January 30, 2007

EA-06-326
EA-01-083

Mr. L. William Pearce
Site Vice President
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P. O. Box 97, 10 Center Road, A290
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT
NRC INTEGRATED INSPECTION REPORT 05000440/2006005
EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Pearce:

On December 31, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings that were discussed on January 5, 2006, with you and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. In addition to the routine NRC inspection and assessment activities, Perry performance is being evaluated quarterly as described in the Assessment Follow-up Letter - Perry Nuclear Power Plant, dated August 12, 2004. Consistent with Inspection Manual Chapter (IMC) 0305, "Operating Reactor Assessment Program," plants in the "Multiple/Repetitive Degraded Cornerstone" column of the NRC's Action Matrix are given consideration at each quarterly performance assessment review for (1) declaring plant performance to be unacceptable in accordance with the guidance in IMC 0305; (2) transferring to the IMC 0350, "Oversight of Operating Reactor Facilities in a Shutdown Condition with Performance Problems," process; and (3) taking additional regulatory actions, as appropriate. On November 7, 2006, the NRC reviewed Perry operational performance, inspection findings, and performance indicators for the 4th quarter of 2006. Based on this review, we concluded that Perry is operating safely. We determined that no additional regulatory actions, beyond the already increased inspection activities and management oversight, are currently warranted.

Based on the results of this inspection, three findings of very low safety significance, two of which involved a violation of NRC requirements, were identified. However, because of their very low safety significance and because the issues have been entered into your corrective action program, the NRC is treating these violations as non-cited violations (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

Also, based on the results of this inspection, materially inaccurate and incomplete information was identified in the mitigating systems performance indicator (MSPI) data submitted to the NRC on July 21, 2006. On October 13, 2006, during your self-assessment review of the MSPI program in response to NRC-identified errors in the MSPI basis document, it was identified that incorrect data was submitted to the NRC in the 2nd and 3rd quarter 2006 MSPI reports due to an error associated with the incorrect use of an outdated Probabilistic Risk Assessment calculation. On November 16, 2006, you submitted corrected data that caused the 2nd quarter 2006 Emergency AC Power System MSPI color to change from Green to White and you entered the issue into your corrective action program. The inaccurate information provided on July 21, 2006, affected the color of the MSPI and therefore affected the timeliness of the NRC's response to the White MSPI. As a result, the NRC determined that a Severity Level IV violation of 10 CFR 50.9, "Completeness and Accuracy of Information," occurred. This violation was evaluated in accordance with the Enforcement Policy, which is included on the NRC's Web site at **www.nrc.gov** and can be accessed by selecting **What We Do, Enforcement**, then **Enforcement Policy**. However, after consultation with the Regional Administrator, NRC Region III, and the Director, Office of Enforcement, I have been authorized to exercise enforcement discretion pursuant to Section VII.B.6, "Violations Involving Special Circumstances," of the Enforcement Policy to refrain from issuing a Notice of Violation. Discretion is warranted in this case because: (1) submission of the incomplete and inaccurate MSPI information was not willful; (2) the incomplete and inaccurate MSPI information was identified within a period of 1 year after the beginning of MSPI data collection, or by April 1, 2007; and (3) in recognition of (a) ongoing Performance Indicator development activities, (b) the time constraints to gather and submit historical data, (c) the large volume of data (12 quarters of data) needed to calculate and verify the MSPIs, and (d) the time needed for licensees to familiarize and adjust to the new reporting guidance.

If you contest the subject or severity of the non-cited violations, 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 a copy 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 Resident Inspector 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/

Mark A. Satorius, Director
Division of Reactor Projects

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

Enclosure: Inspection Report 05000440/2006005
w/Attachment: Supplemental Information

cc w/encl: G. Leidich, President and Chief Nuclear Officer - FENOC
J. Hagan, Senior Vice President of Operations and Chief Operating Officer - FENOC
D. Pace, Senior Vice President, Fleet Engineering - FENOC
J. Rinckel, Vice President, Fleet Oversight
R. Anderson, Vice President, Nuclear Support
Director, Fleet Regulatory Affairs
Manager, Fleet Licensing
Manager, Site Regulatory Compliance
D. Jenkins, Attorney, FirstEnergy
Public Utilities Commission of Ohio
Ohio State Liaison Officer
R. Owen, Ohio Department of Health

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Ohio State Liaison Officer
R. Owen, Ohio Department of Health

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DATE	01/26/07		01/03/07		01/04/07		01/29/07		01/30/07	

OFFICIAL RECORD COPY

[1] NRR concurrence received on January 3, 2007, e-mail from R. Pascarelli, NRR

[2] OE concurrence received on January 4, 2007, e-mail from Leigh Trocine, OE

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	05000440/2006005
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	Perry, Ohio
Dates:	October 1, 2006, through December 31, 2006
Inspectors:	M. Franke, Senior Resident Inspector M. Wilk, Resident Inspector J. House, Senior Radiation Specialist T. Go, Radiation Specialist M. Bielby, Senior Operations Engineer C. Zoia, Senior Operations Engineer R. Clagg, Reactor Engineer R. Ruiz, Reactor Engineer J. McGhee, Reactor Engineer P. Zurawski, Reactor Engineer M. Garza, Emergency Response Specialist
Approved by:	Eric R. Duncan, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000440/2006005; 10/01/2006 - 12/31/2006; Perry Nuclear Power Plant; Licensed Operation Requalification; Other Activities.

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

The emergency preparedness portion of this inspection is administratively tracked as NRC Inspection Report 05000440/2006021.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR 55.46, "Simulation Facilities," when licensee personnel failed to adhere to simulator fidelity requirements prescribed by ANSI/ANS-3.5-1998 for annual steady-state operation testing. Specifically, the licensee failed to provide adequate documentation that demonstrated that heat balance testing was performed and evaluated annually as required. The finding was related to the cross-cutting area of Problem Identification and Resolution because the licensee failed to thoroughly evaluate the simulator model limitations to address extent of condition concerns. The reviews and analyses did not fully analyze the impacts of simulator model limitations on previous testing or identify that some test results were not documented. The correction of the simulator model limitations was expected to be accomplished by a simulator model upgrade, scheduled for completion in July 2007.

  The failure to evaluate and document simulator performance testing was more than minor because it affected the Mitigating Systems cornerstone and did not meet the requirements of 10 CFR 55.46 because of the realistic potential of providing negative training based on significant simulator deficiencies compared to the plant. The finding was considered to be of very low safety significance because it involved simulator fidelity and the simulator did not meet the performance requirements of 10 CFR 55.46 and had the potential to impact operator actions. (Section 1R11)

- Green. The inspectors identified a finding of very low safety significance and an associated non-cited violation of License Condition C(6) for the failure to promptly correct the long-term recurring condition of insufficient CO_2 tank level that was required to support the operability of the reactor recirculation pump CO_2 system. The inspectors noted the reactor recirculation pumps' CO_2 system did not meet fire protection requirements on several occasions since 2001 due to the same failure mechanism. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution because the licensee failed to take appropriate corrective actions to address the recurring condition of low tank level in a timely manner. As part of their immediate corrective actions, the licensee restored tank CO_2 level to restore system operability and performed maintenance on the CO_2 tank to stop the CO_2 leak.

This finding was more than minor because it was associated with the Protection Against External Factors 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 to prevent undesirable consequences. The finding was determined through a Significance Determination Process analysis to be of very low safety significance because of safety functions that were assumed to remain available in the event of a reactor recirculation pump fire even though the finding was assigned a high degradation rating due to inadequate agent concentration required for deep seated fires. (Section 4OA5.2)

- Green. The inspectors identified a finding of very low safety significance associated with the minimum flow settings for the high pressure core spray, low pressure core spray, and residual heat removal pumps. Bulletin 88-04 identified that many pump minimum flow values were too low because they did not account for flow instability concerns. The inspectors identified that when licensee personnel addressed this operating experience item, they failed to properly verify the minimum flow settings with the pump manufacturer in accordance with the bulletin. The licensee's corrective actions included having the manufacturer perform a new analysis, which concluded that the existing minimum flow settings did not allow continuous operation of the pumps, and provided a monitoring and maintenance schedule based on the minimum flow values in order to promptly detect degradation. This performance deficiency was entered into the licensee's corrective action program for resolution. No violation of NRC requirements was identified.

  This finding represented a performance deficiency because the licensee did not verify with the manufacturer that the minimum flow settings for these safety-related pumps were acceptable. The finding was more than minor because these pumps were operated since original plant start-up with an increased potential for unusual wear and aging without establishing increased monitoring and maintenance, or other compensatory actions and, therefore, was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and impacted the cornerstone objective of ensuring the availability and reliability of safety-related pumps. The finding was of very low safety significance based on the results of the licensee's analysis and screened as Green using the Significant Determination Process Phase 1 screening worksheet. (Section 4OA5)

B. Licensee-Identified Violations

None.

REPORT DETAILS

<u>Summary of Plant Status</u>

The plant began the inspection period at 82 percent reactor power as operators conducted power maneuvers for planned maintenance activities. On October 1, 2006, control room operators returned power to 100 percent. On December 13, 2006, at 4:35 a.m, operators manually scrammed the reactor and entered a forced outage in response to indications of degrading instrument air pressure and subsequent feedwater system instability. On December 18, 2006, following forced outage activities, the plant entered Mode 2 at 10:36 a.m. and the reactor was declared critical at 1:16 p.m. On December 19, 2006, the plant entered Mode 1 at 5:15 p.m. and the unit synchronized to the grid at 9:33 p.m. After a series of power maneuvers to support control rod adjustments, the plant reached 100 percent power at 4:18 a.m. on December 24, 2006. With the exception of planned downpowers for routine surveillance testing and rod sequence exchanges, the plant remained at 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 <u>Adverse Weather Protection</u> (71111.01)

a. <u>Inspection Scope</u>

The inspectors reviewed the licensee's procedures and preparations for cold weather conditions. The inspectors reviewed winterization procedures, severe weather procedures, and performed general area and system walkdowns. During walkdowns conducted during the week of November 13, 2006, the inspectors toured selected buildings and areas to determine whether the licensee had identified all discrepant conditions such as damaged doors, windows, or vent louvers. The inspectors reviewed documentation to determine whether licensee Normal Operating Procedure (NOP)-WM-2001, “Work Management Scheduling/Assessment/Seasonal Readiness Processes,” Revision 5, had been completed prior to the onset of cold weather. Additionally, the inspectors observed housekeeping conditions and verified that materials capable of becoming airborne missile hazards during high wind conditions, or impacting snow removal, were appropriately located and restrained. Finally, the inspectors reviewed the licensee’s cold weather readiness to determine whether cold weather protection features such as heat tracing and space heaters were monitored and functional.

This review represented one inspection sample.

b. <u>Findings</u>

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors conducted partial walkdowns of the system trains listed below to determine whether the systems were correctly aligned to perform their designed safety function. The inspectors used valve lineup instructions (VLIs) and system drawings during the walkdowns. The walkdowns 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 walkdowns are listed in the attached List of Documents Reviewed. The inspectors reviewed the following systems:

- Division 2 Emergency Diesel Generator (EDG) during a Division 1 EDG maintenance unavailability in the week of October 2, 2006;
- Division 1 EDG after maintenance activities on October 10, 2006;
- Emergency Service Water (ESW) "A" train during a Division 2 maintenance unavailability in the week of November 6, 2006; and
- Low Pressure Core Spray system following maintenance on November 30, 2006.

This review represented four quarterly partial system walkdowns.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05AQ)

a. Inspection Scope

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

- Fire Zone 0IB-2, Intermediate Building elevation 599';
- Fire Zone 1DG-1C , Unit 1 Division 1 Diesel Generator (DG) Building elevations 620' 6" and 646' 6"
- Fire Zone 1AB-1F, High Pressure Core Spray (HPCS) System elevation 574' 10";
- Fire Zone 0IB-1, Intermediate Building elevations 574' and 585';
- Fire Zones 0CC-2A, 2B, and 2C, Control Complex elevation 599';
- Fire Zone 0FH, Fuel Handling Building; and
- Fire Zone XFMR, Transformer Yard Areas

Emphasis was placed on evaluating the licensee's control of transient combustibles and ignition sources, the material condition of fire protection equipment, and 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 4; Perry

Administrative Procedure (PAP)-1910, "Fire Protection Program," Revision 14; and PAP-0204, "Housekeeping/Cleanliness Control Program," Revision 17; 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 represented seven quarterly inspection samples.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the Division 1 and 2 EDG jacket water heat exchanger performance testing program. The inspectors reviewed the licensee's test data and reviewed historical trending data to verify that current testing frequency was sufficient to detect degradation of heat exchanger performance.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Licensed Operator Requalification Quarterly Inspection

a. Inspection Scope

On October 18, 2006, the resident inspectors observed licensed operator performance in the plant simulator. 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.

The inspectors also observed the licensee's evaluation of crew performance to determine whether the training staff had identified performance deficiencies and specified appropriate remedial actions.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Facility Operating History

a. Inspection Scope

The inspectors reviewed the plant's operating history from November 2004 through November 2006 to identify operating experience that was expected to be addressed by the Licensed Operator Requalification Training (LORT) program. The inspectors assessed whether the identified operating experience had been addressed by the facility licensee in accordance with the station's approved Systems Approach to Training (SAT) program to satisfy the requirements of 10 CFR 55.59(c), "Requalification Program Requirements."

b. Findings

No findings of significance were identified.

.3 Licensee Requalification Examinations

a. Inspection Scope

The inspectors performed a biennial inspection of the licensee's LORT test/examination program for compliance with the station's SAT program which would satisfy the requirements of 10 CFR 55.59(c)(4), "Evaluation." The reviewed operating examination material consisted of seven operating tests, each containing two or three dynamic simulator scenarios (as appropriate) and 10 job performance measures (JPMs). The written examinations reviewed consisted of seven examinations, each with 10 closed reference questions and 30 open reference questions. The senior reactor operator (SRO) examination contained a minimum of six SRO level questions. The inspectors reviewed the annual requalification operating test and biennial written examination material to evaluate general quality, construction, and difficulty level. The inspectors assessed the level of examination material duplication from week-to-week during the current year operating test. The examiners assessed the amount of written examination material duplication from week-to-week for the written examination administered in 2006. The inspectors reviewed the methodology for developing the examinations, including the LORT program 2-year sample plan, probabilistic risk assessment (PRA) insights, previously identified operator performance deficiencies, and plant modifications.

b. Findings

No findings of significance were identified.

.4 Licensee Administration of Requalification Examinations

a. Inspection Scope

The inspectors observed the administration of a requalification operating test to assess the licensee's effectiveness in conducting the test to ensure compliance with 10 CFR 55.59(c)(4), "Evaluation." The inspectors evaluated the performance of one crew in parallel with the facility evaluators during three dynamic simulator scenarios and evaluated various licensed crew members concurrently with facility evaluators during the administration of several JPMs. The inspectors assessed the facility evaluators' ability to determine adequate crew and individual performance using objective, measurable standards. The inspectors observed the training staff personnel administer the operating test, including conducting pre-examination briefings, evaluations of operator performance, and individual and crew evaluations upon completion of the operating test. The inspectors evaluated the ability of the simulator to support the examinations. A specific evaluation of simulator performance was conducted and documented under Section 1R11.9, "Conformance With Simulator Requirements Specified in 10 CFR 55.46," of this report.

b. Findings

No findings of significance were identified.

.5 Examination Security

a. Inspection Scope

The inspectors observed and reviewed the licensee's overall licensed operator requalification examination security program related to examination physical security (e.g., access restrictions and simulator considerations) and integrity (e.g., predictability and bias) to verify compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The inspectors also reviewed the facility licensee's examination security procedure, any corrective actions related to past or present examination security problems at the facility, and the implementation of security and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the examination process.

b. Findings

No findings of significance were identified.

.6 Licensee Training Feedback System

a. Inspection Scope

The inspectors assessed the methods and effectiveness of the licensee's processes for revising and maintaining its LORT program up to date, including the use of feedback from plant events and industry experience information. The inspectors reviewed the licensee's quality assurance oversight activities, including licensee training department self-assessment reports. The inspectors evaluated the licensee's ability to assess the effectiveness of its LORT program and their ability to implement appropriate corrective actions. This evaluation was performed to verify compliance with 10 CFR 55.59(c), "Requalification Program Requirements" and the licensee's SAT program.

b. Findings

No findings of significance were identified.

.7 Licensee Remedial Training Program

a. Inspection Scope

The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the previous biennial requalification examinations and the training from the current examination cycle to ensure that they addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans. This evaluation was performed in accordance with 10 CFR 55.59(c), "Requalification Program Requirements," and with respect to the licensee's SAT program.

b. Findings

No findings of significance were identified.

.8 Conformance With Operator License Conditions

a. Inspection Scope

The inspectors reviewed the facility and individual operator licensees' conformance with the requirements of 10 CFR Part 55. This included a review of the facility licensee's program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53(e) and 10 CFR 55.53(f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators and which control room positions were granted watch-standing credit for maintaining active operator licenses. The inspectors reviewed the facility licensee's LORT program to assess compliance with the requalification program requirements as described by 10 CFR 55.59(c). Additionally, medical records for six licensed operators were reviewed for compliance with 10 CFR 55.53(I).

b. Findings

No findings of significance were identified.

.9 Conformance With Simulator Requirements Specified in 10 CFR 55.46

a. Inspection Scope

The inspectors assessed the Perry Plant-referenced simulator for compliance with 10 CFR 55.46, "Simulation Facilities." This assessment included the adequacy of the licensee’s simulation facility for use in operator licensing examinations and for satisfying experience requirements as prescribed by 10 CFR 55.46. The inspectors also reviewed a sample of simulator performance test records (i.e., transient tests, scenario-based testing and discrepancy resolution validation tests), simulator discrepancy and modification records, and the process used for ensuring simulator fidelity in accordance with 10 CFR 55.46. The inspectors also reviewed and evaluated the discrepancy process to ensure that simulator fidelity was being maintained. Open simulator discrepancies were reviewed for importance relative to the impact on 10 CFR 55.45 and 55.59 operator actions as well as on nuclear and thermal hydraulic operating characteristics. Finally, the inspectors conducted interviews with members of the licensee’s simulator staff about the configuration control process and completed the Inspection Procedure 71111.11, Appendix C, checklist to evaluate whether or not the licensee’s plant-referenced simulator was operating adequately as required by 10 CFR 55.46(c) and (d).

b. Findings

Introduction: The inspectors identified a finding of very low safety significance and a non-cited violation (NCV) of 10 CFR 55.46, “Simulation Facilities,” when licensee personnel failed to adhere to simulator fidelity requirements prescribed by ANSI/ANS-3.5-1998, Section 4.4.3.1, “Simulator Operability Testing,” for steady-state operation testing. In particular, licensee personnel failed to provide adequate documentation that demonstrated that heat balance testing was performed and evaluated annually, as required.

Description: During an operator requalification biennial inspection conducted in December 2006, a review of the steady-state and normal evolution heat balance tests was conducted on the Perry simulator. The inspection revealed that the licensee had not performed and documented all of the testing required to demonstrate simulator fidelity. The inspectors requested all available heat balance test documentation over the last 4 years; however, the licensee could not provide any 2003 testing documentation, the 2005 plant data used to compare plant parameters to the simulator parameters, or any other analysis documentation for the 2005 heat balance test. The licensee provided the available data for the 2004 and 2006 tests, but these had insufficient documentation to demonstrate how the parameter value acceptable bands were determined for comparing simulator data to allowed tolerances. Finally, the inspectors found that not all identified test failures were properly corrected and then retested in either the 2004 or 2006 tests.

Specifically, the single acceptance criteria failure (Megawatts electric (MWe) was found to be about 25 MWe low at 45 percent power) in the 2004 test was addressed by initiating simulator work order (SWO) 04-0031, which was then closed to simulator work request (SWR) 06-0020. However, SWR 06-0020 and condition report (CR) 06-02821 documented that several parameter values were found to be unacceptable at each of three power levels where heat balance data was collected (45 percent, 80 percent and 100 percent) during the performance of the 2006 test. Subsequently, a simulator expert contracted by the licensee determined that due to simulator model limitations, all required values could not be adjusted to be within ANSI tolerance at all three power levels. Therefore, the 100 percent power level was chosen to be the only correct power level, as documented by an analysis attached to CR 06-02821. Accordingly, adjustments to the 100 percent parameter values were made in order to align simulator values to within specified requirements, but only the 100 percent power level parameters were retested afterward. When asked if the other power levels were retested after the adjustments, licensee personnel only recalled noticing that the new values were about the same, but could not provide documentation to support the conclusion.

In addition, the licensee could not explain why the 2004 test found only one simulator parameter out of tolerance, whereas the 2006 test found several values out of tolerance and determined that the values could not be simultaneously adjusted to within specifications due to simulator model limitations. The correction of the simulator model limitations was expected to be accomplished through a simulator model upgrade that was scheduled for completion in July 2007, which would then close SWR 06-0020. Therefore, the 2004 failure, an unknown number of 2005 failures, and numerous 2006 failures will remain uncorrected until after the successful implementation of this upgrade.

Analysis: The performance deficiency associated with this finding was a failure by the licensee to conduct performance testing throughout the life of the simulation facility in a manner sufficient to ensure simulator fidelity.

The inspectors reviewed Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," dated June 22, 2006, and determined that the issue was more than minor based upon example 1.c, when required testing was not performed and subsequent testing indicated that simulator fidelity requirements were not met. In addition, this finding affected the Mitigating Systems cornerstone of Reactor Safety and did not meet the requirements of 10 CFR 55.46 due to the realistic potential of providing negative training based on significant simulator deficiencies compared to the plant. Using IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process," dated August 22, 2005, the inspectors found that the finding was related to simulator fidelity and that the simulator did not meet the performance requirements of 10 CFR 55.46 that had the potential impact of providing negative training. This evaluation resulted in a finding of very low significance. The finding was related to the cross-cutting area of Problem Identification and Resolution because the licensee failed to thoroughly evaluate the simulator model limitations to address extent of condition concerns. The reviews and analyses did not fully analyze the impacts of simulator model limitations on previous testing or identify that some test results were not documented.

Enforcement: 10 CFR 55.46(d)(1) required that a facility periodically conduct performance testing throughout the life of the simulation facility in a manner sufficient to ensure that paragraph (d)(3) was met. Paragraph (d)(3) required that the results of any uncorrected performance test failures that may exist at the time of the requalification program inspection be made available for NRC review. The results of performance tests must be retained for 4 years after the completion of each performance test or until superseded by updated test results. Contrary to these requirements, licensee personnel failed to demonstrate that simulator fidelity was met in accordance with ANSI/ANS-3.5-1998, Section 4.4.3.1, "Simulator Operability Testing," for steady-state operation testing, which was required annually. The ANSI/ANS 3.5-1998 standard, "Nuclear Power Plant Simulators for Use in Operator Training," was endorsed by Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations," Revision 3, and committed to in Perry's "Operator Requalification Programs," Revision 8 established in accordance with 10 CFR 55.59(c). The licensee failed to provide adequate documentation that annual heat balance testing was performed and evaluated annually over the last 4 years. However, because of the very low safety significance of the issue and because the issue has been entered into the licensee's corrective action program (CR 06-11107), the issue is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2006005-01).

.10 Annual Operating Test Results and Biennial Written Examination Results

a. Inspection Scope

The inspectors reviewed the pass/fail results of the individual biennial written examinations, and the annual operating tests (required to be given annually per 10 CFR 55.59(a)(2)) administered by the licensee during calendar year 2006. The overall written examination and operating test results were compared with the significance determination process in accordance with IMC 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process."

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Quarterly Maintenance Effectiveness

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 (SSCs) 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), select surveillance test procedures, and a sample of 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 PRA. During this inspection period, the inspectors reviewed the following SSC:

- annulus exhaust gas treatment system

This review represented 1 quarterly inspection sample.

b. Findings

No findings of significance were identified.

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 50.65(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 SSCs. The following assessments and/or activities were reviewed:

- licensee's management of risk and work control during a Division 1 maintenance outage during the week of October 2, 2006;

- licensee's management of risk and work control during an L20 electrical bus maintenance outage during the week of October 30, 2006;
- licensee's management of risk and work control during a Division 2 maintenance outage during the week of November 6, 2006;
- licensee's management of risk and work control when encountering emergent clearance control issues affecting electrical offsite power indication on November 7, 2006; and
- licensee's management of risk and work control to address emergent maintenance activities associated with reactor power ascension following a forced outage during the week of December 18, 2006.

These reviews represented five inspection samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors selected CRs related to potential operability issues 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 and design criteria in the appropriate sections of the Technical Specifications (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 following issues:

- an operability evaluation associated with discolored oil in the HPCS waterleg pump identified on October 20, 2006;
- an operability evaluation associated with an area radiation monitoring system detector configuration discrepancy during the week of October 23, 2006;
- an operability evaluation associated with a Division 2 EDG cylinder head bolts undertorqued condition identified on November 10, 2006; and
- an operability evaluation associated with ESW sluice gate system material degradation identified on November 27, 2006.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the design change package for the addition of an alternate decay heat removal system and specifically examined modifications made to the service water system. The inspectors reviewed the engineering change package and the design interface evaluations relative to the Perry licensing basis. Finally, the inspectors reviewed the WO documentation and walked down the modification to determine whether it was installed in accordance with design documents.

This review represented one inspection sample

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors evaluated the following post-maintenance testing (PMT) activities for risk-significant systems to assess 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 procedures, and various NRC generic communications. In addition, the inspectors reviewed CRs associated with PMT to determine whether the licensee was identifying problems and entering them in the corrective action program. The specific procedures and CRs reviewed are listed in the attached List of Documents Reviewed. The following PMT activities were reviewed:

- testing of the suppression pool make-up system following a level transmitter replacement during the week of October 16, 2006;
- testing of the Division 1 EDG ventilation system following maintenance during the week of October 23, 2006;
- testing of a reactor core isolation cooling (RCIC) transmitter following maintenance during the week of October 30, 2006; and
- testing of the standby liquid control “A” system following pump sight-glass repair during the week of November 27, 2006.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed activities associated with a forced outage initiated on December 13, 2006. The forced outage continued through December 19, 2006, when the plant was synchronized to the grid. The inspectors assessed the adequacy of forced outage-related activities, including implementation of risk management, conformance to approved site procedures, and compliance with TS requirements. The following major activities were observed or performed:

- On December 13, 2006, plant operators inserted a manual reactor scram in response to feedwater transients caused by an instrument air line break. The inspectors observed operator response to the instrument air line break and operator actions to shut down the plant. The inspectors observed shift briefings, operator performance, and shift management coordination of plant activities.

- From December 13 through December 18, 2006, the inspectors reviewed licensee activities to determine whether emergent issues were appropriately identified and resolved prior to reactor plant mode changes and power ascension. The inspectors observed the licensee's actions in response to high temperatures in the offgas system adsorber beds and hydrogen leaks from the main generator.

- From December 18 through December 19, 2006, the inspectors observed licensee start-up and power ascension activities. The inspectors observed shift briefings, operator performance, and shift management coordination of plant activities, including the synchronization of the turbine generator to the grid.

The observation of these activities represented one inspection sample.

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 ensure 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 surveillance activities assessed were:

- Division 3 EDG routine surveillance testing on October 17, 2006;
- Main Steam Line (MSL) low condenser vacuum channel “C”, routine testing on October 18, 2006;
- Division 1 EDG routine surveillance testing on October 31, 2006;
- Main Steam Line tunnel temperature channel “C”, reactor coolant system (RCS) leakage detection testing on November 7, 2006;
- Standby Liquid Control “A” system in-service testing on November 28, 2006; and
- Residual Heat Removal (RHR) “C” suppression pool suction isolation valve testing on December 4, 2006.

These reviews represented one containment isolation valve; one RCS leak detection; one in-service inspection sample; and three routine inspection samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary plant modifications. The inspectors assessed the acceptability of the temporary configuration change by comparing the 10 CFR 50.59 screening and evaluation information against the design basis, the UFSAR and the TS as applicable. The comparisons were performed to ensure that the new configurations remained consistent with design basis information. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems.

- chemical decontamination of the fuel pool cooling and cleanup system during the week of December 11, 2006; and
- offgas system modification to provide continuous nitrogen purge flow to the offgas system activated carbon adsorber bed trains during the week of December 18, 2006.

This review represented two inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors completed a screening review of Revisions 24, 25, and 26 of the Perry Nuclear Power Plant Emergency Plan to determine whether changes identified in these revisions may have reduced the effectiveness of the licensee's emergency planning. The screening review of Revisions 24, 25, and 26 does not constitute approval of the changes and, as such, the changes are subject to future NRC inspection to ensure that the emergency plan continues to meet NRC regulations.

These activities represented one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 As-Low-As-Is-Reasonably-Achievable Planning And Controls (ALARA) (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history and current exposure trends along with ongoing and planned activities to assess current performance and exposure challenges. This included determining the plant's current 3-year rolling average collective exposure to establish the effects of the plant's source term on radiological exposure to workers.

This review represented one sample.

Site specific trends in collective exposures and source-term measurements were evaluated.

This review represented one sample.

The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and processes used to estimate and track work activity exposures.

This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors evaluated the licensee’s interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling, and engineering groups for interface problems or missing programmatic elements.

This review represented one sample.

The inspectors reviewed work activity planning to determine if the licensee adequately considered certain dose reduction activities when developing the overall schedule. For example, the inspectors evaluated if the licensee’s schedule considered shielding (provided by water-filled components and piping), overall job scheduling and work activity interactions, and the sequence of shielding and scaffolding installation and removal activities, as applicable.

This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Source-Term Reduction and Control

a. Inspection Scope

The inspectors reviewed the licensee’s records and evaluated the historical trends and current status of tracked plant source terms. This included determining if the plant had developed contingency plans for expected changes in the source term due to changes in fuel performance or changes in the plant primary chemistry.

This review represented one sample.

The inspectors determined if the licensee had developed an understanding of the plant source term, which included knowledge of input mechanisms, in order to reduce the source term. This included an evaluation of the licensee’s boiling water reactor radiation assessment and control point dose rate increase during fuel cycle 10 and the licensee’s self-assessment of the cause of the increasing dose rates. The licensee’s source term control strategy, which included a process for evaluating radionuclide distribution plus a shutdown and operating chemistry plan capable of minimizing the source term external to the core, was evaluated. Other methods used by the licensee to control the source term, including component/system decontamination, hotspot flushing and the use of shielding, were evaluated.

This review represented one sample.

The licensee's process for identification of specific sources was reviewed along with exposure reduction actions and the priorities the licensee had established for implementation of those actions. This included the planned chemical decontamination of system piping and low temperature noble metal application for refueling outage 11. Source term reduction results achieved against these priorities since the last refueling cycle were reviewed. For the current assessment period, source term reduction evaluations were reviewed, and actions taken to reduce the overall source term were compared to the previous year.

This review represented one sample.

b. Findings

No findings of significance were identified.

.4 Declared Pregnant Workers

a. Inspection Scope

The inspectors reviewed exposure data for the only declared pregnant worker during the current assessment period. The licensee's monitoring control program was evaluated to determine if it met the requirements contained in 10 CFR Part 20.

This review represented one sample.

b. Findings

No findings of significance were identified.

.5 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and special reports related to the ALARA program since the last inspection to determine if the licensee's overall audit program's scope and frequency for all applicable areas under the Occupational Cornerstone met the requirements of 10 CFR 20.1101(c).

This review represented one sample.

The inspectors determined if identified problems were entered into the corrective action program for resolution, and that they had been properly characterized, prioritized, and resolved. This included dose significant post-job (work activity) reviews and post-outage ALARA report critiques of exposure performance.

This review represented one sample.

Corrective action reports related to the ALARA program were reviewed and staff members were interviewed to determine if follow-up activities had been conducted in an effective and timely manner commensurate with their importance to safety and risk using the following criteria:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation/consideration of risk-significant operational experience feedback.

This review represented one sample.

The inspectors also determined if the licensee’s self-assessment program identified and addressed repetitive deficiencies and significant individual deficiencies that were identified in the licensee's problem identification and resolution process.

This review represented one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

Cornerstones: Barrier Integrity, Occupational and Public Radiation Safety

.1 Reactor Safety Strategic Performance Area

a. Inspection Scope

The inspectors sampled the licensee’s performance indicator (PI) submittals for the periods listed below. The inspectors used PI definitions and guidance contained in Revision 4 of Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment PI Guideline,” to verify the accuracy of the PI data. The following PI was reviewed:

- RCS Specific Activity

  The inspectors reviewed Chemistry Department records and selected isotopic analyses from September 2005 through September 2006 to determine if the greatest Dose Equivalent Iodine (DEI) values obtained during those months corresponded with the values reported to the NRC. The inspectors also reviewed selected DEI calculations to verify that the appropriate conversion factors were used in the assessment. Additionally, the inspectors observed a chemistry technician obtain and analyze a reactor coolant sample for DEI to determine if there was adherence with licensee procedures for the collection and analysis of RCS samples.

This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensee's PI submittals for the periods listed below. The inspectors used PI definitions and guidance contained in Revision 4 of NEI 99-02, "Regulatory Assessment PI Guideline," to verify the accuracy of the PI data. The following PIs were reviewed:

- Occupational Exposure Control Effectiveness

  The inspectors reviewed data associated with the PI for occupational radiation safety, to determine if indicator-related data was adequately assessed and reported during the previous 4 quarters. The inspectors compared the licensee's PI data with the condition report database, reviewed radiological restricted area exit electronic dosimetry transaction records, and conducted walkdowns of accessible locked high radiation area entrances to determine if the controls in place for these areas were adequate. Data collection and analyses methods for PIs were discussed with licensee representatives to determine if there were any unaccounted for occurrences in the Occupational Radiation Safety PI as defined in Revision 4 of NEI 99-02.

  This review represented one sample.

- Radiological Environmental TS/Offsite Dose Calculation Manual (RETS/ODCM) Radiological Effluent Occurrences

  The inspectors reviewed data associated with the RETS/ODCM PI to determine if the indicator was accurately assessed and reported. This review included the licensee's condition report database to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors also reviewed selected gaseous and liquid effluent release data and the results of associated offsite dose calculations generated over the previous 4 quarters. Data collection and analyses methods for PIs were discussed with licensee representatives to determine if the process was implemented consistent with industry guidance in Revision 4 of NEI 99-02.

This review represented one sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.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 corrective action program at an appropriate threshold, whether adequate attention was being given to timely corrective actions, and whether adverse trends were identified and addressed.

This is not an inspection sample.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

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

This review represented one semi-annual trend review inspection sample.

b . Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) LER 05000440/2006-004-00 Oscillation Power Range Monitors (OPRMs) Inoperable

On September 14, 2006, the licensee was notified by General Electric through draft Safety Communication SC06-010 of a potential non-conservative setting of the OPRM enabled region drive flow setpoint that applied during single reactor recirculation loop operation. On September 21, 2006, licensee personnel conservatively determined that all four OPRM channels were inoperable due to this condition and entered TS Limiting Condition for Operation 3.3.1.3. Action B.1 of TS 3.3.1.3 required the use of an alternative method to detect and suppress thermal hydraulic instability oscillations. The licensee determined that the root cause for the problem was that the methodology approved by General Electric and the Boiling Water Reactor Owner's Group that was supplied in NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," dated August 1996, was flawed because it failed to address the maximum indicated drive flow while in single loop operations. This resulted in the potential to not meet the requirement for the OPRMs to be enabled during conditions of core flow of less than 60 percent and reactor power of greater than 23.8 percent. Licensee personnel also identified a missed previous opportunity to detect this error during a review of calculation FM-012, Revision 1, "OPRM Device Settings and Setpoints," that was conducted in accordance with NOP-CC-2001, Revision 4, "Design Verification," step 4.1.2.2. The OPRM system was designed to ensure compliance with General Design Criterion 10 and 12 by providing protection from exceeding the fuel minimum critical power ratio safety limit. The licensee concluded that, while operating with an OPRM enabled setpoint of 56.6 percent, average power range monitor drive flow had an insignificant impact on the core damage frequency or large early release frequency. Corrective actions included operator instructions to declare the OPRMs inoperable upon entry into single loop operation, revision of the setpoint to enable the OPRMs at 63.5 percent total drive flow, and a review of licensee's process for the selection of an appropriate independent verifier for important calculations. No new findings were identified following the inspectors' review. This issue was determined to be of only minor safety significance and was therefore not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee documented the issue in CR 06-00422.

This LER is closed.

This review represented one sample.

.2 <u>Manual Scram Due to Degrading Instrument Air Pressure</u>

On December 13, 2006, with the plant at 100 percent power, the inspectors observed licensee response to a manual reactor scram. The scram was initiated when operators observed lowering reactor water level and feedwater transients caused by a degrading instrument air pressure due to an air line break. The inspectors responded to the control room and observed the licensee's control of reactor vessel water level and reactor pressure. The inspectors reviewed licensee actions to reduce steam loads to maintain compliance with TS cooldown limits. The inspectors determined that the licensee completed notifications as required by 10 CFR 50.72. Finally, the inspectors observed the licensee's actions to restore plant systems affected by the loss of instrument air pressure. No findings of significance were identified.

This review represented one sample.

4OA5 Other Activities

.1 Temporary Instruction 2515/169 - MSPI Verification

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the MSPI guidance for reporting unavailability and unreliability of the monitored safety systems to determine whether it was correctly implemented. The inspectors reviewed licensee procedures listed for unavailability credit and compared these to the guidance contained in NEI 99-02, Revision 4. Using guidance contained in NEI 99–02, the inspectors determined baseline planned unavailability hours and compared these hours to determine whether the hours were correctly translated into the licensee's basis document, "Perry MSPI Basis Document," Revision 0. On a sampling basis for each MSPI system: (1) the inspectors reviewed operating logs, corrective maintenance records, and condition reports to determine whether the actual planned and unplanned unavailability data was accurate; and (2) the inspectors reviewed maintenance and test history to determine the accuracy of failure data for the identified monitored components.

b. Findings and Observations

(1) Basis Document Errors Identified During the Inspection

The inspectors began the inspection of the licensee's MSPI program on October 2, 2006. By October 4, the inspectors had identified numerous issues with the accuracy of the licensee's basis document.

The inspectors determined that licensee personnel had not accurately documented the planned baseline unavailability hours or the actual incurred unavailability hours for the MSPI systems. The inspectors determined that the licensee's translation of 2002-2004 Reactor Oversight Process (ROP) reported unavailability data was incorrect for the Emergency Alternating Current (AC) Power System, HPCS, RCIC, and RHR systems. For example, the licensee reported a total of 328.40 hours of HPCS system unavailability during ROP 2002-2004 submissions. However, In the MSPI basis document, the licensee inputted a significantly higher value of 1278.30 hours for the ROP 2002-2004 submissions. The inspectors identified similar data errors affecting the other systems that were reviewed. The inspectors also identified that the reactor critical hours data was incorrect and noted that this value affected the accuracy of the basis document analysis for all MSPI systems. The licensee entered these issues into their corrective action program as CR 06-7323, "Data Difference from MSPI and SSU for 2002-2004," dated October 4, 2006; and CR 06-7329, "Error In MSPI Basis Document - Reactor Critical Hours," dated October 4, 2006.

During the review of procedures listed in the MSPI basis document for operator action recovery credit, the inspectors identified that 11 of the 14 procedures associated with the HPCS and RHR systems did not satisfy the guidance in NEI 99-02 that was required to warrant this credit. For example, RHR procedure SVI-E12-T1193, "LPCI

Pump A Discharge Low Flow (Bypass) Channel Calibration for 1E12-N052A," Revision 4, was improperly credited since the procedure required a minimum of four dedicated operators (not including independent verifier personnel) stationed in different areas of the plant to implement over 40 procedure steps that included numerous valve manipulations, equipment configuration changes and electrical fuse installations. The inspectors noted that other credited procedures also required a minimum of three or four persons and required numerous steps for completion. The licensee entered this issue into their corrective action program as CR 06-7238, "MSPI Basis Document Information," dated October 4, 2006.

The licensee's initial investigation into the issues raised by the inspectors confirmed the errors. The licensee assembled a team of personnel and performed an assessment of the program. As a result of the assessment, the MSPI basis document was revised to correct the identified errors and to remove references to procedures that inappropriately credited operator action for system availability. In addition, as discussed below, licensee personnel discovered errors associated with the use of the correct revision of a calculation used to determine risk values that were input into the MSPI process.

For the samples selected, the inspectors did not identify any additional significant issues associated with (1) the documentation of actual unreliability information; (2) system boundaries; or (3) monitored components.

(2) Inaccurate Data Submitted for 2nd Quarter 2006 Emergency AC Power Systems MSPI

Based on the results of this inspection, materially inaccurate and incomplete information was identified in the MSPI data submitted to the NRC on July 21, 2006. The correction of this data resulted in a change in the MSPI color for Emergency AC Power Systems from Green to White for the 2nd quarter 2006.

On October 2, 2006, the inspectors began an inspection of the licensee's MSPI program. During the inspection, the inspectors identified errors in the MSPI basis document data used to calculate MSPI values, as discussed in the previous section of this report. In response to the issues identified, the licensee conducted a self-assessment of the MSPI program. Licensee personnel confirmed the errors in the basis document and additionally determined on October 13, 2006, that an outdated calculation had been incorrectly used to determine the MSPI data submitted for the 2nd and 3rd quarter 2006 Emergency AC Power Systems. Specifically, the licensee used an outdated revision of calculation DB-004 for initiating event frequencies in order to determine the PRA model, from which MSPI data was derived. The correct revision of DB-004 that was in effect prior to the 2nd quarter 2006, and that was recorded on the PRA model document in effect at that time, was not used. When the licensee applied the correct revision of DB-004, the MSPI color for the 2nd quarter 2006 Emergency AC Power Systems changed from Green to White.

On November 16, 2006, licensee personnel submitted corrected data for the 2nd and 3rd quarter 2006. This data reflected the change in the 2nd quarter Emergency AC Power System color from Green to White. The 3rd quarter 2006 MSPI color for Emergency AC Power Systems remained Green. The licensee also entered the issue into their corrective action program as CR 06-10069, "NRC PI for Emergency AC Crossed the Threshold from Green to White," dated November 16, 2006.

The inaccurate information provided by the licensee on July 21, 2006, affected the color of the index and therefore affected the timeliness of NRC response, such as additional inspection for a White input to a strategic performance area as described in the Action Matrix of NRC IMC 0305, "Operating Reactor Assessment Program," dated June 22, 2006.

As such, the NRC determined that one Severity Level IV violation of 10 CFR 50.9, "Completeness and Accuracy of Information," occurred. However, submission of the incomplete and inaccurate MSPI information was not willful, and the incomplete and inaccurate MSPI information was identified within a period of 1 year after the beginning of MSPI data collection, or by April 1, 2007. Therefore, in recognition of this, the ongoing PI development activities, the time constraints to gather and submit historical data, the large volume of data (12 quarters of data) needed to calculate and verify the MSPIs, and the time needed for licensees to familiarize and adjust to the new reporting guidance, the NRC is exercising enforcement discretion pursuant to Section VII.B.6, "Violations Involving Special Circumstances, of the Enforcement Policy," to refrain from issuing a Notice of Violation in this case (EA-06-326).

.2 <u>(Closed) Unresolved Item (URI) 05000440/2006-004-01: Failure to Promptly Correct the Degraded Condition of the Reactor Recirculation Pump Carbon Dioxide (CO_2) System</u>

a. <u>Inspection Scope</u>

The inspectors completed a review of URI 05000440/2006-004-01 associated with the licensee's failure to promptly correct the long-term recurring degraded condition of the reactor recirculation pump CO_2 system.

b . <u>Findings</u>

<u>Introduction</u>: The inspectors identified a finding of very low safety significance and an associated NCV of License Condition C(6) for the failure to promptly correct the long-term recurring condition of insufficient CO_2 tank level that was required to support the operability of the reactor recirculation pump CO_2 system.

<u>Description</u>: On August 1, 2006, the inspectors observed that the CO_2 suppression tank level for the reactor recirculation pump fire suppression system was 42 percent. This was below the minimum operability level limit of 45 percent. The inspectors also noted that the tank was leaking CO_2. After further review, the inspectors determined that the tank had been previously identified to be below the minimum operability level

in March 2006, October 2005, August 2005, June 2005, May 2005, April 2005, October 2003, September 2003, August 2002, and July 2001. In addition, during reviews of licensee condition reports, the inspectors determined that the inoperability of the CO_2 tank was unnecessarily extended due to the untimely arrival of CO_2 delivery trucks since 2005. The inspectors also determined that the CO_2 delivery and tank level restoration was also delayed following the identification of the low tank level on August 1, 2006. Licensee personnel performed Perry Technical Instruction (PTI)-P54-P0016, "Carbon Dioxide Storage Tank Pressure and Level Verification," Revision 4, on August 2, 2006, and recorded the tank level at 37 percent. On August 8, 2006, the inspectors observed that the CO_2 tank level had decreased to 10 percent due to the delay in CO_2 delivery.

Perry License Condition C(6) required that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the Perry Final Safety Analysis Report (FSAR). Section 9A.5, Position C.8 of the FSAR stated that measures had been established to ensure conditions adverse to fire protection, such as failures, malfunctions, deficiencies, deviations, defective components, uncontrolled combustible materials, and nonconformances are promptly identified, reported, and corrected. Attachment 3 of PAP-1910, "Fire Protection Program/Fire Protection Functional Specifications," Revision 12, stated that the minimum required level for the reactor recirculation pumps CO_2 system was 45 percent (2460 lbs).

As part of their immediate corrective actions, the licensee restored tank CO_2 level to restore system operability and performed non-complex vendor maintenance on the CO_2 tank that stopped the CO_2 leak and resolved the issue. The licensee entered the issue into their corrective action program as CR 06-03807, "CNRB: Reactor Recirculation Pump CO2 Fire Suppression Tank Leakage," dated August 22, 2006, and CR 07-12202, "Untimely Corrective Action to Repair Drywell CO2 Results In Green NCV," dated January 5, 2007.

Analysis: The inspectors determined that the failure to promptly correct the long-term recurring condition of insufficient CO_2 tank level for operability of the reactor recirculation pump CO_2 system was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on November 2, 2006, because the failure to ensure adequate CO_2 tank levels were maintained could have resulted in not obtaining a sufficient CO_2 concentration to extinguish a reactor recirculation pump fire. As such, this finding affected the Protection Against External Factors attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). In addition, the finding affected the corrective action attribute of the cross-cutting area of Problem Identification and Resolution because the licensee failed to take appropriate corrective actions to address the recurring condition of low tank level in a timely manner commensurate with its safety significance and relative ease of repair.

The inspectors reviewed IMC 0609, “Significance Determination Process,” Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” dated November 22, 2005, and determined that because the finding affected fire protection, a significance determination evaluation using IMC 0609, Appendix F, “Fire Protection Significance Determination Process,” issued February 28, 2005, was required. The inspectors identified at least one instance in which design CO_2 concentrations would not have been achieved. As such, the inspectors assumed that a high degradation existed in accordance with IMC 0609, Appendix F, Attachment 2, “Degradation Rating Guidance Specific to Various Fire Protection Program Elements.” Although the instances where the CO_2 tank was found to be below minimum acceptable tank levels were on the order of a week duration (due to the weekly surveillance frequency), the inspectors conservatively assumed a duration factor of one year to account for the multiple occurrences. For a fire ignition frequency, the inspectors assumed that reactor recirculation pumps had an ignition frequency comparable to reactor coolant pumps (as discussed in IMC 0609, Appendix F, Attachment 4, “Fire Ignition Source Mapping Information: Fire Frequency, Counting Instructions, Applicable Fire Severity Characteristics, and Applicable Manual Fire Suppression Curves”). Due to the location within the drywell, the inspectors did not credit manual suppression. In lieu of performing detailed fire scenarios for each of the pumps, the inspectors conservatively assumed that all equipment and instrumentation process lines within the same drywell quadrant as the affected reactor recirculation pump were adversely affected by a pump fire. The inspectors conservatively assumed that the safety relief valve function would be adversely affected by a fire in either pump. The inspectors noted that the instrument lines for part of the logic for the HPCS system could be adversely affected by a fire in either pump. However, for a single pump fire, a sufficient portion of the logic would be available to ensure functionality of the HPCS system. Due to the proximity to the “B” reactor recirculation pump, the inspectors conservatively assumed that the main steam isolation valves (and associated main steam lines) and the RCIC system could be adversely affected by a fire in the “B” reactor recirculation pump. All other functions were assumed to remain available in the event of a reactor recirculation pump fire. The inspectors evaluated the safety significance using the transient and stuck open relief valve reactor safety worksheets. The inspectors determined that there were a minimum of eight points and seven points of credit for the “A” and the “B” reactor recirculation pumps, respectively. Therefore, the inspectors determined that the finding was of very low safety significance.

Enforcement: License Condition C(6) required the licensee to implement and maintain in effect all provisions of the approved fire protection program as described in the FSAR. Section 9A.5, “Overall Requirement of Nuclear Plant Fire Protection Program”, Position C.8 of the FSAR stated that measures are established to ensure conditions adverse to fire protection, such as failures, malfunctions, deficiencies, deviations, defective components, uncontrolled combustible materials, and nonconformances are promptly identified, reported, and corrected. Revision 12 of PAP-1910, Attachment 3 “Fire Protection Program / Fire Protection Functional Specifications,” stated that the minimum required level for the reactor recirculation pumps CO_2 system was 45 percent (2460 lbs). Contrary to the above, on August 1, 2006, the inspectors identified that the reactor recirculation pumps CO_2 suppression tank level was less than 45 percent and subsequently identified that the licensee had failed to promptly correct a long-term recurring condition of inadequate CO_2 level for operability that existed since 2001. However, because this violation was of very low safety significance and because the condition of low CO_2 level was corrected and was entered into the licensee's corrective action program (CR 07-12202), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2006005-02)

.3 (Closed) URI 05000440/2006009-03: Inadequate Response for Minimum Pump Flow Settings.

a. Inspection Scope

During the Component Design Bases Inspection, the inspectors reviewed the licensee's response to Bulletin 88-04, "Potential Safety-Related Pump Loss," regarding the establishment of minimum flow requirements for safety-related pumps. The inspectors identified two concerns associated with the licensee's response to Bulletin 88-04 in 1988. The first concern was that licensee personnel failed to properly verify the minimum flow settings with the pump manufacturer. The second concern was that procedure revisions were not implemented for those systems that did not contain an adequate caution concerning minimum flow requirements as stated in the licensee's response to Bulletin 88-04. As a result of the licensee's response to Bulletin 88-04, the high pressure core spray (HPCS), low pressure core spray (LPCS), and residual heat removal (RHR) pumps were operated since original plant start-up with an increased potential for unusual wear and aging. Since the pump manufacturer had not completed their assessment of the adequacy of the minimum flow requirements for the RHR, LPCS, and HPCS pumps at the end of the inspection, this issue was identified as a URI.

During this inspection period, the inspectors reviewed the licensee's corrective actions associated with the minimum flow requirements for several safety-related pumps. Specifically, the inspectors reviewed the licensee's corrective action documents and information provided by the pump manufacturer.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance when licensee personnel failed to provide an adequate response to industry operating experience regarding the establishment of appropriate minimum flow settings for the HPCS, LPCS, and RHR pumps. Specifically, the licensee recognized that the conditions reported in Bulletin 88-04 were present in safety-related pumps, but did not determine appropriate minimum pump flow values to minimize and manage, or to eliminate, the potential for pump damage.

Description: NRC Bulletin 88-04,"Potential Safety-Related Pump Loss," identified a concern regarding the adequacy of minimum flow capacities for safety-related centrifugal pumps. The bulletin required licensees to evaluate the capability of safety-related pumps to run long-term at minimum recirculation flow rates. The bulletin stated that many licensees had accounted for thermal considerations in establishing minimum recirculation flow rates, but failed to consider flow instability effects. The latter consideration could necessitate a considerable increase in minimum flow settings, especially for pump operation for extended periods of time. This potential increase occurred because centrifugal pumps demonstrated a flow condition described as hydraulic instability or impeller recirculation at some flow point below approximately 50 percent of the best efficiency point on the characteristic pump curve. These unsteady flow phenomena would become progressively more pronounced if flow was further decreased, and could result in pump damage when operated for extended periods of time.

Unresolved Item 05000440/2006009-03 documented two concerns associated with the licensee's 1988 response to Bulletin 88-04:

- The licensee did not properly verify the minimum flow settings with the pump manufacturer as stated in their response to the bulletin. The licensee had concluded that the original, manufacturer-supplied minimum recirculation flows in the pump purchase specifications were adequate.

  The inspectors questioned whether the current minimum flow settings were reviewed and approved by the pumps' manufacturer (Byron-Jackson), as specified in the licensee's response to the bulletin. The licensee had not contacted the pump manufacturer, but relied upon information provided by General Electric to conclude that no changes were needed for pumps in these three systems. The licensee contacted the pumps' manufacturer (now Flowserve) to perform a new analysis of the HPCS, LPCS, and RHR pumps' minimum flow settings in response to Bulletin 88-04. This issue was entered into the licensee's corrective action program as CR 06-00813.

- In the licensee's response to the bulletin, it was stated "SOI/SVI procedure revisions will be provided for those systems which do not presently contain adequate caution. These cautions will limit pump minimum flow operation to a maximum of 30 minutes and assure that pump discharge is transferred to the full flow test line whenever possible." The bulletin response also stated that the review and approval of necessary procedure changes would be completed by October 5, 1988. When the inspectors reviewed the safety-related pump procedures, there was no evidence that any precautions related to minimum flow were ever implemented in the appropriate HPCS, LPCS, and RHR pump procedures. This issue was entered into the licensee's corrective action program as CR 06-00703 to determine if the 30 minute limitation was still necessary.

In response to these concerns, the inspectors prompted the licensee's Engineering Department to issue on March 2, 2006, a standing order to control room operators to be aware of the concerns associated with the operation of safety-related pumps for extended periods of time while on minimum flow. In addition, licensee personnel contacted the pump manufacturer to perform an analysis to determine the minimum required flows for the HPCS, LPCS, and RHR pumps. In several letters to the licensee, Flowserve provided the minimum flow rates for each pump based upon continuous operation, intermittent operation with a recommended total accumulation of not more than 1500 hours between overhauls, and short periods of operation with a recommended total accumulation of not more than 60 hours between overhauls. These values and the licensee's existing minimum flow rates are included in the following table:

Pump	Minimum flow less than 60 hrs	Minimum flow less than 1500 hrs	Minimum flow unlimited hours	Minimum flow existing
LPCS	1000 gpm	1800 gpm	2600 gpm	1240 gpm
RHR	1000 gpm	1775 gpm	2575 gpm	1230 gpm
HPCS	600 gpm	1500 gpm	2100 gpm	660 gpm

The inspectors concluded that the existing minimum flow rates were not sufficient for continuous pump operations and that other controls were therefore needed to ensure the pumps would not degrade if they were operated on minimum flow for extended periods of time. The existing flow rates were greater than the manufacturer's threshold for short pump duration, but less than the threshold for intermediate operation. Since the inspectors primary concern was associated with the HPCS pump, licensee personnel reviewed plant records and determined that the run time of the HPCS pump on minimum flow since plant startup was only about 35 hours. This time did not exceed the manufacturer's threshold for short term operation to perform a pump overhaul. As such, it did not appear that the pumps would have sustained significant degradation from running at the existing minimum flow rates since plant startup.

The license also identified in CR 06-00703 that corrective action #3, which prescribed that minimum flow operating limits established by the pump manufacturer were incorporated into procedures, were not implemented. This was recognized by the licensee and CR 06-11480 was issued to address the long-term degradation mechanism stated in the bulletin since the minimum flow rates established by the licensee did not meet the manufacturer's recommendation for continuous pump operation. In addition, CR 06-11108 was issued to track the licensee's supplemental response to the NRC to clarify the actions taken in response to the bulletin.

Analysis: The inspectors determined that the failure to adequately address a degradation mechanism identified in Bulletin 88-04, as required by the station's operating experience program, was a performance deficiency warranting a significance evaluation. The inspectors determined that the finding was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Dispositioning Screening," because the finding was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability and reliability of safety-related pumps. Specifically, the HPCS, LPCS, and

RHR pumps continued to be operated with insufficient minimum flow for unlimited operation to avoid unusual wear and aging without establishing increased monitoring and maintenance, or other compensatory actions.

The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it was not a design issue resulting in a loss of function per Part 9900, Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety;" did not represent an actual loss of a system's safety function; did not result in exceeding a TS allowed outage time; and did not affect external event mitigation. The basis for this conclusion was that despite the minimum flows established by the licensee being less than those specified by the manufacturer for unlimited operation, the HPCS pump run time was less than the manufacturer's threshold for short duration operation and the existing minimum flow rate exceeded the associated threshold value, such that the existing minimum flow rates were high enough to allow operation for a sufficiently long time period prior to required pump overhauls to inspect for degradation. The inspectors did not identify a cross-cutting aspect with this finding. As a result of this review, URI 05000440/2006009-03 was closed to a Green finding (FIN 05000440/2006005-03).

Enforcement: No violation of NRC requirements was identified.

.4 Closure of Notice of Violation and Civil Penalty Issued to FENOC

On February 24, 2005, the NRC issued a Notice of Violation (NOV) and proposed a Civil Penalty of $55,000 to FirstEnergy Nuclear Operating Company (FENOC) associated with a violation of 10 CFR 50.7, "Employee Protection," by its contractor Williams Power Corporation (EA-01-083). The violation occurred when the Williams Power Corporation Site Superintendent discriminated against painters employed by Williams Power for having engaged in protected activities. The licensee implemented corrective actions and the NRC evaluated the effectiveness of those corrective actions. No new findings or violations were identified. This violation is closed. (VIO 50-440/2005009-08)

4OA6 Meetings

.1 Interim Exit Meetings

Interim exit meetings were conducted for:

- Biennial Operator Requalification Program Inspection with Mr. T. Evans, Training Manager, on December 7, 2006;
- Biennial Operator Requalification Program Inspection with Mr. W. O'Malley, Licensed Operator Requalification Training Supervisor, on December 21, 2006, via telephone; and

- Emergency Preparedness inspection with Mr. J. Beavers on December 26, 2006.
- ALARA planning and controls program under the occupational radiation safety cornerstone, and PI verifications under the occupational and public radiation safety cornerstones and the barrier integrity cornerstone with Mr. L. Pearce on November 9, 2006.
- Closure of URI 05000440/2006009-03, “Inadequate Response for Minimum Pump Flow Settings,” with Mr. T. Hilston, Design Engineering Supervisor, and other members of the licensee’s staff on January 16, 2006.

.2 Exit Meeting

On January 5, 2007, the resident inspectors presented the inspection results to Mr. L. Pearce, Site Vice President, and other members of his staff who acknowledged the findings. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

None.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>
L. Pearce, Vice President-Nuclear
B. Allen, General Manager, Nuclear Power Plant Department
T. Evans, Training Manager
R. D. Gray, Maintenance Rule Program Engineer
G, Halnon, Director, Performance Improvement Initiative
H. Kelley, Emergency Preparedness Manager
T. Kledzik, Regulator Affairs Engineer
J. Lausberg, Manager, Regulatory Compliance
W. O'Malley, Licensed Operator Requalification Training Supervisor
M. Wayland, Director, Maintenance
J. Shaw, Director, Nuclear Engineering
S. Thomas, Manager, Radiation Protection
M. Wayland, Maintenance Director

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened and Closed</u>

05000440/2006005-01	NCV	Failure to Demonstrate Simulator Fidelity for Steady State Operations (Section 1R11)
05000440/2006005-02	NCV	Failure to Promptly Correct Degraded Condition of the Reactor Recirculation Pump CO2 system (Section 4OA5.2)
05000440/2006005-03	FIN	Minimum Pump Flow Settings Not Sufficient for Unlimited Operation (Section 4AO5.3)
2515/169	TI	MSPI Verification (Section 4OA5.1)

<u>Closed</u>

05000440/2006-004-00	LER	Oscillation Power Range Monitors (OPRMs) Inoperable (Section 4OA3.2)
05000440/2006-004-01	URI	Failure to Promptly Correct Degraded Condition of the Reactor Recirculation Pump CO2 system (Section 4OA5.2)
05000440/2006-009-03	URI	Inadequate Response for Minimum Pump Flow Settings (Section 4OA5.3)
05000440/2005-009-08	URI	Notice of Violation and Civil Penalty issued to FENOC (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather that selected sections of 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.

<u>Section 1R01 Adverse Weather</u>

NOP-WM-2001, "Work Management Scheduling/Assessment/Seasonal Readiness Processes," Revision 5
Winter Preparations Status Sheet, dated November 13, 2006
CR 06-09849; SVI-M26-T1264-A Not Scheduled for Weather Considerations; dated November 13, 2006

<u>Section 1R04 Equipment Alignment</u>

VLI-R44; Division 1 and 2 DG Starting Air System (Unit 1); Revision 4
CR 06-07087; Conflicting Torque Values; dated October 2, 2006
CR 06-03200; Division 2 DG Flywheel Cover Fasteners Missing; dated July 19, 2006
CR 06-07860; NRC Questions on Division 1 Diesel; dated October 11, 2006
CR 06-07647; Division 2 DG Fuel Oil Transfer Pump #1 Failed; dated October 10, 2006
VLI-P45; Emergency Service Water System; Revision 7
CR 06-09645; Emergency Service Water Pumphouse Issues Identified; dated November 8, 2006
VLI-E21; Low Pressure Core Spray System; Revision 7
CR 06-10645; LPCS Minimum Flow Valve; dated November 28, 2006

<u>Section 1R05 Fire Protection</u>

FPI-1DG; DG Building; Revision 5
FPI-0IB; Intermediate Building; Revision 4
FPI-1AB; Auxiliary Building; Revision 2
FPI-0CC; Control Complex; Revision 6
FPI-0FH; Fuel Handling; Revision 3
FPI-XFMR; Transformer Yard Areas; Revision 2
PAP-1910, Fire Protection Program; Revision 12

<u>Section 1R07 Heat Sink</u>

R46-023; Division 1 EDG Jacket Water Heat Exchanger Performance Test Evaluation; Revision 0; dated September 10, 2003
R46-024; Division 2 EDG Jacket Water Heat Exchanger Performance Test Evaluation; Revision 0; dated September 18, 2003
PTI-R46-P0001-A; EDG Heat Exchanger Performance Testing Trend Chart; Revision 1; dated September 29, 1994

Section 1R11 Licensed Operator Requalification

Simulator Scenario for the Week of October 16, 2006
ANSI/ANS-3.4-1983; Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants; dated April 29, 1983
ANSI/ANS-3.5-1998; Nuclear Power Plant Simulators for Use in Operator Training; dated April 15, 1998
CR 06-02708; Operations Training Self-Assessment Identified Area for Improvement; dated June 15, 2006
CR 06-02821; Simulator Test Results Do Not Meet ANSI Acceptance Criteria; dated June 5, 2006
CR 06-03412; Simulator Does Not Respond Like Plant; dated July 30, 2006
CR 06-03414; Simulator Deficiency; dated July 30, 2006
CR 06-03842; Periodic Log in the Simulator Is Not The Same as the Periodic Log in The Plant; dated August 23, 2006
CR 06-11060; NRC 71111.11 Inspection Comment on License Reactivation; dated December 7, 2006
CR 06-11107; 71111.11B Debrief Regarding Simulator Testing Documentation; dated December 7, 2006
Feedback and Attendance Sheets; Continuing Operator Training; Cycle 2006-07; dated March - April 2006
License Event Reports 2004-2006; various dates
License Operator Requalification Exam; Sample Plan 2005 - 2006, Cycles 1 -11
Licensed Operator Requalification Training Curriculum Review Committee Meeting Minutes; dated various 2005 - 2006
Master License Operator Requalification Schedule (2005-2006); dated April 2, 2006
Perry Plant NRC Integrated Inspection Reports; dated variously from January 2004 through October, 2006
Perry Reactor Oversight Process Plant Issue Matrix from January 1, 2004, to October 31, 2006
PYBP-POS-1-5; Operations Training Guidelines; Revision 2; dated January 16, 2006
PYBP-PTS-0005; Operator Continuing Training Program Administration; Revision 12
PYBP-PTS-0007; Simulator Scenario Guide Preparation, Review, Revision, and Approval; Revision 0
PYBP-PTS-0015; JPM Preparation, Review, Revision, Approval and Administration; Revision 0
PYBP-PTS-0033; Simulator Configuration Control; Revision 0; dated October 25, 2006
Regulatory Guide 1.134; Medical Evaluation of Licensed Personnel for Nuclear Power Plants; Revision 2; dated April 1987
Regulatory Guide 1.149; Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations; Revision 3; dated October 2001
Self-Assessment Report SA# 750 PYTM 2005; dated June 13 - 17, 2005
Self-Assessment Report SA# 826 PYTM 2006; dated June 5 - 9, 2006
Self-Assessment Report SA# 850 PYTM 2006; dated October 16 - 20, 2006
Simulator Core Performance Tests; various dates
Simulator Minor Work Item Summary; Open Items; dated December 2006
Simulator Real Time Test; ANSI Section 4.1.1 Test; dated June 2, 2006
Simulator Review Board Meeting Minutes; dated various 2006

Simulator/Scenario Performance Validation (various); dated September 2003 - August 2006
Simulator Steady State Tests; dated various
Simulator Testing; Heat Balance - 45 percent; 80 percent; 100 percent (Year 2003; 2004; 2005; 2006)
Simulator Transient Tests; dated various
SWO Summary; Items Closed After December 2004 to December 2006; dated December 2006
SWO Summary; Open Items; dated December 2006
SWO# 04-0028; Corrective Action for CR 03-06127, New Suppression Pool Model; dated July 2, 2004
SWO# 05-0021; Inability to Test the Mechanical Overspeed Trip; dated June 1, 2005
SWO# 05-0026; Reactor Feed Pump Turbines Don’t Windmill in the Same Way as the Plant; dated July 1, 2005
SWO# 05-0104; LPRM Lights Bypass, Downscale and Upscale on APRMs D and H; dated December 16, 2005
SWO# 06-0018; Fuel Element Failure Malfunction Does Not Give Expected Results (TH15); dated July 21, 2006
SWO# 06-0020; Update (Simulator) BOP Model - Generator Megawatts Electric Discrepancy for 80 Percent and 45 Percent Power Heat Balance; dated August 17, 2006
Simulator Work Request Summary; Open Items; dated December 1, 2006
Six Licensed Operator Medical Records; dated various
Snapshot Assessment Topic: Review of Operator Training Material Open Action Items January 1 - November 6, 2006 # 919 PYTM 2006; dated November 29, 2006
SRO and RO License Operator Requalification 2006 Biennial Written Examinations; Weeks 1 -7; dated various
SRO and RO License Operator Requalification 2006 Annual Operating Test JPMs; Weeks 1 - 7; dated variously
SRO and RO License Operator Requalification 2006 Annual Operating Test Scenarios; Weeks 1 - 7; dated various
TMA-4206; Operator Requalification Programs; Revision 8
Training PIs - 2006; dated various

Section 1R12 Maintenance Effectiveness

CR 06-01629; Unexpected Annulus Differential Low Alarm Causes Entry into PEI-N11; dated April 10, 2006
CR 06-03880; Unexpected Low Flow Alarm on 1M15C0001A During Normal Operations; dated August 28, 2006
CR 06-06226; Unable to Complete Shift of AEGTS [Annulus Exhaust Gas Treatment System] Fan Due to Low Flow; dated September 12, 2006
CR 06-06379; 1M15C0001A Low Flow Alarm Being Received with Fan Normal Operation; dated September 15, 2006
CR 06-06826; Annulus Exhaust Gas Treatment System Flow Indication/Alarm OOS [Out of Service]; dated September 26, 2006
CR 06-10106; AEGTS Fan B Auto Started During Shift After Being Secured; dated November 16, 2006
CR 06-10465; Annulus Exhaust Gas Treat Unresolved Long Standing Equipment Reliability Issues; dated November 22, 2006

CR 06-10588; Annulus Exhaust Gas Treatment System High Flow Discovered During PMT; dated November 22, 2006

Section 1R13 Maintenance Risk Assessments and Emergent Work Control

Perry Work Implementation Schedule; Week 2, Period 7
Maintenance Risk Evaluation; Week 2, Period 7; Revision 1
Division 1 Outage Plan; Week 2, Period 7; dated October 5, 2006
Perry Work Implementation Schedule; Week 6, Period 7
Maintenance Risk Evaluation; Week 6, Period 7; Revision 2
Perry Work Implementation Schedule; Week 7, Period 7
PYBP-POS-2-2; Division 2 Outage (Yellow) Protected Equipment Posting Checklist; Revision dated June 19, 2006
SVI-R10-T5227; Off-Site Power Availability Verification; November 7, 2006; Revision 2
CR 06-09552; SVI R10T5227 Required Alternate Instrumentation Due to Division 2 Outage Clearance; dated November 7, 2006

Section 1R15 Operability Evaluations

CR 06-08404; Suspect Oil in 1E22C003 Oil Bubbler; dated October 19, 2006
Pump, Bearing and Oiler Facts; dated October 20, 2006
CR 06-08765; Apparent Non-compliance with UFSAR Licensing Basis; dated October 25, 2006
CR 06-09781; Division 2 DG Head Nut Torques; dated November 10, 2006
CR 06-09890; Division 2 DG Head Bolts Found Under Torqued; dated November 13, 2006
Control Room Operator Logs; dated on November 10, 13, and 14, 2006
Replacement and Inspection of Division 1 and 2 EDG Liners; dated November 14, 2006
CR 06-10596; Corroded ESW Sluice Gate; dated November 27, 2006
Prompt Operability Determination Form for CR 06-10596; dated November 28, 2006

Section 1R17 Permanent Plant Modifications

ECP 04-0270-01, Alternate Decay Heat Removal System Design and Installation; dated March 25, 2006
ECR 04-0270, Conceptual Design Report for the Alternate Decay Heat Removal System; dated January 26, 2005
ECR 04-0270, Initiation Report; dated October 22, 2004

Section 1R19 Post-Maintenance Testing

WO 200105076; Replace Suppression Pool Level Transmitter 1G43N0070A; dated October 17, 2006
SVI-G43-T1305-C; Suppression Pool Water Level Channel Calibration for 1G43-N070A; Revision 4
CR 06-8702; Additional Fuse Pulled During Clearance Hang; dated October 24, 2006
WO 200143828; Standby Liquid Control System A; dated November 30, 2006
WO 200214205; Standby Liquid Control System A; dated November 30, 2006

SVI-C41-T2001A; Standby Liquid Control A Pump and Valve Operability Test; Revision 10
FTI-F-0036; Post Maintenance Test Manual; Revision 4

Section 1R22 Surveillance Testing

WO 200217204; DG Start and Load Division 3; dated October 17, 2006
WO 200114671; Main Steam Line Low Condenser Vacuum Channel C Calibration; dated October 18, 2006
CR 04-02738; NRC 95002 Inspection Team - Maximum EH Bus Voltage; dated May 26, 2004
CR 04-03032; Voltage Droop Verses Isochronous LOOP [Loss of Offsite Power] Scenario Question For Division 1 and 2 DG; dated June 9, 2004
CR 04-03263; DG Response to a LOCA Signal While in Test Mode; June 22, 2004
CR 04-03637; Unable to Complete SVI-R43T1317 Due to EH11 Bus Voltage; July 14, 2004
WO 200192055; MSL Tunnel Temperature High Channel C Functional for 1E31-N604C; dated November 7, 2006
SVI-E31-T0078-C; MSL Tunnel Temperature High Channel C Functional for 1E31-N604C and 1E31-N605C; Revision 6
WO 920000642; Standby DGs Initial Calibration of Relays; dated April 2, 1992
WO 920000735; Standby DGs Division 2 Wiring to Support Division 1 Wiring; dated May 7, 1992
WO 200030466; Relay 81 Division #2 D/G Load Test Overload; dated September 17, 2003
WO 910001149; Standby DGs ISO Division 1 From Offsite Power/DCP 87-785; dated April 2, 1992
CR 06-09670; Division ½ DG LOOP Response While in Parallel Operations with the Grid; dated November 9, 2006
WO 200151755; Standby Liquid Control A Pump and Valve Operability Test; dated November 28, 2006
CR 06-10404; SLC [Standby Liquid Control] Vibration Data Not Taken With Correct Equipment During Last Surveillance; dated November 21, 2006
WO 200242776; RHR [Residual heat Removal] C Pump and Valve Operability Test; dated December 4, 2006

Section 1R23 Temporary Plant Modifications

TM06-0013; Chemical Decontamination Equipment into the Auxiliary Building

Section 1EP4 Emergency Action Level and Emergency Plan Changes

Perry Nuclear Power Plant Emergency Plan; Revisions 24, 25, and 26

Section 2OS ALARA Planning and Controls

RWP 060355; OG [Offgas] System Repairs Include Valve Disassembly, Boroscopic Inspection, Replace Valves, Grinding and Welding of Associated Drain Piping - High Radiological Risk; Revision 4
RWP 065404; High Radiation Forced Outage Drywell at Power/Repair 833 Leak; Revision 5
RWP 060310; High Radiation Examination of LP [Low Pressure] Condenser “D” Waterboxes, Tube Testing, Tube Leak Repair and Support Work, High Radiological Risk; Revision 4

RWP 060317; High Radiation Condensate Demineralizer Septa Filter Changeout - High Radiological Risk; Revision 3
RWP 060333; High Radiation - FPCC [Fuel Pool Cooling and Cleanup] Chemical Decontamination; Revision 0
AAP-06-001; Chemical Decontamination Filter Transfer Plan; dated October 31, 2006
CR 06-03482; Contamination Events to Fuel Pool Cooling and Cleanup Demineralizer Breach; dated August 2, 2006
CR 06-03094; Contractor Person Continue Working After Rate Alarm; dated July 12, 2006
CR 06-03584; Plant Engineering Section Has Exceed Annual Dose Budget; dated August 8, 2006
CR 06-03463; Locked High Radiation Area Barricade is Not Well Defined; dated August 1, 2006
CR 06-03120; Radiological Control Peer Checking of Turbine Project; dated July 13, 2006
CR 06-03375; Condensate Demineralizer Septa Change-out Robot Difficulties; dated July 28, 2006
CR 06-03787; Discrete Particle Found on PCM2 Foot Detector; dated August 20, 2006
CR 06-06130; Ion Chamber Failure; dated September 6, 2006
CR 06-06410; Feedwater Iron Trend Outside Optimum Range for Dose Reduction; dated September 15, 2006
CR 06-02894; Missed Dose Saving Opportunity; dated June 28, 2006
Source Reduction Charter 2005-2009; Revision 1
NOP-WM-7001; ALARA Program; Revision 0
NOP-WM-7002; Operational ALARA Program; Revision 1
FS-SA-06; FENOC ALARA Snapshot Assessment Plan; dated July 10, 2006
Preliminary BRAC Survey Results; EOC 9 and 10
PY-C-05-03; Perry Nuclear Oversight Assessment; dated November 23, 2005

71151 PI Verification

Dose Equivalent Iodine Data dated September 2005 through September 2006
Access Control Records dated November 2005 through October 2006
M-35 Drains, Monthly Composite; dated October 10, 2006
Gaseous Effluent Dose Data; dated October 20, 2006
P06-156-L; Liquid Release; dated October 18, 2006
P06-160-L; Liquid Release; dated October 22, 2006

Section 4OA2 Identification and Resolution of Problems

CR 06-09517; Trending of Prompt Alert Siren Failures - L37; dated November 6, 2006
CR 06-07097; Trend In Human Performance Errors; dated October 2, 2006
CR 06-07082; PY-PA-06-03; Negative Trend - Lapse of Emergency Response Qualifications; dated September 29, 2006
CR 06-06106; Suppression Pool Level Trending Lower; dated September 8, 2006
Perry Plant Health Report 2006-3; dated November 30, 2006
CR 06-10514; Oil Leaking from 1R43N0711B After Recent Replacement; dated November 24, 2006
CR 06-10418; Unsatisfactory Pump Rework; dated November 21, 2006

Section 4OA3 Event Followup and Notices of Enforcement Discretion

LER 05000440/2006-004, “Oscillation Power Range Monitors (OPRMs) Inoperable”, dated November 13, 2006
FM-012, “OPRM Device Settings and Setpoints”, Revision 1
CR 06-11338; Post Reactor Scram Evaluation, Manual Scram, Degrading Instrument Air Pressure; dated December 13, 2006
CR 06-11339; Manual Scram - Instrument Air Joint Failure Leading to Loss of Instrument Air; dated December 13, 2006

Section 4OA5 Other Activities

CR 06-07329; Error in MSPI Basis Document - Reactor Critical Hours; dated October 4, 2006
CR 06-07238; Mitigating Systems Performance Index Basis Document Information; dated October 4, 2006
CR 06-07323; Data Difference from MSPI and SSU [Safety System Unavailability] for 2002-2004; dated October 4, 2006
SVI-E12-T1193; LPCI Pump A Discharge Low Flow (Bypass) Channel Calibration for 1E12-N052A; Revision 4
SVI-E22-T1199; HPCS Pump Discharge Pressure - High Channel Calibration for 1E22-N051; Revision 8
PTI-E12-P0002; RHR Heat Exchangers A and C Performance Testing; Revision 6
SVI-E22-T2001; HPCS Pump and Valve Operability Test; Revision 20
SVI-E12-T0146; ECCS [Emergency Core Cooling System]/LPCI Pump A Start Time Delay Relay Channel Functional/Calibration for 1E12A-K70A; Revision 4
CR 06-06983; MSPI Data Entry Changes; dated September 29, 2006
CR 06-07480; NRC PI Data May Be Incorrect; dated October 7, 2006
CR 06-07630; MSPI Inspection Preparation Was Less Than Adequate; dated October 10, 2006
913 PYDM; MSPI Data Submittal Process; dated October 12, 2006
CR 06-08969; Adequacy of MSPI Implementation; dated October 27, 2006
FENOC Memo; Mid-quarter Correction, Submittal of NRC PI Data; dated November 16, 2006
PRA-MSPI-001; MSPI Basis Document (Revision 3) PRA Input; dated November 13, 2006
Mitigating Systems Performance Index Perry Nuclear Power Plant Basis Document; Revision 0
Mitigating Systems Performance Index Perry Nuclear Power Plant Basis Document; Revision 2
Mitigating Systems Performance Index Perry Nuclear Power Plant Basis Document; Revision 3
CR 06-10069; NRC PI For Emergency AC Crossed the Threshold from Green to White; dated November 16, 2006
Consolidated Data Entry 3.0 MSPI Derivation Report; dated December 12, 2006
CR 07-12202; Untimely Corrective Action To Repair Drywell CO2 Results In Green NCV; dated January 5, 2007
CR 06-03807; CNRB: Reactor Recirculation Pump CO2 Fire Suppression Tank Leakage; dated August 22, 2006
CR 06-00703; Commitment in IEB 88-04 Response Not Implemented; dated February 13, 2006

CR 06-00813; Question on Basis of Min Flow Adequacy; dated February 16, 2006
CR 06-11108; Discrepancy Identified in Response to NRC Bulletin 88-04; dated December 7, 2006
CR 06-11480; Low Flow Operation of ECCS Pumps; dated December 5, 2006
Flowserve Letter to First Energy; Minimum Flow Analysis Residual Heat Removal (RHR) Pumps; dated May 3, 2006
Flowserve Letter to First Energy; Minimum Flow Analysis High Pressure Core Spray (HPCS) Pumps; dated April 19, 2006
Flowserve Letter to First Energy; Minimum Flow Analysis Low Pressure Core Spray (LPCS) Pumps; dated April 26, 2006

LIST OF ACRONYMS USED

AC	Alternating Current
ALARA	As-Low-As-Is-Reasonably-Achievable
ANSI/ANS	American National Standard Institute/American Nuclear Society
CFR	Code of Federal Regulations
CR	Condition Report
DEI	Dose Equivalent Iodine
DG	Diesel Generator
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FIN	Finding
FPI	Fire Protection Instruction
FSAR	Final Safety Analysis Report
gpm	Gallons Per Minute
HPCS	High Pressure Core Spray
IMC	Inspection Manual Chapter
JPM	Job Performance Measure
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LORT	License Operator Requalification Training
LPCS	Low Pressure Core Spray
MSL	Main Steam Line
MSPI	Mitigating Systems Performance Index
MWe	Megawatts Electric
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NOP	Normal Operating Procedure
NRC	Nuclear Regulatory Commission
OPRM	Oscillation Power Range Monitor
PAP	Perry Administrative Procedure
PI	Performance Indicator
PMT	Post-Maintenance Testing
PRA	Probabilistic Risk Assessment
PTI	Perry Technical Instruction
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RETS/ODCM	Radiological Environmental Technical Specifications/Offsite Dose Calculation Manual
RHR	Residual Heat Removal
ROP	Reactor Oversight Process
SAT	Systems Approach to Training
SRO	Senior Reactor Operator
SSC	Structures, Systems, and Components
SWO	Simulator Work Order
SWR	Simulator Work Request
SVI	Surveillance Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report

URI	Unresolved Item
USAR	Updated Safety Analysis Report
VLI	Valve Lineup Instruction
WO	Work Order