January 18, 2001

Mr. John K. Wood
Vice President - Nuclear
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-440/00-14(DRP)

Dear Mr. Wood:

On December 31, 2000, the NRC completed an inspection at the Perry Nuclear Power Plant. The enclosed report presents the results of that inspection which were discussed on January 9, 2001, 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 one issue of very low safety significance (Green). The issue was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because the issue has been entered into your corrective action program, the violation is being treated as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with a copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the Perry facility.

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Sincerely,

/RA/

Thomas J. Kozak, Chief
Reactor Projects Branch 4

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

Enclosure: Inspection Report 50-440/00-14(DRP)

cc w/encl: B. Saunders, President - FENOC
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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	50-440/00-14
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	P.O. Box 97 A200 Perry, OH 44081
Dates:	November 16 - December 31, 2000
Inspectors:	C. Lipa, Senior Resident Inspector R. Vogt-Lowell, Resident Inspector S. Sheldon, Reactor Engineer, Region III
Approved by:	Thomas J. Kozak, Chief Reactor Projects Branch 4 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner, which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

IR 05000440-00-14; on 11/16-12/31/2000; FirstEnergy Nuclear Operating Company; Perry Nuclear Power Plant; Event Followup.

The inspection was conducted by resident inspectors and a regional inspector. This inspection identified one green issue, which was a Non-Cited Violation. The significance of the issue is indicated by the color (green, white, yellow, red) and was determined by the Significance Determination Process.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- Green. As a result of an inadequate test procedure, one safety relief valve unexpectedly opened during testing on December 18, 2000. The procedure failed to provide instructions to reset the low low set logic before applying an input signal to the trip unit. A Non-Cited Violation was identified for the inadequate procedure.

  The finding was of very low safety significance because, although the issue increased the frequency of an initiating event, all mitigation systems were available during the event. The inspectors used the Perry-specific worksheets in the Phase 2 Significance Determination Process (SDP) analysis to assess the safety significance of the issue. (Section 4OA3)

Report Details

Summary of Plant Status: The plant began this inspection period with Unit 1 administratively limited to 98.5 percent power. There were several short downpowers to about 75 percent reactor power during the inspection period for sequence exchanges and repair of a moisture separator reheater drain line. Beginning around November 27, 2000, full reactor power could no longer be maintained with the maximum available core flow and standard balance of plant configuration. As a result, the reactor started to coast down in power. By the end of the inspection period on December 31, the reactor was at 91 percent power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

Reactor (R)

1R01 Adverse Weather Protection

a. Inspection Scope (71111.01)

The inspectors completed a review of the licensee's cold weather readiness via an examination of the following documents:

PNSD Operational Surveillance Report No.: 00-044, "Cold Weather Preparation"

CR 00-3728, "Inspection Results of CST Piping"

Through this review, the inspectors evaluated the adequacy of the protection provided for components/sensing lines located in areas exposed to outside weather or located in outside structures. Included in the review was an evaluation of the adequacy of heat tracing for cold weather protection of piping and equipment, as well as the adequacy of cold weather protection of CST level sensing lines. The inspectors verified that the licensee has identified weather related problems that could affect mitigating systems and support systems in their corrective action program and that the problems are being properly addressed for resolution.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment-Partial Walkdown

a. Inspection Scope (71111.04)

On December 18 and 19, 2000, the inspectors reviewed equipment alignment on the "B" train of the control complex chilled water system and control room ventilation while train "A" was inoperable for maintenance. The inspectors performed the walkdown of the system to verify equipment alignment and identify any discrepancies that could impact the function of the system and therefore potentially increase overall risk to the plant. The inspectors ensured that the configuration of the train was in accordance with applicable operating procedures and checklists and appropriate for the existing conditions. The inspectors reviewed the system operating instructions and valve lineup instructions prior to walking down the train.

b. Findings

No findings of significance were identified.

Equipment Alignment-Complete Walkdown of Emergency Closed Cooling System (ECC)

a. Inspection Scope (71111.04)

The inspectors reviewed equipment alignment on the ECC system to identify any discrepancies that could impact the function of the system and therefore potentially increase overall risk to the plant. The inspectors reviewed open work requests and condition reports for the system. The inspectors performed a complete walkdown of both trains of ECC. During the walkdown, a portion of the system was tagged out for maintenance and subsequently returned to service. The inspectors ensured that the configuration of the system was in accordance with applicable operating procedures and checklists, and appropriate for the existing conditions.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope (71111.05)

The inspectors walked down selected risk significant areas looking for any fire protection issues related to: the control of transient combustibles, ignition sources, fire detection equipment manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, and barriers to fire propagation. Areas walked down were the 3 Divisional EDG rooms and the 620' elevation in the Control Complex building, Intermediate building, Auxiliary building, and Turbine Power Complex building.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

The inspectors reviewed activities associated with a heat exchanger performance test on the “B” residual heat removal (RHR) system, conducted from November 28 through 30, 2000. This inspection included a review of the test set-up and control established in Periodic Test Instruction, PTI-E12-P0003, “RHR Heat Exchangers B and D Performance

Testing," Revision 5. The inspectors observed the prerequisites for the test and actual performance of the test in the plant. The licensee had preliminarily reviewed the data and found the results to be acceptable; however, final review of the data and evaluation of results was not expected to be completed until mid-January, 2001. The inspectors planned to review the final conclusions regarding heat exchanger performance in the next inspection period.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope (71111.11)

On December 5, 2000, the inspectors observed two licensed operator evaluated scenarios as part of the annual requalification program. The inspectors verified that training personnel properly evaluated the performance of the operators during the critique.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope (71111.14)

On November 21, 2000, the inspectors observed operator performance during a non-routine evolution associated with work on the reactor water level control system. The scope of the work order was to replace a circuit card on the master level controller. In order to facilitate the activity, the reactor water level control system was realigned from automatic control to manual control and then realigned back to automatic control following the activity. The inspectors observed that operators reviewed and followed appropriate procedures and held thorough briefings as needed.

On December 18, 2000, during safety/relief valve surveillance testing, a safety/relief valve unexpectedly opened. The inspectors observed part of the licensee's response which was in accordance with Off-Normal Instructions (ONI) ONI-B21-1, "SRV Inadvertent Opening/Stuck Open" and ONI-C51, "Unplanned Change in Reactor Power or Reactivity." The operators reduced plant power to 90 percent as specified in the ONI and closed the SRV without incident. The inspectors also reviewed plant logs and computer data to verify proper licensed operator response to the event. The licensee documented the condition on CR 00-3901. See Section 4OA3 for further discussion of this issue and the associated Non-Cited Violation.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

- The inspectors reviewed an operability evaluation involving the safety-related Control Complex Chillers (P47). The inspectors reviewed CR 00-2915 and CR 00-3095, which were initiated to address a low cooling water flow condition to the Control Complex Chillers. The engineering review of this condition evaluated the minimum Emergency Closed Cooling (ECC) system water flow required to adequately cool the chillers during a postulated loss of coolant accident, which places the largest heat load on the chillers. The evaluation utilized Calculation P47-1, Rev 0, "P47 System Operating Temperatures", and Calculation P42-025, Rev 2, "ESW Winter Bypass Line Flow to ECC." The inspectors verified that the operability evaluation was technically justified and that the appropriate compensatory measures were taken.

- The inspectors reviewed CR 00-2538 which contained an operability evaluation to address past and present operability of the ECC system with high as-found seating torque on one motor operated butterfly isolation valve in the system. The valve had been tested on December 13, 1999; however, the data had been incorrectly analyzed and the potential operability issue was not discovered until August 22, 2000 as part of a follow-up to an NRC inspection finding. The valve is an isolation valve between the safety-related ECC system and the non safety-related nuclear closed cooling (NCC) system. The licensee had performed additional dynamic flow testing of the valve and also revised calculation MOV-0P42-05 to reduce the required differential pressure requirement for the valve. The revised calculation assumed that catastrophic failure of the NCC piping downstream of the isolation valves would not have occurred and only the effects of a leakage crack need to be postulated. The inspectors questioned the basis for the revised differential pressure calculation and planned to submit a request for assistance to the Office of Nuclear Reactor Regulation. Pending NRR review of the assumptions in the revised differential pressure calculation, this will be an Unresolved Item **(URI 50-440/2000-014-01(DRP))**. The inspectors verified that the valve was closed to maintain ECC system operability during additional investigation of the issue.

- The inspectors reviewed an operability evaluation involving the Reactor Core Isolation Cooling (RCIC) system (E51). The inspectors reviewed CR 00-2440, which was initiated to address the lack of full thread engagement on three RCIC system air operated valves (AOVs). This operability determination was chosen due to the potential for leakage outside containment should concurrent failure of two of these valves (1E51F0025 and 1E51F0026) occur. The license performed a minimum thread engagement/bolt stress calculation to determine the actual stress levels for the worst case yoke-to-bonnet stud. The calculation revealed that the tensile stress on the stud and the shear stress on the stud threads and the nut were within the allowable values for the material type involved. The inspectors verified that the operability evaluation was technically justified and that the appropriate corrective actions for long term resolution of the issue were taken.

- The inspectors reviewed an operability evaluation associated with CR 00-2946 relating to the installation of a potentially incorrect relief valve on the Reactor Recirculation Hydraulic Power Unit (HPU) subloop A2 of the B33 Reactor Recirculation system. This HPU provides the motive force for the recirculation loop flow control valve (FCV) utilized to change reactor power level. Technical Specification (TS) 3.4.2 requires that FCVs fail "as-is" on loss of hydraulic power and also requires that FCV motion be limited to less than 11 percent of stroke per minute. As part of the operability evaluation the licensee confirmed that the correct relief valve had been installed and therefore concluded that there was no impact on the functions required by the TS. The inspectors verified that the operability evaluation was technically correct and that no adverse condition existed and no compensatory measures were required.

- The inspectors reviewed an operability evaluation associated with CR 00-2947 relating to a loose tubing strap on the Division 2 Emergency Diesel Generator (EDG). The tubing involved is mainly associated with the diesel trip instrumentation. If the tubing were to break, either the diesel would not start due to insufficient air pressure or it would shutdown due to activation of the trip functions. One of the tubing lines measures lube oil pressure. If the line were to break, the diesel would lose lube oil, and would eventually shutdown. The licensee measured the distance between existing, properly installed support straps on either side of the loose tubing strap, and recalculated allowable spans based on that measurement. The stress values resulting from the calculation were found to be acceptable and below the value that would cause breakage of the tubing following a seismic event. The inspectors verified that the operability evaluation was technically correct and that no adverse condition existed and no compensatory measures were required.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope (71111.16)

The inspectors reviewed the licensee's operator workaround list and selected the item discussed below for review.

- Inadequate Position Indication for Central Deicing Valves: This workaround involved the failed position indication for the mid-way position for central deicing valves (circulating water system) 1N71-F355, F365, F360, and F370. When making valve position adjustments, the operators were required to time the valves to determine the mid-way position because the mid position lights were failed. This workaround had no affect on system function or operator action in responding to an event. The inspectors verified that replacement of the valve operators was scheduled for the next refueling outage as Work Order 00-1479.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19)

The inspectors selected the two activities listed below for review. The work package was reviewed to determine test requirements and then the test was observed to verify that test requirements were met. The inspectors also reviewed other documents, such as the USAR, TSs, and Maintenance Procedures to determine that the testing was sufficient to demonstrate that the systems and components were capable of performing their intended safety functions.

- On November 17, the inspectors selected Work Order 00-9541, that controlled maintenance on the Division 2 emergency diesel generator (EDG), for review. The post maintenance testing consisted of starting and operating the Division 2 EDG for 5 hours at different loads, starting with no load for 15 minutes and then ending with operating at 5800KW for 2 hours. The inspectors observed portions of the testing from the control room and portions from the diesel room and verified that the testing was properly completed.

- On November 17, the inspectors selected Work Order 00-9535, that controlled maintenance on the jacket water/lubricating oil heat exchanger, for review. The post maintenance testing consisted of verifying no leaks at the heat exchanger flanged connections during operation of the EDG. The inspectors observed that the testing was properly completed.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspectors witnessed the below listed surveillance tests and verified that requirements were met and were consistent with applicable sections of TSs, USAR, and Plant Procedures. The inspectors verified that test control was properly coordinated with the control room and that the testing was properly performed in the sequence specified in the surveillance instruction. Also, the inspectors verified that test equipment was properly calibrated and installed to support the surveillance tests.

- SVI-R43-T1318, “Diesel Generator Start and Load, Division 2,” on November 18, 2000.

- SVI-E51-T2001, “RCIC Pump and Valve Operability Test,” on November 29, 2000.

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications

a. Inspection Scope (71111.23)

The inspectors selected a recent Temporary Modification (TM) for review based on the potential impact on the leakage detection system for drywell unidentified leakage. Temporary Modification 1-00-0006 was initiated due to degradation of the drywell equipment drain sump low level switch (1G61-N0025). The failed switch prevents the sump pump from automatically starting on high sump level. The TM installed a jumper to allow operation of the sump pump in the manual mode. The inspectors reviewed Work Order 00-9178 associated with this TM and noted that repair of the switch was scheduled for the next refueling outage. This would allow clearing the TM and restoration of the condition back to design. The inspectors reviewed other associated documents, such as the TM Technical Evaluation, 10 CFR 50.59 Applicability Check, and Design Interface Evaluation. The inspectors verified that the TM did not adversely affect any important safety functions.

b. Findings

No findings of significance were identified.

Emergency Preparedness (EP)

1EP6 Drill Evaluation

a. Inspection Scope (71114.06)

On December 5, 2000, the inspectors observed 2 simulator-based training evolutions, which the licensee had declared as contributing to the Performance Indicator for drill performance. The inspectors determined that the correct classification and notifications were made within the required times during the evolutions.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Cornerstones: Mitigating Systems

a. Inspection Scope (71151)

The inspectors verified the licensee's data for the Performance Indicator (PI) listed below. For the time periods indicated, the inspectors reviewed Operator Logs and Daily Plant Status Reports and conducted interviews with licensee personnel to review the data collected and reported for the indicator. The inspectors also verified that the licensee's data met the guidance in NEI 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 0.

- Residual Heat Removal (includes Shutdown Cooling and Suppression Pool Cooling), Q2-Q3 2000

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

Inadvertent Safety Relief Valve (SRV) Opening During Testing

a. Inspection Scope (71153)

The inspectors observed the licensee's response to an unexpected SRV opening on December 18, 2000. The inspectors arrived in the control room shortly after the SRV was closed and observed the follow-up licensed operator actions, including operator briefings, actions required by the off-normal procedures, and the monitoring of plant conditions. As part of the follow-up to this event, the inspectors observed plant chart recorders, reviewed off-normal procedure requirements and the surveillance test procedure, and held discussions with plant personnel.

The following documents were reviewed:

ONI-B21-1, "SRV Inadvertent Opening/Stuck Open"

ONI-C51, "Unplanned Change in Reactor Power or Reactivity"

SVI-B21-T0369F, "SRV Pressure Actuation Channel F Functional Test for 1B21-N668F"

CR 00-3901, "Unanticipated Opening of SRV 1B21F0051D During Surveillance Test"

CR 00-3903, "SRV Weeping After Being Opened and Closed"

b. Findings

During the performance of SVI-B21-T0369 on December 18, 2000, SRV 1B21F0051D opened unexpectedly. The licensed operator response to the event included, promptly following ONI-B21-1, which required reducing power to 90 percent and then closing the SRV. The SRV was closed successfully within 2 minutes of opening. As expected, there was an increase in the suppression pool temperature and level, however, these parameters remained within TS limits. The review of operator performance in response to this event is discussed in Section 1R14 of this report.

The licensee's investigation determined the cause to be an inadequate Surveillance Instruction (SVI). The SVI did not have a step to reset the low low set logic before applying an input signal to the trip unit. The licensee's investigation also determined that there were missed opportunities to prevent the event. One opportunity was during the identical testing the previous week, when licensed operators and instrumentation technicians questioned the low low set logic lights being lit and thoroughly evaluated the condition and decided to reset the logic before continuing with the test. This action was not documented and the procedure weakness was not recognized at the time. A second opportunity was during the testing on December 18, when operators questioned the low low set logic lights, but they did not throughly review the potential plant impact before continuing with the next steps of the procedure.

The inspectors used Phase 1 and 2 of the Significance Determination Process (Manual Chapter 0609) to assess the significance of the issue. Since the inadequate procedure resulted in the SRV opening during the test, the issue had an impact on safety and increased the frequency of an initiating event. However, all mitigation systems were available during the event and using the worksheets in the Phase 2 analysis, the finding was determined to be of very low safety significance (Green).

Technical Specification 5.4.1.a requires written procedures be established and maintained covering the activities specified in Regulatory Guide 1.33, Appendix A. Item 8b of Appendix A requires procedures for the surveillance tests listed in the TS. The inadequate Surveillance Procedure is a violation of TS 5.4.1.a. However, because of the very low safety significance, this issue is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (**NCV 50-440/2000-014-02(DRP)**)). This violation is in the licensee's corrective action program as Condition Report 00-3901.

4OA6 Management Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. Wood, Vice President, Nuclear, and other members of licensee management at the conclusion of the inspection on January 9, 2001. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee

J. Wood, Vice President-Nuclear
B. Boles, Operations Manager
N. Bonner, Director, Nuclear Maintenance Department
S. Davis, Superintendent, Plant Operations
G. Dunn, Manager, Regulatory Affairs
D. Gudger, Supervisor, Compliance
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T. Rausch, Director, Nuclear Engineering Department
K. Russell, Compliance Engineer
R. Schrauder, General Manager, Nuclear Power Plant Department

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

440/2000-014-01	URI	Inadequate evaluation of MOV test data
440/2000-014-02	NCV	Inadequate Test Procedure for SRV Logic

Closed

440/2000-014-02	NCV	Inadequate Test Procedure for SRV Logic

Discussed

None.

LIST OF ACRONYMS AND INITIALISMS USED

ADAMS	Agencywide Documents Access and Management System
AOV	Air Operated Valve
CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
ECC	Emergency Closed Cooling
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FCV	Flow Control Valve
FENOC	FirstEnergy Nuclear Operating Company
HPU	Hydraulic Power Unit
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
ONI	Off-Normal Instruction
PARS	Publicly Available Records
PI	Performance Indicator
PNSD	Perry Nuclear Services Department
PTI	Periodic Test Instruction
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RHR	Residual Heat Removal
SDP	Significance Determination Process
SRV	Safety Relief Valve
SVI	Surveillance Instruction
TS	Technical Specification
USAR	Updated Safety Analysis Report