March 14, 2002

Mr. Guy Campbell
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P. O. Box 97, A200
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT
NRC INSPECTION REPORT 50-440/01-16

Dear Mr. Campbell:

On February 17, 2002, the NRC completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on February 26, 2002, 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 issues of very low safety significance (Green) that were determined to involve a violation of NRC requirements. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC' s Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with a basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Perry Nuclear Power Plant.

In accordance with 10 CFR 2.790 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). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/Christine A. Lipa

Christine A. Lipa, Chief
Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 50-440/01-16

cc w/encl: B. Saunders, President - FENOC
T. Rausch, Director, Nuclear Maintenance Department
G. Dunn, Manager, Regulatory Affairs
K. Ostrowski, Director, Nuclear Services Department
J. Powers, Director, Nuclear Engineering Department
W. Kanda, General Manager, Nuclear Power Plant Department
Public Utilities Commission of Ohio
Ohio State Liaison Officer
R. Owen, Ohio Department of Health

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	50-440/01-16
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	P.O. Box 97 A200 Perry, OH 44081
Dates:	January 1, 2002 through February 17, 2002
Inspectors:	Ray Powell, Senior Resident Inspector John Ellegood, Resident Inspector Steve Campbell, Senior Resident Inspector, Fermi Robert Jickling, Emergency Preparedness Analyst
Approved by:	Christine A. Lipa, Chief Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000440-01-16; on 01/1-02/17/2002; First Energy Nuclear Operating Company; Perry Nuclear Power Plant. Temporary Modifications.

This report covers a 7-week routine inspection. The inspection was conducted by resident inspectors and a regional inspector. Two findings of very low risk significance were identified during this inspection and were considered to be Non-Cited Violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, “Significance Determination Process” (SDP). The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at: http://www.nrc.gov/NRR/OVERSIGHT/index.html. Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violations.

A. Inspection Findings

Cornerstone: Initiating Events

GREEN. The inspectors identified a Non-Cited Violation of 10CFR50 Appendix B, Criterion III, for failure to remove temporary lighting from the reactor water cleanup room after one-cycle as required by Field Clarification Request. The lights eventually degraded and caught fire.

The finding was greater than minor because it had an actual impact of causing a small fire in a room containing plant operating, fire protection and safety-related equipment. The event was of very low safety significance because, although the finding contributed to the likelihood of an external event initiator, no equipment was damaged from the event. (Section 1R23.1).

Cornerstone: Mitigating Systems

GREEN. The inspectors identified a Non-Cited Violation of 10CFR50 Appendix B, Criterion V, for failing to follow plant procedures to maintain electrical separation between Class 1E and Non-class 1E cables and conduits.

The finding was greater than minor because if left uncorrected, routing the extension cords near safety-related power cables increased the likelihood of rendering multiple trains of safety-related equipment inoperable given a fire from those temporary cables. Further, the multiple examples of violating the electrical separation criteria indicated a lack of plant personnel knowledge of the requirement. The finding was of low safety significance because an actual fire had not occurred that rendered the associated equipment unavailable. (Section 1R23.2).

B. Licensee Identified Violations

None

Report Details

Summary of Plant Status: The plant began the inspection period with Unit 1 at 100 percent power. The unit remained at 100 percent power until January 19, 2001 when power was reduced to 60 percent for condenser tube plugging. The unit was returned to 100 percent power on January 21. The unit remained at 100 percent power until January 27 when power was reduced to 80 percent for control rod alignment. The unit returned to 100 percent power later that same day. On February 10, the licensee reduced power to 90 percent in order to recover an inadvertently scrammed control rod. The rod scrammed because a fuse had blown on one of its reactor protection system scram solenoids and a second channel was tripped as part of surveillance testing. Following rod recovery, the unit returned to 100 percent power. Power effectively remained at 100 percent for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04Q)

a. Inspection Scope

The inspectors conducted a partial alignment walkdown of the Division 2 Emergency Service Water (ESW), a risk important system, to evaluate its readiness while the Division 1 train was declared inoperable due to ESW pumphouse ventilation maintenance. The walkdown included selected switch and valve position checks, review of associated effective operating procedures, and verification of electrical power to critical components. The inspectors reviewed sections of the Updated Safety Analysis Report (USAR) and Technical Specifications (TS) as applicable to the walkdown. The documents used for the walkdown are listed in the attached List of Documents Reviewed.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

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

- Fire Zone 1CC-4a, Unit 1, Division 2 Cable Spreading Area
- Fire Zone 1CC-4e, Unit 1, Division 1 Cable Spreading Area
- Fire Zone 1RB-1b, Reactor Water Cleanup Heat Exchanger Room
- Fire Zone 1AB-1a, Low Pressure Core Spray Pump Room
- Fire Zone 1AB-1f, High Pressure Core Spray Pump Room

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. Finally, the inspectors toured the reactor water heat exchanger room to assess the extent of damage to components in the room following the January 7, 2002 lighting string fire. 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. Circumstances that caused the lighting string fire in the reactor water cleanup heat exchanger room are discussed in Section 1R23 of this report.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On February 5, 2002, the resident inspectors observed licensed operator performance in the plant simulator. The evaluated scenario included severe weather, a pressure regulator failure, and a loss of reactor water level indication.

The inspectors evaluated crew performance for clarity and formality of communication; the ability to take timely action in the safe direction; the prioritizing, interpreting, and verifying of alarms; the 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.

a. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12Q)

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the maintenance rule requirements to verify that component and equipment failures were identified, entered, and scoped within the maintenance rule and that select structures, systems and components were properly categorized and classified as (a)(1) or (a)(2) in accordance with 10 CFR 50.65. The inspectors reviewed station logs, maintenance work orders, selected surveillance test procedures, and a sample of Condition Reports (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:

- Feedwater Control System.
- Feedwater System

The problem identification and resolution condition reports (CR) reviewed are listed in the attached List of Documents Reviewed.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (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 structures, systems, and components. The following specific activities were reviewed:

- The maintenance risk assessment for replacement of static inverter 1R41K0090A on January 23, 2002. The work was risk significant due to the resulting Division 1 Emergency Diesel Generator (EDG) unavailability.

- The maintenance risk assessment for ESW pumphouse 'A' train ventilation system rework which occurred from January 19 through February 3, 2002. The work was potentially risk significant due to potential impacts on ESW system operability.

- The maintenance risk assessment for planned Division 3 EDG ventilation system work. The inspectors verified the impact of the work on EDG availability was appropriately characterized.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions (71111.14)

.1 Personnel Response to Smoke in Containment

a. Inspection Scope

On January 7, 2002, the licensee identified smoke in containment. The resulting investigation discovered a smoldering lighting string in a locked high radiation area - the reactor water cleanup heat exchanger room. The inspectors reviewed personnel performance including fire brigade and operator response to determine if operators had entered off-normal instructions properly. The inspectors reviewed procedures to determine whether the condition was reportable and whether the event should have been classified as an unusual event.

b. Findings

No findings of significance were identified. Circumstances that caused the lighting string fire in the reactor water cleanup heat exchanger room are discussed in Section 1R23 of this report.

.2 Personnel Response to Scrammed Rod During Surveillance Testing

a. Inspection Scope

The inspectors evaluated operator response to a single scrammed rod which occurred during reactor protection system manual scram channel functionality testing on February 10, 2002. The rod scrammed when operators manually initiated the ‘B’ channel scram signal while a fuse for the rods “A” channel scram solenoid was blown. The inspectors evaluated operator performance to verify that actions were taken in a timely manner in accordance with Off-Normal Instruction (ONI) C51, “Unplanned Change in Reactor Power or Reactivity,” Rev. 8 and that rod recovery actions were appropriate. Additionally, the inspectors reviewed the associated condition report CR 0-0416, “Rod 38-43 Inserted During SVI-C71-T0051.” Finally, the inspectors reviewed procedures to determine whether the event was reportable

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the operability determination associated with ESW pump house ventilation subsystem unavailability. The inspector’s reviewed the licensee’s evaluation, as documented in CR 02-0329, that for ambient conditions existing at the time, the ESW system remained operable during maintenance on the pump house ventilation subsystem.

The inspectors reviewed the operability determination performed for CR 02-0151. The CR described an issue involving potential problems associated with the computer program utilized to evaluate heat exchanger performance. The inspectors reviewed the licensee's evaluation that the potential errors were bounded by analytical uncertainty compensation.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (OWAs)

.1 Nuclear Island Radiologically Restricted Area (RRA) Operator Rounds Accompaniment

a. Inspection Scope

The inspectors accompanied a plant operator, nuclear island RRA, during the performance of a normal rounds tour on February 25, 2002. The inspectors observed all log readings and equipment manipulations made by the operator. Any actions which indicated a potential problem that could increase initiating event frequencies, impact multiple mitigating systems, or affect the ability to respond to plant transients and accidents were considered as possible OWAs. Additionally, the inspectors discussed the effect of active OWAs with the operator.

b. Findings

No findings of significance were identified.

.2 Nuclear Island Non-RRA Operator Rounds Accompaniment

a. Inspection Scope

The inspectors accompanied a plant operator, nuclear island non-RRA, during the performance of a normal rounds tour on February 24, 2002. The inspectors observed all log readings and equipment manipulations made by the operator. Any actions which indicated a potential problem that could increase initiating event frequencies, impact multiple mitigating systems, or affect the ability to respond to plant transients and accidents were considered as possible OWAs.

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 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 post-maintenance testing 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 specific post-maintenance activities evaluated included:

- Diesel Fire Pump Operability Test following planned maintenance
- ESW Pump House Ventilation System Train A Damper Stroking following planned maintenance

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing or reviewed test data for risk-significant systems or components to assess compliance with TS, 10 CFR 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 surveillance instruction, 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 specific surveillance activities assessed included:

- Surveillance Instruction (SVI) E22-T1319, Diesel Generator Start and Load Division 3
- Periodic Test Instruction (PTI) P54-P0045, Fire Detection Instrumentation Functional Test

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications (71111.23)

.1 Temporary Lighting Overheats, Causing Smoke and Fire in Containment

a. Inspection Scope

The inspectors reviewed the engineering justification that allowed a string of temporary lights to be installed in the reactor water cleanup (RWCU) heat exchanger room within the containment. The power cord was part of the lighting system that smoked and caught fire on January 7, 2002. The inspectors reviewed engineering procedures, the licensee's root cause determination that was documented in CR 02-0057, and conducted interviews with station personnel.

b. Findings

GREEN. A Non-Cited Violation (NCV) of 10 CFR 50 Appendix B, Criterion III for not removing a string of temporary lights from the RWCU heat exchanger room after one cycle as specified on Field Clarification Request (FCR) 24726. A temporary 100-foot string of 10 lights was affixed to scaffolding using plastic tie-wraps and installed during Refueling Outage 6 (ended October 23, 1997) in the RWCU heat exchanger room. The scaffolding also supported lead blankets, used to reduce the radiation dose rate around the reactor water cleanup heat exchangers. Since the permanent lights had been disabled, site personnel requested that the temporary lights remain in the room for at least one cycle. An engineer wrote a design change via FCR 024726, "Temporary Lighting in the RWCU Heat Exchanger Room," to justify leaving the lights installed for one cycle.

Field Clarification Request 024726 evaluated the lights under electrical, environmental qualification, mechanical and fire protection considerations. Of particular note was that the environmental qualification evaluation focused on seismic concerns regarding decreased strength from radiation exposure of the plastic tie wraps supporting the lighting. The FCR did not consider radiation aging of the rubber conductor insulation, which, when subjected to high radiation fields, ages and degrades over time. The highest dose in the room exceeded 10 rads.

The engineer who wrote the FCR evaluated installation of these lights for a limited time and required removal of the lights after one cycle. The cycle ended when Refueling Outage 7 started March 27, 1999. Removal of the lights was not tracked in the design change process and the temporary lights remained in the room greater than one cycle. The room contains the following safety related and shutdown components:

- Power cables for reactor and remote shutdown panel pressure and level transmitters and indications
- Power cables for drywell pressure transmitters
- Power cables for suppression pool temperature indication
- Power cables for Reactor Water Cleanup solenoid operator for Valve 1B33-F019, containment spray, containment pool cooling, and safety related instrument air containment isolation valves
- Division 1 Non regenerative heat exchanger tube outlet temperature element

- Division 1 Leak Detection Temperature Elements
- Division 2 Leak Detection Temperature Elements
- Division 1 Hydrogen Igniters
- Division 2 Hydrogen Igniters

These components were in the overhead of the reactor water cleanup heat exchanger room and were not damaged by the fire.

On January 7, 2002, at about 3:00 p.m., a radiation protection technician plugged the lights into a receptacle to illuminate the room so the technician and an operator, who assisted, could conduct a leak inspection inside the heat exchanger room. After completing the inspection, the technician did not unplug the lights, rather, he told decontamination personnel in the area that the lights remained powered. Based on interviews, he assumed they would unplug the fixture. Unfortunately, the decontamination personnel thought the operator would request the oncoming crew to unplug the lights. Therefore, the lighting string remained energized and caught fire. The cord smoldered and burned for several hours until the fire brigade unplugged the lights and discharged a fire extinguisher on the lighting string. Other than heat damage to temporary lead shielding and the lighting string, no equipment damage occurred from the fire.

The performance deficiency associated with this event was that the design control process to remove the temporary lighting after one-cycle as specified on FCR 024726 was inadequate. The finding was greater than minor because it had an actual impact of causing a small fire in a room containing plant operating, fire protection and safety-related equipment. Using the SDP phase 1 worksheet for the seismic, fire, flooding, and severe weather screening criteria, the finding was of very low safety significance because, although the finding contributed to the likelihood of an external event initiator, no equipment was damaged from the event. 10 CFR 50, Appendix B, Criterion III, states, in part, that measures shall be established for the identification and control of design interfaces and for coordination between participating organizations and that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to 10 CFR 50 Appendix B, Criterion III, no design control process (coordination among organizations or established procedures) existed to remove the temporary lights from the reactor water cleanup heat exchanger room after one-cycle as required by FCR 024726. However, because of the very low safety significance and because the issue is in the licensee’s corrective action program, it is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-440/01-16-01). This violation is in the licensee's corrective action program as CR 02-00057.

.2 Electrical Separation Criteria Involving Extension Cords

a. Inspection Scope

Following the temporary lighting fire in the reactor water cleanup heat exchanger room (Section 1R23.1), the inspectors reviewed industry events involving the routing of extension cords. The inspectors reviewed the licensee’s design specifications for maintaining separation criteria between Class 1E and Non-Class 1E power cables. The inspectors conducted tours of the reactor building to determine if the licensee maintained electrical separation between safety-related power sources and extension cords.

b. Findings

GREEN. A Non-Cited Violation of 10 CFR 50 Appendix B, Criterion V for failing to implement the following: 1) Plant Administrative Procedure (PAP)-0204, "Housekeeping/Cleanliness Control," 2) Drawing D-214-004, "Conduit and Cable Tray Separation Criteria," and, 3) Installation Specification 2250, "Electrical Work and Equipment Specification," for installing temporary power cables in a manner to prevent violating the electrical separation criteria.

On January 8, 2002, the inspectors reviewed the NRC event database and found Event 33314 (dated November 26, 1997) that described a condition at the Pilgrim Nuclear Plant regarding extension cords being draped over or tie-wrapped to Class 1E Conduits in the reactor building, in violation of electrical separation criteria. Subsequently, the inspectors reviewed the licensee's investigation of the fire caused by a light string in the reactor water cleanup heat exchanger room (Section 1R23.1). The licensee's report documented a walkdown of the plant to determine if other temporary power cables were overheating. This walkdown did not include evaluation of electrical separation criteria.

The inspectors toured the plant and found two temporary power cords routed in violation of separation requirements. The first cord powered a portable airborne radiation monitor. This power cord and ran from a wall receptacle and routed around a safety-related electrical conduit in the reactor core isolation cooling system room. The power cord was routed within one-inch of conduits containing power cables for the open and close limit switches and solenoid for air operated steam supply drain pot drain line valve 1E51F026. During standby conditions, the valve remains open to drain condensate from a drain pot on the steam line entering the Reactor Core Isolation Cooling (RCIC) turbine. This fail-close valve closes on RCIC startup to prevent diverting steam from the RCIC turbine to the condenser.

The inspectors also identified a temporary power cord draped over safety related cable trays near the ceiling on the 620 foot level of the intermediate building. Safety-related Cable Trays A 660 and A 147 were closest to the power cord and contained power cables for the following systems:

- Control Rod Drive
- Redundant Reactor Control System
- Standby Liquid Control
- Airborne Radiation Monitor System
- Fuel Pool Cleanup and Cooling System
- Containment Vessel and Drywell Purge
- Annulus Exhaust Gas Treatment System
- Emergency Service Water Pump House Ventilation System
- Hydrogen Igniter

- Service Water System
- Emergency Service Water
- Emergency Service Water Screen Wash
- Fire Protection System
- 480 Volt Alternating Current Electrical Distribution System
- Uninterruptible Power Supply
- Direct Current Electrical

The licensee initiated CR 02-00091 to document the inspector's findings and promptly corrected the conditions.

The licensee conducted a walk down of the plant to identify other examples of violations of electrical separation criteria. Condition Report 02-00069 was initiated when the licensee found a power cord wrapped around safety-related conduit 1N27R189A, which supplied power to a pressure transmitter for the feed water leakage control transmitter. The feed water leakage control system is used during a loss of feed water event, when the plant is shut down and when the main feed water isolation valves are closed. The transmitter provides a pressure permissive at 35 psig feed water pressure to allow operators to manually start the feed water leakage control system which provides a water seal to the main feed water isolation valve bonnet area. The water supply is from the residual heat removal keep fill system and provides a post Loss of Cooling Accident seal to prevent escape of airborne contaminants. An interlock signal to automatically stop the feed water control system occurs at 45 psig main feed water pressure. Upon loss of the transmitter, a zero pressure signal is generated, allowing manual operation of the feed water leakage control system. However, the ability to automatically stop the feed water control system is lost, requiring the operator to stop the system manually.

The performance deficiency associated with this event is failure to follow plant procedures which resulted in several examples of electrical separation criteria violations. The finding was greater than minor because, if left uncorrected, routing the extension cords near safety-related power cables increased the likelihood of rendering multiple trains of safety-related equipment inoperable given a fire from those temporary cables. Further, the multiple examples of violating the electrical separation criteria indicated a lack of plant personnel knowledge of the requirement. The finding was determined to be of very low safety significance using the phase 1 SDP screening criteria for seismic, fire, flooding, and severe weather because an actual fire had not occurred that would render the associated equipment inoperable. 10 CFR 50, Appendix B Criterion V, states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Procedure PAP-204, "Housekeeping/Cleanliness Control," Step 6.1.1.8a states that those temporary power cords, including extension cords, are treated as Non-Class 1E Cables and shall conform to separation criteria of ISS-2250. Both Installation Specification 2250 and Drawing D-214-004 requires that the preferred minimum separation distance between Class 1E and Non- Class 1E conduit be one inch. The failure to follow PAP-204 is a violation. However, because of the low safety significance and because the issue is in the licensee's corrective action program, it is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-440/01-16-02). This violation is in the licensee's corrective action program as CRs 02-00091 and 02-00069.

Cornerstone: Emergency Preparedness (EP)

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

a. Inspection Scope

The inspectors reviewed Revisions 14 and 15 of the Perry Nuclear Power Plant emergency Plan to determine whether changes identified in Revision 15 reduced the effectiveness of the licensee’s emergency planning, pending onsite inspection of these changes.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

The inspectors reviewed reported fourth quarter 2001 data for the Unplanned Scrams and Scrams with Loss of Normal Heat Removal PIs using the definitions and guidance contained in Nuclear Energy Institute 99-02, “Regulatory Assessment Indicator Guideline,” Revision 1. The inspectors reviewed station logs, event notification reports, condition reports, licensee cause analysis, and personnel statements for selected 2001 scrams to verify the accuracy of the licensee’s data submission.

b. Findings

The inspectors reviewed the licensee’s determination that the December 15, 2001 scram did not involve a loss of normal heat removal. The inspectors did not reach the same conclusion as the licensee.

On December 15, 2001, a failure of the feedwater control system circuitry resulted in high reactor water level and generated a level 8 scram signal. The Reactor Feed Pump Turbines (RFPTs) tripped, as designed, at Level 8 and reactor water level dropped rapidly (less than 60 seconds) to level 2 due to loss of feedwater. As documented in personnel statements after the event, there was confusion during the initial stages as to what caused the transient. A Reactor Operator (RO) noted trips of both RFPT A&B, noted the Motor Feed Pump (MFP) failed to auto start, and noted that both the red and green indicating lights for the MFP were extinguished. The Unit Supervisor later documented that “it was announced in the control room that we had no feed pumps.” The RO did not attempt to start the MFP. RCIC and High Pressure Core Spray (HPCS) auto started as designed at level 2.

An incident investigator who interviewed the RO after the event told the inspectors that since RCIC and HPCS had auto started and were increasing reactor water level, the RO deferred pursuing immediate troubleshooting of the MFP. A plant operator was, however, dispatched to walkdown the MFP and the MFP breaker (which was accomplished prior to eventually starting the pump). As the transient response continued, a second level 8 trip was received approximately 4 minutes after the first. At some point after the second level 8 trip, the RO determined the indicating light problem was due to a bulb problem.

Control room logs indicate the pump and breaker walkdowns were completed approximately 16 minutes after the “no feed pump announcement.” After the second level 8 cleared, the trips were reset and several minutes later the MFP was started on the startup controller. Per the control room logs, the MFP was started 30 minutes after the start of the transient.

The licensee concluded that all systems functioned as designed and, as a result, there was no loss of normal heat removal. Licensee personnel, regulatory affairs, informed the inspectors that had the operators required the MFP they would have attempted to start it and it would have functioned as designed and therefore was always available.

The inspectors reviewed personnel statements, interviewed an incident investigator, and discussed the event with licensed operators. The inspectors concluded that during a transient such as the December 15, 2001 reactor scram, equipment is only available if operators consider it to be available. Based on personnel statements and control room logs, for some finite period of time, perhaps, fifteen minutes, control room operators believed they had no feed pumps available and took action accordingly based on the indications available to them. The inspectors concluded the uncertainty of the scram initiator combined with the lack of local indicating lights created sufficient doubt as to MFP availability. The operator did not attempt to start the MFP while level fell from level 8 to level 2.

In hindsight, with the transient fully understood, the inspectors agreed that, with the exception of a light bulb, equipment functioned as designed. The MFP did not start on loss of RFPTs because of the sealed-in initial level 8 scram signal; however, this was not initially recognized by the operators. The operators would need to recognize and reset the level 8 signal to allow the MFP to start. The fact remains, however, that the operators considered and treated the MFP as unavailable and utilized an alternate method of heat removal (RCIC).

NEI 99-02, Rev. 1 guidance stated that the “indicator monitors that subset of unplanned and planned automatic and manual scrams that necessitate the use of mitigating systems and are therefore more risk-significant than uncomplicated scrams.” The guidance also defined normal heat removal path as “the path from the main condenser through the main feedwater system, steam generators (or reactor vessel), the main steam isolation valves, and back to the main condenser.” Finally, the guidance stated that complete loss of all main feedwater constitutes a loss of normal heat removal path condition if it "cannot be easily recovered from the control room without the need for diagnosis or repair."

With respect to the NEI guidance, the inspectors noted that the December 15 event necessitated the use of mitigating systems in that RCIC and HPCS were used to restore reactor water level. Further, the operators announced the unavailability of the feedwater system and did not attempt to use it until the cause of the failure to automatically start on trip of the RFPTs was understood, the indicating light issue was resolved, and a field walkdown of the pump and breaker were completed. The inspectors concluded that diagnosis was required prior to recovering the normal heat removal path and that, therefore, this event should be counted as a scram with loss of normal heat removal.

Since this event, if counted as a scram with loss of normal heat removal, would result in the PI crossing the green to white threshold, the inspectors considered this issue an Unresolved Item (URI) (URI 50-440/01-16-03). The inspectors submitted a Reactor Oversight Process Feedback form in accordance with established procedures to document the disagreement with the licensee.

4OA3 Event Follow-up(71153)

(Closed) Licensee Event Report (LER) 50-440/2001-02: Oscillation Power Range Monitors Inoperable Due to Non-Conservative Setpoints. On June 27, 2001, General Electric notified the licensee that oscillation power range monitor (OPRM) instrumentation scram setpoints were non-conservative due to non-conservative analysis. Upon notification, the licensee declared the system inoperable, completed TS required actions which required initiation of an alternate method to detect and suppress thermal hydraulic instability oscillations, and entered the issue into their corrective action program as CR 01-2582. The inspectors noted that operators were trained to monitor for potential instability (power-to-flow montioring) both prior to and after OPRM installation. The inspector's review identified no new issues. This is a minor violation not subject to formal enforcement.

4OA6 Meetings

Exit Meeting

The inspectors presented the inspection results to Mr. Guy Campbell, Site Vice President and other members of licensee management at the conclusion of the inspection on February 26, 2002. The licensee acknowledged the findings presented. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee

G. Campbell, Vice President-Nuclear
B. Boles, Operations Manager
G. Dunn, Manager, Regulatory Affairs
D. Gudger, Supervisor, Compliance
T. Lentz, Manager, Design Engineering
K. Ostrowski, Director, Nuclear Services Department
D. Phillips, Manager, Plant Engineering
T. Rausch, Director, Nuclear Maintenance Department
W. Kanda, General Manager, Nuclear Power Plant Department
R. Strohl, Superintendent, Plant Operations

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-440/01-16-01	NCV	Failure to Remove Temporary Lights From the Reactor Water Cleanup Heat Exchanger Room After One Cycle
50-440/01-16-02	NCV	Failure to Follow Procedures for Maintaining Electrical Separation Criteria
50-440/01-16-03	URI	Scrams With Loss of Normal Heat Removal Reporting Criteria

Closed

50-440/2001-02	LER	Oscillation Power Range Monitors Inoperable Due to Non-Conservative Setpoints
50-440/01-16-01	NCV	Failure to Remove Temporary Lights From the Reactor Water Cleanup Heat Exchanger Room After One Cycle
50-440/01-16-02	NCV	Failure to Follow Procedures for Maintaining Electrical Separation Criteria

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
CR	Condition Report
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FCR	Field Clarification Request
FENOC	FirstEnergy Nuclear Operating Company
HPCS	High Pressure Core Spray
LER	Licensee Event Report
MFP	Motor Feed Pump
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
ONI	Off-Normal Instruction
OPRM	Oscillation Power Range Monitor
OWA	Operator Workarounds
PAP	Plant Administrative Procedure
PARS	Publicly Available Records
PI	Performance Indicator
PTI	Periodic Test Instruction
RCIC	Reactor Core Isolation Cooling
RFPT	Reactor Feed Pump Turbines
RO	Reactor Operator
RRA	Radiologically Restricted Area
SDP	Significance Determination Process
SVI	Surveillance Instruction
TS	Technical Specifications
USAR	Updated Safety Analysis Report

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

SOI-P45/49	Emergency Service Water and Screen Wash Systems, Rev. 2	September 19, 1995
VLI-P45	Emergency Service Water System, Rev. 4	August 22, 1989
Drawing D-302-791	Emergency Service Water System	July 25, 2001
Drawing D-302-792	Emergency Service Water System	April 17, 2000
Drawing D-912-630	Emergency Service Water Pumphouse Ventilation System	April 17, 2000
USAR Section 9.2.1	Emergency Service Water System	
USAR Section 9.4.5	Engineered Safety Features Ventilation System	
TS 3.7.1	Emergency Service Water (ESW) System - Divisions 1 and 2	

1R05 Fire Protection

Drawing E-023-015	Fire Protection Evaluation - Control Complex and Diesel Generator Roof Plan, El. 638'-6" and 646'-6"	September 2001
Drawing E-023-018	Fire Protection Evaluation - Unit 1 Reactor Building Plan, El. 654'-0"	March 1991
Drawing E-023-002	Fire Protection Evaluation - Unit 1 Auxiliary and Reactor Building Plan, El. 574'-10"	September 2001
USAR Section 9A.4.2.1.1.2	Fire Zone 1RB-1b	
USAR Section 9A.4.2.1.1	Fire Zone 1AB-1a	
USAR Section 9A.4.2.1.6	Fire Zone 1AB-1f	
USAR Section 9A.4.4.4.1.1	Fire Zone 1CC-4a	

USAR Section 9A.4.4.4.1.5	Fire Zone 1CC-4e	
National Fire Protection Association	Fire Protection Handbook, Edition 15.	
Condition Report 02-0057	Extension Cord Overheats Causing Smoke in Containment	January 7, 2002

1R11 Licensed Operator Requalification

ONI C51	Unplanned Change in Reactor Power or Reactivity, Rev. 8	March 14, 2001
ONI C71-1	Reactor Scram, Rev. 3	May 21, 2001
ONI C85-2	Pressure Regulator Failure - Open, Rev. 3	May 22, 1989
ONI ZZZ-1	Tornado or High Wind, Rev. 2	June 30, 1995

1R12 Maintenance Rule Implementation

CR 01-0060	RFPT ‘B’ Control Power Fuse Blown	January 7, 2001
CR 01-0440	RFPT ‘B’ Work	February 7, 2001
CR 01-0864	Motor Feedwater Pump Cracks	February 26, 2001
CR 01-1113	FM Found in Valve 1N27F160B	March 5, 2001
CR 01-1228	Motor Feed Pump Did Not Trip as Expected	March 8, 2001
CR 01-1586	Maintenance Rule Evaluation Required on Reactor Feedwater Booster Pump	March 21, 2001
CR 01-1606	RPV Level Control	March 22, 2001
CR 01-1983	Motor Feedpump Run With Minimum Flow Of Approximately 800GPM	April 29, 2001
CR 01-2081	Repeat Failure of Motor Feed Pump Recirc Flow Control Valve to Stroke	May 4, 2001
CR 01-2112	Maint Rule Evaluation is Required on Reactor Feed Pump ‘A’ Min Flow Valve	May 7, 2001
CR 01-2779	1N27F0170 [Motor Feed Pump Recirc Valve], Failed to Stroke Properly	July 17, 2001
CR 01-2827	Damaged 2nd Stage Diffuser Vanes	July 21, 2001

CR 01-3966	Collective Maintenance Rule Evaluation of Motor Feed Pump Component Failures	November 14, 2001
System Health Report	Feedwater Control System Status Report	1st Quarter 2001
System Health Report	Feedwater Control System Status Report	2nd Quarter 2001
System Health Report	Feedwater Control System Status Report	3rd Quarter 2001
System Health Report	Feedwater Control System Status Report	4th Quarter 2001
System Health Report	Feedwater System Status Report	1st Quarter 2001
System Health Report	Feedwater System Status Report	2nd Quarter 2001
System Health Report	Feedwater System Status Report	3rd Quarter 2001
System Health Report	Feedwater System Status Report	4th Quarter 2001
PAP-1125	Monitoring the Effectiveness of the Maintenance Program Plan, Rev. 6	April 4, 2001
Logs	Control Room Logs	01/01/01 - 12/31/01
NUMARC 93-01, Revision 2	Nuclear Energy Institute Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

	Week 9, Period 4 Forecast Risk Profile	January 21, 2002
	Week 10, Period 4 Forecast Risk Profile	January 28, 2002
	Week 11, Period 4 Forecast Risk Profile	February 4, 2002
WO 02-000122-000	Replace Static Inverter 1R41K00090A	
WO 00-005257-000	Rework Damper IAW SMRF 00-5027	
WO 01-16229-000	Replace the Auto and Manual Status Indicating Lamps for Div. 3 DG Room Fan 2C	

SOI-M43	Diesel Generator Building Ventilation System, Rev. 5	May 24, 1990
USAR 9.4.5.2.4	Diesel Generator Building Ventilation System	
PAP 1924	On-Line Safety Assessment and Configuration Risk Management, Rev. 2	November 30, 2000

1R14 Personnel Performance During Non-Routine Evolutions

	Plant Narrative Logs	January 7, 2002
CR 02-0057	Extension Cord Overheats Causing Smoke in Containment	January 7, 2002
NUREG 1022	Event Reporting Guidelines 10 CFR 50.72 and 50.73, Rev. 2	
	Plant Narrative Logs	February 10, 2002
ONI C51	Unplanned Change in Reactor Power or Reactivity, Rev. 8	March 14, 2001
SVI-C71-T0051	Reactor Protection System Manual Scram Channel Functional, Rev. 2	April 27, 1988
CR 0-0416	Rod 38-43 Inserted During SVI-C71-T0051	February 10, 2002

1R19 Post-Maintenance Testing

WO 01–17117-000	Repair Air Leak at Gasket on Intake Manifold Blanking Plate	November 29, 2001
SOI-P54 (WTR)	Fire Protection System Water, Rev. 0	August 14, 2001
USAR 9.5.1	Fire Protection System	
WO 00-005257-000	Rework Damper IAW SMRF 00-5027	
PTI-M32-P0004	Emergency Service Water Pump House Ventilation System Train A Damper Stroking, Rev. 0	August 7, 1987
USAR Section 9.4.5	Engineered Safety Features Ventilation System	
	NH90 Series Hydramotors Maintenance Manual, Rev. 9	March 19, 1997

1R22 Surveillance Testing

SVI-E22-T1319	Diesel Generator Start and Load Div. 3, Rev. 10	December 14, 2000
SOI-E22B	Division 3 Diesel Generator, Rev. 6	May 11, 1995
TS 3.8.1	AC Sources - Operating	
USAR Section 8.3.1	Onsite Power Systems	
PTI-P54-P0050	Unit One Fire Detection Instrumentation Functional Test for SDP-1H51-P929	January 9, 2002
PTI-P54-P0045C	Fire Detection Instrumentation Functional test for SDP-H51-P219	January 8, 2002
NFPA 72E	Standard for Automatic Fire Detectors	1974
NFPA 72	Chapter 7, Inspection, Testing and Maintenance	5-79

1R23 Temporary Modifications

Field Clarification Request 024726	Temporary Lighting in the RWCU Heat Exchanger Room	October 13, 1997
CR 02-0057	Extension Cord Overheats Causing Smoke in Containment	January 7, 2002
	Plant Narrative Logs	January 7, 2002
NEI 0674	Specification Changes, Rev. 8	
CR 02-0069	Temporary Power Cable Separation Violation	January 8, 2002
CR 02-0091	Temporary Extension Cord Routing	January 9, 2002
PAP 0204	Housekeeping/Cleanliness Control, Rev. 10	
Drawing D-214-004	Conduit and Cable Tray Separation Criteria, Rev. T	
Installation Specification 2250	Electrical Work and Equipment Specification, Rev. 1	

1EP4 Emergency Action Level and Emergency Plan Changes

	Perry Nuclear Power Plant Emergency Plan, Rev. 14	
	Perry Nuclear Power Plant Emergency Plan, Rev. 15	

4OA3 Event Follow-up

CR 01-2582	OPRM Operability - Potentially Non-Conservative Stability Reload Calculation	June 27, 2001
IOI-3	Power Changes, Rev. 7	November 8, 1993
TS 3.3.1.3	Oscillation Power Range Monitor (OPRM) Instrumentation	