April 16, 2004

Mr. William R. Kanda
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P. O. Box 97, A210
10 Center Road
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT
NRC INTEGRATED INSPECTION REPORT 05000440/2004002

Dear Mr. Kanda:

On March 31, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on April 8, 2004, with you and other members of your staff.

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

Based on the results of this inspection, the inspectors identified two findings of very low safety significance (Green). One of the findings was determined to involve a violation of NRC requirements. However, because of its very low safety significance, and because it had been entered into your corrective action program, the NRC is treating the finding as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of this Non-Cited Violation, 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 the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at **http://www.nrc.gov/reading-rm/adams.html** (the Public Electronic Reading Room).

Sincerely,

/RA/

Mark A. Ring, Chief
Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report 05000440/2004002
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	05000440/2004002
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	P.O. Box 97 A210 Perry, OH 44081
Dates:	January 1 through March 31, 2004
Inspectors:	R. Powell, Senior Resident Inspector J. Ellegood, Resident Inspector T. Ploski, Senior Emergency Preparedness Inspector
Approved by:	M. A. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000440/2004002; 01/01 - 03/31/04; Perry Nuclear Power Plant; Equipment Availability and Functional Capability; Identification and Resolution of Problems.

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

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was self-revealed when the normal power supply to reactor protection system (RPS) bus 'B' was lost on November 29, 2003. A comprehensive investigation by the licensee determined that an age-related failure of a contactor in the circuitry resulted in a blown fuse which de-energized RPS bus 'B.' The licensee's investigation also identified that General Electric (GE) Service Information Letter (SIL) 508 issued in 1990, if properly implemented, would have prevented the event. The licensee's immediate actions included restoration of RPS bus 'B' by transfer to the alternate power supply. The failed contactor was replaced. The primary cause of this finding was related to the cross-cutting area of Human Performance because the licensee's review of GE SIL 508 failed to identify all affected plant components.

  This finding was more than minor because it was associated with reactor safety/initiating event cornerstone attribute of equipment performance and affected the cornerstone objective of limiting the likelihood of events that upset plant stability. The finding was of very low safety significance because mitigating system availability was unaffected. The affected contactors were not safety-related components. Therefore, no violation of regulatory requirements occurred. (Section 1R/EP.4)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors for a violation of Technical Specification 5.4, "Procedures." A licensee procedure required that unless risk-assessed, no items shall be left unattended below the 623' 4" level in containment at any time. On February 5, 2004, the inspectors observed a large sheet of permalon by the 'A' hydraulic power unit with no workers in the area. The licensee removed the material later that same day. The primary cause of this finding was related to the cross-cutting area of Human Performance because plant personnel failed to

follow licensee procedures and left material unattended in the swell region of containment.

This finding was more than minor because the inspectors concluded that it could reasonably be viewed as a precursor to a more significant event. Specifically, leaving unattended items in containment can lead to the items falling into the suppression pool without being noticed or being transported into the pool during an actual event. This material can then clog suppression pool strainers thereby reducing emergency core cooling system flow. Since no material fell into the suppression pool and no actual loss of safety function occurred, the inspectors determined the finding to be of very low safety significance. This issue was a Non-Cited Violation of Technical Specification 5.4 which required implementation of procedures for performing maintenance that can affect the performance of safety-related equipment. (Section 4OA2.2)

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The unit began the inspection period by reducing power on January 2 to 60 percent for power suppression testing. The unit returned to maximum achievable power, approximately 97 percent, on January 4 and subsequently reduced power to 65 percent on January 5 for a load line adjustment and scram time testing. The unit returned to 100 percent power on January 7. The unit remained at or near 100 percent, until March 6 when power was reduced to 75 percent for a rod pattern sequence exchange and insertion of an additional control rod for fuel defect management. The unit returned to maximum achievable power, approximately 96 percent, on March 7 and subsequently reduced power to 69 percent on March 9 for a load line adjustment and scram time testing. The unit returned to 100 percent power on March 10 and remained at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R/EP Equipment Availability and Functional Capability (71111.EP)

.1 Operability Evaluations (OEs)

a. Inspection Scope

The inspectors selected condition reports (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 (TSs) and Updated Safety Analysis Report (USAR) to the licensee's evaluations, to verify that the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors verified that the measures were in place, would work as intended, and were properly controlled. Additionally, the inspectors verified, where appropriate, compliance with bounding limitations associated with the evaluations. The inspectors reviewed:

- an OE associated with incomplete seismic analysis of the testable rupture disks for the Division 1 and 2 emergency diesel generators (EDGs) completed on December 19;
- an OE associated with a broken body to bonnet stud on a reactor sample isolation valve completed on December 15;
- an OE associated with an inadequate seismic qualification of EDG testable rupture disks on January 27;
- an immediate investigation of steam leaks on the reactor core isolation cooling (RCIC) turbine and turbine control valve completed on February 1; and
- an OE on containment penetrations associated with the fire protection water supply to containment completed on February 8.

b. Findings

No findings of significance were identified.

.2 Operator Workarounds (OWAs)

a. Inspection Scope

Semi-Annual Cumulative Review

During the week of March 8, the inspectors evaluated the overall effectiveness of the licensee OWA program. The inspectors reviewed the licensee's CRs in order to determine if issues identified in these documents were also reviewed as part of and captured in the licensee's OWA program. Additionally, the inspectors evaluated the interaction between the OWAs to determine if cumulative effects existed. The inspectors reviewed the licensee's plans to eliminate the need for the identified OWAs. Finally, the inspectors accompanied licensed and non-licensed operators during the performance of their rounds. The inspectors observed all log readings and equipment manipulations made by the operators. The inspectors discussed the effect of active OWAs and other equipment deficiencies with the operators.

Fuel Defects

The inspectors reviewed the OWAs associated with identified fuel defects. During the week of February 2, the inspectors reviewed operational changes to minimize degradation of the leaking fuel pins and methods established to detect further degradation of the fuel pins. As a result of identified fuel defects, the licensee has established limits on power ascension as well as power generation in the affected fuel bundles.

Leakage into the RCIC Room

The inspectors reviewed OWAs associated with leakage into the RCIC pump room from the reactor water cleanup (RWCU) pump and valve room. Leaks in the RWCU room can enter the RCIC pump room from a grating in the RCIC pump room overhead. In order to prevent water from leaking onto safety-related motor-operated valves and other equipment, the licensee installed catch basins to collect the water and direct it to the RCIC room sump. The licensee increased the frequency of sump level inspections as well as sump pump operations to account for the increased leakage into the sump.

b. Findings

No findings of significance were identified.

.3 Temporary Plant Modifications

a. Inspection Scope

During the week of February 23, the inspectors reviewed the temporary modification package associated with Temporary Modification 1-03-001, “Temporary Mod for FME Protection of Div. 3 Diesel Exhaust Rupture Disk Opening” to verify that the modification was properly installed, had no effect on the operability of the safety-related equipment, and met design basis requirements. The inspectors assessed the acceptability of the temporary modification to the facility by comparing the 10 CFR 50.59 screening evaluation and supporting operating procedures to the design basis documents and plant drawings. The inspectors also checked temporary modification tags and walked down the system to ensure the temporary modification did not impact the operability of interfacing systems.

b. Findings

No findings of significance were identified.

.4 Maintenance Effectiveness

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule requirements to verify that 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, selected surveillance test procedures, and a sample of CRs to verify that 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 verify that the criteria adequately monitored equipment performance and to verify that licensee changes to performance criteria were reflected in the licensee’s probabilistic risk assessment. During this inspection period, the inspectors reviewed the following areas:

- nuclear boiler system;
- reactor protection system; and,
- containment and suppression pool.

The problem identification and resolution CRs reviewed are listed in the attached List of Documents Reviewed.

b. Findings

Introduction: A finding of very low safety significance was self-revealed when the normal power supply to reactor protection system (RPS) bus ‘B’ was lost on November 29, 2003. A comprehensive investigation by the licensee determined that an

age-related failure of a contactor in the circuitry resulted in a blown fuse which de-energized RPS bus 'B.' The licensee's investigation also identified that a service information letter (SIL) issued in 1990, if properly implemented, would have prevented the event.

Description: In 1990, the licensee received General Electric (GE) SIL 508 which contained recommendations on scram contactor coil replacement. Although the SIL specifically referenced scram contactors, the contactor models identified served other functions in the RPS and were in use in numerous other Perry plant systems. The licensee reviewed the SIL and took action to inspect and replace the scram contactors consistent with the recommendations.

On November 29, 2003, the loss of the normal power supply to the RPS 'B' bus resulted in the de-energization of the RPS 'B' bus loads. The bus de-energization caused a half-scram and multiple system isolations. The licensee, per procedure, transferred the bus to an alternate power supply and restored the affected systems. Subsequent investigation by the licensee determined that an age-related failure of a contactor in the circuitry resulted in a blown fuse which de-energized the RPS 'B' bus. The failed contactor was replaced. The licensee determined the root cause of the event to be the failure to replace the contactor within the time frame recommended by the component vendor. The root cause also identified that the licensee's review and implementation of the GE SIL failed to include an adequate scope.

Analysis: The inspectors determined that the licensee's failure to implement the recommendations of GE SIL 508 was a performance deficiency warranting a significance evaluation. The inspectors determined that the finding was more than minor because it: (1) was associated with reactor safety/initiating event cornerstone attribute of equipment performance; and (2) affected the cornerstone objective of limiting the likelihood of events that upset plant stability. This finding also affected the cross-cutting area of Human Performance because the licensee's review of GE SIL 508 failed to identify all affected plant components.

The inspectors completed a significance determination of this issue using Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors answered "no" to all questions in the initiating event column of the Phase 1 Screening Worksheet because no safety-related mitigation systems were affected by the issue. The inspectors therefore concluded that the issue was of very low safety significance.

Enforcement: The affected contactors were not safety-related components. As such, the requirements of 10 CFR Part 50, Appendix B did not apply. Therefore, no violation of regulatory requirements occurred. This issue was considered a finding of very low safety significance **(FIN-05000440/2004002-01)**. The licensee entered the event into its corrective action program as CR 03-06398.

1R04 Equipment Alignment (71111.04)

.1 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete walkdown of accessible portions of the control rod drive hydraulic system (CRDHS) to verify system operability during the week ending January 10. The inspectors conducted a representative sampling of individual accumulators. The CRDHS was selected due to its risk significance and current system health status. The inspectors used CRDHS valve lineup instructions (VLIs) and system drawings to accomplish the inspection.

The inspectors observed selected switch and valve positions, electrical power availability, system pressure and temperature indications, component labeling, and general material condition. The inspectors also reviewed open system engineering issues as identified in the licensee's quarterly system health reports, outstanding maintenance work requests, and a sampling of licensee CRs to verify that problems and issues were identified, and corrected, at an appropriate threshold. Finally, the inspectors reviewed recently completed surveillance procedures to assess equipment performance demonstrations. The documents used for the walkdown and issue review are listed in the attached List of Documents Reviewed.

b. Findings

No findings of significance were identified.

.2 Partial System Walkdowns

a. Inspection Scope

The inspectors conducted partial walkdowns of the system trains listed below to verify that the systems were correctly aligned to perform their designed safety function. The inspectors used licensee 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 three systems:

- the Division 3 emergency diesel generator (EDG) and associated support systems on January 26 during a planned reactor core isolation cooling (RCIC) system maintenance outage;
- the high pressure core spray system on January 28 during a planned RCIC system maintenance outage; and,
- the residual heat removal (RHR) system 'B' train on March 24 while the RHR 'A' containment spray function was inoperable for planned maintenance.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

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

- Fire Area 1CC-3a, Division 2 Switchgear Room;
- Fire Area 1CC-3b, Division 3 Switchgear Room;
- Fire Area 1CC-3c, Division 1 Switchgear Room;
- Fire Area 1CC-4a, Unit 1, Division 2 Cable Spreading Area;
- Fire Area 1CC-4e, Unit 1, Division 1 Cable Spreading Area;
- Fire Zone 1DG-1B, Division 3 EDG Room;
- turbine building fire zones;
- turbine power complex fire zones; and,
- emergency service water pumphouse.

Emphasis was placed on the 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 looked at fire hoses, sprinklers, and portable fire extinguishers to verify that 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 verify that they were in good physical condition. The documents listed at the end of the report were used by the inspectors during the assessment of this area.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

During the week of March 1, the inspectors reviewed the plant underdrain system. The system was designed to provide a method of controlling groundwater level around plant structures. The inspector's reviewed system health reports, CRs, and maintenance orders to assess overall system condition. The inspector's reviewed completed surveillance instructions (SVIs) to assess component and system performance. Finally, the inspectors reviewed the associated alarm response instructions (ARIs) to verify that they can reasonably be used to achieve the desired results.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On January 13, 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 verify that the training staff had observed important performance deficiencies and specified appropriate remedial actions.

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 verify that scheduled and emergent work activities were adequately managed. In particular, the inspectors reviewed the licensee's program for conducting maintenance risk assessments to verify that 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 verify that the actions were accomplished when on-line risk was increased due to maintenance on risk-significant SSCs. The following six assessments and/or activities were reviewed:

- the maintenance risk assessment and work execution associated with emergent work to replace the normal and alternate RPS 'A' bus contactors on January 22;

- the maintenance risk assessment and work execution associated with a planned RCIC system outage the week of January 26;
- work management and execution during a planned outage of RPS 'B' conducted the week of February 2;
- work management and execution during emergent work on RPS 'A' conducted the week of February 22;
- the maintenance risk assessment and work execution associated with planned work on Division 1 systems and subsequent risk management during emergent unavailability of Division 1 EDG during the week of March 21; and,
- the maintenance risk assessment and work execution associated with emergent work on the diesel driven fire pump performed on March 30.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14)

.1 Fuel Defect Localization and Isolation

a. Inspection Scope

On January 2 and January 3, the inspectors observed licensee efforts to localize and isolate a fuel defect. The inspectors observed infrequently performed test or evolution briefings, pre-shift briefings, and reactivity control briefings to verify the briefings met criteria specified in the Perry Operations Section Expectations Handbook and PAP-1121, "Conduct of Infrequently Performed Tests or Evolutions," Rev. 1. Additionally, the inspectors observed test performance to verify that procedure use, crew communications, and coordination of activities between work groups similarly met established station expectations and standards.

b. Findings

No findings of significance were identified.

.2 Response to Loss of RPS 'A' Bus

a. Inspection Scope

On February 22, the licensee experienced a loss of RPS 'A' bus concurrent with the performance of a routine weekly manual scram SVI. The inspectors responded to the site and reviewed licensee immediate and supplemental actions. Specifically, the inspectors verified the licensee's actions were consistent with operating instructions (OIs), ARIs, and off-normal instructions (ONIs).

b. Findings

No findings of significance were identified.

.3 Response to Loss of Battery Room Exhaust Fan M24A

a. Inspection Scope

On February 24, the licensee was shifting ventilation system trains in preparation for scheduled surveillance testing. While attempting to shift from the 'B' to 'A' trains, annunciators were received associated with the battery room exhaust fan M24A. Subsequent licensee investigation identified a blown fuse. The inspectors were in the control room at the time of the event and observed licensee immediate and supplemental actions. Specifically, the inspectors verified the licensee's actions were consistent with OIs and ARIs.

b. Findings

No findings of significance were identified.

.4 ONI-C51 Entry Due to Unexpected Increase in Indicated Thermal Power

a. Inspection Scope

On March 10, the inspectors reviewed operator response to a prompt increase in indicated thermal power. With no known change in plant conditions, indicated thermal power increased by approximately 80 megawatts. After being informed of the entry into ONI-C51, "Unplanned Change in Reactor Power or Reactivity," the inspectors responded to the control room and reviewed licensee immediate and supplemental actions. Specifically, the inspectors verified the licensee's actions were consistent with OIs, ONIs, and TSs. The inspectors also reviewed the licensee's immediate investigation which determined that a feedwater temperature input to the integrated computer system's heat balance calculation was providing erroneous low temperature input. As such, the licensee determined that no actual increase in thermal power had occurred.

b. Findings

No findings of significance were identified.

.5 Transfer of Recirculating Pump Seal Purge from Control Rod Hydraulics to Internal Flow

a. Inspection Scope

On March 18, the inspectors observed a brief on shifting the source of seal purge water from control rod hydraulics to internal flows in order to support maintenance activities. After the brief, the inspectors performed a walkdown of the valves that would be manipulated as part of the evolution. Inspectors verified that the licensee considered and planned for conditions that could result in loss of a recirculating pump. The inspectors observed operator performance during the evolution to verify that procedure use, crew communications, and coordination of activities between work groups met established station expectations and standards.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

During the week of March 1, the inspectors reviewed the design change package and the work associated with installation of an exhaust fan in the Division 3 EDG room to verify the adequacy of the design change for interfaces with the fire protection system and the safety-related portions of the ventilation system. The inspectors reviewed the regulatory applicability determination and 10 CFR 50.59 evaluation to determine if all current licensing basis and justifications were accurate. The inspectors compared the design with that of similar systems installed for the Division 1 and 2 EDGs to verify the newer design corrected any problems noted with the earlier designs.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (PMT) (71111.19)

a. Inspection Scope

The inspectors evaluated the following 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 TSs, 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 if 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 six post-maintenance activities were reviewed:

- testing of the Division 1 EDG starting air compressor ‘A’ conducted January 16 and 19;
- overspeed testing of the RCIC turbine conducted on January 30;
- controller testing for control room heating, ventilation and air conditioning (HVAC) conducted on February 11;
- RPS motor generator testing following voltage regulator replacement on February 25;
- combustible gas mixing compressor testing following fuse replacement on February 28; and,

- restoration of the 'B' control rod drive pump to standby readiness following rework on seal water supply unions on March 5 and subsequent in-service leak test conducted March 11.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing or reviewed test data for risk-significant systems or components to assess compliance with TSs, 10 CFR Part 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 reviewed whether test control was properly coordinated with the control room and performed in the sequence specified in the 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 five surveillance activities assessed were:

- safety/relief valve pressure actuation channel functional testing commenced January 8 and completed January 10;
- test discharge of the Unit 2 Division 1 battery conducted January 12;
- primary containment air lock door seal leakage testing conducted February 17;
- control room ventilation train 'A' heat removal testing commenced February 24 and completed February 25; and
- reactor pressure vessel level instrumentation response time testing conducted March 8.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspector reviewed Revision 18 of the Perry Nuclear Power Plant Emergency Plan to determine if changes identified in Revision 18 reduced the Plan's effectiveness, pending on-site inspection of the implementation of these changes.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the simulator control room and technical support center during an emergency preparedness drill conducted on February 18. The inspection focused on the ability of the licensee to appropriately classify emergency conditions, complete timely notifications, and implement appropriate protective action recommendations in accordance with approved procedures.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors sampled the licensee's submittals for performance indicators (PIs) listed below. The inspectors used PI definitions and guidance contained in Revision 2 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," to verify the accuracy of the PI data. The following three PIs were reviewed:

- reactor coolant system leakage;
- safety system functional failures; and,
- unplanned scrams.

The inspectors reviewed station logs, CRs, TS logs, and Licensee Event Reports (LERs) to verify the accuracy of the licensee's data submission.

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 verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed.

b. Findings

No findings of significance were identified.

.2 Annual Sample Review - Procedure Use and Adherence

a. Inspection Scope

The inspectors reviewed procedure use and adherence in relation to maintenance activities on safety-related equipment. The inspection included review of recent CRs related to failure to use or adhere to procedures as well as observation of work activities.

b. Findings

Introduction: The inspectors identified an NCV of TS 5.4, “Procedures,” for leaving a sheet of permalon unattended in the pool swell region in containment. Licensee procedure PAP-0204 states that unless risk assessed, no items shall be left unattended at any time below 623' 4". On February 5 the inspectors observed a large sheet of permalon by the 'A' hydraulic power unit with no workers in the area.

Description: On February 5, the inspectors performed a tour of containment. During this tour, the inspectors noted that work was in progress on the 'A' hydraulic power unit and an unattended sheet of permalon had been placed on the 620' level grating to prevent small parts from dropping through the grating and into the suppression pool. The inspectors were concerned the roughly 8'x10' sheet of plastic could clog a portion of the suppression pool strainer. After exiting containment, the inspectors reviewed the applicable procedure for controlling foreign material in containment and confirmed that this use did not conform to procedural requirements. When the licensee was notified of the unattended permalon, procedural compliance was promptly restored. The licensee discussed the condition with their staff to preclude further material from being left unattended.

The licensee developed PAP-0204, “Housekeeping,” in part, to provide instructions to control foreign material within containment. This procedure prohibits unattended flexible impermeable material anywhere within containment and any unattended material below the 623' 4" level. Although the procedure allows use of the 50.59 process or risk assessment process to grant relief from these requirements, they were not used. In addition, the work supervisor noted that while material was removed at the end of the day, work practices left the material unattended during lunch and other short breaks.

The inspectors have noted other recent examples where procedures were not followed. For example:

- During a test discharge of the Division 1, Unit 2 battery, the steps were not performed in the sequence written. The surveillance procedure specifically required performance of all steps in the order written; however, the licensee performed steps for installation of test equipment out of sequence in order to expedite procedure

performance. Subsequent review of the procedural steps performed out of sequence confirmed that they did not affect the results of the surveillance.

- On February 8, following performance of a surveillance on the fuel pool cooling and cleanup system, operators opened an incorrect valve. As a result, approximately 1800 gallons of water drained to the associated backwash receiver tank. The improper lineup lasted several hours until identified by an operator on the next shift.

- On February 11, while performing a radioactive waste discharge, the operator opened the wrong valve and sent approximately 2700 gallons of water to the condenser. No unmonitored discharge occurred and reactor water chemistry was not adversely affected. This event occurred despite an operations standdown on procedural use and adherence following the February 8 improper valve positioning.

The inspectors note that procedural non-compliance affected most Perry organizations and corrective actions focusing on the operations organization would not completely address the problem. In addition to the examples listed, numerous other examples can be found in the corrective action program.

Analysis: This finding was more than minor because the inspectors concluded that it could reasonably be viewed as a precursor to a more significant event. Specifically, leaving unattended items in containment can lead to the unattended items falling into the suppression pool without being noticed or being transported into the pool during an actual event. This material could then clog suppression pool strainers thereby reducing emergency core cooling system (ECCS) flow. This finding also affected the cross-cutting area of Human Performance because plant personnel failed to follow licensee procedures and left material unattended in the swell region of containment.

Using Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors answered "no" to all five screening questions in the Phase 1 Screening Worksheet under the mitigating systems column. Specifically, no material fell into the suppression pool and the quantity of material observed was not sufficient to result in a loss of safety function. The inspectors therefore determined the finding to be of very low safety significance.

Enforcement: Technical Specification 5.4 states, in part, that procedures shall be established, implemented, and maintained as recommended in Regulatory Guide 1.33. Regulatory Guide 1.33 recommended the establishment of procedures for performing maintenance that can affect the performance of safety-related equipment. Contrary to these requirements, the licensee failed to implement the requirements of PAP-0204 for control of material in containment, and left material in containment that could affect the performance of the ECCS systems. Because of the very low safety significance and because the issue has been entered into the licensee's corrective action program (CR 04-00708), the issue is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000440/2004002-02)**.

.3 Annual Sample Review - Review of "CF" Condition Reports

a. Inspection Scope

The inspectors reviewed a sample of licensee CRs classified as conditions adverse to quality, but not requiring root or apparent cause evaluations because the cause of the issue can be easily determined ("CF" designation). The inspectors reviewed selected CRs to verify that they were dispositioned in accordance with the licensee's corrective action program as defined in NOP-LP-2001, "Condition Report Process," Rev. 4, Rev. 5, and Rev. 6; that classification and prioritization of the resolution of the identified problems was commensurate with safety significance; and that corrective actions were completed in a timely manner commensurate with safety significance.

b. Findings

In accordance with licensee procedures, an evaluation methodology of "fix" can be selected for issues involving conditions for which "a likely cause can easily be determined." Guidance stated "the intent of this evaluation method is to correct the condition, if needed and enter coding that can be trended." The user was instructed that "if an Evaluation Code of "F" (Fix) is assigned, then determine the most likely cause without a formal evaluation, generate actions to correct the condition and document the cause only for trending purposes."

The inspectors performed a statistical review of CRs generated between March 1, 2003, and December 31, 2003. During the period, over 900 CRs were designated as "CF" meaning, by program definition, that although the issue was a condition adverse to quality, the cause was easily determined. Despite being dispositioned as "cause easily identified," the inspectors identified that over 50 of the 900 CRs were assigned a cause code of "Unknown."

The inspectors performed a review of 20 selected "CF" CRs and identified the following issues:

- CR 03-03169 documented a jacket water leak on the Division 2 EDG. Corrective action was assigned to "verify work activity to correct leakage is scoped into [refueling outage 9] RFO9." The work order was approved for RFO9 and the corrective action was closed. This was not consistent with licensee procedure which required all corrective actions for "CRs categorized as SCAQ [significant condition adverse to quality] and CAQ [condition adverse to quality] shall be tracked in the CREST database from initiation until the approved Corrective Action(s) are implemented, have corrected the deficiency and their implementation is documented in the database."

- CR 03-04929 documented an inner calibration select command switch which was found out of position. During an instrumentation and calibration (I&C) surveillance, the switch was found to be selected to the 4 position instead of OFF. Since the switch was not activated (pushed in), no immediate safety concern existed. Licensee review identified the switch had been manipulated during an I&C

surveillance 4 days prior to finding it out of position. Since the technician who performed the earlier work stated he recalled putting the switch in the OFF position and received independent verification of such action, the licensee closed the CR with cause unknown and no corrective actions. Use of the "F" evaluation was therefore not consistent with licensee program requirements. The inspectors also questioned the adequacy of the licensee's review which concluded that there was no apparent or conclusive reason for the switch being found out of position. In that selector switches are under cover plates which are locked in place, inadvertent mis-positioning of the switch was extremely unlikely. An independent evaluation of the issue should have concluded mis-positioning of the switch during the prior surveillance at least as the likely cause of the as-found condition. As such, the inspectors concluded that corrective action should have been assigned to review the issue with I&C technicians and reenforce independent verification expectations.

- CR 03-6539 documented problems entering substitute data into the plant computer system for RWCU inlet temperature. The computer accepted an invalid number and, as a result, stopped the computation of the nuclear heat balance. The unit reduced power approximately one percent as a result of the condition. The problem was resolved by restarting a desktop computer. Licensee review did not identify a cause. Use of the "F" evaluation was therefore not consistent with licensee program requirements. Additionally, CR closure comments stated the condition "has been noted in the computer support unit's tracking log for future evaluation." Use of tracking mechanisms other than CREST for corrective action associated with SCAQ and CAQ CRs was not consistent with licensee program requirements.

- CR 03-02496 documented finding 5-amp fuses in two locations in the Division 1 EDG power panel. System drawings identified the locations as having 2-amp fuses. The licensee's investigation revealed that the system drawings were correct and that 2 amps was the proper size. The documented investigation did not state when, or if, the fuses had last been replaced. The licensee's review did not identify a cause. Use of the "F" evaluation was therefore not consistent with licensee program requirements. The licensee's extent of condition review was limited to the same panel on the Division 1 EDG. Corrective action was assigned to "initiate a work request to inspect fuses F5 & F6 in the Division 2 panel." The work request was initiated and the corrective action closed. At the time of this inspection, the inspection had not been completed. Use of tracking mechanisms other than CREST for corrective action associated with SCAQ and CAQ CRs was not consistent with licensee program requirements.

- CR 03-03907 documented a degraded seal on a control room boundary door. The licensee promptly replaced the seal. The licensee's investigation determined that the seal damage was not attributable to normal wear, but most likely due to a cart impacting the door and causing the damage. Despite clearly identifying the cause, no corrective actions were assigned to address it.

- CR 03-04797 documented a half-scram which was received while a jumper shorted to ground during installation. The jumper was being installed to bypass the downscale rod block function for I&C surveillance testing. Corrective actions were

assigned to review the jumper installation methodology and discuss heightened awareness during jumper installation with I&C personnel. The event occurred in August 2003. The I&C discussions are documented to have been completed in December 2003. The installation methodology review identified that if the jumpers for bypassing the downscale rod block function were installed neutral side first, and removed positive side first, the half-scram would not have occurred. The inspectors discussed corrective action timeliness with the licensee since the affected surveillance test is performed quarterly, yet corrective action which would prevent recurrence is not due until March 2005.

The licensee generated CR 04-01402, "NRC Senior Resident Inspector Observations Regrading Condition Report Process," to address the individual issues identified during the inspection. Although the inspectors determined the individual issues identified to be minor in nature and therefore not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy, the inspectors concluded that corrective action program implementation remained an issue at Perry.

At the end of the inspection period, the licensee informed the inspectors that a revision to NOP-LP-2001 was being drafted to re-define "fix" CRs. Specifically, the licensee informed the inspectors that the program will be changed to state that the intent of a "fix" CR is to remediate the condition and not to determine cause. Thus, the only determining factor in classifying an issue as CF would be significance, not required evaluation method.

4OA3 Event Followup (71153)

(Closed) Licensee Event Report (LER) 05000440/2003-006-00: Loss of Safety Function and Limiting Condition for Operation 3.0.3 Entry Due to Support System Inoperability. On December 21, two main line control fuses for the Division 1 emergency closed cooling water (ECCW) system temperature control valve blew due to shorted windings in the hydraulic actuator motor. As a result of the loss of power, the temperature control valve failed in the full-cooling position. Per system OIs, the licensee declared the Division 1 control complex chilled water (CCCW) system chiller inoperable since ECCW system temperatures were less than 55 °F. Concurrently, the Division 2 CCCW chiller was inoperable for planned maintenance. The loss of both divisions of safety-related CCCW chillers resulted in a declaration of inoperability for the supported systems, including alternating current and direct current distribution and ECCSs. As a result of the inoperable equipment, the licensee entered LCO 3.0.3. The Division 2 CCCW chiller was restored to standby 1 hour and 45 minutes after the Division 1 ECCW temperature control valve failed and was declared operable after completion of PMT. Due to the short time period both divisions of safety-related CCCW chillers were inoperable, the licensee concluded no actual loss of safety function occurred based on postulated heat-up rates of supported equipment in post-accident scenarios. The LER did not identify any new performance deficiencies. No violations of regulatory requirements were identified. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 1R/EP.4 of this report had, as its primary cause, a human performance deficiency, in that the licensee's review of GE SIL 508 failed to identify all affected plant components. This caused numerous contactors to not be properly reviewed for inclusion in the preventative maintenance program for periodic replacement. On November 29, 2003, an age-related failure of a contactor in the normal power supply circuitry resulted in a blown fuse which de-energized the RPS 'B' bus.

.2 A finding described in Section 4OA2.2 of this report had, as its primary cause, a human performance deficiency, in that plant personnel failed to follow licensee procedures and left material unattended in the swell region of containment.

4OA5 Other Activities

Spent Fuel Material Control and Accounting At Nuclear Power Plants (TI 2515/154)

The inspectors completed Phase I and Phase II of the subject TI (Temporary Instruction) and provided the appropriate documentation to NRC management as required by the TI.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. W. Kanda, Site Vice President and other members of licensee management at the conclusion of the inspection on April 8. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meeting

Emergency Preparedness inspection with Mr. D. Cleavenger on January 29, 2004.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

W. Kanda, Vice President-Nuclear
R. Coad, Radiation Protection Manager
D. Cleavenger, Senior Emergency Planning Specialist
V. Higaki, Manager, Regulatory Affairs
T. Lentz, Director, Nuclear Engineering
T. Rausch, General Manager, Nuclear Power Plant Department
R. Strohl, Superintendent, Plant Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000440/2004002-01	FIN	Loss of Normal Power Supply to RPS Bus 'B' (Section 71111.EP.4)
05000440/2004002-02	NCV	Unattended Items Left in Containment (Section 71152)

Closed

05000440/2004002-01	FIN	Loss of Normal Power Supply to RPS Bus 'B' (Section 71111.EP.4)
05000440/2004002-02	NCV	Unattended Items Left in Containment (Section 71152)
05000440/2003-006-00	LER	Loss of Safety Function and Limiting Condition for Operation 3.0.3 Entry Due to Support System Inoperability (Section 4AO3)

LIST OF DOCUMENTS REVIEWED

1REP Equipment Availability and Functional Capability

CR 03-06691; Inconsistent Weight of EDG Division 1 and 2 TRD; dated December 18, 2003

CR 03-06639; Broken Body to Bonnet Bolt on B33-F020; dated December 13, 2003

DWG 302-0355-00000; HPCS and Standby Diesel Generator, Intake and Crankcase; Rev. S

DWG 302-0602-00000; Reactor Water Recirculation System; Rev. S

DWG 55A7119; 1" Body 40 Actuator 667NS-DBQNS Diaphragm Actuated Control Valve; Rev. E

Maintenance Rule Database, Rev. 5.08

Maintenance Rule Monitor Database, dated January 4, 2004

Perry Nuclear Power Plant Health Report, Third Quarter 2003

Perry Nuclear Power Plant Health Report, Fourth Quarter 2003

CR 92-028; [Spurious Reset of Slave Trip Unit During Surveillance]; dated February 21, 1992

CR 00-3901; Unanticipated Opening of SRV 1B21F0051D; dated December 18, 2000

CR 99-2319; [Voltage spike Due to Trip Unit Removal]; dated September 16, 1999

SVI-B21-T0369-F; SRV Pressure Actuation Channel F for 1B21-N668F; Rev. 3

CR 03-05745; Lifted 2 SRVs During the Performance of SVI-B21-T0369B; dated October 15, 2003

CR 03-00322; Div. 1 and Div. 2 Testable Rupture Disc Qualification; dated December 23, 2004

Operations Standing Instruction; Fuel Defect Monitoring; dated January 7, 2004

CR 04-00476; VT-2 Rejectable Leakage; dated January 31, 2004

CR 04-00471; RCIC System Operation with Minimum Flow Valve Cycling; dated January 31, 2004

GMI-0125; Reactor Core Isolation Cooling Turbine Overhaul; Rev. 2

Perry Nuclear Power Plant Cycle 10 Fuel Defect Operational Action Plan; Rev. 0

CR 04-00768; High RPS B MG Voltage Upon Shifting To MG Set B; dated February 13, 2004

CR 03-06398; Trip of RPS B System; dated November 29, 2003

CR 03-04797; Unexpected ½ Scram; dated August 18, 2003

CR 03-03565; Inadvertent ½ Scram Due To Mis-Ranging IRM Range Switch; dated May 28, 2003

CR 03-03446; Half Scram Reset Did Not Fully Reset Half Scram; dated May 21, 2003

CR 03-02585; Inadvertent Half Scram During Surveillance SVI-C71T0254C; dated April 28, 2003

CR 03-02015; Loose Connections On Fuse Terminals; dated April 11, 2003

CR 03-01338; Thermography Concern With Div 1 RPS Fuses C71-F18A & C71-F18E; dated March 18, 2003

CR 03-00198; OE 15054 Response RPS EPA Circuit Breakers; dated January 16, 2003

CR 04-00634; Rx Bld. P54 Standpipe not to License Basis when Drained; dated February 6, 2004

CR 04-00471; RCIC System Operation with Minimum Flow Valve Cycling; dated January 31, 2004

TM 1-03-001; Temporary Mod for FME Protection of Div 3 Diesel Exhaust Rupture Disc Opening; Rev. 1

CR 02-04855; Inspection of Concrete Structures at Div 3 Diesel Exhaust Rupture Disc; dated December 19, 2002

CR 02-04282; Inspection of Concrete Structures at EDG Div 3 TRD; dated November 12, 2002

CR 02-01110; Div 3 Rupture Disk Concrete Inspection; dated April 15, 2002

CR 02-04440; Water Intrusion below the Containment Annulus Floor Level; dated November 21, 2002

CR 03-06250; Valves 1D23F0050 and 1G43F0060 Downpowered Open During Div 3 Scheduled Outage; dated November 18, 2003

DCP 96-0042; Design Report ECCS Suction Strainer; Rev. 2

CR 03-01168; 1M14F602 Found Open; dated March 10, 2003

CR 03-01290; Valve 1P52F0645 Found Out of Position; dated March 10, 2003

1R04 Equipment Alignment

Perry Nuclear Engineering Department Action Plan; Control Rod Drive Hydraulic System; dated January 9, 2004

Perry Nuclear Power Plant System Health Report; Third Quarter 2003

VLI-C11 (CRDH); Control Rod Drive Hydraulic System (CRDH); Rev. 11

SOI-C11 (CRDH); Control Rod Drive Hydraulic System; Rev. 14

DWG 302-0871-0000; Control Rod Drive Hydraulic System; Rev. CC

DWG 302-0872-0000; Control Rod Drive Hydraulic System; Rev. Y

CR 02-03448; Scram Disch. Volume Vent and Drain Valve Couplings Missing Lockwashers; dated September 24, 2002

CR 02-03589; RFA With Anticipatory Rod Stroking; dated October 2, 2002

CR 02-03636; Control Rod Triple Notched Out; dated October 4, 2002

CR 02-03793; Unusual Indications During Performance of SVI-C11-T1003-A; dated October 11, 2002

CR 02-03795; Abnormally High Number of Control Rods Required Alternate Control Methods; dated October 11, 2002

CR 02-04554; Erosion of CRD Pump A Casing Discovered During Mechanical Seal Replacement; dated December 3, 2002

CR 02-04623; C11 A Pump Union Leak, PMT Failure; dated December 8, 2002

CR 03-00881; Basis For 30 Second Settle Time In TXI-0355; dated February 23, 2003

CR 03-03089; FCV 1C11F0002A Oscillations; dated May 9, 2003

CR 03-03331; CRD Flow Controller Failed to Respond In Auto; dated May 17, 2003

CR 03-03549; Scram Discharge Volume Vent and Drain Operability Surveillance Failed; dated May 27, 2003

CR 03-05205; Settle Time Testing Required Prior Than Previously Anticipated; dated September 4, 2003

CR 03-05228; 1C11-F010 Stroke Time Too Fast; dated September 12, 2003

CR 04-00044; Trending Info Regarding Improved Control Rod System Performance; dated January 5, 2004

PEI-SPI 1.4; Venting the Scram Air Header; Rev. 1

PEI-SPI 1.5; Venting CRD Overpiston Volumes; Rev. 0

PEI-SPI 4.1; CRD Alternate Injections; Rev. 1

VLI-E22A; High Pressure Core Spray; Rev. 6

VLI-R44/E22B; Division 3 Diesel Generator Starting Air System; Rev. 5

VLI-R45/E22B; Division 3 Diesel Generator Fuel Oil System (Unit 1); Rev. 3

VLI-R46/E22B; Division 3 Diesel Generator Jacket Water System (Unit 1); Rev. 5

VLI-R47/E22B; Division 3 Diesel Generator Lube Oil System (Unit 1); Rev. 3

VLI-E12; Residual Heat Removal; Rev. 5

DWG 302-0642-00000; Residual Heat Removal; Rev. CC

1R05 Fire Protection

FPI-A-I01; Fire Rated Assemblies and Detector Inspection Guidelines; Rev. 0

FPI-0CC; Control Complex; Rev. 3

FPI-1DG; Diesel Generator Briefing; Rev. 1

FPI-TB; Turbine Building; Rev. 1

FPI-TPB; Turbine Power Building; Rev. 1

FPI-1DG; Diesel Generator Building; Rev. 1

FPI-0EW; Emergency Service Water Pumphouse; Rev. 3

1R06 Flood Protection Measures

USAR Figure 2.4-71; Plant Foundation Underdrain System; Rev. 13

CR 03-04196; Maintenance Rule Walkdown Underdrain Manholes; dated July 8, 2003

CR 03-04631; Plant Underdrain System; dated August 6, 2003

CR 04-00602; Underdrain Manhole #7 Cover; dated February 5, 2004

CR 04-00909; ESW Pumphouse Max Lake Water Level Higher Than P72 Piping; Flow Back Concern; dated February 23, 2004

PTI-P72-P0002; Plant Underdrain Groundwater Inflow Test; Rev. 5

PTI-P72-P0005; Plant Underdrain Groundwater Level Readings; Rev. 2

SOI-P72; Plant Foundation Underdrain System; Rev. 7

Perry Nuclear Power Plant Health Report, Fourth Quarter 2003

ARI-H13-P970-0001; Common Long Response Benchboard; Rev. 5

1R13 Maintenance Risk Assessments and Emergent Work Control

PAP-1924; On-Line Safety Assessment; Rev. 3

PDB-C0011; PSA Pre-Solved Configurations For On-Line Risk; Rev. 2

Probabilistic Safety Assessment; Period 4 Week 6; Rev. 2

Probabilistic Safety Assessment; Period 4 Week 10; Rev. 1

Order 200079481; Replace the Normal and Alternate RPS Contactors Due to Extend Time in Service; dated January 22, 2004

SOI-C71; RPS Power Supply Distribution; Rev. 9

PAP-0204; Housekeeping/Cleanliness Control Program; Rev. 12

CR 03-03518; RFA-PSA Analysis of Temporary Alterations on the Refuel Floor; dated May 24, 2003

Problem Solving Plan; Loss of RPS ‘A’ NORM During Weekly Manual Scram SVI; dated February 23, 2004

CR 04-00901; Loss of RPS ‘A’ NORM During Weekly Manual Scram SVI, Causes Entry Into ONI-C71-2; dated February 22, 2004

Probabilistic Risk Assessment; Period 5, week 2 Rev. 1

Probabilistic Risk Assessment; Period 5, week 2 Rev. 2

Probabilistic Risk Assessment; Period 5, week 3 Rev. 2

1R14 Operator Performance During Nonroutine Evolutions and Events

Operator Logs; dated February 22 through February 24, 2004

PAP-1121; Conduct of Infrequently Performed Tests or Evolutions; Rev. 1

ARI-H13-P904-2; Common Heating, Ventilation, and Air Conditioning Control Panel; Rev. 4

SDM B33; Reactor Recirculation and Recirculation Flow Control System; Rev. 9

ONI-C51, Unplanned Change in Reactor Power or Reactivity, Rev. 13

1R17 Permanent Plant Modifications

ECR 99-5036; HPCS Diesel Generator Room Auxiliary Exhaust Fan; dated April 4, 2002

DWG 208-0135-00012; Div. 3 Diesel Generator Room; dated October 28, 2003

DWG 208-0135-00011; Div 2 Diesel Generator Room; Rev. C

S02-00222; 10 CFR 50.59 Screen HPCS Diesel Generator Room High Ambient Room Temperature; Rev. 1

R02-00222; Regulatory Applicability Determination HPCS Diesel Generator Room High Ambient Room Temperature; Rev. 1

1R19 Post-Maintenance Testing

Order 200064427; Troubleshoot/Rework/Overhaul Air Compressor 1R44C0001A; dated January 11, 2004

GMI-0073; V-Belt and Sheave Maintenance; Rev. 5

PMI-0026; Standby Diesel Generator Starting Air Compressors; Rev. 0

PTI-E51-P003; RCIC Terry Turbine Overspeed Trip Test; Rev. 3

RCIC Steam Turbine; Rev. 36

GMI-0125; Reactor Core Isolation Cooling; Rev. 2

Perfector Series 600 Installation, Adjustment and Maintenance Manual; dated 1981

WO 200082970; 0M25C0001A Flow Lower Than Design; dated February 11, 2004

PTI-C71-P0002; RPS Motor/Generator Set Voltage Regulator Calibration and Functional Test; Rev. 4

GEK 42296; Motor Generator Package Set; Rev. 1

WO 200040726; Periodic Fuse replacement (1M51C0001A); dated February 28, 2004

CR 01-1711; Broken Fuse Block for Gas Mixing Compressor A; dated April 1, 2001

SOI-C11 (CRDH); Control Rod Drive Hydraulic System; Rev. 14

1R22 Surveillance Testing

CR 04-00095; Unexpected Alarm During SVI-B21-T0369A; dated January 8, 2004

CR 03-05745; Lifted 2 SRV's During Performance of SVI-B21-T0369B; dated October 15, 2003

SVI-B21-T0369-A; SRV and Low-Low Set Pressure Actuation Channel Functional For 1B21-N668A; Rev. 5

Root Cause Analysis Report; SRV's Lifted (2) During the Performance of SVI-B21-T0369B; dated December 4, 2003

Problem Solving Plan; CR 04-0095 System Response Anamolies During SVI-B21-T0369A; dated January 9, 2004

DWG 208-0011-0006; Automatic Depressurization System - Relay Logic; Rev. F

DWG 208-0011-0009; Automatic Depressurization System - Relay Logic Analog Circuits; Rev. J

DWG 208-0011-0010; Automatic Depressurization System - Relay Logic; Analog Circuits; Rev. K

GEI-0039; Full Battery Equalizing Charge for Lead-Calcium Batteries; Rev. 6

Vendor File Number 0136; Operation and Maintenance Instructions for Plant Batteries; Rev. 6

SVI-R42-T5217; Performance Test of Battery Capacity; Rev. 5

SVI-P53-T6305; Lower Primary Containment Air Lock (Penetration P305), In Between the Seals Test; Rev. 5

SVI-P53-T6305; Upper Primary Containment Air Lock (Penetration P312), In Between the Seals Test; Rev. 5

SVI-M25-T1270-A; Control Room Ventilation Heat Removal Test; Rev .1

Calculation M25-01; Control Room Simulation of 95 Degree Day; Rev. 0

SVI-B21-T0252-A; RPV Level 3 and 8 Channel A Response Time For 1B21-N680A; Rev. 5

1EP4 Emergency Action Level and Emergency Plan Changes

Perry Nuclear Power Plant Emergency Plan; Revision 18

4OA2 Identification and Resolution of Problems

NOP-LP-2001; Condition Report Process; Rev. 6

NOP-LP-2001; Condition Report Process; Rev. 5

NOP-LP-2001; Condition Report Process; Rev. 4

NOP-LP-2001; Condition Report Process; Rev. 3

NOBP-LP-2011; FENOC Root Cause Analysis Reference Guide; Rev. 2

SOI-G41; Fuel Pool Filter Demineralizer System; Rev. 6

SVI-G41-T2001; Fuel Pool Cooling and Cleanup System Pump and Valve Operability; Rev. 6

CR 04-00652; Valve Found Out of Position; dated February 8, 2004

CR 04-00681; Lack of Compliance to the Procedure Implementation Process; dated February 9, 2004

CR 04-00696; Personal Error During Relay Replacement; dated February 10, 2004

DWG 302-0653-00000; Fuel Pool Filter and Demineralizer; Rev. S

DWG 302-0655-00000; Fuel Pool Cooling and Cleanup System; Rev. T

DWG 302-654-00000; Fuel Pool Cooling and Cleanup System; Rev. M
CR 04-00733; Control Switch Mispositioning Event Radwaste Control Room dated February 11, 2004

CR 03-02153; Bus H12 Breakers Racked to “Disconnect” with Control Power Fuses Installed; dated April 17, 2003

CR 03-03130; Exciter Field Breaker not in tagged Position; dated April 16, 2003

CR 01-1319; Div. 1 BOP Isolation while Replacing Relays Resulting in Loss of SDC; dated March 12, 2001

CR 03-05819; Procedure Step Missed while Racking Out a Breaker; dated October 20, 2003

CR 03-04734; During SVI-C51-T0050F a Test Cable was not Hooked Up Correctly; dated August 13, 2003

CR 03-03518; RFA-PSA Analysis of Temporary Alterations on the Refuel Floor; dated May 24, 2003

CR 03-02496; Installed Fuses Do Not Match the Drawing; dated April 26, 2003

CR 03-02822; 1N27-F739A Feedwater Leakage Control Check Valve Fails High Pressure Seat Leak; dated May 2, 2003

CR 03-03001; 1B21F0019 Seat Leakage; dated May 6, 2003

CR 03-03009; Vacuum Breaker Check Valves Installed in Reverse; dated May 7, 2003

CR 03-03075; Error During SVI-B21-T1317B; dated May 9, 2003

CR 03-03132; Broken Heat Shield Bolt Div 1 DG; dated May 10, 2003

CR 03-03133; Unexpected Valve Movement During SVI-R43-T5366, Div 1 LOOP/LOCA Testing; dated May 10, 2003

CR 03-03169; Jacket Water Leakage at LB #5 Div 2 Diesel Generator; dated May 11, 2003

CR 03-03907; Degraded Control Room Boundary Door Seal; dated June 16, 2003

CR 03-04018; EOF Outside Air Dampers; dated June 25, 2003

CR 03-04069; HPU B Fryquel Leak To Suppression Pool; dated June 30, 2003

CR 03-04219; Battery Charger Output Fuses Blown During Functional Testing; dated July 10, 2003

CR 03-04281; C41 Tank Heater Leak; dated July 17, 2003

CR 03-04582; Diesel Fire Pump Inoperability Due to Electrolyte Level; dated August 4, 2002

CR 03-04797; Unexpected ½ Scram; dated August 17, 2003

CR 03-04929; Switch Out of Position; dated August 23, 2003

CR 03-05198; Division 1 Feedwater Leakage Control Piping Found Partially Air Filled; dated September 10, 2003

CR 03-05242; Temporary Power Installation During Temp Mod 03-0022; dated September 13, 2003

CR 03-05574; Maintenance Rule Program Procedure Requirements Not Followed; dated October 2, 2003

CR 03-06539; Problems With Entering Substitute Values in ICS Computer; dated December 7, 2003

CR 03-06662; Unexpected Alarm RCIC Suction Pressure; dated December 16, 2003

4OA5 Other Activities

FTI-A0017; Non-Special Nuclear Material Pool Inventory Mechanism; Rev. 0

PAP-0802; Control of Special Nuclear Material; Rev. 0

PAP-0802; Control of Special Nuclear Material; Rev. 6

FTI-D2; SNM Physical Inventory; Rev. 0

FTI-D02; Special Nuclear Material Physical Inventory; Rev. 2

FTI-D0006; Preparation of Fuel Movement Checklist; Rev. 4

FTI-D0009; Use of the Fuel Movement Checklist; Rev. 7

FTI-E0036; Inspection of Irradiated Bundles; Rev. 1

TXI-0299; Fuel Rod Accountability; Rev. 0

Pool Inventory Log; dated February 5, 2004

WO 890006954; Rework Fuel Bundles; dated January 30, 1990

WO 920002782; Inspect/Reconstitution - Fuel Bundles; dated November 3, 1992

WO 920005245; Inspect/Reconstitution - Fuel Bundles; dated January 13, 1993

WO 950005569; Perform Inspection of Failed Fuel; dated February 29, 1996

WO 98-008430-000; Reconstitute Leaking Fuel Bundles(s) in Accordance With Attached TXIs; dated April 15, 1999

WO 00-001452-000; Perform Post-Irradiated Fuel Bundle Examinations Per TXI in FHB; dated June 8, 2000

WO 00-008337-000; Perform Post-Irradiated Fuel Bundle Examinations Per TXI in FHB; dated February 24, 2001

WO 03-002758-000; Perform Post-Irradiated Fuel Bundle Examinations Per TXI in FHB; dated April 7, 2003

CR 04-00280; Vendor Incorrectly Completed Procedural Attachment; dated January 22, 2004

CR 04-00776; Documentation Errors Found On Fuel Bundle Inspection Sheets; dated February 13, 2004

CR 04-00988; OE 11903 - Two Fuel Rods Could Not Be Located At Millstone; dated February 26, 2004

LIST OF ACRONYMS USED

ARI	Alarm Response Instruction
CAQ	condition adverse to quality
CCCW	control complex chilled water
CFR	*Code of Federal Regulations*
CR	condition report
CRDHS	control rod drive hydraulic system
ECCW	emergency core cooling water
ECCS	emergency core cooling system
EDG	emergency diesel generator
FENOC	FirstEnergy Nuclear Operating Company
GE	General Electric
HVAC	heating, ventilation, and air-conditioning
I&C	instrument and control
LER	licensee event report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
OI	Operating Instruction
OE	Operability Evaluation
ONI	Off-Normal Instruction
OWA	operator workaround
PI	performance indicator
PMT	post-maintenance testing
RCIC	reactor core isolation cooling
RFO9	Refueling Outage 9
RHR	residual heat removal
RPS	reactor protection system
RWCU	reactor water clean up
SCAQ	significant condition adverse to quality
SDP	significance determination process
SIL	Service Information Letter
SSC	structure, system & component
SVI	surveillance instruction
TI	Temporary Instruction
TS	Technical Specification
USAR	Updated Safety Analysis Report
VLI	valve lineup instruction