April 19, 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 INTEGRATED INSPECTION REPORT 50-440/01-04

Dear Mr. Wood:

On March 31, 2001, the NRC completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings, which were discussed on April 10, 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.

No findings of significance were identified.

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
Reactor Projects Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 50-440/01-04

See Attached Distribution

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No: License No:	50-440 NPF-58
Report No:	50-440/01-04
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	P.O. Box 97 A200 Perry, OH 44081
Dates:	February 25 - March 31, 2001
Inspectors:	C. Lipa, Senior Resident Inspector (SRI) R. Vogt-Lowell, Resident Inspector P. Louden, SRI, Clinton M. Farber, Reactor Engineer D. Simpkins, Resident Inspector, Davis-Besse
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	**Radiation Safety**	**Safeguards**
• Initiating Events	• Occupational	• Physical Protection
• Mitigating Systems	• Public	
• Barrier Integrity		
• Emergency Preparedness		

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 05000440-01-04; on 02/25-03/31/2001; FirstEnergy Nuclear Operating Company; Perry Nuclear Power Plant; Integrated Inspection Report.

The inspection was conducted by resident inspectors and a regional inspector. There were no findings identified in this report.

Report Details

Summary of Plant Status: At the beginning of the inspection period on February 25, the reactor was in Mode 5, Refueling. Mode 4, Cold Shutdown, was reached at 11:38 p.m. on March 15 when the reactor vessel head closure bolts were fully tensioned. On March 21, the licensee entered Mode 2, Startup, and initiated a reactor startup at 9:00 a.m. The generator was synchronized to the grid at 9:10 a.m. on March 23 and full power was reached on March 31. At the end of the inspection period on March 31 the reactor was at 100 percent power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

Reactor (R)

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 drywell, refuel floor in containment, turbine building, and heater bay.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06)

The inspectors reviewed the licensee's documentation containing design flood levels for areas containing safety equipment to assess whether flooding mitigation plans and equipment were consistent with the design requirements and risk analysis assumptions. The inspectors also assessed the current relevance and continued adequacy of corrective actions previously implemented for two significant pipe cracks which occurred in 1991 and 1993 (Licensee Event Reports 50-440/91-27 and 93-10). These included sealing of equipment below flood-lines, holes or unsealed penetrations in floors and walls between flood areas, watertight doors, common drain systems and sumps, and level alarm circuits. Documents reviewed were: USAR Sections 2.4, 3.4, and 3.6; Condition Reports (CRs) 00-2313, 01-1723, 01-1752; ARI-H13-P870-3 (windows B1, C1, D1, E1, E3, F1, F2, F3, G1, G2, G3, H1, and H2); ARI-H13-P970-1 (windows F5 and F6); ONI-ZZZ-6 (Leak in Underground Piping); PTI-P72-P0001, "Plant Underdrain Continuity

Test," PTI-P72-P0002, "Plant Underdrain Groundwater Inflow Test," PTI-P72-P0005, "Plant Underdrain Groundwater Level Readings," and EPI-A1, "Emergency Action Levels."

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope (71111.07)

The inspectors completed this module by reviewing the licensee's final evaluation of the test results for the "B" residual heat removal heat exchanger test performed on November 29, 2000. (The initial review of the heat exchanger test was documented in Inspection Report 50-440/2000-014). The final results were documented within calculation E12-98, Revision 1, that was finalized on February 27, 2001. The inspectors verified how the licensee accounted for instrument uncertainties and how the differences between test conditions and design conditions were accounted for.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12Q)

The inspectors reviewed equipment issues, surveillance 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 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: 01-1502 (evaluation required on hydrogen analyzer), 01-1503 (evaluation required on suppression pool temperature indicator), 01-1170 (Local Leak Rate Test (LLRT) failure of valve 1P51F0530), 01-1175 (ESW valve 1P45F0040A will not stroke closed), 01-1112 (LLRT failure on 1P11F545), 01-0973 (LLRT failure on 1B21F019), and 01-0946 (evaluation required for radiation monitor failure).

- Plant underdrain system. The inspectors reviewed the following condition reports and maintenance rule failure assessment sheets: 00-3622 (degraded pumps will not support scheduled system test), 00-1991 (underdrain system test not completed when required by USAR), 00-3065 (suction and discharge of sump pump found nearly plugged solid with calcium scale) 00-2694 (system has degraded pumps which prevent successful completion of periodic testing), and 00-1631 (testing was scheduled past its late date).

- Feedwater control system. The inspectors reviewed the following condition reports and maintenance rule failure assessment sheets: 00-1983 (feedwater control recorder indication not tracking properly), 00-3446 (master level control station needs replacement), 00-3588 (wide range level indication is lower than plant data book graph), and 00-3342 (several level recorders indicated a change in level, cause was drift in the master level controller setpoint station).

- Neutron monitoring system, including average power range monitors (APRM), intermediate range monitors (IRM), and source range monitors (SRM). This review included the following condition reports and maintenance rule failure assessment sheets: 00-1852 (IRM A has a higher noise than the other 7 IRM's), 00-1240 (LPRM 4A-56-33 failing downscale), 00-1761 (SRM "B" signal to noise ratio had drifted over 8 hours), 00-2063 (as-found value for APRM count circuit outside the allowable value), 01-0490 (as found data outside the allowable value), 01-0580 (IRM "A" channel is noisy at both meter and recorder), 01-0645 (SRM "D" increased with no change in reactivity), 01-1424 (missed action for inoperable IRM's).

- Emergency service water (ESW) system. This review included the following condition reports and maintenance rule failure assessment sheets: 00-1364 (determine if the P45B pump has degraded), 00-1406 (the post-maintenance test for valve 1P45F0140 failed), 00-3433 (ESW "B" through wall pipe leak downstream of valve 1P45F0541B), and 01-0474 (ESW "B" flow during SVI-P45-T2002 is in Alert range low).

b. Findings

No findings of significance were identified.

Maintenance Rule Periodic Evaluation

a. Inspection Scope (71111.12B)

The objective of the inspection was to:

- Verify that the periodic evaluation was completed within the time restraints defined in the maintenance rule (once per refueling cycle, not to exceed two years), ensuring that the licensee reviewed its goals, monitoring, preventive maintenance activities, industry operating experience, and made appropriate adjustments as a result of that review;

- Verify that the licensee balanced reliability and unavailability during the previous refueling cycle, including a review of safety significant structures, systems, and components, (SSC);

- Verify that (a)(1) goals were met, corrective action was appropriate to correct the defective condition including the use of industry operating experience, and (a)(1) activities and related goals were adjusted as needed; and

- Verify that the licensee has established (a)(2) performance criteria, examined any SSCs that failed to meet their performance criteria, or reviewed any SSCs that have suffered repeated maintenance preventable functional failures including a verification that failed SSCs were considered for (a)(1).

The inspector examined the current periodic evaluation, “Maintenance Rule Monitoring Program Periodic Assessment Report of Maintenance Effectiveness for Operating Cycle 7,” dated July 26, 2000. To evaluate the effectiveness of (a)(1) and (a)(2) activities, the inspector examined approximately 40 Condition Reports (CR) (contained in the list of documents in Attachment 2 of this report) associated with the following systems: control rod drive, and turbine building chilled water. The inspector also reviewed the following condition report to assess licensee identification and resolution of problems associated with maintenance rule implementation: 01-1271, Maintenance Rule Program Deficient in Monitoring Operating Systems, March 10, 2001.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope (71111.13)

The inspectors reviewed the licensee’s risk assessment 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: 01-1606 (feedwater level control shift may challenge operators), 01-1651 (plant startup with high condensate temperatures), 01-1659 (performance of non-logic work activities affected safety-related system maintenance schedule), 01-1726 (maintenance risk assessment error), 01-1726 (maintenance rule unavailability time for ESW screen wash), 01-1788 (test instruction did not fully anticipate effects of partial arc modification).

- The inspectors reviewed the licensee’s risk assessment associated with on-line maintenance activities during the week of March 26 through 30. Equipment out-of-service included the “A” train of standby liquid control, the “A” train of emergency service water screen wash, and the Division 1 emergency diesel generator.

b. Findings

No findings of significance were identified.

1R14 <u>Personnel Performance During Nonroutine Plant Evolutions</u>

a. <u>Inspection Scope (71111.14)</u>

- On March 16, 2001, the inspectors attended a pre-job Control Room brief where system re-alignment from shutdown cooling train “B” to train “A” was discussed. Subsequent to the brief, the inspectors observed the licensed operators carry out the realignment by using Step 5.4.1 of SOI-E12, “Residual Heat Removal System.“

- On March 20, 2001, the inspectors observed performance of procedure SVI-T23-T0401, “Drywell Integrity Verification Test” carried out under WO 99-10531. This procedure uses the combustible gas mixing system to increase drywell pressure and ascertain no gross leakage exists following a refueling outage, thereby providing an indication of the ability of the drywell to perform its design function.

- On March 20, 2001, the inspectors attended a control room brief on Infrequently Performed Test/Evolution (IPTE) 2001-003. This IPTE concerned the upcoming performance of several tests to confirm adequate plant response to the Partial Arc Modifications implemented during the refueling outage. These special tests were associated with the 5 percent power uprate approved by NRC in 2000. The testing discussed during the brief would be conducted under the following procedures: a) TXI-313, “Turbine Control Valve Testing Operation”; b) TXI-317, “3579 MWth to 3758 MWth Power Uprate Implementation”; c) TXI-318, “Turbine Bypass Valve Testing Operation”; d) TXI-333, “Pressure Regulator Linearity Testing”; e) TXI-334, “Turbine Admission Mode Conversion Testing Data”; and f) PTI-C85-P001, “Pressure Control Tuneup Instruction.” Test termination criteria were discussed by the IPTE Manager.

- On March 23, 2001, the inspectors observed Safety/Relief Valve (SRV) testing per Step 4.8.9 of IOI-1, “Cold Startup” and SVI-B21-T2005, “SRV Exercise Test.” To demonstrate SRV operational readiness, this SVI exercises the main steam line SRVs by using the remote manual switches and testing both the “A” (Division 1) series and “B” (Division 2) series solenoids for each SRV cycling.

- On March 23, 2001, the inspectors observed synchronization of the main generator to the grid in accordance with Step 4.3 of IOI-3, “Power Changes.”

- On March 18 and 19, 2001, the inspectors observed control room briefings and operators establishing plant conditions to support ISI-B21-T1300-1, “Reactor Coolant System Leakage Pressure Test.” This evolution was controlled as an IPTE.

b. <u>Findings</u>

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

- The inspectors reviewed an operability evaluation involving the restrainer bracket set screws for Jet Pumps 1 through 20. Each jet pump has a vessel side set screw, a shroud side set screw and a wedge assembly that align and hold the mixer assembly. During the refueling outage, Condition Reports 01-1068 and 01-1076 were initiated by the licensee to document as-found set screw gaps that did not meet acceptance criteria. The inspectors reviewed the licensee's operability determination associated with Condition Report 01-1068 to determine whether the licensee's conclusions were technically justified.

- The inspectors reviewed the licensee's operability evaluation associated with CR 01-0487, "Leakage Outside Containment Exceeds USAR Limit," dated February 13, 2001. This issue was identified during the performance of Periodic Test Instruction, PTI-GEN-P0015, "Emergency core Cooling system Header Drains Seat Leakage Test."

- The inspectors reviewed the licensee's operability evaluation associated with CR 01-1175, emergency service water strainer blowdown isolation valve P45-F040A failing to fully close upon demand. The inspectors reviewed the system operating instruction, USAR description for the strainer blowdown, and Technical Specifications.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope (71111.19)

- The inspectors reviewed post-maintenance testing activities on fuel handling equipment after a spent fuel assembly that was being raised from the fuel transfer carriage dropped approximately 7 inches back into the carriage seat on March 2, 2001. The licensee performed extensive testing of the fuel handling system prior to re-commencing fuel movements. The inspectors reviewed for adequacy the Periodic Testing Instruction PTI-F11 "Fuel Handling Platform, System Operating Instruction SOI-F11 "Fuel Handling Platform," and actions taken associated with Condition Report 01-1047 and Work Order 00-005463.

- The inspectors reviewed the post-maintenance testing activities associated with reseating of the jet pump #5 inlet mixer after the licensee identified that the inlet mixer was not properly seated on March 1, 2001. The inspectors reviewed Condition Report 01-0998 and Work Order 00-9228 which were initiated to reseat and test the inlet mixer and the work package and documentation for the associated testing.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope (71111.20)

The inspectors conducted inspections of the following areas during Perry's eighth refueling outage which commenced February 17 and ended on March 23, 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.

- 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.

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

- Containment integrity control as required.

- Refueling activities, including fuel handling and core verification.

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

- Reactor restart activities including approach to critical, turbine startup, recirculation system motor speed change, and ascension to 100 percent reactor power.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspectors witnessed or reviewed the test data for the below listed surveillance tests to determine whether requirements were met, consistent with applicable sections of Technical Specifications, 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 and 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: 01-1497 (SRM unable to be tested per surveillance instruction), 01-1506 (precautions and limitations not followed during surveillance), 01-1506 (acceptance criteria not met during surveillance), 01-1133 (reactor pressure transmitter failure during surveillance), 01-0979 (degraded snubber on low pressure core spray system), 01-0970 (carbon adsorber sample failed on auxiliary building ventilation system)

- SVI-R43-T5366, "LPCS/LPCI A Initiation and Loss of EH-11 Response Time Test"
- SVI-R43-T1337, "Division 1 Standby Diesel Generator Loss of Offsite Power Test"
- SVI-B21-T1400, "Main Steam Isolation Valves Logic System Functional Test"
- SVI-B21-T9000, "Type C Local Leak Rate Test of 1B21 Main Steam Line Penetrations"

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications

a. Inspection Scope (71111.23)

The inspectors reviewed the licensee's list of active temporary modifications following the completion of the refueling outage. The inspectors determined that there were no significant temporary modifications warranting a detailed review.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA5 Other

Review of World Association Of Nuclear Operators (WANO) Report

The inspectors reviewed the licensee's final report, dated January 3, 2001, of the May 2000 evaluation performed by WANO. No additional followup is planned.

(Closed) URI 50-440/98005-01: Engineering Assumptions for Fire Protection Seals.

This issue involved engineering assumptions used by the licensee to justify a 3-hour fire rating for large-gap penetration seals in the plant. The inspector reviewed the following documents associated with this issue:

- Potential Issue Form (PIF) 96-3243, Construction Gap Fire Seals and Compartmentalized Blockout Seals are not Installed as Depicted on the Design Details, October 21, 1996

- NRC Perry Fire Protection Follow-Up Inspection Report, 50-440/98005(DRS)

- CR 00-1307, Engineering Assumptions Used in Evaluating Fire Penetration Seals, April 28, 2000

In response to the issues raised in PIF 96-3243, the licensee conducted a non-conforming condition investigation which determined that the seals were acceptable. NRC review of this investigation, as documented in the above inspection report, identified several concerns with assumptions made in the investigation. CR 00-1307 was issued to assess the concerns and determine corrective action. In this CR, the licensee staff acknowledged that the 3-hour fire rating for these seals was not adequately supported. Engineering work necessary to correct these conditions is addressed in CR 00-1307 and is expected to be completed in early 2001. After a review of the group 1 and 2 questions from Manual Chapter 0610*, this was determined to be a minor violation of 10 CFR Part 50, Appendix R, Section 3.G.2. Based on the fact that the issue and corrective actions are addressed in CR 00-1307, no further NRC action is planned and the item is closed.

4OA6 Meeting(s)

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 April 10, 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.

PARTIAL LIST OF PERSONS CONTACTED

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
T. Rausch, Director, Nuclear Engineering Department
R. Schrauder, General Manager, Nuclear Power Plant Department

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

50-440/98005-01	URI	Engineering Assumptions for Fire Protection Seals

Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
APRM	Average Power Range Monitor
ARI	Annunciator Response Instruction
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CR	Condition Report
DCP	Design Change Package
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
FENOC	FirstEnergy Nuclear Operating Company
IFTS	Inclined Fuel Transfer System
IPTE	Infrequently Performed Test/Evolution
IOI	Integrated Operating Instruction
IRM	Intermediate Range Monitor
LLRT	Local Leak Rate Test
LPRM	Local Power Range Monitor
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
PIF	Potential Issue Form
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
SRM	Source Range Monitor
SSC	Systems, Structures, and Components
SVI	Surveillance Instruction
TIA	Task Interface Agreement
TS	Technical Specification
URI	Unresolved item
USAR	Updated Safety Analysis Report
VLI	Valve Line-up
WANO	World Association of Nuclear Operators

PARTIAL LIST OF DOCUMENTS REVIEWED

PAP-1125	Monitoring the Effectiveness of Maintenance Program Plan, Revision 5, July 13, 2000
PAP-1125	Maintenance Rule Reference Guide, Version: 02, February 17, 2001
	Maintenance Rule Monitoring Program Periodic Assessment Report of Maintenance Effectiveness for Operating Cycle 6, August 10, 1998
	Maintenance Rule Monitoring Program Periodic Assessment Report of Maintenance Effectiveness for Operating Cycle 7, July 26, 2000
	Maintenance Rule Functions, Performance Criteria and Classifications, Revision 3.02, February 22, 1999
	Maintenance Rule Functions, Performance Criteria and Classifications, Revision 4.06, March 16, 2001
PIF 98-773	Difficulties Encountered with Rod Sequence Exchange and Scram Timing in Withdrawing Control Rods from Position 00, April 18, 1998
CR 99-0531	P46A Chiller Started, Would Not Load, No Refrigerant Seen in Bullseye and Oil Temp Began to Rise, March 9, 1999
CR 99-0562	Number of Problems Noted by Operations on Turbine Bldg Chilled Water Chiller 1P46B0001A, March 12, 1999
CR 99-0610	HCU 38-51 was Hydraulically Isolated per SOI-C11 HCU Sect 7.5, at the Time Rod Was Isolated, it Was at Notch Position 48, March 18, 1999
CR 99-0699	Control Rods 58-31, 46-39, and 38-55 Were Found Low to Position 43 During Performance of SVI-C11-T2006 Following Reactor Scram, March 27, 1999
CR 99-1266	WO 96-5866 Replaced O-Rings and Internals for 1C11F0009. During PMT (SVI-C11-T2004) it Was Noted That When Switch S2A Was Depressed an Unexpected SDV Valve Movement Occurred, April 25, 1999
CR 99-1330	Observations During OPSI Scram Timing per SVI-C11-T1006. Suspect Bad SRI Switches, April 28, 1999
CR 99-1377	Control Rod 30-31 Could Not Be Withdrawn from Notch Position 00 Using Normal Drive Water Pressure, May 4, 1999
CR 99-1430	Scram Discharge Volume First Drain Valve 1C11F0011 Failed Stroke Time Test with an Excessive Stroke Time, May 10, 1999
CR 99-1433	Sequence of Events for Control Rod 22-19 That Resulted in Loss of Position Indication on Both Channels, May 11, 1999
CR 99-1434	Loss of Position Indication for Control Rod 22-19 at Position 48, May 8, 1999
CR 99-1435	During SVI-C11-T2204, Valves 1C11F0010 and 1C11F0181 Failed Their Close Stroke Times, this Required Entry in T.S. 3.1.8, May 11, 1999
CR 99-1748	During Clearance Activities on 1C11C001B (CRDH B Pump), the Pump Suction Piping between the Pump Suction Valve and the Pump May Have Exceeded Design Pressure for the Piping, July 7, 1999
CR 99-1857	P46 Chillers Have Been Tripping on Low Refrigerant Temperatures at an Increased Frequency during Hot Weather, July 25, 1999
CR 99-2220	CRD Oil Pump Tripped as a Result of a PSIG Setting That Was Made Using a Gauge Found to be Reading 3 PSIG Low, September 14, 1999

CR 99-2244	During Performance of SVI-C11-T2004, 1C11F0010 Failed to Close in the Required Time, September 16, 1999
CR 99-2250	During Performance of SVI-C11-T2004, 1C11F0180 Failed its Close Stroke Time by Closing Too Fast, September 16, 1999
CR 99-2595	Repeat Issue Where Position Indication Problems Were Experienced with Control Rod 22-19, October 20, 1999
CR 99-2607	1C11F0150A Opened at 10 PSIG Rather Than between 60 and 66 PSIG, November 1, 1999
CR 99-2634	Rod 38-35 Was Not Deselected and Reselected to Reset the RWL Prior to Withdrawing the Rod. Rod was Pulled 3 Notches. Resetting RWL would Prevent Pulling Rod More than 2 Notches but Did Not, November 3, 1999
CR 99-2643	Rod Withdrawal Limiter Function of RCIS requires Manual Operator Action to Ensure It Functions as Designed, November 4, 1999
CR 99-2699	Rod 14-19 Double Notched from Position 34 to 38 While at Increased Drive Pressure Per SOI-C11 (RCIS) Alternate Control Methods, November 3, 1999
CR 00-0122	SDV Level Setpoint was Found Outside of the LAIZ during SVI C11-T5376D, January 13, 2000
CR 00-0241	C11 Pump A Discharge Pressure Decreased from 1743 PSIG to 1656 PSIG over a 60-Day Period, January 22, 2000
CR 00-0940	C11B Pump Inboard Seal Supply Line Coupling Leakage Has Increased to 500ml/min, March 27, 2000
CR 00-1098	Unacceptable Increasing Vibration Trend on Speed Increaser - Control Rod Drive 1C11C0001B Pump/Motor Assembly, April 6, 2000
CR 00-1182	While Withdrawing Control Rod 14-19, Using Single Notch Withdrawal Commands, Double Notch Movements were Occurring, April 14, 2000
CR 00-1189	During Rod Pattern Adjustment Rod 18-39 Double Notched from 00 to 04, April 16, 2000
CR 00-1714	ICS Scram Data for Rods 30-11, 54-35, 50-39, and 54-39 Were Noted as Not Having Enough Pulses, June 5, 2000
CR 00-1827	PTI-C11-P0005 was Changed to Incorporate CRRA-99-2634-2 Actions to Deselect a Rod after Inserting to Ensure the Rod Withdrawal Limiter Would Function Properly, June 13, 2000
CR 00-1837	While Performing HCU Venting of Rod 30-19, the Rod Double-Notched from 8 to 4 at Drive Pressure of 400 psi and Single-Notch Insert Signal, June 13, 2000
CR 00-1887	CRDM 14-19 Double-Notched from Position 10 to 14 during Rod Withdrawal During Plant Startup, June 16, 2000
CR 00-1957	During SVI C11 T1003B, Rod 22-19 Failed to Indicate Position 48 for 2 to 3 minutes. The Full Out Indication Was Lit, and a Data Fault, Channel Disagree, and Rod Withdrawal Block Was Received, June 22, 2000
CR 00-2301	Turbine Building Chill Water Chiller “A” Tripped on Low Oil Pressure, July 31, 2000
CR 00-2406	Control Rod Drive System (C11 CRDH) Has a Trend of Degrading Parameters Over the Past Several Months, August 9, 2000
CR 00-2511	Cooling Water Check Valves (C11-138) for Control Rod Drives 30-31, 30-35, and 38-31 Failed Their Exercise Open Test During SVI-GEN-T2001, August 19, 2000
CR 00-2739	Cavitation-Induced Erosion Damage to the Valve Body and Cage of 1C11F002A/B, September 6, 2000
CR 00-2778	Control Rod 22-31 Double-Notched from 40 to 44, September 10, 2000

CR 00-2779	Control Rod 46-39 Double-Notched from 32 to 36, September 10, 2000
CR 00-2960	Control Rod Drive Hydraulic Control Unit Level High/Pressure Low Alarms are Being Received on a More Frequent Basis than in the Past, September 26, 2000
CR 00-2962	While Shifting FCVs from the A to B per the SOI, Adjusting the C11R600 was Unable to Match the Red and Black Needles on the B Manual Load Regulator, September 26, 2000
CR 00-3007	During the Period of 8/21/2000 to 9/29/2000 an Average of 3 to 4 HCU Accumulator Faults Were Received per Day, September 30, 2000
CR 00-3106	Unable to Perform Stall Flow Testing of 3 CRDMs, October 8, 2000
CR 00-3708	Unable to Obtain Prescribed Preload on Mechanical Seal for 1P46C0001B, November 30, 2000
CR 00-3736	Jumpers Installed at Wrong Terminal Board During SVI-C11-T2004, December 3, 2000
CR 01-0126	Turbine Building Chilled Water Pump “B” Motor Casing Extremely Hot to Touch, January 11, 2001
CR 01-0285	Replacement Capacitor Appears to be Non-Compatible-RFA, January 25, 2001
CR 01-0295	RCIS BJM Connector Problem, January 26, 2001
CR 01-0471	TBCW Chiller A Trip on Low Oil Pressure, February 9, 2001
CR 01-1125	System Engineer Identified Potential Damage Resulting from Procedure Change, March 5, 2001
CR 01-1214	1C11F002A Reassembly Discrepancies Noted During RSE Walkdown, March 8, 2001
CR 01-1356	Significant Pitting/Erosion was Noted in the CRD Flow Control Valve Body, March 12, 2001
CR 01-1358	Misleading WIP Log Entry Discovered During C11 FCV Troubleshooting, March 13, 2001
CR 01-1513	Rods Not in Signal Generated Although all Control Rods Fully Inserted, March 18, 2001
NOP-LP-2001	Condition Report Process, Revision 1
CR 01-1271	Maintenance Rule Program Deficient in Monitoring Operating Systems, March 10, 2001
PIF 96-3243	Construction Gap Fire Seals and Compartmentalized Blockout Seals are not Installed as Depicted on the Design Details, October 21, 1996
	NRC Perry Fire Protection Follow-Up Inspection Report, 50-440/98005(DRS)
CR 00-1307	Engineering Assumptions Used in Evaluating Fire Penetration Seals, April 28, 2000