow of the magnitude noted on November 28, 2007. The surveillance procedure did provide bounding analysis values for other test parameters based on previous successful RPV injections as basis for operability.

Licensee CR 08-34777 noted that GE/Hitachi Report 0000-0079-1103, Revision 1, delineates that RCIC operability of the flow control loop can be assured by maintaining RCIC flow controller tuning settings demonstrated during successful RPV injection. As part of their corrective actions, the licensee implemented administrative controls for the RCIC flow controller settings.

Enforcement: Criterion XI, "Test Control," of 10 CFR Part 50, Appendix B, required the implementation of a test program that assures that written test procedures incorporate requirements and acceptance limits contained in design documents and that the system will perform satisfactorily when required. Contrary to this requirement, the licensee's test program did not assure that the RCIC system would perform as designed as evidenced by its failure on November 28, 2007. The cause of the failure was determined to be an improperly tuned flow controller which occurred during January 2006 and test procedures failed to detect this deficiency. This issue is related to URI 05000440/2007005-10 which is now closed. The issue has been entered into the licensee's CAP (CR 07-31441) and the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2008002-05)

b.3 Failure to Implement Appropriate Procedures for RCIC Instrument Line Testing

Introduction: The inspectors identified a finding of very low safety significance for a violation of 10 CFR Part 50, Appendix B, Criterion V. The inspectors determined that the licensee failed to implement appropriate procedures during testing of the RCIC system for identification and quantification of system gas accumulation.

Description: On February 14, 2008, the inspectors observed the licensee's performance of WO 200299311, "RCIC Instrument Line Venting and Transmitter Venting." The WO was performed to provide data on possible gas intrusion into the RCIC system and to verify operability of the RCIC system following the RCIC failure on November 28, 2007.

The licensee was performing the test on a weekly basis and considered RCIC inoperable during the test because the testing affected instrument lines associated with RCIC system functions. The test procedure involved the connection of a clear hose from an instrument vent line to a bucket with water in it. The procedure then had personnel open the instrument line valve to open a path from the instrument to the bucket. The inspectors observed the test and noted air traveling through the hose to the bucket. The inspectors' initial assessment was that the air bubbles could be from the instrument line. The technicians attributed the air to be from the hose and did not document the air bubbles as required by the WO. The inspectors further noted that the procedure did not provide a method to distinguish between air in the hose and any gas that may have been coming from the instrument line, such as instructions to ensure the hose was full of water prior to the test. As such, the procedure did not provide an adequate method to characterize or quantify any gas that may have been observed.

The inspectors determined that the procedure would most likely have been adequate for a fill and vent operation, but noted that the purpose of the testing was to detect and quantify gas accumulation. Because the purpose of the WO was to detect and quantify gas accumulation in the RCIC system and the WO did not provide an adequate method to accomplish this, the inspectors determined that the weekly performance of the procedure unnecessarily affected RCIC system capability to respond to an event.

As part of their corrective actions, the licensee revised the test procedure. The revision included instructions to fill the hose with water prior to opening the instrument line vent valves so personnel could detect gas from the instrument line and isolate the source of gas to the instrument line.

Analysis: The inspectors determined that the licensee's failure to specify adequate testing protocol and measurement methods was a performance deficiency warranting a significance determination. The inspectors determined the issue was more than minor because it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the RCIC system was unnecessarily made inoperable to perform this evolution. The primary cause of this finding was related to the cross-cutting area of Human Performance per IMC 0305 H.2(c), because the licensee failed to provide complete and accurate procedures related to nuclear safety.

The inspectors performed a significance determination of this issue using IMC 0609, "Significance Determination Process," dated January 10, 2008, and IMC 0609.04, "Initial Screening and Characterization of Findings," dated January 10, 2008. The finding was of very low safety significance because, the time RCIC was inoperable was less than TS allowed inoperability time.

Enforcement: Criterion V, "Procedures," of 10 CFR Part 50, Appendix B, required in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to these requirements, on February 14, 2008, the licensee failed to use procedures that were appropriate to the circumstances for quantifying the amount of gas accumulation in the RCIC system. Because of the very low safety significance and because the issued has been entered into the licensee's CAP (CR 08-37980), the issue is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000440/2008002-06).

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors selected emergency preparedness exercises that the licensee had scheduled as providing input to the Drill/Exercise PI. The inspection activities included, but were not limited to, the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critiques. Observations were compared with the licensee's observations and CAP entries. The inspectors verified that there were no discrepancies between observed performance and PI reported statistics.

The inspectors selected the following emergency preparedness activity for review for a total of one sample:

- March 25, 2008, emergency plan drill to evaluate the drill conduct and the adequacy of the licensee's critique of performance to identify weaknesses and deficiencies.

b. Findings

No findings of significance were identified.

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed licensee event reports (LERs), licensee data reported to the NRC, plant logs, and NRC inspection reports to verify the following PIs for the time periods indicated:

- Unplanned Scrams for the period from the First Quarter of 2007 through the Fourth Quarter 2007 for a total of four quarters; and
- Unplanned Scrams with Complications for the period from the First Quarter of 2007 though the Fourth Quarter of 2007 for a total of four quarters

The inspectors determined whether the licensee accurately reported performance as defined by the applicable revision of Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline."

The NRC's Frequently-Asked-Questions (FAQs) process reviewed two June 2007 scram events and the NRC considered these Unplanned Scrams per NEI 99-02. During the inspection period, the licensee updated the Unplanned Scram PI and the PI met the White Threshold for the second quarter of 2007. An IP 95001 Supplemental Inspection is planned to address the White Unplanned Scram PI.

These reviews represent two inspection samples.

b. Findings

b.1. Failure To Report A Manual Scram

On June 21, 2007, the licensee commenced a planned shutdown to effect repairs to the 'A' reactor recirculation system flow control valve. The operators were following the preplanned sequence of shutdown in accordance with the documented reactivity plan. On June 22, 2007, the operators attempted to shift reactor recirculation pumps from fast to slow speed. The 'B' reactor recirculation pump failed to shift to slow speed, stopped, and began to unexpectedly start in fast speed several times due to the failure of an improperly installed breaker relay. Operators were able to secure the 'B' pump.

The documented preplanned shutdown sequence called for operators to reduce power to 20 percent and remove the main generator from the grid, while the plant was in dual recirculation loop mode of operation, prior to inserting a manual scram. Operators determined that it would not be prudent to continue following the documented sequence because the plant had lost flow in one of the recirculation loops. The operators inserted the full manual scram within three hours of the 'B' reactor recirculation pump failure. The plant was scrammed from about 23 percent power while the main generator was on the grid.

NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Revision 2, stated, "Preplanned actuations are those which are expected to actually occur. Such actuations are those for which a procedural step or other appropriate documentation indicates the specific actuation is actually expected to occur. Control room personnel are aware of the specific signal generation before its occurrence. However, if during the evolution, the system actuates in a way that is not part of the preplanned evolution, that actuation should be reported." Additional NUREG-1022 guidance stated, "The staff also considers intentional manual actions, in which one or more system components are actuated in response to actual plant conditions resulting from equipment failure or human error, to be reportable because such actions would usually mitigate the consequences of a significant event."

The inspectors determined that the June 22, 2007, manual scram was not in accordance with the licensee's documented preplanned sequence.

Enforcement: As stated in 10 CFR 50.72(b)(2)(iv)(B), "Four Hour Reports," any event or condition that results in actuation of the RPS when the reactor is critical except when the actuation results from and is part of a preplanned sequence during reactor operation, requires a four-hour report. Contrary to these requirements, on June 22, 2007, the licensee failed to make the appropriate four-hour and LER reports when a manual scram was inserted not in accordance with the preplanned sequence. Because the licensee has entered this issue into their CAP (CR 08-35944), the issue is being treated as a Severity Level IV NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000440/2008002-07).

b.2. Untimely Resolution of PI Data Submitted For Second Quarter 2007

Introduction: The inspectors identified a finding associated with the licensee's reporting of Unplanned Scram Performance Indicator (PI) data for the second quarter 2007. The NRC disagreed with the licensee's characterization of two scrams that occurred in June 2007. The inspectors determined that the licensee failed to pursue resolution of the reporting discrepancy in a timely manner in accordance with established industry standards.

Description: On July 23, 2007, the licensee submitted Unplanned Scram PI data that included a total of one Unplanned Scram for the second quarter of 2007. The inspectors reviewed the PI data and questioned the licensee on whether scrams that occurred on June 22, 2007, and June 29, 2007, should have been included in the Unplanned Scram PI submittal.

Following further review of the licensee's basis for not reporting the June 2007 scrams, the inspectors informed licensee management that the second quarter 2007 PI report of one Unplanned Scram did not appear to be accurate. Specifically, the inspectors considered that the June 22 and 29 scrams were Unplanned Scrams and that the correct total of Unplanned Scrams for the second quarter 2007 appeared to be three. On August 23, 2007, the licensee entered the issue into their CAP as CR 07-25590.

Revision 5 of NEI 99-02, “Regulatory Assessment Indicator Guideline,” addressed licensee treatment of NRC PI reporting. Appendix E of NEI 99-02, stated that FAQs should be submitted as soon as possible once the licensee and resident inspector or region have identified an issue on which there is not agreement.

On October 22, 2007, the licensee submitted PI data for the third quarter 2007 and included a comment that referenced a licensee review of two events that occurred in June; however, the June Unplanned Scrams remained unreported and without FAQs.

On or about November 26, 2007, the licensee submitted FAQs to the NRC to address the June scrams.

On February 20, 2008, after review through the NRC’s FAQ resolution process, the NRC denied the licensee’s positions in FAQs and responded that the June scrams should count as Unplanned Scrams. On March 7, 2008, the licensee submitted a revision to the NRC for the second quarter 2007 Unplanned Scram PI data to include the June scrams and reflect a total of three Unplanned Scrams.

Analysis: The inspectors determined that the licensee’s failure to initiate an FAQ in accordance with NEI 99-02 as soon as possible once a disagreement was identified was a performance deficiency warranting significance determination. The finding was considered more than minor because it was related to a PI and caused the PI to exceed a threshold. Had all three unplanned scrams been reported in July 2007, the Unplanned Scram PI would have crossed the Green to White threshold. The finding was determined to be of very low safety significance after management review because the NRC was aware of the scram circumstances during the time they were not reflected in the PI data.

Enforcement: No violations of regulatory requirements occurred.
(FIN 05000440/2008002-08)

4OA2 Identification and Resolution of Problems (71152)

Cornerstone: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to determine whether they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed.

This is not an inspection sample

b. Findings

No findings of significance were identified.

.2 In-Depth Review - Operator Workarounds

a. Inspection Scope

During the week of March 3, 2008, the inspectors performed a review of the cumulative effects of operator workarounds, operator burdens, and control room deficiencies. The list of open operator workarounds, operator burdens, and control room deficiencies was reviewed to identify any potential effect on the functionality of mitigating systems. Inspection activities included, but were not limited to, a review of the cumulative effects on the availability and the potential for improper operation of systems, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents. The inspectors conducted a review of CRs to ensure that workaround-related issues were entered into the CAP when required.

b. Findings and Observations

The inspectors identified that the licensee completed a review of its workaround process which identified a programmatic weakness. The licensee identified that workarounds were not always identified at the appropriate threshold and listed in the appropriate database. The databases included operator workarounds, operator burdens, and control room deficiencies. The licensee reviewed issues related to workarounds after redefining the thresholds and increased the number of items in the operator burdens and control room deficiency databases. The inspectors noted that several of these items were not classified as directed by the licensee procedure, "Work Around Process," Revision 1, which required the items to be classified a Priority 4 to ensure timely correction. From a total of 58 operator burdens and control room deficiencies, 26 items were classified as a lower priority than required. The licensee reclassified the items to a Priority 4.

The licensee identified a negative cumulative effect of deficient recorders in the control room operators' area after the threshold change. The operators noted these recorders affected tracking, trending, and independent verification of plant parameters. The licensee placed these issues into their control room deficiency system and was in the process of replacing the recorders. The inspectors noted that a number of the deficient recorders have existed since 2006 and some since 2001. This indicated that the licensee threshold for identification of burdens was not appropriate prior to 2007. This represented a lack of timeliness in resolving known issues because the licensee's goal was to resolve issues within one operating cycle.

During CR review, the inspectors identified two additional issues that were not included in the operator burden list. The first issue was documented CR 07-20443, which described an operator burden associated with main turbine temperature control during synchronization which required manual action to ensure proper main turbine operation. The second issue was documented in CR 07-20487 and described a condition where both RWCU and plant air systems tripped when an auto transfer of bus loads occurs during a plant scram, which requires operator action to restore these systems when responding to a plant trip. The licensee entered both items into the operator burden list.

This review represents the first of three in-depth inspection samples.

.3 In-Depth Review of Licensee Root Cause Evaluations

a. Inspection Scope

The inspectors selected recent root cause evaluations for a more in-depth review. These issues included a dropped fuel channel, RCIC system trip following plant scram, DFWCS scram, AEGTS failure, and operator training programs placed on probation. The inspectors considered the nature and significance of the issues with respect to safety, risk, and licensee corrective action procedural requirements. Attributes reviewed included: complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. The inspectors implemented a collective assessment of these root causes for possible organizational issues.

b. Findings and Observations

The inspectors questioned whether the licensee fully considered the underlying causes that appeared common to the reviewed root cause evaluations. The inspectors noted a common theme of inadequate change management processes. In reviewing the root causes, the inspectors determined that the licensee had taken appropriate actions to address the root and contributing causes identified by the root cause evaluation teams. However, there were no indications that the licensee had looked at the results collectively from the root cause analyses as an indicator that the site's change management process was not being used effectively to prevent the types of issues being reviewed.

Change management processes are designed to ensure effective organizational performance is sustained. Change management as described by IMC 0305, "Operating Reactor Assessment Program," dated November 27, 2007, is a systematic process for planning, coordinating, and evaluating the impacts of decisions related to major changes in organizational structures and functions, leadership, policies, programs, procedures, and resources. In addition, a February 27, 2008, operating experience notice from NRR offered additional amplification on change management. The operating experience included a discussion, among other items, of changes in equipment and changes in roles and responsibilities being issues which are part or change management.

During the review of the events listed in the Scope section, the inspectors identified that each root cause identified issues that can be traced back to the basic concepts of a change management process. For example, the licensee's evaluation for the dropped fuel channel (CR 07-28851), dated November 12, 2007, identified that the method used for verifying the new grapple's engagement to the fuel channel tabs was less than adequate. Change management would indicate that the use of new tools can introduce new hazards and that their use needs to be fully understood.

In another example, the analyses completed for the May and November 2007 reactor scrams identified that the newly installed DWFCS in one case did not match actual plant conditions and if licensee personnel were more familiar with the system, they may have responded to earlier indications of system degradation, and in the other that the depth of site resources, expertise, and knowledge of digital systems needed improvement. Again, both issues are captured within the context of a change management process; a new “tool,” i.e. the DFCWS, and changes in roles and responsibilities, that includes knowledge and expertise, of site engineering and operations have changed.

The conclusion of the root cause evaluation for RCIC system trip following plant scram (CR 07-30660/CR 07-31441), dated January 28, 2008, identified several issues which fall into the change management realm. Specifically the evaluation identified problems in configuration control, procedure changes, loss of knowledge and experience, and lack of training for new individuals. These issues map well to change management areas of process changes, procedures, and roles and responsibilities.

The root cause evaluation for the AEGTS failure (CR 07-31871), dated February 18, 2008, listed as a contributing cause of less than adequate procedures and training for proper shaft alignment. The roles and responsibilities area of change management encompasses appropriate training.

The inspectors engaged licensee personnel on the use of the change management process on March 27, 2008, in order to improve performance.

This review represents the second of three in-depth inspection samples

.4 In-Depth Review of RCIC Controller Issues

a. Inspection Scope

The RCIC system was declared inoperable on January 14, 2007, due to flow controller voltage drifts. The licensee performed repairs to the flow control system and the system failed post-maintenance testing on January 19, 2007. The inspectors reviewed the licensee’s evaluation of the events to determine whether the licensee appropriately identified and resolved the issues associated with the flow controller failures

b. Findings and Observations

Unresolved Item (URI) 05000440/2008002-12: Reactor Core Isolation Cooling System Flow Controller Reliability.

At the end of the inspection period, the inspectors continued to evaluate the reliability of the RCIC flow controller. The inspectors noted that the licensee had identified that the controller was in need of replacement due to reliability concerns and intended to replace the controller with a new design when parts became available. However, the inspectors questioned whether appropriate measures were in place to compensate for the currently in-service flow controller in light of recent controller failures.

This review represents the third of three in-depth inspection samples.

.5 Human Performance (Semi-Annual Trend Review)

a. Inspection Scope

The inspectors reviewed monthly performance reports, self-assessments, quality assurance assessment reports, performance improvement initiatives and CRs to identify any trends that had not been adequately evaluated or addressed by proposed corrective actions.

b. Observations

The inspectors focused this sample on Perry's declining performance in the area of human performance. The NRC's end-of-cycle assessment determined that a substantive cross-cutting issue existed in human performance, work control. In response to the assessment, the licensee assembled a root cause evaluation team to investigate the cause for this negative trend.

During the course of the first quarter 2008 the negative trend continued for the licensee. The following lists human performance issues identified by the CAP:

Seven Events in the Operations Department

- A RWCU system work control procedure required an isolation valve to be tagged shut to provide equipment/human protection during the replacement of the RWCU 'A' pump. The isolation valve was tagged open during the work window. Protection was provided by an administratively closed valve. During system restoration, the administratively closed valve was opened and reactor coolant water was released into the room via a vent valve. The clearance was subject to an independent verification process. This resulted in a site clock reset in human performance.

- Lake water drained into the Radwaste building due to an improper Fire Protection lineup.

- A non-licensed operator entered the Radiological Controlled Area with his electronic dosimeter on pause; therefore, the operator entered in violation of radiation worker permit requirements.

- A licensed operator filled an Emergency Response Organization position with an expired qualification.

- Emergency closed cooling 'B' was inoperable for 60 hours due to less than minimum flow requirements unbeknownst to the shift manager and unit supervisor. Licensee personnel failed to perform an adequate impact review prior to issuing a clearance and the shift manager discovered this issue nearly 3 days later. Technical Specification actions for this prolonged condition require plant shutdown.

- A one-hour TS Action Statement was missed for the Division 1 EDG when work was performed on ESW 'A' loop. The shift manager and unit supervisor expected technicians to restore ESW 'A' within the one-hour, but the technician performing the task was 'pulled away' to perform another task. Another technician stepped in to perform the task, but the equipment was not properly calibrated.

- Operators removed an incorrect fuse while performing a clearance in the Radwaste building. Initially, both workers believed that an incorrect fuse type was previously installed.

Two Events in the Instrument and Calibration Department

- While performing average power range monitor channel calibration the first performer mistakenly marked steps 95 – 107 as "not applicable", when the procedure only required steps 95 – 106 to be marked "not applicable." This action was peer checked by the second performer who did not notice the mistake. Step 107, which was not performed, called for the installation of a jumper, and required an independent verification of the jumper installation. The licensee normally required the independent verification to be a separate procedure step, but in this instance the independent verification requirement was embedded in step 107. Without the jumper an actual half-scram signal was received by RPS.

- During radiation monitor maintenance in the control room, a qualified supervisor walked away from an under-instruction technician. The under-instruction technician believed he understood the correct performance of the instruction, and without required supervision, performed work on an active radiation monitor. This resulted in an unexpected radiation monitor alarm in the Control Room.

Eight Events During RCIC Outage for Maintenance Department

- While correcting 'Alert' and 'Expedite' vibration readings on the RCIC room air-handling unit, a tensioning bolt broke. The technicians attributed this to a modification made earlier to the bolt-motor bracket assembly.

- Electricians working on RCIC: 1) failed to ensure that quality assurance material used in the field had the required identification labeling for verification; 2) did not have the WO or in-field reference guides at the job site; and 3) did not accurately document parts used.

- Maintenance engineering technicians working on the RCIC air-handling unit: 1) failed to meet work-in-progress log expectations in that as-found conditions and vibration & flow test results were not documented; 2) performed inadequate place-keeping; and 3) did not enter a condition of a low flow discovered during testing into the corrective action program.

- Licensee personnel improperly impaired a RCIC room water-tight door and left a Coppus blower hose through the doorway.

- A procedure associated with the RCIC leak detection system was found to have inappropriate instructions for restoration of a system transmitter that could have led to inadequate response of the system.

- Despite verifying prior to the outage that all scaffold/ladders were acceptable, it was determined that the scaffold/ladders for two WOs were inadequate. This required rebuild and modification. This issue introduced delay in the outage and increased dose by 83 mrem.

- The rework of the scaffold/ladders and the associated dose estimates were not properly communicated to radiation protection and this impacted radiation protection personnel's ability to assure implementation of as-low-as-reasonably-achievable practices.

- A mechanical technician left his electronic dosimeter in his protective clothing when undressing and lost control of the electronic dosimeter. The dosimeter was later retrieved and radiation protection technicians performed actions to recover from this incident.

Four Human Performance Events by Operations During the RCIC Outage

- Operations personnel failed to appropriately control a limiting condition for operation (LCO) when both channels of the RCIC leak detection system were inoperable. One channel was inoperable by an earlier clearance, and the second was inoperable by maintenance. The shift manager and unit supervisor expected notification when work commenced, even though they already gave permission to maintenance personnel to start the work. This communication did not occur, and TS requirements (shutting of the isolation valve) were met by other administrative processes.

- Operators were directed to fill and vent RCIC for restoration. Later, when other operators were removing a RCIC clearance they noticed a RCIC valve was still red-tagged shut. This signified that the fill and vent could not have been completed since the downstream piping was not accessible. The RCIC fill and vent was re-performed.

- An operator was performing timed valve stroke surveillance testing and did not understand adequately the appropriate way to receive time data. Instead of using a stop watch, the operator decided to use a computer point, which was allowed by the procedure. The operator did not understand that the computer point changes state after 25 percent stroke travel, and thus could lead to inaccurate timing information. The surveillance had to be repeated twice to confirm operability requirements.

- Another surveillance associated with a leak detection system transmitter provided the incorrect restoration sequencing.

The inspectors noted that licensee management took the following corrective actions:

- The site conducted a human performance stand-down February 18 through 19, 2008.

- The licensee commenced a root cause evaluation after the resident inspectors informed Perry senior management of a potential cross-cutting issue in Human Performance. Initial investigations from the root cause determined that during the last two quarters of 2007, the licensee failed to properly classify several CAP issues as human performance errors. This may indicate an inadequacy with Perry's self-assessment process.

- Perry supervision and management have recently performed more observations in the field with a renewed focus on human performance.

- The licensee assigned and began training for a new full-time site human performance advocate. The position was not staffed from May 2007 until January 2008.

The inspectors were concerned that human performance had declined at Perry and noted that the licensee commenced initiatives to address this issue.
This review represents one semi-annual trend review inspection sample.

4OA3 Follow-up of Events and Notices of Enforcement Discretion (71153)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

.1 (Closed) LER 05000440/2007-004-00 and 05000440/2007-004-01: "Automatic Reactor Protection System Actuation Due to Feedwater Control Power Supply Failure."

On November 28, 2007, the reactor scrammed from 100 percent power due to an automatic actuation signal from the RPS upon receipt of a turbine control valve fast closure signal. The automatic scram signal was the failure of two power supplies in the DFWCS. These failures caused an errant RPV high water level signal to be sent to the main turbine generator control system, resulting in the turbine control valves closing. The valve closures produced an RPS automatic scram signal, and all control rods fully inserted into the reactor core.

The DFWCS high RPV water level signal caused both turbine driven reactor feedwater pumps to trip and prevented the motor feedwater pump from starting. The total loss of feedwater caused RPV water level to lower to the point of actuating RCIC and HPCS systems to provide water inventory to the RPV. Both RCIC and HPCS initiated and commenced injecting into the RPV; however RCIC tripped on low suction pressure about 13 seconds after initiation. The HPCS continued to operate and restored RPV water level. The lowest RPV water level was approximately 109 inches above the top of active fuel. Due to the HPCS actuation signal, the Division 3 EDG started normally, but was not required to provide electrical power. Both reactor recirculation pumps tripped off as designed and all required containment isolation valves closed.

The NRC determined that the cause of the RCIC failure was an improperly tuned flow controller which was confirmed by the licensee. The flow controller was replaced on January 20, 2006, and the settings made the controller over reactive when injecting into RPV and this caused suction flow oscillations. The NRC Region III staff conducted a Special Inspection for this event and the issue of RCIC operability was documented as URI 05000440/2007010-02.

The improper tuning was caused by exempting the controller settings from configuration control procedures. In 1987 the flow controller setpoints were removed from the master setpoint list without establishing adjustment limits in the tuning procedure. In 1999, start-up performance data was removed from the tuning procedure to allow for a 'stream-lined' process. Licensee staff turnover resulted in a knowledge deficiency in proper controller tuning. The 'streamlined' procedures coupled with a lack of experience and training of the instrumentation and control technicians on the Bailey controller resulted in unrecognized controller adjustments outside of allowable values. With an improperly tuned flow controller, RCIC was inoperable from January 21, 2006 to November 28, 2007. This issue (URI 05000440/2007010-02, "RCIC Operability Between January 2006 and December 2007") is being resolved by NCV 05000440/2008002-04.

With RCIC inoperable from January 21, 2006 to November 28, 2007, the plant was in violation of TS 3.5.3, "RCIC System," Condition A. Since January 21, 2006, the plant was in violation of TS 3.5.1, "ECCS-Operating," Condition B each time HPCS was declared Inoperable for greater then one hour. TS required verifying RCIC operable within one hour of placing HPCS inoperable.

The licensee's corrective actions included replacing the DFWCS power supplies with newer style Foxsboro power supplies and the addition of a third Lambda power supply for diversity. The RCIC flow controller was retuned to the original pre-operational settings established when RCIC successfully injected into the RPV from the suppression pool. A two-second time delay on the RCIC low suction pressure trip signal was installed. The licensee flow control tuning procedures were revised to incorporate appropriate acceptance criteria. The licensee planned to revise the training regimen for both engineering and technicians. A licensee-identified violation associated with this issue is documented in Section 4OA7 of this report. No additional findings were identified in the inspectors' review. This LER is closed.

This review represents the first of six samples for this inspection procedure.

.2 Steam Leak from Reactor Water Cleanup System

a. Inspection Scope

On January 4, 2008, licensee personnel were opening a valve associated with the RWCU 'A' pump. Reactor steam unexpectedly issued from the system in the pump room. Operators responded to the event and isolated the leak. A danger-tagged valve was found out of position. The licensee determined that no equipment damage or personnel injury occurred due to the event. The inspectors reviewed the licensee's response to the event. The inspectors reviewed the licensee's actions to determine whether procedural and TS requirements were met.

b. Findings

A finding and an NCV associated with this event is documented in Section 1R13 of this report.

This review represents the second of six samples for this inspection procedure.

.3 (Closed) LER 0500440/2007-005-00: "Plant Startup With Inoperable Reactor Core Isolation Cooling System."

Findings

Introduction: The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, when the licensee declared RCIC inoperable on December 12, 2007. The cause was attributed to improper flow controller tuning parameters, and this issue was previously evaluated as the cause of RCIC inoperability from January 21, 2006 to November 28, 2007.

Description: On December 12, 2007, the RCIC system was declared inoperable when it was determined that the RCIC flow controller settings would not assure operability. Specifically, the licensee determined that the controller settings would challenge stable flow control during system operation. The licensee was recovering from the November 28, 2007, scram where RCIC did not perform its safety function due to improper flow controller settings. The licensee had adjusted flow controller settings prior to startup on December 6, 2007.

Investigations after the November 28, 2007, RCIC failure determined that the appropriate acceptance criteria for flow controller tuning were removed through licensee processes dating to 1987. During the evaluation period prior to December 6, 2007, the licensee adjusted controller acceptance criteria in accordance with the Bailey vendor controller manual. These criteria were associated with the Rate and Reset values of the controller. The staff was to include review of applicable OpE for proper RCIC flow controller settings.

The licensee set up a RCIC test regimen to verify system operability and commenced plant startup on December 6, 2007. Based on initial test results, the licensee declared RCIC operable on December 7, 2007, prior to the plant exceeding 150 psig reactor operating pressure when RCIC was required to meet TS requirements. Testing continued through December 10, 2007, and the licensee preliminarily accepted all test results. On December 12, 2007, after additional engineering review of the results, the licensee determined that RCIC may not be able to perform its safety function. Specifically, the licensee determined through OpE review from Limerick, that high controller gain values would cause flow oscillations. Additional OpE existed from Brunswick and Peach Bottom that attributed inadequate RCIC performance to improper controller gain settings.

With technical assistance from General Electric engineers, the licensee restored RCIC flow controller settings to startup test values that were established when RCIC was successful at injecting into the RPV at operating pressures. The licensee placed this issue into their CAP as CR 07-31441. The licensee has revised the RCIC tuning procedures with the appropriate acceptance criteria.

Analysis: The inspectors determined that the inoperability of RCIC due to inadequate flow controller settings was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," dated September 20, 2007. The finding was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events in order to prevent undesirable consequences. Specifically, the finding resulted in the RCIC system not meeting all design requirements.

The inspectors performed a significance determination of this issue using IMC 0609, "Significance Determination Process," dated January 10, 2008, and IMC 0609.04, "Initial Screening and Characterization of Findings," dated January 10, 2008. The issue screened as a loss of system safety function and required analysis using IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," dated January 10, 2008. During Phase 2 analyses, the inspectors used an interval of inoperability between 3 and 30 days with operator recovery and determined that the finding was of very low safety significance. The primary cause of this finding was related to the cross-cutting area of Problem and Identification and Resolution per IMC 0305 P.2(a), because the licensee failed to communicate relevant external OpE in a timely manner.

Enforcement: Criterion XVI, "Corrective Action," of 10 CFR Part 50, Appendix B, stated that measures shall be established to assure that for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to this requirement, the licensee failed to prevent recurrence of the RCIC system's inoperability due to inappropriate flow controller settings on December 6, 2007. Specifically, the RCIC flow controller settings used between December 6 and 12, 2007, could not be shown to assure stable RCIC flow conditions. Because this violation was of very low safety significance and was entered into the licensee's CAP as CR 07-30930, this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000440/2008002-09). This LER is closed.

This review represents the third of six samples for this inspection procedure.

.4 Local Flooding and Road Closures on February 6, 2008

a. Inspection Scope

On February 6, 2008, Lake County Ohio experienced flooding due to heavy snow and rains. The flooding affected roads that were designated as emergency response routes for the licensee. The inspectors observed the licensee's response to the event to determine whether appropriate procedures were followed, compensatory measures were established, and communications with local emergency responders were made.

b. Findings

No findings of significance were identified.

This review represented the fourth of six samples for this inspection procedure.

.5 (Closed) URI 0500440/2007010-01: Inadequate Classification of Condition Report for RCIC

Findings

Introduction: The inspectors identified a finding of very low safety significance and an NCV of 10 CFR Part 50, Appendix B, Criterion V, during a review of the licensee's treatment of the safety-related RCIC system's failure to perform its safety function when called upon during an event (URI 05000440/2007010-01).

Description: On November 28, 2007, during the loss of feedwater event, the RCIC system failed to operate as designed. Operators realigned the RCIC system to take suction from the condensate storage tank and restarted the pump using automatic flow control. The pump again tripped shortly after it was started. The licensee was able to successfully use the RCIC system with the flow controller in manual.

Following the event, the inspectors were concerned with the adequacy of the licensee's evaluation effort to address the RCIC system failure. On November 29, 2007, the inspectors noted that the licensee had classified the issue as a condition adverse to quality within the CAP. The inspectors questioned the licensee on whether the classification was consistent with the safety significance of the issue and whether it was consistent with the licensee's CAP procedure Normal Operating Procedure (NOP)-LP-2001, "Corrective Action Program," Revision 17. Procedure NOP-LP-2001 addressed the identification and classification of conditions adverse to quality. Procedure NOP-LP-2001 provided examples of conditions that should be considered significant conditions adverse to quality. These examples included multiple failures in systems required to mitigate accidents.

The inspectors considered that the failures of the RCIC system to fulfill its safety role when called upon during a scram with complications were consistent with the guidance in the NOP-LP-2001 for classification as a significant condition adverse to quality. The licensee conducted a review of the appropriateness of the classification to address the inspectors' questions. Following this review, the licensee reclassified the condition as a significant condition adverse to quality and initiated a root cause investigation.

Analysis: The inspectors determined that the failure to identify the issue as a significant condition adverse to quality was a performance deficiency warranting significance determination. The inspectors concluded that the finding was greater than minor in accordance with Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," dated September 20, 2007. The inspectors determined that the failure to identify significant conditions adverse to quality would become a more significant safety concern if left uncorrected.

Although not suitable for SDP review, regional management determined that the finding was of very low safety significance because the finding did not result in an actual safety consequence. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution per IMC 0305 P.1(a) because the licensee failed to identify the issue completely, accurately, and in a timely manner commensurate with its safety significance.

Enforcement: Criterion V, "Procedures," of 10 CFR Part 50, Appendix B, required in part that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Contrary to the above, on November 29, 2007, the licensee failed to adhere to procedure NOP-LOP-2001. Specifically, licensee personnel classified the failures of the RCIC system on November 28, 2007, as a condition adverse to quality when procedure NOP-LOP-2001 described multiple failures of a system designed to mitigate accidents as a significant condition adverse to quality. However, because of the very low safety significance of the issue and because the issue has been entered into the licensee's CAP (CR 08-34762) the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2008002-10) This LER is closed.

This review represents the fifth of six samples for this inspection procedure.

.6 (Closed) LER 0500440/2007-006-00: "Loss of Safety Function and Condition Prohibited by TSs due to Annulus Exhaust Gas Treatment System Inoperability."

Findings

Introduction: A finding of very low safety significance and an NCV of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," was self-revealed when a loss of the AEGTS safety function occurred on December 21, 2007.

Description: On December 21, 2007, the AEGTS train 'A' was inoperable in order to obtain a charcoal sample at 8:25 a.m. When the 'A' charcoal plenum was opened operators received the 'B' train low flow alarm and containment annulus low differential pressure alarm. Annulus differential pressure decreased to 0.3 inches of vacuum gauge, which was below the TS limit of 0.66 inches of vacuum water gauge.

Licensee personnel closed the 'A' charcoal plenum and containment annulus differential pressure was returned to TS limits. Operators then restored AEGTS 'A' to operable status at 10:12 a.m. on December 21, 2007. Maintenance personnel commenced testing of the AEGTS 'B' damper system when AEGTS 'A' auto started on low flow conditions. The licensee determined the cause of the low flow condition was a failure of the 'B' train discharge damper.

Operators declared AEGTS 'B' inoperable on December 21, 2007 at 12:03 p.m. During post-maintenance testing the licensee discovered that the AEGTS 'B' recirculation damper failed and could not operate manually. Licensee personnel completed repairs and declared AEGTS 'B' operable at 2:19 a.m. on December 23, 2007.

The licensee entered these issues into their CAP as CR 07-31871. The licensee determined that the failure of the discharge damper was due to side-load wear induced binding of the shaft and shaft extension. The side-loading was likely caused by misalignment of the shaft and components during assembly. It was determined that maintenance instructions lacked adequate provisions to ensure proper alignment of hydramotor shafts and associated linear converters. The licensee conducted an extent-of-condition review and determined several procedures did not include adequate instructions.

The licensee determined that the 'B' recirculation damper failed to stroke five months after vendor rebuild and that a vendor issued part associated with the motor assembly failed.

The licensee commenced a failure analysis program for AEGTS failures to address recurrent problems. The licensee planned to update procedures to provide adequate guidance for damper assembly and acceptance criteria to ensure proper shaft alignment.

Both trains of AEGTS were inoperable on December 21, 2007, and the system could not perform its safety function.

Analysis: The inspectors determined that the loss of AEGTS safety function due to inadequate maintenance procedures was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with Appendix B, "Issue Screening," of IMC 0612, "Power Reactor Inspection Reports," dated September 20, 2007. It was greater than minor because it was associated with the Procedure Quality attribute related to maintenance of containment function of the Barrier Integrity Cornerstone and affected the cornerstone objective of reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, the finding was determined to have resulted in a degraded condition of secondary containment.

The inspectors performed a significance determination of this issue using IMC 0609, "Significance Determination Process," dated January 10, 2008, and IMC 0609.04, "Initial Screening and Characterization of Findings," dated January 10, 2008. The finding was of very low safety significance because the finding only represented a degradation of the radiological barrier function. The primary cause of this finding was related to the cross-cutting area of Human Performance per IMC 0305 H.2(c), because the licensee failed to provide complete and accurate procedures related to nuclear safety.

Enforcement: Criterion V, "Procedures," of 10 CFR Part 50, Appendix B, required procedures appropriate to the circumstances. Contrary to this requirement, licensee maintenance procedures were not appropriate to the circumstances in that they did not provide adequate instruction or acceptance criteria for hydramotor assembly and this resulted in a loss of safety function of AEGTS on December 21, 2007. However, because of the very low safety significance of the issue and because the issue has been entered into the licensee's CAP as CR 07-31871, the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2008002-11)

This review represented the sixth of six samples for this inspection procedure.

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted the following observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

- Multiple tours of operations within the Central and Secondary Security Alarm Stations;
- Tours of selected security towers/security officer response posts;
- Direct observation of personnel entry screening operations within the plant's Main Access Facility; and
- Security force shift turnover activities.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 IP 92702 Follow-up on Independent CAP Assessment.

a. Inspection Scope

The inspectors evaluated the licensee's independent assessment of its CAP and the response to it. The inspectors reviewed the report of the CAP's implementation, sampled a number of CRs reviewed by the independent team, and reviewed the CRs generated by the licensee to address the issued identified by the independent assessment. The specific report and CRs are listed in the attachment to this report.

b. Observations

The independent assessment looked into the following areas of CAP implementation:

- identification, classification, and categorization of conditions adverse to quality;
- evaluation and resolution of problems;
- corrective action implementation and effectiveness;
- trending program implementation and effectiveness;
- effect of program backlogs;
- effectiveness of internal assessment activities; and
- validation of continuing progress with the implementation of the CAP.

The independent assessment rated each of the above areas, with the exception of the last item, as effective. The last area was rated as "Continuing to Improve." In addition to the overall ratings, the independent assessment identified "areas for improvement" and "areas in need of attention." The more significant areas for improvement were: continued re-enforcement of the site expectation that CRs be written at a low threshold, improvement in Generic Implication Evaluations specifically regarding extent of condition and extent of cause.

In reviewing the licensee's response to the independent assessment the inspections noted the following conditions:

- CR 07-25260, "2007 IND. CAP Assessment: CR 06-8758 Evaluation Requires Clarification," regarding extent of cause and condition reviews. The CR identified a number of CRs in which the extent of cause/condition was deficient; however, no mention was made as to whether the deficiencies were corrected. Information provided by the licensee indicated that where possible the reviews were conducted.

- CR 07-25720, "2007 IND. CAP Assessment: Area in Need of Attention for Task #3," regarding electrical penetrations and operator burden/workarounds. The CR did not support its conclusions and while providing the definitions of burden and work-around the CR write-up misused the terms. Discussion with the operations manager identified that the CR's conclusions were appropriate however none of the points raised by the manager were included in the CR.

- CR 07-25452, "2007 IND. CAP Assessment: Improper Closure of Corrective Actions for CR 06-11339," regarding alternate closure approvals, i.e., vice president approval. This issue was also identified during the IP 95003 inspections in 2005 and 2006. Given the small number of individuals needing to understand the requirement for the vice president's signature to effectively implement this procedural requirement, the inspectors were concerned that the issue had not been corrected.

- CR 07-25258, "2007 IND. CAP Assessment: Improper Categorization of condition reports," regarding CR categorization. The "barrier analysis" used to assess this CR appeared not to have been conducted in accordance with the licensee's guidelines. Specifically, the analysis form had not been properly filled out.

- CR 07-19009, "Emergency AC NRC Performance Indicator is White," regarding CR categorization. This CR dealt with an EDG deficiency and had been categorized as a "condition adverse to quality." The independent assessment believed that a more appropriate categorization would have been a "significant condition adverse to quality." It was noted that the licensee's guidance would allow either categorization. While agreeing with the licensee's conclusion that because a root cause evaluation had been performed, the categorization of the CR did not make a difference, the inspectors questioned whether the Management Review Board was making conservative calls in cases where the guidance would allow a "condition adverse to quality" or a "significant condition adverse to quality" classification.

Based on the results of our reviews, the inspectors concluded that the independent CAP assessment adequately assessed the licensee's CAP implementation in response to equipment issues. However, the inspectors noted that the independent assessment did not address the CAP's response to human and organization performance issues. While the independent review included individual human performance errors for specific events, it did not look at the licensee's actions to address the cumulative nature of the errors. This is significant from the stand point that many of the CRs at Perry were the result of human performance errors and the independent team did not address how the CAP addressed the larger issue of human performance.

c. Findings

No findings of significance were identified.

.2 (Closed) URI 05000440/2007006-01: Failure to Adequately Correct and Evaluate a Condition Affected the ESW Pump and Its Associated Discharge Valves

An URI was opened during the 2007 Modification/50.59 Inspection associated with the licensee's failure to adequately evaluate and implement appropriate corrective actions to address a condition that affected ESW Pump 'A' and its associated discharge valve. Based on the information discussed in Section 1R17.1.b.1 of this report, an NCV of 10 CFR Part 50, Appendix B, Criterion XVI was identified. Therefore, this URI is closed.

.3 (Closed) URI 05000440/2007006-02: Emergency Diesel Generators Non-Critical Bypass Circuits Modification

During the Modification/50.59 Inspection at Perry Power Plant, which was conducted August 27 through September 14, 2007, the inspectors identified an URI concerning the adequacy of a 10 CFR 50.59 safety evaluation, which was completed for Engineering Change Package (ECP) 05-0229, "Division 1 and 2 Emergency Diesel Generator (EDG) Bus Under/Degraded Voltage Start Logic Modification."

The licensee implemented modification ECP 05-0229 in August 2007 which altered the operation of the EDGs, such that during a bus under/degraded voltage (LOOP) event, the associated EDG non-critical trips would be bypassed. These trips would continue to provide alarms on abnormal engine conditions to alert the operator of the engine abnormal condition. Prior to the implementation of the modification, these non-critical trips were not bypassed under these conditions. The non-critical trips were only bypassed during a LOCA signal.

The inspectors also noted that TS Surveillance Requirement 3.8.1.13 required verifying that each of the EDGs' automatic trips are bypassed on an actual or simulated emergency core coolant system initiation signal except for engine overspeed and generator differential current. The TS Surveillance Requirement did not address verifying the bypass of these non-critical trips on a LOOP signal.

The issue was unresolved pending further NRC review of Perry's licensing basis to verify whether the design change that bypassed the non-critical trips on a LOOP start signal had not affected the accident analysis, did not result in more than minimal increase in the likelihood of occurrence of a malfunction of the EDGs, and to evaluate if the activity has also not affected the current TS Surveillance Requirement 3.8.1.13.

The inspectors discussed this issue with the Office of Nuclear Reactor Regulation staff and based on the available information, the inspectors determined that the design change modification did not affect the EDGs' design basis for concurrent LOOP/LOCA accident analysis. The inspectors also determined that a TS amendment was not necessary because the licensee's current TS surveillance procedures indirectly verify that the EDG non-critical trips are bypassed on LOOP signal. Therefore, the inspectors concluded that no findings of significance were identified and no violation of NRC requirements occurred. This URI is closed.

.4 Quality of Immediate Investigations

The inspectors completed reviews of two Immediate Investigations that were considered to have provided inadequate evaluation to support the investigations' conclusions (URI 05000440/2007010-04). While the inspectors determined that the Immediate Investigations associated with this URI were representative of a weakness in technical rigor associated with the investigations, no findings of significance were identified. This issue is considered closed.

.5 Review of Institute of Nuclear Power Operations (INPO) Report

The inspectors completed a review of the interim report for an INPO October 2007 Evaluation. No additional follow-up is planned.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to the Site Vice President, Mr. Mark Bezilla and other members of licensee management on April 15, 2008. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit

- IP 92702 exit meeting with Mr. Barry Allen on February 21, 2008; and
- The closure of URI 05000440/2007006-01 as a Non-Cited Violation and Unresolved item 05000440/2007006-02 (no violation) with Mr. Elberfeld, Nuclear Compliance Supervisor, on April 2, 2007, via telephone.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

Cornerstone: Mitigating Systems

- As outlined in 10 CFR 50.36(c)(3), "Surveillance Requirements," are requirements relating to testing that assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met. Contrary to this requirement, on January 18, 2008, the licensee discovered plant procedures allowed preconditioning of the EDG jacket water system during surveillance testing. In reviewing Standard Operating Instruction (SOI)-R43, "Division 1 and 2 Diesel Generator System," Revision 31, licensee personnel determined that preventative maintenance could be performed to address jacket water leaks. Part 9900: Technical Guidance, Maintenance, Section C.1.c, addressed as unacceptable the alteration or adjustment of the physical conditions of a structure system or component during a surveillance test which results in acceptable test results. The inspectors determined that the finding was of very low safety significance because the finding did not result in a loss of safety function. The licensee entered the issue into their CAP as CR 08-33768.

- Technical Specifications 5.4, “Administrative Controls,” required procedures to be established, implemented, and maintained for plant safety equipment. Contrary to this requirement, on February 20, 2008, the licensee discovered that plant personnel failed to appropriately implement procedures while conducting work on the RCIC system. The licensee documented several instances of failure to implement procedures in CRs 08-35811, 08-35725, and 08-35803. The inspectors determined that the finding was of very low safety significance because the finding did not result in a loss of safety function greater then its TS-allowed outage time.

- Technical Specification 3.0.4, “Limiting Condition for Operation (LCO) Applicability,” required that a mode change cannot occur when an LCO is not met, unless certain conditions are fulfilled. Contrary to this requirement, the plant changed modes on numerous occasions between January 21, 2006, and November 28, 2007, when TS LCO 3.5.1 requirements B.1 and TS LCO 3.5.3 A.2 were not met. Specifically, the RCIC system was in an inoperable status from January 21, 2006 to November 28, 2007. Technical Specifications prohibited mode changes when the RCIC system is inoperable. Not knowing that TS LCO 3.5.3 was not met, licensee personnel allowed the mode changes.

- Technical Specification 3.0.4, “Limiting Condition for Operation (LCO) Applicability,” required that a mode change cannot occur when an LCO is not met, unless certain conditions are fulfilled. Contrary to this requirement, on December 6, 2007, the plant changed modes when TS LCO 3.5.3 requirements were not met. Specifically, on December 12, 2007, the licensee discovered through engineering analysis and OpE review that the RCIC flow controller settings were improper. The improper flow controller settings would preclude reliable system performance in accordance with RCIC safety design criteria. The plant entered Mode 2 on December 6, 2007, and subsequently Mode 1 with RCIC inoperable. Not knowing that TS LCO 3.5.3 was not met, licensee personnel allowed the mode changes.

Cornerstone: Barrier Integrity

- Technical Specification 3.0.3, “Limiting Condition for Operation (LCO) Applicability,” requires the unit to be placed in a Mode in which the LCO is not applicable, and action shall be initiated within one hour to meet the requirements of TS 3.0.3. Contrary to this requirement, on December 21, 2007, the plant continued operation in Mode 1 when TS LCO 3.0.3 requirements were met and did not initiate action for Mode 2 entry. Specifically, on December 21, 2007, the licensee discovered through maintenance results that both trains of AEGTS were inoperable for a period of one hour and 46 minutes. This condition met TS 3.6.4.3 Condition D and required immediate TS 3.0.3 entry. The licensee was unaware of the TS 3.0.3 condition until after the 'A' train was returned to operable status on December 1, 2007, at 10:12 a.m.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Allen, Vice President Nuclear
M. Bezilla, Vice President Nuclear
C. Elberfeld, Nuclear Compliance Supervisor
K. Krueger, Plant General Manager

Nuclear Regulatory Commission

D. Passehl, Senior Reactor Analyst, Division of Reactor Safety, RIII
D. Hills, Chief, Engineering Branch 1, Division of Reactor Safety, RIII

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000440/2008002-01	NCV	Adequacy of Maintenance Associated with Emergency Service Water Strainer Failure (Section 1R12)
05000440/2008002-02	NCV	Failure to Implement Adequate Configuration Control Affecting 'A' Reactor Water Cleanup System (Section 1R13)
05000440/2008002-03	NCV	Failure to Adequately Correct and Evaluate a Condition Affecting the ESW Pump and Its Associated Discharge Valves (Section 1R17)
05000440/2008002-04	NCV	Adequacy of Reactor Core Isolation Cooling System Flow Controller Tuning Procedures (Section 1R22.b.1)
05000440/2008002-05	NCV	Inadequate Test Control Program to Ensure Reactor Core Isolation Cooling System Operability (Section 1R22.b.2)
05000440/2008002-06	NCV	Failure to Implement Testing of the Reactor Core Isolation Cooling Instrument Lines With Appropriate Procedures (Section 1R22.b.3)
05000440/2008002-07	NCV	Failure to Make 10 CFR 50.72 Report (Section 4OA1.b.1)
05000440/2008002-08	FIN	Failure to Report Timely Performance Indicator Information (Section 4OA1.b.2)
05000440/2008002-09	NCV	Failure to Ensure Recurrence of Reactor Core Isolation Cooling Inoperability Due to Improper

		Controller Settings (Section 4OA3.3)
05000440/2008002-10	NCV	Inadequate Classification of Condition Report for Reactor Core Isolation Cooling (Section 4OA3.5)
05000440/2008002-11	NCV	Loss of Safety Function of the Annulus Exhaust Gas Treatment System (Section 4OA3.6)

Opened

05000440/2008002-12	URI	Reactor Core Isolation Cooling System Flow Controller Reliability (Section 4OA2.4)

Closed

05000440/2007-004-00 05000440/2007-004-01	LER	Automatic Reactor Protection System Actuation Due to Feedwater Control Power Supply Failure (Section 4OA3.1)
05000440/2007-005-00	LER	Plant Startup with Inoperable Reactor Core Isolation Cooling System (Section 4OA3.3)
05000440/2007005-09	URI	Adequacy of Maintenance Associated with Emergency Service Water B Strainer Failure (Section 1R12)
05000440/2007005-10	URI	Adequacy of Reactor Core Isolation Cooling System Surveillance Testing (Section 1R22.b.2)
05000440/2007005-11	URI	Adequacy of Reactor Core Isolation Cooling System Flow Controller Tuning Procedures (Section 1R22.b.1)
05000440/2007010-01	URI	Inadequate Classification of Condition Report for RCIC Failure to Run (Section 4OA3.5)
05000440/2007010-02	URI	RCIC Operability Between January 2006 and December 2007 (Section 1R22.1)
05000440/2007010-04	URI	Quality of Immediate Investigations (Section 4OA5.4)
05000440/2007006-01	URI	Failure to Adequately Correct and Evaluate a Condition Affecting the ESW Pump and its Associated Discharge Valves (Section 4OA5.2)
05000440/2007006-02	URI	Emergency Diesel Generator Non-Critical Trips Bypass Circuits Modification (Section 4OA5.3)
05000440/2007-006-00	LER	Loss of Safety Function and Condition Prohibited by TS Due to Annulus Exhaust Gas Treatment System Inoperability

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R04 Equipment Alignment

GCI-0016; Scaffolding Erection, Modification, or Dismantling Guidelines; Revision 10
VLI-P57; Safety-Related Instrument Air System; Revision 7
VLI-P42; Emergency Closed Cooling System; Revision 14

1R05 Fire Protection

FPI-0IB; Intermediate Building; Revision 5
FPI-1RB; Unit 1 Reactor Building; Revision 4
FPI-0CC; Control Complex; Revision 7
FPI-1AB; Auxiliary Building; Revision 2

1R11 Operator License Qualification

OTLC-305800806_PY-SGB; Scenario Guide Reference

1R12 Maintenance Effectiveness

CR 07-31994; ESW B Discharge Strainers Failed; dated December 27, 2007
Control Room Operator Logs; dated December 27, 2007
WO 200188470; Emergency Service Water Pump; dated April 20, 2007
Perry Nuclear Power Plant, Plant Health Report 2007-02
CR 08-34625; Shutdown of Division 2 Diesel Generator When Performing Governor Check; dated January 30, 2008
Plant Health Report 2007-04

1R13 Maintenance Risk Assessments and Emergent Work Control

VLI-E22A; High Pressure Core Spray; Revision 7
VLI-E12; Residual Heat Removal System; Revision 8
CR 07-30965; Packing Leak 1P45-F068A; dated December 4, 2007
CR 08-32531; Valve Found Out of Position and Near Miss; dated January 4, 2008
CR 08-32583; RWCU Pump A Exceeded Allowable Warmup Rate; dated January 4, 2008
Clearance PY1-G33-0004; RWCU Pump A; dated January 4, 2008
CR 08-35163; Unplanned Tech Spec Entry Which Declared ECC B and Associated Systems Inop; dated February 10, 2008
CR 08-32905; RWCU A Pump Casing Temperature Reading Low; dated January 8, 2008
On-Line Probabilistic Risk Assessment; Period 4 Week 2; Revision 1
Reactivity Plan; Downpower for Power Suppression Testing, Control Rod Pattern Change, and Scram Time Testing; Revision 1
IPTE Worksheet; Fuel Defect Localization and Suppression per FTI B0013; dated March 13, 2008

Perry Work Implementation Schedule; Week 7, Period 3
Perry Work Implementation Schedule; Week 9, Period 3
Perry Work Implementation Schedule; Week 11, Period 3
Perry Work Implementation Schedule; Week 2, Period 4
Perry Work Implementation Schedule; Week 5, Period 4

1R15 Operability Evaluations

CR 07-31441; RCIC Flow Controller Dynamic Response – Immediate Investigation; dated December 21, 2007
EQ-195; Qualified Life of Bailey Controls 701 Controller; Revision 0
CR 08-33381; RCIC Flow Control Loop Step Changes – Immediate Investigation; dated January 20, 2008
CR 08-33913; RCIC Flow Controller 1E51R0600 Output Voltage Not Checked After Reinstallation; dated January 20, 2008
CR 08-34502; Packing Leak on 1P45-F068A, RHR ‘A’ HX ESW Outlet Valve; dated January 29, 2008
SVI-P45-T2001; ESW Pump A and Valve Operability Test; Revision 17
WO 200292540; ESW Outlet Isolation Valve from RHR HX – Packing Leak; dated January 17, 2008
CR 08-35931; Packing Leak, ESW 'A' Loop Discharge Valve; dated February 26, 2008
CR 08-35817; RHR A Suction Pressure Low Alarm Received on Pump Start; dated February 23, 2008
CR 08-36674; Inspection of the ESW Piping at P45-F068A; dated March 11, 2008
CR 08-37130; Div. 1 D/G Control Transfer Switch SW8 (1R22-S51) Hardware Found Loose; dated March 20, 2008

1R17 Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications

CR 0726346, Perry TS SR 3.8.1.13 Does Not Match the Standard TS, dated September 10, 2007
CR 07-26412, Potential for Control Room Fire Result in Unavailability of ESW Pump A, dated September 11, 2007
Drawing D-240-012, Front and Side Views – Reactor Core Cooling Benchboard -1H13-P601, Revision B
D-240-039, Front View – Diesel Generator Benchboard – 1H13-P877, Revision M
10 CFR 50.59 Evaluations/SCREENINGS, SE 06-00185, Division 1 and 2 Diesel Generator Bus Under/Degraded Voltage Start Logic Modification, dated August 26, 2007
10 CFR 50.59 Evaluations/SCREENINGS, Screen 06-03964, Loss of DC Bus ED-1-A, dated December 11, 2006
CR 0726346, Perry TS SR 3.8.1.13 Does Not Match the Standard TS, dated September 10, 2007

1R19 Post-Maintenance Testing

WO 200295394; PY-1P42 Emergency Closed Cooling Pump A; dated January 6, 2008
P42 Emergency Closed Cooling System Lubrication Manual
WO 200171293; PY-1P42 Emergency Closed Cooling; dated November 7, 2006
WO 200297675; Partial RCIC PMT for WO 200296765; dated January 19, 2008
WO 200297676; Partial RCIC PMT for WO 200296765; dated January 20, 2008

CR 08-35578; RCIC Governor Tubing Dimensions Different from Drawing; dated February 19, 2008
CR 08-33829; Failed PMT for RCIC Pump Flow Controller WO 200296765; dated January 19, 2008
WO 200315433; Hydraulic Control Unit (06-47); dated March 16, 2008
CR 08-36844; Control Rod 06-47 Movement Issues; dated March 14, 2008
GMI-0019; Disassembly and Reassembly of CRD HCU Directional Control Valves; Revision 3

<u>1R22 Surveillance Testing</u>

WO 200296765; RCIC Pump Flow Controller Tuning; dated January 18, 2008
ICI-C-E51-3; RCIC Control System Tuning; Revision 2
ICI-C-E51-3; RCIC Control System Tuning; Revision 3
ICI-C-E51-3; RCIC Control System Tuning; Revision 5
CR 07-31441; RCIC Flow Controller Dynamic Response Immediate Investigation; dated December 21, 2007
ICI-B16-15; Plant Instrument Calibration Instruction – Bailey Type 701 Controller; Revision 3
Bailey Controls Service Manual, 4570K11-300G, Type 701 Basic Controller
CR 07-30660; RCIC Tripped on Actuation; dated November 28, 2007
SVI-E51-T2001; RCIC Pump and Valve Operability Test; Revision 27
WO 200295224; SVI-E51-T2001; dated December 29, 2007
WO 200256262’ SVI-E51-T2001; dated December 26, 2007
WO 200294676’ SVI-E51-T2001; dated December 19, 2007
WO 200294675; SVI-E51-T2001; dated December 16, 2007
WO 200293121; SVI-E51-T2001; dated December 9, 2007
WO 200029281; SVI-E51-T2001; dated December 7, 2007
WO 200293120; SVI-E51-T2001; dated December 8, 2007
WO 200256261; SVI-E51-T2001; dated October 2, 2007
WO 200232269; SVI-E51-T2001; dated August 5, 2007
WO 200217300; SVI-E51-T2001; dated May 28, 2007
WO 200202791; SVI-E51-T2001; dated February 22, 2007
CR 08-33381; RCIC Flow Control Loop Step Changes; dated January 20, 2008
CR 08-33913; RCIC Flow Controller 1E51R0600 Output Voltage not Checked After Reinstallation; dated January 20, 2008
WO 200299311; dated February 14, 2008
WO 200194433; RHR/RCIC Steam Line Flow High Channel Calibration; dated February 21, 2008
WO 200194437; RCIC Steam Line Flow High & Timer Channel Calibration; dated February 21, 2008
CR 08-35794; Incorrect Restoration Step in SVI-E31T00124 A/B; dated February 22, 2008
CR 08-35791; Incorrect Restoration Step for RCIC SVI E31T5396A; dated February 22, 2008
CR 08-35788; Incorrect Restoration Step for RCIC SVI; dated February 22, 2008
CR 08-3477; NRC Special Inspection Report Identifies URI for Operability of RCIC System Between January 2006 and December 2007; dated February 1, 2008
CR 08-36522; NRC URI: Adequacy of RCIC System Surveillance Testing; dated March 7, 2008
Reactivity Plan; Downpower for Power Suppression Testing, Control Rod Pattern Change, and Scram Time Testing; Revision 1
IPTE Worksheet; Fuel Defect Localization and Suppression per FTI B0013; dated March 13, 008

1EP6 Drill Evaluation

2008 ERO Plan; Self Assessment Plan PY-SA-08-068 (PYER); dated March 20, 2008
03-25-08 ERO Drill Mini Scenarios 1-13 Descriptions
PNPP ERO Training Drill 03/25/2008; dated March 20, 2008

4OA1 Performance Indicator Verification

Control Room Operator Logs; dated 2007
NOBP-LP-4012-01; Unplanned Scrams Per 7000 Critical Hours, Revision 1; dated 2007

4OA2 Identification and Resolution of Problems

NOP-LP-2001; CAP; Revision 17
NOP-ER-3001; Problem Solving and Decision Making; Revision 4
IOI-3, Power Changes, Revision 35
IOI-4, Shutdown, Revision 12
IOI-5, Maintaining Hot Shutdown, Revision 8
IOI-8, Shutdown by Manual Reactor Scram, Revision 5
CR 08-34293; Integration of IOI's Can Be Improved; dated January 24, 2008
PYBP-SITE-0057; Work Around Process; Revision 1
CR 07-12789; Design Deficiency with the Aux Condensers A & B Level Instrumentation; dated January 16, 2007
CR 07-12821; Operational Focus Self Assessment Identified Area for Improvement; dated January 17, 2007
CR 07-20487; RWCU and Plant Air System Response During Turbine Trip and Station Load Transfer; dated May 13, 2007
CR 07-20443; Turbine Temperature Control Response Requires Manual Action; dated May 13, 2007
CR 08-35160; Determine Correct Hot Surge Tank Level Controller Setting; dated February 10, 2008
CR 08-35744; Too Many Recorders in the Control Room with Deficiencies; dated February 22, 2008
CR 08-35799; Concerns with Low Air Flow Conditions on RCIC Room Cooler; dated February 22, 2008
CR 08-35804; 1M39B0004 Vibration in Expedited Maintenance Range; dated February 22, 2008
CR 08-34293; Integration of IOI's Can Be Improved; dated January 24, 2008
CR 08-35923; BV-PA-08-01-NUREG-0612 Heavy Load Document Control Issue; dated February 26, 2008
CR 08-32972; Cross-cutting Theme for Human Performance Aspect H.3.A Work Control; dated January 7, 2008
CR 08-33768; Potential Preconditioning Permitted In EDG Surv. Procedures; dated January 18, 2008
CR 08-34489; CR 07-31218 Did Not Identify The HU Errors Made During SOI-E51 7.17; dated January 29, 2008
CR 08-35672; Woodward EGM Would Not Calibrate; dated February 20, 2008
CR 08-35942; Maintenance RCIC Outage Human Performance Issues; dated February 26, 2008
CR 08-35833; Potential Modification of Adjusting Bolt on RCIC Room Air Handling Unit Motor; dated February 24, 2008

CR 08-35579; ECCS Room Flood Watch Requirements Not Properly Implemented; dated February 19, 2008
CR 08-35826; e51f0045 Stroke Time Did Not Meet Stroke Time in SVI-E51-T2001; dated February 24, 2008
CR 08-35794; Incorrect Restoration Step in SVI-E31T00124A/B; dated February 22, 2008
CR 08-35791; Incorrect Restoration Step in SVI E31T5396A; dated February 22, 2008
CR 08-35788; Incorrect Restoration Step in for RCIC SVI; dated February 22, 2008
CR 08-35560; Additional Scaffold Builds Leads to Unplanned Dose; dated February 18, 2008
CR 08-35562; Maintenance Dose Estimates not Provided; dated February 19, 2008
CR 08-35588; MG Left in PCs at Step-Off-Pad; dated February 19, 2008
CR 08-35942; Maintenance RCIC Outage Human Performance Issues; dated February 26, 2008
CR 08-35833; Potential Modification of Adjusting Bolt on RCIC Room Air Handling Unit Motor; dated February 24, 2008
CR 08-35579; ECCS Room Flood Watch Requirements not Properly Implemented; dated February 19, 2008
CR 08-35826; E51F0045 Stroke Time Did Not Meet Stroke Time in SVI-E51T2001; dated February 24, 2008
CR 08-35560; Additional Scaffold Builds Leads to Unplanned Dose; dated February 18, 2008
CR 08-35562; Maintenance Dose Estimates not Provided; dated February 19, 2008
CR 08-35588; MG Left in PCs at Step-off Pad; dated February 19, 2008
CR 08-35811; QC ID: Fundamental Work Practices not Implemented; dated February 20, 2008
CR 08-35813; Documentation in Work Order Did Not Meet Requirements of NOP-WM-4300; dated February 22, 2008
CR 08-35725; RCIC Isolation Instrumentation Tech Spec Entry; dated February 20, 2008
CR 08-35803; Improper Fill and Vent of RCIC System; dated February 23, 2008
RCE 07-28851; Dropped Fuel Channel; dated November 12, 2007
RCE 07-30660 / 07-31441; RCIC System Trip Following Plant Scram; dated January 28, 2008
RCE 07-30642; DFWCS Scram; dated January 18, 2008
RCE 08-35515 / 08-35665; Operator Training Programs Placed on Probation; dated February 13, 2008
RCE 07-31871; AEGTS Failure; dated February 18, 2008
NOP-NF-1102 Action Level 1 – Indication of a Low Release Defect; March 2008 Staff Recommendations

4OA3 Event Follow-Up

CR 08-32531; Valve Found Out of Position and Near Miss; dated January 4, 2008
Drawing D-303-0672-00000; Reactor Water Cleanup System; Revision GG
Radiological Survey Form; RWCU Pump and Valve Rooms; dated January 4, 2008
CR 08-35943; NRC Staff Response to Perry NRC Performance Indicator FAQs; dated February 26, 2008
CR 07-31871; AEGTS B Discharge Damper is Not Functioning Correctly; dated December 21, 2007
CR 07-31923; M15-F070B Damper Failure; dated December 22, 2007
Operational Decision Making Sheet; Cycle 12 Fuel Defect Operation; dated March 10, 2008

4OA5 Other Activities

PY-SA-07-67, Independent Assessment of the CAP Implementation at Perry Nuclear Power Plant; dated September 24, 2007

CR 07-14481; VP Approval Signatures are Missing in Crest; dated February 13, 2007
CR 07-19009; Emergency AC NRC Performance Indicator is White; dated April 21, 2007
CR 07-20446; Unexpected Turbine Trip during Startup; dated May 13, 2007
CR 07-20576; Reactor Scram During Digital Feedwater Control System Testing Under TXI-373; dated May 15, 2007
CR 07-24458; Common Cause of Site Issues; dated July 31, 2007
CR 07-25102; 1st Half 2007 Site Rollup IPA Trend: Supervisory Team Alignment; dated August 3, 2007
CR 07-25209; 2007 IND. CAP Assessment: Limited ACE Lacks Review for Transportability; dated April 19, 2007
CR 07-25222; Trend of Organization Misalignment on Initiating Condition Reports; dated August 16, 2007
CR 07-25258; 2007 IND. CAP Assessment: Improper Categorization of Condition Reports; dated August 15, 2007
CR 07-25260; 2007 IND. CAP Assessment: CR 06-8758 Evaluation Requires Clarification; dated August 16, 2007
CR 07-25395; 2007 IND. CAP Assessment: Limited ACE 07-22981 Generic Implication; dated August 20, 2007
CR 07-25415; 2007 IND. CAP Assessment: Potential Incomplete Evaluation of CR 07-15934; dated August 21, 2007
CR 07-25452; 2007 IND. CAP Assessment: Improper Closure of Corrective Actions for CR 06-11339; dated August 21, 2007
CR 07-25716; 2007 IND. CAP Assessment: AFI for Task #2, Evaluation & Resolution of Problems; date August 27, 2007
CR 07-25717; 2007 IND. CAP Assessment: Task #3 Area In Need of Attention; dated August 27, 2007
CR 07-25720; 2007 IND. CAP Assessment: Area in Need of Attention for Task #3; dated August 27, 2007
CR 07-25723; 2007 IND. CAP Assessment: Area in Need of Attention for Task #3; dated August 27, 2007
CR 07-25725; 2007 IND. CAP Assessment: Area in Need of Attention for Task #4; dated August 27, 2007
CR 07-25726; 2007 IND. CAP Assessment: Area in Need of Attention for Task #6; dated August 27, 2007
CR 07-31363; Declining Quality in Cause Evaluations; dated December 11, 2007

4OA7 Licensee-identified Violations

CR 08-33768; Potential Preconditioning Permitted in EDG Surveillance Procedures; dated January 18, 2008
CR 08-35811; Fundamental Work Practices not Implemented; dated February 20, 2008
CR 08-35813; Documentation in Work Order Did Not Meet Requirements of NOP-WM-4300; dated February 22, 2008
CR 08-35725; RCIC Isolation Instrumentation Tech Spec Entry; dated February 20, 2008
CR 08-35803; Improper Fill and Vent of RCIC system; dated February 23, 2008

LIST OF ACRONYMS USED

AC	alternating current
AEGTS	Annulus Exhaust Gas Treatment System
CAP	corrective action program
CDF	core damage frequency
CCDF	conditional core damage frequency
CFR	*Code of Federal Regulations*
CR	condition report
DC	direct current
DFWCS	Digital Feedwater Control System
ECP	Engineering Change Package
EDG	emergency diesel generator
ESW	emergency service water
FAQ	Frequently-Asked-Question
FENOC	FirstEnergy Nuclear Operating Company
FPI	Fire Protection Instruction
IMC	Inspection Manual Chapter
IP	Inspection Procedure
LCO	limiting condition for operation
LER	Licensee Event Report
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
NCV	non-cited violation
NEI	Nuclear Energy Institute
NOP	Normal Operating Procedure
NRC	Nuclear Regulatory Commission
ONI	Off-Normal Instruction
OpE	Operating Experience
PAP	Perry Administrative Procedure
PI	Performance Indicator
RASP	Risk Assessment Standardization Project
RCIC	reactor core isolation cooling
RHR	residual heat removal
RPS	Reactor Protection System
RPV	reactor pressure vessel
RWCU	reactor water cleanup
SDP	Significance Determination Process
SOI	Standard Operating Instruction
SPAR	Standardized Plant Analysis Risk
SRA	Senior Reactor Analyst
SVI	Surveillance Instruction
TS	Technical Specification
USAR	Updated Safety Analysis Report
URI	unresolved item
VLI	Valve Lineup Instruction
WO	work order