March 16, 2001

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

SUBJECT: PERRY NUCLEAR POWER PLANT
NRC INSPECTION REPORT 50-440/01-02(DRP)

Dear Mr. Wood:

On February 24, 2001, the NRC completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on February 28, 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.

There was one finding of very low safety significance (Green) identified in the report.

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/

Thomas J. Kozak, Chief
Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 50-440/01-02(DRP)

See Attached Distribution

cc w/encl: B. Saunders, President - FENOC
N. Bonner, Director, Nuclear
Maintenance Department
G. Dunn, Manager, Regulatory Affairs
K. Ostrowski, Director, Nuclear
Services Department
T. Rausch, Director, Nuclear
Engineering Department
R. Schrauder, General Manager,
Nuclear Power Plant Department
A. Schriber, Chairman, Ohio Public
Utilities Commission
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:	50-440/01-02
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 through February 24, 2001
Inspectors:	C. Lipa, Senior Resident Inspector R. Vogt-Lowell, Resident Inspector
Approved by:	Thomas J. Kozak, Chief 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 05000440-01-02(DRP); on 01/01 through 02/24/2001; FirstEnergy Nuclear Operating Company; Perry Nuclear Power Plant; Event Follow-up.

The inspection was conducted by resident inspectors. This inspection identified one Green issue. The significance of the issue is indicated by the color (Green, White, Yellow, Red) and was determined by the Significance Determination Process.

Inspector Identified Findings

Cornerstone: Mitigating Systems

Green. As a result of inadequate engineering reviews, the inventory in the suppression pool makeup system was potentially impacted when the inclined fuel transfer system blind flange was removed at power. This issue was reported to the NRC as LER 50-440/2000-001.

The finding was of very low safety significance because, although the issue potentially impacted a mitigating system, the duration was small and there was a nonsafety-related valve in the system that maintained the water inventory (Section 4OA3).

Report Details

Summary of Plant Status:

On January 1, 2001, the reactor was in a coast down condition, with reactor power at 91 percent. The licensee implemented weekly downpowers to conduct control rod exercise testing as a conservative measure due to a suspected fuel defect. On January 5 to January 7, the plant power was reduced to approximately 60 percent for a fuel defect localization test. One fuel defect was identified and Control Rod 42-31 was inserted for fuel defect suppression. On January 21, the licensee reduced power to remove the "B" reactor feed pump from service for planned maintenance and to remove the second stage reheaters from service. When power was restored later on January 21, the feed water pump configuration was one motor feed pump and one turbine feed pump. The licensee implemented an administrative power limit based on feed water flow rate limitations of approximately 92 percent reactor power. On February 16 at 11:00 a.m. the licensee began reducing reactor power to prepare to shut down for Refueling Outage 8 and at 12:04 a.m. on February 17, the licensee removed the main generator from the grid and then subsequently scrammed the reactor. At the end of inspection period on February 24, the reactor was in Mode 5, Refueling.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment-Partial Walkdowns

a. Inspection Scope (71111.04)

On January 22, 2001, the inspectors reviewed equipment alignment on the "B" train of the emergency closed cooling water system while motor operated valve testing was in progress on the "A" train. The inspectors performed the walkdown of the system to verify equipment alignment and to identify any discrepancies that could impact the function of the system and therefore potentially increase overall risk to the plant. The inspectors inspected the configuration of the train to determine whether it was in accordance with applicable operating procedures and appropriate for the existing conditions. The inspectors reviewed the system operating instructions and valve lineup instructions prior to conducting the walkdown.

On January 25 and 26, the inspectors reviewed equipment alignment on the "B" train of the residual heat removal system while snubber work was in progress on the "A" train. The inspectors performed the walkdown of the system to verify equipment alignment and to identify any discrepancies that could impact the function of the system and therefore potentially increase overall risk to the plant. The inspectors inspected the configuration of the train to determine whether it was in accordance with applicable operating procedures and appropriate for the existing conditions. The inspectors reviewed the system operating instructions and valve lineup instructions prior to conducting the walkdown.

b. Findings

No findings of significance were identified.

Equipment Alignment-Complete Walkdown of High Pressure Core Spray System (HPCS)

a. Inspection Scope (71111.04)

The inspectors reviewed equipment alignment on the high pressure core spray (HPCS) 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 the accessible portions of the HPCS system, the HPCS Diesel Generator, and the HPCS Emergency Service Water system. The inspectors compared the configuration of the systems to the applicable operating procedures to determine whether the configuration was appropriate for the existing conditions. Portions of the following documents were used to confirm proper system status and configuration:

SOI-E22A, High Pressure Core Spray System (Unit 1)
SOI-E22B, Division 3 Diesel Generator (Unit 1)
SOI-P45/49, Emergency Service Water and Screen Wash System
VLI-R44/E22B, Division 3 Diesel Generator Starting Air System (Unit 1)
VLI-R45/E22B, Division 3 Diesel Generator Fuel Oil System (Unit 1)
VLI-R46/E22B, Division 3 Diesel Generator Jacket Water System (Unit 1)
VLI-R47/E22B, Division 3 Diesel Generator Lube Oil System (Unit 1)
DCP 98-5062, HPCS DG Air Dryers - Dewpoint Requirements
PTI-R44-P0001C, Division 3 HPCS Diesel Generator Starting Air System

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 emergency service water pump house and the “B” residual heat removal pump room.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

The inspectors reviewed equipment issues, surveillance test failures, and other performance problems for the systems listed below. The inspectors reviewed whether the components were properly scoped in accordance with the Maintenance Rule, whether any failures were properly characterized, and whether the performance criteria were appropriate. In addition, the inspectors reviewed condition reports associated with maintenance rule to determine if the licensee was identifying problems and entering them in the corrective action program. The problem identification and resolution (PIR) Condition Reports reviewed were: 01-380 (incorrect log entries), 00-3351 (relief valve performance issues), and 01-534 (maintenance rule failure assessment not revised to correspond to root cause).

- Suppression pool makeup system. This review included the following condition reports that documented the equipment issues and the associated maintenance rule failure assessment sheets: 00-2531, 00-3383, and 00-3661.

- Standby liquid control. This review included the following condition reports and maintenance rule failure assessment sheets: 00-2593, 00-2611, 00-2612, and 00-2613.

- Division 2 emergency diesel generator. This review included the following condition reports and maintenance rule failure assessment sheets: 00-3561, 00-3582, 00-3500.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope (71111.13)

The inspectors reviewed several of the licensee's risk assessments associated with maintenance and surveillance activities, as discussed below. In addition, the inspectors reviewed condition reports associated with maintenance-related risk assessment and management or emergent work control to determine if the licensee was identifying problems and entering them in the corrective action program. The problem identification and resolution (PIR) condition reports reviewed were: 01-0254, 01-0322, 01-0243, 01-414, and 01-450.

- The inspectors reviewed the licensee's risk assessment for planned and emergent work the week of January 22 thru 26. One work order rendered the "A" train of the suppression pool make up system inoperable and unavailable on January 23. Other items unavailable at the time included one of two trains of emergency diesel ventilation for both Division 1 and Division 2. Another system out-of-service later during the week was residual heat removal (RHR) "A" due to emergent work on a snubber. The inspectors also reviewed associated Condition Report 01-0243.

- The inspectors reviewed the licensee's risk assessment for the week of December 31, 2000, to January 6, 2001. Equipment out-of-service during that week included the RHR "A" minimum flow valve, a snubber on the Division 2 emergency diesel generator (EDG), standby liquid control "A," and one train of EDG ventilation.

- The inspectors reviewed the licensee's risk assessment for the week of January 28 to February 3. Equipment out-of-service during that week included the reactor core isolation cooling system and the ED-2-B 125 VDC electrical bus.

- The inspectors reviewed the licensee's risk assessment for the week of February 5 thru 9. Equipment out-of-service during the week included the Unit 2 Division 2 battery, several switchyard breakers, standby liquid control "B," RHR "A," and the Division 2 EDG fuel oil system.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope (71111.14)

- On January 21, the inspectors observed operator performance during a nonroutine evolution associated with a modified feedwater pump alignment. The System Operating Instruction was recently revised to allow operation above 85 percent power with one motor feed pump and one turbine feed pump in service. As a result of the procedure changes, the operators were required to monitor feedwater suction flow rates and motor feed pump amps. The inspectors observed operator briefings and operator use of procedures. The inspectors also independently monitored the control room indications.

- On January 25, the plant operators entered ONI-D51, "Earthquake," due to a report of seismic shock in the Administration and Service Buildings. The inspectors reviewed the operator logs to determine what occurred and how the operators responded. The inspectors also reviewed the ONI and interviewed plant operators to determine if operator response was in accordance with procedure requirements.

- On February 1, the inspectors reviewed control room operator performance during a nonroutine evolution associated with overspeed testing of the reactor core isolation cooling (RCIC) turbine, uncoupled from the RCIC pump. Testing was conducted in accordance with procedure PTI-E51-P0003, "RCIC Terry Turbine Overspeed Trip." Set up of test conditions required operator actions to isolate the reactor steam from the auxiliary steam such that auxiliary steam could be used to spin the RCIC turbine. The RCIC turbine tripped out of specification low during the initial automatic trip test. After adjustments were made, the RCIC turbine trip system testing was completed.

- On February 17, the inspectors observed operator performance during the removal of the main generator from the grid and the subsequent planned scram to shutdown the plant for the refueling outage. Procedures reviewed included: IOI-8, “Shutdown by Manual Scram,” IOI-3, “Power Changes,” IOI-7, “Cooldown Following a Reactor Scram Main Condenser Available,” PEI-B13, “Reactor Pressure Vessel Control,” and ONI-C71-1, “Reactor Scram.”

- On February 17, the inspectors reviewed the operators’ performance of activities associated with a 5:32 p.m. entry that day into off normal instruction ONI-D17, “High Radiation Levels Within Plant (Unit 1).” Entry into ONI-D17 was prompted by a high alarm on the drywell gaseous activity monitoring channel. At 10:38 p.m. on February 17, plant emergency instruction PEI-N11, “Containment Leakage Control,” was entered due to an annulus exhaust gas treatment system radiation monitor reading 340 cpm. The plant remained in PEI-N11 until 2:00 a.m. on February 20, at which time the entry conditions had cleared. ONI-D17 was exited shortly thereafter at 2:14 a.m. once the source of the gas had been resolved by filter changes on the reactor water cleanup system. Condition Report 01-599 was written to document the issue.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspectors reviewed an operability evaluation involving a test failure of the residual heat removal “A” minimum flow valve. The as-found thrust was below the minimum acceptable value. The inspectors reviewed the licensee’s operability determination associated with Condition Report 01-0018 to determine whether the licensee’s conclusions were technically justified.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

Cumulative Effects of Operator Workarounds

a. Inspection Scope (71111.16)

The inspectors reviewed the licensee’s operator workaround list and reviewed the cumulative effect of the workarounds. The inspectors discussed the eleven workarounds on the “Operator Work Around List” with licensed operators and operations department management and walked down portions of the affected systems. The inspectors reviewed the cumulative effects of the workarounds to determine if there was any impact on the reliability and availability of systems, any potential to increase the frequency of initiating events, and any effect on the ability of operators to respond to plant transients and accidents. The inspectors also reviewed associated condition reports, System Operating Instructions, and the USAR. The inspectors also reviewed degraded conditions that operators were compensating for an a daily basis to determine whether they should be included in the licensee's list of workarounds. The inspectors reviewed the following documents:

- M&C-14, "Work Around Policy," dated February 15, 2000
- "FENOC Work-Around and Control Room Deficiency Guidelines" dated January 17, 2000
- Daily LCO Surveillance Requirements Sheets for January 23, 2001
- Condition Reports 00-2354 (RHR minimum flow piping), 00-3608 (EDG fuel level instrument)
- "Operator Work Arounds," dated January 23, 2001

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19)

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

- On January 4, the inspectors selected Work Order 01-0081, that controlled maintenance on the residual heat removal pump minimum flow valve, 1E12-F064A, for review. The post maintenance testing consisted of static and dynamic MOVATS testing to demonstrate that the valve would close under design basis conditions. The test procedure was PTI-E12P0005A, "Dynamic Diagnostic Testing for RHR A Valves." The inspectors observed portions of the testing from the control room and reviewed the results upon completion of the test. The inspectors also reviewed associated Condition Reports 01-0049, 01-0040, 01-0018.

- On February 9, the inspectors selected Work Order 99-2967, that controlled replacement of a Division 2 diesel generator fuel oil system relief valve, for review. The post maintenance testing consisted of verifying that fuel pressure was between 30-45 psig from no load to full load during the engine run subsequent to valve replacement. The inspectors observed the post maintenance testing noting, as did the licensee, that the results did not meet the pressure range specified in the acceptance criteria. As required by the post maintenance testing program, the licensee issued CR 01-473 and initiated an immediate investigation. The immediate investigation concluded that, based on the characteristics of the new valve, the supply header pressure values obtained during the post maintenance testing were acceptable, and that no other actions were required to declare the diesel operable with the higher supply header pressure.

- The inspectors performed an in-office review of completed Work Order 00-8377, that controlled static testing of standby liquid control system pump suction Valve “B” in accordance with procedure FTI-F-0016, “Operation of the MOVATS 3500 Motor Operated Valve Diagnostic System.” The post maintenance testing prescribed in Work Order 00-8377, consisted of stroking the valve full open and closed, twice, from the control room, in accordance with applicable sections of SVI-C41-T2001B, “Standby Liquid Control B Pump and Valve Operability Test.” The inspectors reviewed the sections of SVI-C41-T2001B that were completed in support of the post maintenance testing.

- The inspectors performed an in-office review of the post maintenance testing requirements of Work Orders 00-7843, 00-7846, and 00-7847. These work orders controlled the installation of replacement outside air intake dampers which provide tornado protection for the diesel generator building. The inspectors reviewed CR 00-3739, written to address the inability to perform full flow, post maintenance, testing on the new dampers, due to low outside ambient air temperature, and its potential adverse impact to the safety-related and non-safety-related equipment in the supplied area. From a review of the original design calculations, the licensee determined that the fan and damper equipment was sized to maintain the room temperature less than or equal to 122 °F with a maximum outside ambient temperature of 95 °F and the diesel generator fully loaded. From this information, the licensee concluded that full flow testing was not necessary below 50 °F. To consider the diesel generator building ventilation system operable below 50 °F, verification that design parameters were within design ratings would be accomplished by damper stroking to verify damper control and position and by measurement of fan motor amperage and voltage. The licensee planned to conduct confirmatory full flow testing after the outside ambient temperature increased above 50 °F but before it reached 90 °F. Potential LCOs 01-56, 57, and 58 were created to track completion of the confirmatory flow testing.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors conducted inspections of the following areas during the refueling outage which commenced February 17, 2001:

- Outage risk assessment of planned activities to determine if defense-in-depth would be maintained and if high risk activities were appropriately controlled and reviewed by station management.

- Operational evolutions such as, plant shutdown and establishing shutdown cooling.

- Component and equipment configuration management control to ensure equipment relied on to perform a key safety function would not be adversely affected by outage activities.

- Clearance and special operating permit programs.

- Reactor coolant system instrumentation to ensure operators maintained a clear understanding of accuracy of measurement and contingencies if the instrument indications were lost.

- Decay heat removal system operability and protection during key times of the outage, and during special surveillance testing.

- Containment integrity control as required.

- Review of selected outage related maintenance and surveillance activities to ensure the activities were conducted in accordance with station procedures and TS requirements.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspectors witnessed the surveillance tests listed below to determine whether requirements were met, consistent with applicable sections of TS, USAR, and Plant Procedures. The inspectors reviewed whether test control was properly coordinated with the control room and performed in the sequence specified in the surveillance instruction. Also, the inspectors determined if test equipment was properly calibrated and installed to support the surveillance tests. In addition, the inspectors reviewed condition reports associated with surveillance testing to determine if the licensee was identifying problems and entering them in the corrective action program. The problem identification and resolution (PIR) condition reports reviewed were CR 01-265 on snubber issues and CR 01-474 on low flow in the emergency service water (ESW) “B” train.

- SVI-E12-T2001, “RHR A Pump and Valve Operability Test”

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications

a. Inspection Scope (71111.23)

The inspectors reviewed temporary plant configuration changes that had been implemented as a result of degraded equipment to determine whether the conditions were properly controlled as temporary plant modifications. The issues included: (1) several failed room temperature indicators that were being compensated for by plant operators taking local readings on plant rounds; and (2) manual backwashing of the ESW “B” pump discharge strainer, performed once every 24 hours by plant operators, and required as a result of disabling the automatic strainer backwash feature due to erratic operation.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

a. Inspection Scope (71151)

The inspectors verified the licensee’s data for the Performance Indicators (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 reviewed the data compared to the guidance in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 0. Also reviewed were the following CR’s and Licensee Event Reports: CRs 00-3561, 00-3582, 00-1326, LERs 2000-001, 2000-003, 2000-004, and 2000-005.

- Safety System Functional Failures, Q1-Q3, 2000

- Emergency Diesel Generator Unavailability, Q2-Q3, 2000

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up

(Closed) Licensee Event Report 50-440/2000-001: Potential for Inadequate Suppression Pool Make-up for the Emergency Core Cooling Systems. Subsequent to approval of License Amendment No. 100, which allowed removal of the inclined fuel transfer system (IFTS) blind flange with the reactor at power, questions were raised regarding the adequacy of the engineering reviews that supported the amendment. After further evaluation, the internal engineering reviews were determined to be inadequate and the use of the amendment to remove the blind flange at power resulted in a condition that did not meet design criteria. Specifically, on March 25 through March 27, 1999, with the blind flange removed and the upper IFTS pool gate removed at power, there was no safety-related barrier to prevent loss of water from the upper pool. Although there was always a nonsafety-related closed valve in the IFTS system, a potential loss of water from the upper pool could reduce the inventory below that volume required by the accident analysis for suppression pool makeup. The licensee implemented interim controls to assure that the IFTS flange would not be used until further engineering reviews were completed. Additionally, the licensee submitted a letter to the NRC, dated February 1, 2001, describing their intent to modify the amendment to include additional necessary controls and to not the remove the blind flange at power until the revised amendment is approved.

The inadequate engineering reviews resulted in a credible impact on safety because the barriers to maintain necessary inventory in the upper pool were reduced to only a single nonsafety-related valve. This issue affects a mitigating system because an adequate suppression pool makeup system volume is required to ensure adequate net positive suction head for the emergency core cooling systems following a postulated design basis accident. A phase 3 risk assessment was performed by both regional and headquarters risk analysts and reviewed both the potential impact of loss of containment integrity and loss of suppression pool makeup. The risk assessment was performed for the 57 hours the blind flange was removed, even though a longer time period would have been allowed by the TS. Potential accident scenarios were postulated for the operation of the IFTS while the plant was at power. Even when conservative assumptions were applied, the estimated increase in core damage frequency (CDF) from the scenarios was small when compared to the guideline value of 1E-6 per year in Regulatory Guide 1.174. Even if the increase in large early release frequency is assumed to be equal to that for the CDF, Regulatory Guide 1.174 guidelines for “small” increases are still adhered to. Finally, the estimated core damage probability for operation of the IFTS for a period of 57 hours conforms to the guideline value of 5E-7 provided in Regulatory Guide 1.177. Therefore, it can be concluded that the risk from the removal of the IFTS blind flange during power operations was very low (Green).

There was no violation associated with this finding because the TS allowed the removal of the blind flange at power. The condition is in the licensee’s corrective action program as Condition Report 99-3035. This item is closed.

4OA5 Other

(Closed) Unresolved Item 50-440/2000-014-01: Inadequate evaluation of emergency closed cooling system MOV test data. In reviewing the associated Operability

Evaluation, the inspectors questioned an assumption in the licensee's evaluation of past operability. Specifically, the licensee's revised calculation assumed that catastrophic failure of the non-safety-related, non-seismic piping downstream of the isolation valves would not have occurred and only the effects of leakage from a pipe crack needed to be postulated. The inspectors verified that the valve was closed to maintain the emergency closed cooling system operability during additional investigation of the issue. The valve was scheduled for replacement in refueling outage 8 (March, 2001). The inspectors submitted Task Interface Agreement (TIA) No. 2000-18 to the Office of Nuclear Reactor Regulation for review. This item is closed.

4OA6 Meetings

Exit Meeting

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 February 28, 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 included in the Inspection Report.

Management Meeting

On February 8, 2001, the NRC Region III Administrator and other staff members conducted a public meeting with several licensee management individuals in the Region III Office. Topics included the licensee's corrective action program and management focus areas. The handout from this meeting is included as Attachment 1 to this inspection report.

KEY POINTS OF CONTACT

Licensee

J. Wood, Vice President-Nuclear
B. Boles, Operations Manager
R. Strohl, Superintendent, Plant Operations
G. Dunn, Manager, Regulatory Affairs
D. Gudger, Supervisor, Compliance
T. Lentz, Manager, Design Engineering
K. Ostrowski, Director, Nuclear Services Department
D. Philipps, Manager, Plant Engineering
T. Rausch, Director, Nuclear Engineering Department
K. Russell, Compliance Engineer
R. Schrauder, General Manager, Nuclear Power Plant Department

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Closed

440/2000-014-01	URI	Inadequate evaluation of MOV test data
440/2000-001-00	LER	Potential for Inadequate Suppression Pool Make-up for the Emergency Core Cooling Systems

Discussed

None.

LIST OF ACRONYMS AND INITIALISMS USED

ADAMS	Agencywide Documents Access and Management System
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
DCP	Design Change Package
DRP	Division of Reactor Projects
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FENOC	FirstEnergy Nuclear Operating Company
IFTS	Inclined Fuel Transfer System
IOI	Integrated Operating Instruction
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
PIR	Problem Identification and Resolution
PTI	Periodic Test Instruction
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SDP	Significance Determination Process
SOI	System Operating Instruction
SVI	Surveillance Instruction
TIA	Task Interface Agreement
TS	Technical Specification
URI	Unresolved item
USAR	Updated Safety Analysis Report
VLI	Valve Line-up