February 22, 2001

Mr. R. P. Powers
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: D. C. COOK INSPECTION REPORT 50-315/01-02(DRP); 50-316/01-02(DRP)

Dear Mr. Powers:

On February 10, 2001, the NRC completed a baseline inspection at your D. C. Cook Units 1 and 2 reactor facility. The inspection results were discussed on February 14, 2001, with the Plant Manager and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules, regulations, and the conditions of your license. Within these areas, the inspection consisted of reviews of selected procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance (Green). The issue involved the failure to incorporate requirements and acceptance limits from design documents into postaccident hydrogen monitoring system surveillance test procedures and was determined to be 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 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 D. C. Cook facility.

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/***

Geoffrey E. Grant, Director
Division of Reactor Projects

Docket Nos. 50-315; 50-316
License Nos. DPR-58; DPR-74

Enclosure: Inspection Report 50-315/01-02(DRP);
50-316/01-02(DRP)

cc w/encl: A. C. Bakken III, Site Vice President
J. Pollock, Plant Manager
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REGION III

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| Docket Nos: | 50-315; 50-316 |
| License Nos: | DPR-58; DPR-74 |
| Report No: | 50-315/01-02(DRP); 50-316/01-02(DRP) |
| Licensee: | American Electric Power Company<br>1 Cook Place<br>Bridgman, MI  49106 |
| Facility: | D. C. Cook Nuclear Generating Plant |
| Location: | 1 Cook Place<br>Bridgman, MI  49106 |
| Dates: | January 1, 2001 through February 10, 2001 |
| Inspectors: | B. L. Bartlett, Senior Resident Inspector<br>K. A. Coyne, Resident Inspector<br>J. D. Maynen, Resident Inspector |
| Approved by: | A. Vegel, Chief<br>Reactor Projects Branch 6<br>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 05000315-01-02, IR 05000316-01-02, on 01/01-02/10/2001, Indiana Michigan Power Company, D. C. Cook Nuclear Plant Units 1 & 2. Resident inspector report.

The inspection was conducted by the resident inspectors. 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

**Barrier Integrity**

- GREEN. A non-cited violation was identified for the failure to ensure that test procedure acceptance criteria associated with the PostAccident Hydrogen Monitoring System (PACHMS) backup air supply incorporated the requirements and acceptance limits contained in applicable design documents. The inspectors concluded that this was a violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control." Design requirements for the PACHMS backup air supply included a twelve hour postaccident air capacity. The minimum air bottle pressure required to meet this design requirement for the Unit 2, train A, PACHMS air bottles was determined to be 2420 psig. Contrary to this design limit, the minimum acceptable bottle pressure limits contained in the PACHMS test procedures was 2000 psig.

  The inspectors concluded that this failure had a credible impact on safety and was more than a minor violation of NRC requirements because early failure of the PACHMS backup air supply could result in the inability to operate containment hydrogen sample valves. Emergency operating procedures were written to utilize results obtained from the PACHMS system to determine appropriate postaccident follow-up actions. This issue did not result in an actual open pathway in the physical integrity of the reactor containment or an actual reduction of the atmospheric pressure control function of the reactor containment. (Section 1R22.3)

## Report Details

Summary of Plant Status:

Unit 1 operated in Mode 1 (Power Operation) during the inspection period. At the beginning of this inspection period, Unit 1 was operating at 86 percent power. On January 3, 2001, Unit 1 achieved full power. Unit 1 power was reduced to approximately 58 percent on January 12, 2001, to support cleaning of the Unit 1 East main feedwater pump waterbox. Unit 1 power was restored to 100 percent power January 14, 2001. On February 9, 2001, Unit 1 power was reduced to approximately 50 percent to support repairs to a failed reactor protection system power supply. Unit 1 was returned to full power on February 10, 2001 and continued full power operation for the remainder of the inspection period.

Unit 2 began the inspection period operating at full power. On January 22, 2001, Unit 2 commenced a power reduction due to rod control system problems. Unit 2 entered Mode 3 (Hot Shutdown) on January 23 and returned to Mode 2 (Startup) on January 26, 2001. Unit 2 achieved full power on January 31, 2001 and continued full power operation throughout the remainder of the inspection period.

### 1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

.1 Partial Equipment Alignment of Unit 2 Containment Airlocks

a. Inspection Scope

The inspectors conducted a partial system walkdown of the Unit 2 upper and lower containment airlocks to verify that operation of the containment airlocks was consistent with the Technical Specifications and licensing basis.

- Updated Final Safety Analysis Report (UFSAR) Section 5.2.4.3, "Equipment Hatched and Personnel Locks"
- Technical Specification 3.6.1.1, "Containment Integrity"
- Technical Specification 3.6.1.3, "Containment Airlocks"
- 12-MHP [Maintenance Head Procedure] 5021.001.030, "Airlock Door Repairs," Revision 4
- 02-OHP [Operations Head Procedure] 4030.STP.010, "Containment Isolation," Revision 9a
- Plant Managers Procedure (PMP) 4010.CAC.001, "Containment Access and Cleanliness," Revision 0
- Attachment 1 to 02-OHP 4030.STP.010, "Containment and Personnel Airlock Checklist," partially performed January 23, 2001
- Attachment 1 to 02-OHP 4030.STP.010, "Containment and Personnel Airlock Checklist," performed January 24, 2001
- Job Order 01023055, Correct interlock malfunction on Unit 2 lower containment airlock

- CR 01023055, Lower containment access interlock did not prevent opening the inner door while the outer door was open
- CR 01023065, Containment equipment and personnel airlock checklist not completed

b. Issues and Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed fire protection walkdowns of the following risk-significant plant areas: the Unit 1 essential service water (ESW) pumps and motor control center (Fire Zones 29A, 29B, and 29E), the spent fuel pool heat exchanger pump room (Fire Zone 36), the Unit 1 “A” Train and “B” Train 4kV switchgear rooms (Fire Zones 40A and 40B), and the Unit 1 600V switchgear mezzanine area (Fire Zone 41). The inspectors verified that fire zone conditions were consistent with assumptions in the licensee’s fire hazard analysis. The inspectors walked down fire detection and suppression equipment, assessed the material condition of fire control equipment, and evaluated the control of transient combustible materials. The following documents were reviewed during this inspection:

- UFSAR Section 9.8.1, “Fire Protection System”
- D. C. Cook Nuclear Plant Fire Hazards Analysis, Units No. 1 and 2, Revision 8
- D. C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook, February 1995
- Administrative Technical Requirement 1-FP-1, “Unit 1 Fire Detection”
- PMP 2270.CCM.001, “Control of Combustible Materials,” Revision 0
- PMP 2270.FIRE.002, “Responsibilities for Cook Plant Fire Protection Program Document Updates,” Revision 0
- PMP 2270.WBG.001, “Welding, Burning and Grinding Activities,” Revision 0
- Plant Mangers Instruction (PMI) 2270, “Fire Protection,” Revision 26

b. Issues and Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

In NRC Inspection Reports 50-315/00-20; 50-316/00-20 and 50-315/00-22; 50-316/00-22, the NRC identified several non-cited violations associated with the licensee’s implementation of Maintenance Rule requirements. Specifically, the inspectors identified issues associated with identification of maintenance preventable functional failures, monitoring of system unavailability, scoping of emergency operating procedure functions, and effectiveness of corrective actions for (a)(1) structures, systems, and components (SSC’s). Although the licensee’s corrective actions for several of these issues were still in progress at the time of this inspection, the licensee stated that, with the exception of reviews of recent historical data, compliance with the requirements of 10 CFR 50.65 for SSC's categorized under paragraph (a)(2) had been restored. The inspectors reviewed the licensee's implementation of 10 CFR 50.65 (the Maintenance Rule) requirements for selected systems categorized under Maintenance Rule paragraph (a)(2). The inspectors reviewed the following systems classified as (a)(2): steam generator blowdown, switchgear room ventilation, and component cooling water.

For the selected (a)(2) systems, the inspectors reviewed; (1) scoping in accordance with 10 CFR 50.65; (2) the safety significance classification; (3) the basis for (a)(2) classification for the SSCs; and (4) the appropriateness of performance criteria for SSCs classified as (a)(2). The inspectors also discussed system scoping and expert panel evaluation with the Maintenance Rule coordinator. The inspectors reviewed the following general guidance documents for this review:

- NUMARC 93-01, Revision 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"
- Regulatory Guide 1.160, Revision 2, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"
- D. C. Cook Nuclear Plant Units 1 and 2, Probabilistic Risk Assessment, Component Cooling Water System Notebook, October 1991
- CR 00356032, Component Cooling Water system Maintenance Rule history review
- CR 00320067, Discrepancies between the scope determination in document text and the matrix in the approved Steam Generator Blowdown Maintenance Rule Scoping Document
- CR 00353050, Documentation of historical review of work requests/job orders and control room logs on the blowdown system maintenance rule functions
- 01-OHP 4030.STP.011, "Containment Isolation and ISI Valve Operability Test," Revision 21
- D. C. Cook Nuclear Plant Units 1 and 2, Probabilistic Risk Assessment, 4160 and 600 V AC Electric Power System Notebook, September 1991
- CR 00355035, Switchgear Ventilation Maintenance Rule history review
- PMP 4030.001.001, "Impact of Safety Related Ventilation on the Operability of Technical Specification Equipment," Revision 3
- CR 01043042, Inconsistencies identified in Maintenance Rule scoping documents
- CR 01043055, Plant level performance criteria does not reflect all unplanned increases in shutdown risk
- Maintenance Rule Scoping Documents for Component Cooling Water, Steam Generator Blowdown, and Switchgear Ventilation systems

b. Issues and Findings

No findings of significance were identified.

1R13 Maintenance and Emergent Work Control

.1 Unit 2 Rod Control and Instrumentation Problems

a. Inspection Scope

On January 23, 2001, Shutdown Bank "D" was inserted 8 steps in accordance with TS 4.1.3.1.2 but could not be returned to the full out position. In addition, on January 24, 2001, control rod K-10 did not move with the rest of Control Bank "C". The inspectors reviewed the licensee's work planning and risk analysis for the emergent work on both rod control problems. (Additional discussion of events associated with this failure are discussed in Sections 1R15.1 and 1R19 of this report.) The following documents were reviewed as part of this inspection:

- Job Order 01023053, Troubleshoot rod control problems
- Schedule Addition Request Form for Job Order 01023053
- Job Order 01025001, Individual Rod Position Indication (IRPI) for rod K-10 does not appear to be withdrawn
- Schedule Addition Request Form for Job Order 01025001
- PMP 2291.SCH.001, Revision 3, "Work Management Activity Scheduling Process"
- CR 01024033, Wrong data sheet used for work scope addition requests
- CR 01026049, Job Order on Main Feedwater Pump, seal water pump, not reviewed for schedule risk specifically but only reviewed as part of the overall forced outage schedule
- Performance Assurance Field Observation FO-01-A-035, Risk Assessment During Unit 2 Forced Outage

b. Issue and Findings

No findings of significance were identified.

.2 Replacement of Unit 1 Reactor Protection System Power Supply

a. Inspection Scope

On February 8, 2001, the control room operators received a reactor protection system (RPS) Train "B" trouble annunciator due to the failure of one of the two redundant 48 VDC power RPS Train "B" power supplies. In accordance with annunciator response procedure 01 OHP 4024.110, the licensee declared RPS Train "B" inoperable and entered TS 3.3.1.1 "Reactor Trip System Instrumentation," Action 1. Action 1 required that the unit be placed in Mode 3 (Hot Standby) within 6 hours. The licensee commenced a reactor shutdown, evaluated RPS operability, and revised the annunciator response procedure to recognize that the RPS train remained operable with a loss of one redundant power supply. The licensee exited the TS action statement approximately 4 hours after declaring the Train "B" RPS inoperable. Reactor power was reduced to approximately 50 percent. The licensee restored the failed "B" RPS power supply to service on February 9, 2001. The inspectors observed control room activities during the power reduction, observed portions of the maintenance activities on the RPS power supply, and verified licensee compliance with applicable Technical Specifications. The inspectors reviewed the following documents during this review:

- CR 01039047, Input breaker for reactor protection Train "B" 48 volt power supply tripped open
- JO C01029047, Remove failed RPS power supply and replace with new power supply
- 01 IHP 40030.STP.411, "Train "B" Reactor Protection System and Engineered Safety Features Reactor Trip Break and Solid State Protection System Automatic Trip/Actuation Logic Functional Test," Revision 5
- CR 01040013, During power supply replacement on Unit 1 Train "B" logic cabinet, the 15 volt power supply failed causing reactor trip breaker B to open (the bypass breaker was closed preventing a reactor trip)
- CR 01040002, Source range detector N-31 and N-32 inadvertently energized during reactor protection system power supply maintenance
- Unit 1 control room logs for February 8 - 9, 2001.
- 10 CFR 50.72 Daily Event Number 37728, February 8, 2001

b. Issues and Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 Operability Evaluation of the Failure of Two Unit 2 Shutdown Banks to Move

a. Inspection Scope

On January 22, 2001, the licensee was performing a routine surveillance test of the Unit 2 rod control system. During the surveillance test, Shutdown Bank "C" would not respond to movement commands. The licensee entered TS action statement 3.1.3.1.b, which required that the plant be placed in Mode 3 (Hot Standby) within 6 hours. Additional testing identified that Shutdown Bank "D" also would not respond to movement commands. Subsequently, the licensee performed an operability review and decided that the shutdown banks remained operable and that TS action statement 3.1.3.1.b should be exited. The inspectors reviewed the licensee's operability evaluation of the failure of two shutdown banks to move. The following documents were reviewed as part of this inspection:

- 02-OHP 4030.STP.015, Revision 9, "Full Length Control Rod Operability Test"
- Control room logs
- TS 3.1.3.1, Movable Control Assemblies
- CR 01022037, Malfunction of the rod control system
- CR 01029009, Three separate electrical connection problems resulted in an extended forced outage for Unit 2
- CR 01024035, NRC disagrees with declaring rod banks operable and not completing the TS 3.1.3.1. Action b required shutdown to hot standby

b. Issue and Findings

No findings of significance were identified.

.2 Operability of the PostAccident Containment Hydrogen Monitoring System (PACHMS) Backup Air Bottles

a. Inspection Scope

On November 30, 2000, the licensee identified that an incomplete design verification for the PACHMS backup air supply had been performed. The licensee documented this issue in CR 00335112. As discussed in Licensee Event Report (LER) 50-315/92-011, the licensee had previously identified that a backup air supply to the control air system, was required to meet the guidelines of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident." The licensee installed a backup air system to support postaccident cycling of the air operated hydrogen sampling isolation valves during a loss of control air. Condition Report 00335112 documents that the PACHMS backup air system might not have had sufficient air capacity to meet design requirements. During the investigation related to CR 00335112, the licensee identified that the Unit 2, Train "A" PACHMS backup air supply pressure was less than the minimum required pressure specified in design calculations. As discussed in Section 1R22.3 below, the acceptance criteria contained in the backup air system periodic testing procedures was non-conservative relative to the design specification. The licensee performed an operability determination to evaluate PACHMS performance with backup air system pressure less than design requirements. The inspectors reviewed the basis for the operability determination, supporting procedures, licensing basis requirements, and performed a field walkdown to identify any conditions that were inconsistent with the operability evaluation. The inspectors reviewed the following documents during this review:

- NUREG-0737, "Clarification of TMI Action Plan Requirements"
- NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident"
- 12 THP [Technical Head Procedure] 6020 PAS.618, PACHMS Backup Air, Revision 0a
- 12 EHP [Engineering Head Procedure] 4030.STP.254, "Unit 1 and Unit 2 PostAccident Containment Hydrogen Monitoring System (PACHMS) Backup Air Supply (BUAS) Check Valve Leak Test," Revision 1
- System Description SD-12-PAS-110, "PostAccident Containment Hydrogen Monitoring System (PACHMS)," Revision 0
- Calculation CHEP 921020, "Determine Design Parameters for PACHMS Backup Air Supply (Unit 1)"
- Calculation CHEP 921120, "Determine Design Parameters for PACHMS Backup Air Supply (Unit 1)"
- CR 00335112, Incomplete design verification for PostAccident Containment Hydrogen Monitoring System Backup air supply
- CR 00342053, Change the Unit 2 Train "A" PACHMS Back-up air supply bottles to install two bottles in excess of 2420 psig

- CR 00-1104, Control Air Leak found at inlet to 2-XSO-622, Containment Hydrogen recombiner HR1 Area Sample containment isolation valve ECR-22 control air solenoid
- AR A0207033, Air leak at 2-XSO-622 inlet fitting
- LER 50-315/92-011-00, “Inoperability of the PostAccident Containment Monitoring System (PACHMS) to Operate per Licensee Commitments”
- Drawing No. OP-2-5120KK, “Control Air System Auxiliary Bldg. Tapoffs Unit #2"
- Drawing No. OP-2-5141D, “Flow Diagram PostAccident Sampling Containment Hydrogen Unit No. 2"
- CR 01023027, During followup to NRC questions, identified that there is no procedure to determine containment hydrogen concentration in accordance with NUREG 0737, Item II.B.3

b. Issues and Findings

No findings of significance were identified.

.3 Unit 2 Aggregate Operability Determination for Mode Ascension

a. Inspection Scope

On January 22, 2001, the licensee initiated a shutdown due to a rod control system failure that resulted in the inability to meet rod insertion limitations specified in TS 3.1.3.5, “Shutdown Rod Insertion Limit.” Additional discussion of events associated with this failure are discussed in Section 1R13.1, 1R15.1, and 1R19 of this report. In accordance with PMP 7030.OPR.001, Section 3.8.3, the licensee performed an aggregate operability determination prior to ascension to Mode 2. The purpose of this operability determination was to determine the combined impact of degraded or nonconforming conditions and associated compensatory measures. The inspectors reviewed the following documents during this review:

- CR 01023049, Aggregate operability determination evaluation for Unit 2 startup following repairs to the rod control system
- PMP 7030.OPR.001, “Operability Determination,” Revision 4

b. Issues and Findings

No findings of significance were identified.

.4 Hydraulic Locking of Containment Isolation Valves

a. Inspection Scope

On October 5, 2000, the licensee initiated CR 00279011 to identify that hydraulic locking of certain containment isolation valves was possible under some circumstances. Hydraulic locking could result in the failure of some containment isolation valves to fully close if the pressure in the associated system was sufficiently high. The licensee evaluated the operability of these containment isolation valves in CR 00279011 (Unit 2) and CR 00295013 (Unit 1). The inspectors reviewed the operability evaluations and discussed the evaluation conclusions with operations and engineering personnel. The inspectors reviewed the following documents during this inspection:

- CR 00279011, the evaluation for CR 00-6696 improperly evaluated the possibility of hydraulic locking of non-essential service water containment isolation valves
- Vendor Technical Manual (VTM) ITEV-0002, Vendor Technical Manual for ITT Engineered Valves
- MPR Associates Technical Report MPR-2131, "Evaluation of D. C. Cook Unit 2 Piping Segments For Potential Thermal Overpressurization (GL 96-06)," Revision 2
- CR 00-6696, Containment isolation valves such as non-essential service water can hydraulic lock open when required to close if outermost isolation valve closes first
- Flow Diagram OP-2-5114A, "Flow Diagram Non-Essential Service Water"
- CR 00295013, Hydraulic locking of containment isolation valves (air operated diaphragm) can occur due to system alignment or CIV closure sequence

b. Issues and Findings

No findings of significance were identified.

1R16 Operator Workarounds

.1 Review of Selected Operator Workarounds

a. Inspection Scope

The inspectors evaluated the following operator workarounds (OWAs) to determine if the applicable system function was impacted or if the OWA affected the operator's ability to implement abnormal or emergency operating procedures:

- OWA 00-02 Residual Heat Removal vibrations when aligned to the normal cooldown flow path.
- OWA 99-06 Placing the Startup flashtank in service requires realignment of the other unit to prevent water hammer.

b. Issues and Findings

No findings of significance were identified.

.2 Review of the Cumulative Effect of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative effect of OWAs on equipment availability, initiating event frequency, and the ability of the operators to implement abnormal or emergency operating procedures. As part of this inspection, the inspectors reviewed the following licensee documents:

- PMP 4010.OWA.001, “Oversight and Control of Operator Workarounds”, Revision 1
- Workaround list for Unit 1, 2, and common

The inspectors interviewed the Workaround Coordinator to discuss the oversight and control of operator workarounds. The inspectors also observed the periodic management meeting in which operator workarounds are discussed and statused.

b. Issues and Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

.1 Post Maintenance Testing Following Troubleshooting of the Unit 2 Shutdown Banks

a. Inspection Scope

On January 22, 2001, during the performance of Surveillance Test 02-OHP 4030.STP.015, “Full Length Control Rod Operability Test,” the control room operators identified that Unit 2 Shutdown Bank “C” failed to move. The licensee declared Shutdown Bank “C” inoperable and commenced a reactor shutdown in accordance with Action Statement b of TS 3.1.3.1.

The licensee determined that the immovable control rods could be tripped; therefore, the licensee believed that the rods were capable of performing their intended safety function. The licensee’s operability determination is discussed above in Section 1R15.1.

The licensee determined that the troubleshooting effort on the rod control system potentially placed the plant in a condition that could result in an unexpected reactor trip. During the follow-up troubleshooting for the immovable control rods, the licensee inserted Shutdown Bank “D” but the control room operators were not able to withdraw Shutdown Bank “D”. Because Shutdown Bank “D” was inserted below the rod insertion limits of TS 3.1.5.1, the licensee commenced a reactor shutdown in accordance with TS 3.0.3. The Unit 2 reactor was shutdown on January 23, 2001.

The inspectors reviewed the licensee’s troubleshooting efforts, both prior to and after the reactor shutdown, to verify that the rod control system and rod position indication were returned to an operable condition prior to restart of the unit. The inspectors interviewed members of the licensee’s troubleshooting team and discussed the troubleshooting and post maintenance test results. In addition, the inspectors reviewed the following documents:

- TS 3.1.3.1, "Moveable Control Rod Assemblies - Group Height"
- TS 3.1.3.5, "Shutdown Rod Insertion Limit"
- UFSAR Section 7.3, "Control Systems"
- 02-IHP [Instrument Head Procedure] 4030.STP.517, "Rod Control Logic Slave Cycler Current Order Test," Revision 3
- 02-IHP 4030.STP.518, "Rod Control Coil Current Test," Revision 0
- 02-OHP 4030.STP.015, "Full Length Control Rod Operability Test," Revision
- 02-OHP SP.227, "Control Rod Testing in MODE 3," Revision 0
- PMP 2291.PMT.001, "Work Management Post Maintenance Testing Matrices," Revision 2
- PMP 2291 TRS.001,"Troubleshooting," Revision 0a
- JO 01023009, Troubleshoot and repair rod control circuits
- JO 01025001, Troubleshoot control rod drive mechanism K-10
- CR 01022037, During 02-OHP 4030.STP.015, Shutdown Bank C did not indicate movement
- CR 01025001, Rod K-10 does not appear to be withdrawing
- CR 01036023, NRC identified that the troubleshooting plan moved control rods in excess of TS required rod alignment limits

b. Issues and Findings

No findings of significance were identified.

.2 Post Maintenance Testing Following Troubleshooting of Unit 2 Control Rod Bank A

a. Inspection Scope

On January 26, 2001, the licensee was attempting to start up the Unit 2 reactor following the correction of the rod control problems discussed above. When the operators selected Shutdown Bank "A" for withdrawal, Control Bank "A," moved instead. The operators fully inserted all of the control rods after withdrawing Control Bank "A" 12 steps.

The inspectors verified that the licensee's troubleshooting and post maintenance testing restored the rod control system to an operable status. The inspectors also discussed the licensee's troubleshooting efforts with members of the licensee's team. In addition, the following documents were reviewed:

- 02-OHP SP.227, "Control Rod Testing in MODE 3," Revision 0
- PMP 2291.PMT.001, "Work Management Post Maintenance Testing Matrices," Revision 2
- JO 01025069, Clean control rod bank selector switch
- CR 00-8989, Control Bank "A" withdrew during startup instead of Shutdown Bank "A"
- CR 01025069, While commencing Unit 2 reactor startup, Control Bank "A" rods moved with the selector switch selected for Shutdown Bank "A"

b. Issues and Findings

No findings of significance were identified.

.3 Water Found in Unit 1 West Motor Driven Auxiliary Feed Pump Outboard Pump Bearing

a. Inspection Scope

On February 1, 2001, the licensee sampled the Unit 1 West Motor Driven Auxiliary Feed Pump (MDAFP) outboard pump bearing oil as part of a routine surveillance. The samples indicated that water was present in the bearing lubricating oil; however, the operators reported that the bearing temperatures remained steady during past operation of the pump. This was documented in Condition Report 01032008. The inspectors verified that the licensee's post maintenance testing adequately tested the Unit 1 West MDAFP following oil replacement. The inspectors also discussed the licensee's post-maintenance testing with the system engineer, observed portions of the testing following the packing replacement and reviewed the following documents:

- PMP 2291.PMT.001, "Work Management Post Maintenance Testing Matrices," Revision 2
- JO 00348124, Adjust packing on Unit 1 West MDAFP (Outboard end is spraying during run)
- CR 00348124, Unit 1 West MDAFP outboard pump packing is leaking
- CR 01032008, Water was found in the oil from the Unit 1 West MDAFP outboard pump bearing

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Unit 2 Full Length Control Rod Operability Test

a. Inspection Scope

The inspectors reviewed the results of a scheduled quarterly control rod operability test performed on January 22, 2001. The operability impact of this test was discussed in Section 1R15, above. At about 2:05 p.m. the licensee determined that Shutdown Bank "C" would not move and the operators entered TS 3.1.3.1, Action b. Subsequently, the licensee continued the surveillance test and determined that Shutdown Bank "D" also would not move. The licensee exited the limiting condition for operation after performing an operability review. Following repairs to the rod control system, the TS surveillance was performed successfully. The inspectors assessed procedural compliance, communication, worker performance, and work control associated with the surveillance test performance. The following documents were reviewed during this inspection:

- CR 01022037, Malfunction of the rod control system
- Control room logs

- PMP 4030.EXE.001, "Conduct of Surveillance Testing", Revision 1
- 02-OHP 4030.STP.015, "Full Length Control Rod Operability Test", Revision 9
- TS 3.1.3.1, Movable Control Assemblies

b. Issues and Findings

No findings of significance were identified.

.2 Semi-Annual Fast Start of the Unit 1 "A" Train Diesel Generator

a. Inspection Scope

On January 17, 2001, the licensee performed a routine 6 month surveillance of the Unit 1, CD ("A" Train) Diesel Generator. The inspectors observed a portion of the surveillance test, reviewed the surveillance results and the following documents:

- 01-OHP 4030.STP.027CD, "CD Diesel Generator Operability Test (Train "A")," Revision 15a
- Administrative Technical Requirement 1-EDG-1
- Administrative Technical Requirement 2-EDG-1
- 01-OHP 4030.132.217A, "DG1CD Load Sequence and ESF Testing," Revision 0a

b. Issues and Findings

No findings of significance were identified.

.3 Non-Conservative Acceptance Criteria in PostAccident Containment Hydrogen Monitoring System Test Procedures

a. Inspection Scope

Units 1 and 2 Technical Specification 3.6.4.1, "Combustible Gas Control Hydrogen Analyzers," requires two operable hydrogen analyzers in Modes 1 and 2. The licensee performed monthly and quarterly testing of the backup air supply to the PACHMS containment isolation sample valves to verify functionality of the PACHMS sample isolation valves in the event of a loss of control air. The inspectors reviewed the procedures associated with PACHMS backup air supply testing and discussed PACHMS operation with operations and engineering personnel. The inspectors reviewed the following documents during this inspection:

- 01 OHP 4023.E-1, "Loss of Reactor or Secondary Coolant," Revision 8
- 01 OHP 4023.E-0, "Reactor Trip or Safety Injection," Revision 15
- 12 EHP 4030 STP.254, "Unit 1 and Unit 2 PostAccident Containment Hydrogen Monitoring System (PACHMS) Back-Up Air Supply (BUAS) Check Valve Leak Test," Revision 6
- 12 THP 6020 PAS.618, "PACHMS Backup Air," Revision 0a
- 12 THP 6020 PAS.002, "PACHMS Operation," Revision 7
- Calculation CHEP 921020, "Determine Design Parameters for PACHMS Backup Air Supply (Unit 1)"

- Calculation CHEP 921120, “Determine Design Parameters for PACHMS Backup Air Supply (Unit 1)”
- CR 00335112, Incomplete design verification for PACHMS Backup air supply
- CR 00342053, Change the Unit 2 Train “A” PACHMS Backup air supply bottles to install two bottles in excess of 2420 psig
- PMP 2081 EPP.208, “Emergency Radiation Protection,” Revision 3
- CR 01024057, NRC identified that Unit 2, Train “A” PACHMS backup air system was tested with an acceptance criteria that was non-conservative to the calculation of record

b. <u>Issues and Findings</u>

The inspectors identified one GREEN barrier integrity non-cited violation associated with the adequacy of PACHMS backup air system surveillance testing. Each train of the PACHMS backup air supply consists of two compressed air bottles and a pressure regulating panel. The calculations of record for the PACHMS backup air supply, CHEP 921120 and CHEP 921020, determined the minimum acceptable backup air supply bottle pressure assuming a minimum 12 hour postaccident air capacity. The 12 hour capacity was consistent with the expected PACHMS air bottle performance stated in Section 6.1.16 of Procedure 12 THP 6020 PAS.003, “PACHMS Operation.” Additionally, Section 4.5 of Procedure 12 EHP 4030 STP.254, “Unit 1 and Unit 2 PostAccident Containment Hydrogen Monitoring System (PACHMS) BackUp Air Supply (BUAS) Check Valve Test,” stated that the “design basis for the PACHMS BUAS include[d] 12 hr operability.” The minimum backup air bottle pressure limits contained in the periodic PACHMS BUAS test procedures, 12 EHP 4030 STP.254 and 12 THP 6020 PAS.618, were inconsistent with the above design requirement. Specifically, the minimum bottle pressure for the Unit 2, Train “A” BUAS contained in calculation CHEP 921120 was 2420 psig, while steps 4.5 and 5.4.17 of 12 EHP 4030.STP.254 and Step 4.4.18 of 12 THP 6020.PAS.618 required a minimum backup air bottle pressure of 2000 psig. The minimum required pressure for both of the Unit 1 PACHMS BUAS trains and the remaining Unit 2 train were bounded by the test procedure acceptance limits.

As discussed in Section 1R15.2 above, the licensee previously identified the discrepancy between BUAS design requirements and test acceptance criteria during the evaluation of CR 00335112 in December 2000. The licensee concluded that BUAS bottle pressures between 2000 psig and 2420 psig resulted in an operable but non-conforming condition. The major impact of lower than design bottle pressure was the potential to require bottle change out earlier than expected following a postulated accident. Because the backup air bottles were located in close proximity to the lower containment access airlock, early change out of the BUAS bottles following a postulated accident would result in unnecessary radiation dose to workers. The licensee estimated the dose associated with early change out of the air bottles was greater than 5 rem, but less than the 10 rem procedural limit for corrective action emergency exposures. The licensee failed to correct the non-conservative acceptance criteria for Unit 2, Train “A” of the BUAS in 12 EHP 4030.STP.254 prior to a subsequent performance of the test on January 24, 2001. After the inspectors identified that the surveillance procedure was performed again with non-conservative test acceptance criteria, the licensee revised 12 EHP 4030.STP.254 to reflect the Unit 2 Train “A” minimum design pressure of 2420 psig and initiated CR 01024057.

Appendix B of 10 CFR 50, Criterion XI, “Test Control,” required, in part, that test procedures incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, the licensee failed to incorporate the applicable design limits for the PACHMS backup air supply in the minimum acceptable bottle pressure limits of procedure 12 EHP 4030.STP.254 and 12 THP 6020.PAS.618. The inspectors concluded that this failure constituted a Non-Cited Violation (50-316/01-02-01) of 10 CFR 50, Appendix B, criterion XI. This violation is in the licensee’s corrective action system as CR 01024057. Additionally, the licensee has developed Engineering Action Plan PAS 01-591, to address resolution of design and licensing basis issues related to PACHMS. This issue was screened as GREEN (very low safety significance) after a Phase 1 Significance Determination Process.

The inspectors concluded that this failure had a credible impact on safety and was more than a minor violation of NRC requirements. Early failure of the PACHMS backup air supply could result in the inability to operate containment hydrogen sample valves. Emergency operating procedures were written to utilize results obtained from the PACHMS system to determine appropriate postaccident follow-up actions. Additionally, the Updated Final Safety Analysis Report Section 7.8, “PostAccident Monitoring Instrumentation,” designated PACHMS as a system providing primary information required to permit the control room operator to take specific manual controlled actions for which no automatic control was provided and that are required for safety systems to accomplish their safety systems for design basis events. Because control of containment hydrogen concentration potentially impacts the capability of the containment building to perform its design basis function, the inspectors determined that this failure impacted the barrier systems cornerstone. This issue did not result in an actual open pathway in the physical integrity of the reactor containment or an actual reduction of the atmospheric pressure control function of the reactor containment.

.4 Reactor Thermal Power Measurement

a. Inspection Scope

The inspectors reviewed the reactor thermal power measurement surveillance procedure. Technical Specification Table 4.3-1, “Reactor Trip System Instrumentation Surveillance Requirements,” required a daily comparison between indicated nuclear instrumentation power and heat balance power in Mode 1 above 15 percent power. Additionally, license condition 2.C.(1), “Maximum Power Level,” limited the steady state reactor core power level to 3250 MW (Unit 1) and 3411 MW (Unit 2). The inspectors reviewed the reactor thermal power measurement procedure and methodology, reviewed approximately 2 weeks of thermal power data for each unit, and verified that the measurement method was consistent with TS requirements. The inspectors reviewed the following documents during this review:

- 02-OHP-4030.STP.029, “Reactor Thermal Power,” Revision 13
- 01-OHP-4030.STP.029, “Reactor Thermal Power,” Revision 12
- Plant Process Computer Nuclear Steam Supply System Program Reference Manual, Chapter 11, Calorimetric Program
- I&C Information Change Package (ICP) 00059, Uncertainty Evaluation of Manual Calorimetric Procedure
- UFSAR Table 4.1-5, “Steam Generator Design Data”

- UFSAR Table 4.5-6, “Reactor Coolant Pumps Design Data”
- CR 99-25754, Discrepancies in how nuclear steam supply system net heat losses are used in analyses, procedures and UFSAR
- CR 00-7469, Reactor thermal power determination does not demonstrate required accuracy
- CR 00280011, Cumbersome thermal power procedure does not give clear guidance when PPC thermal power and manual calorimetric power are outside specified tolerance
- CR 01038038, NRC identified potentially nonconservative error in the plant computer calorimetric calculation
- CR 01044033, Administrative requirements for compensatory actions were not followed for maximum reactor power limit of 99.8 percent following NRC identification of PPC calculation error

b. Issues and Findings

No findings of significance were identified.

**4. OTHER ACTIVITIES (OA)**

4OA1 Performance Indicator Verification

a. Inspection Scope

Using Inspection Procedure 71151, the inspectors reviewed the licensee’s program for the gathering and submittal of data for Unit 2:

- Unplanned Scrams per 7,000 Critical Hours portion of the Initiating Events cornerstone;

- Scrams with Loss of Normal Heat Removal portion of the Initiating Events cornerstone;

- Unplanned Power Changes per 7,000 Critical Hours portion of the Initiating Events cornerstone;

- Safety System Unavailability - Emergency AC Power System portion of the Mitigating Systems cornerstone;

- Safety System Unavailability - Heat Removal System (AFW) portion of the Mitigating Systems cornerstone;

- Safety System Functional Failures portion of the Mitigating Systems cornerstone;

- Reactor Coolant System Activity portion of the Barrier Integrity cornerstone; and

- Reactor Coolant System Identified Leak Rate portion of the Barrier Integrity cornerstone.

The inspectors utilized the following documents during this review:

- PMP 7110.PIP.001, "Regulatory Oversight Program Performance Indicators," Revision 0
- PMI 7110, "Regulatory Oversight Program," Revision 0
- Performance Assurance Field Observation FO-01-A-027, "Review of Source Data for RHR and Aux Feedwater NRC Cornerstone Performance Indicators"
- Regulatory Oversight Performance Indicators for Second Quarter 2000
- Regulatory Oversight Performance Indicators for Third Quarter 2000
- Data Sheets for Third Quarter Information per PMP 7110.PIP.001
- Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0
- 02-OHP 4030.STP.050W, "West Residual Heat Removal Train Operability Test Modes 1-4," Revision 8
- 02-OHP 4030.STP.050E, "East Residual Heat Removal Train Operability Test Modes 1-4," Revision 8
- 02-OHP 4030.STP.027CD, "CD Diesel Generator Operability Test (Train "A")," Revision 16
- 02-OHP 4030.STP.027AB, "AB Diesel Generator Operability Test (Train "B")," Revision 14
- Control Room logs
- CR 01029040, NRC inspectors questioned the licensee's basis for not counting unavailability time when rolling a diesel engine over to check for moisture in the cylinders.

b. Issues and Findings

Due to the extended plant shutdown, the licensee had not gathered historical data required for the calculation of certain Performance Indicators (PIs). Following restart of Unit 2, the licensee began collecting data and the inspectors reviewed the data for the third quarter of 2000.

Following discussions with the NRC inspectors, the licensee decided to change their unavailability monitoring program to commence counting the time spent checking the diesel generators for moisture prior to testing as unavailable time. The licensee stated that they would submit a request for interpretation for counting unavailability under this circumstance to the Nuclear Energy Institute (NEI). The additional unavailability associated with moisture checks would not cause the performance indicator to cross a threshold.

4OA3 Event Follow-Up

.1 Licensee Event Reports

a. Inspection Scope

The inspectors reviewed the corrective actions associated with the following licensee event reports.

b. Issues and Findings

b.1 (Closed) Licensee Event Report 50-316/96008-00: Two cam follower springs found broken on Unit 2 CD emergency diesel generator due to manufacturing flaw, Part 21 follow-up report. On April 13, 1996, the Unit 2 CD emergency diesel generator was manually tripped during a test run when the licensee determined that the number 4 rear bank cylinder was not producing any power. The licensee's subsequent investigation found that two cam follower springs were broken due to a manufacturing defect. This event and the licensee's follow-up actions were discussed in Inspection Report 50-315/96006; 50-316/96006. This issue was entered into the licensee's corrective action program as Condition Report 96-0622. This LER is closed.

(Closed) Licensee Event Report 315/94-004-00: Failure of three pressurizer safety valves to lift within tolerance. On April 6, 1994 with Unit 1 in Mode 5 (Cold Shutdown) it was determined that all three of the pressurizer safety valves, which were sent to an off site laboratory for testing, were found with lift settings outside of the Technical Specification acceptance criteria. Acceptable settings were between 2461 psig and 2509 psig. Valve 1-SV-45A was found to have a lift setpoint of 2536 psig, valve 1-SV-45B had a lift setpoint of 2535 psig and 1-SV-45C had a lift setpoint of 2538 psig.

There was minimal safety-significance since the worst case (1-SV-45C lift setpoint of 2538 psig) would have resulted in a maximum transient pressure of 2615 psig (2538 psig plus 3 percent accumulation to attain its full rated lift). This was below the Technical Specification safety limit of 2735 psig.

The licensee installed pressurizer safety valves that were set to within allowed TS tolerance. The inspectors reviewed the LER and the licensee's corrective actions for safety valve setpoint drift and did not identify any significant findings. The licensee had changed out all of the pressurizer and main steam safety valves during the extended outage and implemented disc modifications on some valves. The licensee was to evaluate the new disc material during the next refueling outage and determine if the modifications were successful in reducing setpoint drift. This LER is closed.

(Closed) Licensee Event Report 50-316/95-003-00: Failure of two pressurizer safety valves to meet Technical Specification requirements. On October 5, 1994 with Unit 2 in Mode 6 (no fuel), the pressurizer safety valves were removed for shipment to an offsite lab for set point testing. On February 3, 1995, with Unit 2 in Mode 1 at 100 percent reactor thermal power, the testing vendor determined that two of three Unit 2 pressurizer safety valves were found with lift settings outside the Technical Specification acceptance criteria. Acceptable settings are between 2461 psig and 2509 psig. Valve 2-SV-45A was found to have a lift setpoint of 2524 psig, and valve 2-SV-45C had a lift setpoint of 2538 psig. There was minimal safety significance because the safety valves would still have limited the peak transient pressure to 2615 psig in the event of an over pressure transient. This is below the Technical Specification safety limit of 2735 psig.

The licensee installed pressurizer safety valves that were set to within allowed TS tolerance. The inspectors reviewed the LER and the licensee's corrective actions for safety valve setpoint drift and did not identify any significant findings. The licensee had changed out all of the pressurizer and main steam safety valves during the extended outage and implemented disc modifications on some valves. The licensee was to evaluate the new disc material during the next refueling outage and determine if the modifications were successful in reducing setpoint drift. This LER is closed.

(Closed) Licensee Event Report 50-316/98-003-00: Failure of two pressurizer safety valves to meet Technical Specification requirements. On March 4, 1998, two of three Unit 2 Pressurizer safety valves, 2-SV-45A and 2-SV-45B, were found to have lift set-points that exceeded their Technical Specification value by more than the one percent tolerance allowed Technical Specification 3.4.3. The valves failed to lift within tolerance due to setpoint drift. The valves have been disassembled and inspected, and no cause of the setpoint drift was identified. The observed lift set-points would not have allowed the Reactor Coolant System to exceed 110 percent of the design pressure, 2735 psig. There was minimal safety significance because the safety valves would still have limited the peak pressure to 2647 psig in the event of an over pressure transient.

The inspectors reviewed the LER and the licensee's corrective actions for safety valve setpoint drift and did not identify any significant findings. The licensee had changed out all of the pressurizer and main steam safety valves during the extended outage and implemented disc modifications on some valves. The licensee was to evaluate the new disc material during the next refueling outage and determine if the modifications were successful in reducing setpoint drift. This LER is closed.

(Closed) Licensee Event Report 50-315/94-006-01: Seismic gaps found filled with untreated Styrofoam behind fire seal. Revision 1 of this LER superceded Revision 0 as analysis and evaluation by the licensee had determined that the as found configuration of the seismic gaps was within the design and licensing basis. The inspectors reviewed the LER and did not identify any significant findings. This LER is closed.

(Closed) Licensee Event Report 50-315/94-015-00 and 50-315/94-015-01: The motor for the East motor driven auxiliary feedwater pump (MDAFWP) was not capable of extended operation to support postaccident conditions. Revision 1 of this LER superceded Revision 0 as analysis and evaluation by the licensee had determined that the MDAFWP would have performed its intended safety function in the as found condition. The inspectors' review of the LER did not result in any questions. This LER is closed.

(Closed) Licensee Event Report 50-316/95-007-00: Fire watch tour not conducted due to personnel error. On September 12, 1995, with Unit 2 in Mode 1 at 7 percent rated thermal power, the required hourly fire watch tour for the Unit 2 4kV Switchgear Complex was not completed. This tour had been established in support of equivalent shutdown capability for Appendix R on August 24, 1995. It was determined that from 5:10 a.m. to 7:13 a.m. no fire watch tour was conducted for the area. As this time period of 123 minutes was in excess of the required hourly tour, the Technical Specification Action Statements 3.1.2.3.b and 3.7.1.2.b were not met. The event was of minimal safety significance since fire detection systems for the area were operable and the length of time the area was without a fire tour was short. This failure constituted a violation of minor significance and is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The inspectors reviewed the LER and did not identify any significant findings. This LER is closed.

(Closed) Licensee Event Report 50-315/95-009-00: Fire protection compensatory actions incorrectly established due to personnel error. On September 13, 1995, with Unit 2 at 100 percent power the fire detection system for the Unit 2 Pressurizer Heater Transformer Room and Diesel Generator Ramp/Corridor Areas, Detection Zone 31 was declared inoperable. An hourly fire watch patrol was established to patrol these areas as required by TS 3.3.3.8. With the loss of detection in the area, a continuous patrol was required in this area. A continuous post was not established until 16 hours after Detection Zone 31 was declared inoperable. The event was caused by Fire Protection Section personnel error. During this 16 hour period the affected area was toured approximately every half hour. During that time Unit 2 continued to operate at 100 percent power with its shutdown equipment operable. The inspectors concluded that the failure to comply with TS 3.3.3.8 constituted a violation of minor significance and is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The inspectors reviewed the LER and did not identify any significant findings. This LER is closed.

(Closed) Licensee Event Report 50-315/98-021-01: Oil Drip Pans Not Installed on Reactor Coolant Pump Motors Results in Appendix R Noncompliance. On March 11, 1998, it was determined that potential oil leakage sites existed on the Reactor Coolant Pumps (RCP's) that had not been equipped with an oil collection system. Section III of Appendix R to 10 CFR 50, required RCP's to be equipped with an oil collection system to reduce the fire hazard due to possible oil contact with ignition sources.

The root cause for this condition was the failure to fully implement the requirements of 10 CFR 50, Appendix R, and failure of system walk-downs to identify the condition. A licensee review of the condition concluded that there was no significant compromise to plant safety as no direct leak path to hot surfaces existed.

The inspectors concluded that the failure to install RCP drip pans in accordance with 10 CFR 50, Appendix R, Criterion III, constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the licensee's corrective action program as Condition Report 98-01339. This LER is closed.

b. Closure of Severity Level IV Violations Under Revised Enforcement Policy

On May 1, 2000, the NRC revised NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions," (Enforcement Policy). Section VI.A, "Non-Cited Violation (NCV)," of the Enforcement Policy discusses the NRC's enforcement approach for Severity Level IV violations. The Policy allows dispositioning of a Severity Level IV violation as a non-cited violation provided certain requirements are met. These requirements include entry of the violation into the licensee's corrective action program and restoration of compliance with NRC requirements, as well as other considerations described in the Enforcement Policy. In accordance with the Enforcement Policy, dispositioning of a Severity Level IV violation as an NCV allows closure of the violation without a written response from the licensee. The NRC has conducted a review of the following Severity Level IV violations, and considers it appropriate to close these violations consistent with Section VI.A of the Enforcement Policy:

| | Violation Number | Condition Report Number |
|---|---|---|
| • | 50-315/97004-01<br>50-316/97004-01 | CR 97-00748 |
| • | 50-315/97024-02<br>50-316/97024-02 | CR 97-03644 |
| • | 50-315/97025-01<br>50-316/97025-01 | CR 98-0069 |
| • | 50-315/97025-02<br>50-316/97025-02 | CR 97-02457, 99-18122 |
| • | 50-316/98008-03 | CR 99-18112 |
| • | 50-316/99017-01 | CR 99-21510, 99-14175 |

4OA5 Other

As part of the NRC Inspection Manual Chapter (IMC) 0350 process for the restart of the Cook Nuclear Plant, the IMC 0350 Panel and Region III Senior Reactor Analysts reviewed and assessed the risk significance associated with each open item. Items with high risk significance were added to each units' respective Restart Action Matrix (RAM). Items with low risk significance were evaluated for restart significance, and those items which were required to be resolved prior to restart were also added to each unit's RAM. The closure of the Unit 2 RAM was documented in a letter dated June 13, 2000. The closure of the Unit 1 RAM was documented in a letter dated December 12, 2000. The IMC 0350 Panel determined that a number of the low risk significant open items did not impact the safety of the restart; therefore, these items remained open while the units were restarted.

The inspectors reviewed the low risk significant items to determine if any new information had been obtained which might have changed the original NRC risk determinations. The inspectors also used the following questions from IMC 0610* to determine if any of the open items were of more than minor significance:

- Does the issue have an actual or credible impact on safety?
- Could the issue be reasonably viewed as a precursor to a significant event?
- If left uncorrected, would the same issue under the same conditions become a more significant safety concern?
- Does the issue relate to collecting or reporting performance indicators that would have caused a PI to exceed a threshold?

The inspectors determined that each of the following open items was of minor significance. Based on the minor significance determination, the inspectors did not use the IMC 0609 Significance Determination Process for these open items.

.1 Inspectors' Review of Low Risk Significant Open Items Against Revised Reactor Oversight Process Inspection Procedures

Following the restart of Unit 2, the NRC transitioned the Cook Nuclear Plant to the Revised Reactor Oversight Process (RROP). The RROP implemented a number of periodic inspection procedures to continually assess various aspects of nuclear plant operation. The inspectors also reviewed the open items against the RROP inspection procedures and determined that, for the following low risk significant open items, a RROP inspection procedure covered the area included in the open item.

- (Closed) Inspection Follow-up Item 50-315/94010-01; 50-316/94010-01: Fire watch lesson plan had not been approved or validated. Inspection Procedure 71111.05, "Fire Protection," was implemented to provide periodic assessments to determine if the licensee has implemented a fire protection program that adequately controls combustibles and ignition sources within the plant, provides effectively maintained fire detection and suppression capability, maintains passive fire protection features in good material condition, and puts adequate compensatory measures in place for out-of-service, degraded or inoperable fire protection equipment, systems or features; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96004-04; 50-316/96004-04: Review of training for communications PARs [protective action recommendations]. Inspection Procedures 71114.01, "Exercise Evaluation," and 71114.06, "Drill Evaluation," were implemented to provide periodic observations of emergency plan drills and training evolutions to identify weaknesses and deficiencies in classification, notification and PAR development activities; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96006-08; 50-316/96006-08: Inconsistencies were identified between the operating crews. Inspection Procedure 71111.11, "Licensed Operator Requalification," was implemented to provide periodic assessments of licensed operator performance; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96006-09; 50-316/96006-09: Administrative duties were distracting the Shift Supervisor from oversight responsibilities. Inspection Procedure 71111.11, "Licensed Operator Requalification," was implemented to provide periodic assessments of licensed operator performance; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96006-18; 50-316/96006-18: Weaknesses in licensee's process for identifying rework. Inspection Procedure 71111.12, "Maintenance Rule Implementation," was implemented to provide periodic assessments of maintenance effectiveness; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96006-19; 50-316/96006-19: Foreign material exclusion practices considered weakness. Inspection Procedure 71111.13, “Maintenance Risk Assessments and Emergent Work Control,” was implemented to provide periodic assessments to verify that troubleshooting evolutions and maintenance activities are adequately controlled at the job site to minimize risk to the system/component being worked and that all activities are within the approved work control boundary; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96006-20; 50-316/96006-20: Licensee failed to replace the emergency diesel generator quick exhaust valve diaphragms at the scheduled interval. Inspection Procedure 71111.12, “Maintenance Rule Implementation,” was implemented to provide periodic assessments of maintenance effectiveness; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96006-21; 50-316/96006-21: Procedure adherence issues or inadequate maintenance procedures were identified. Inspection Procedure 71111.12, “Maintenance Rule Implementation,” was implemented to provide periodic assessments of maintenance effectiveness; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96006-22; 50-316/96006-22: Reliance on electronic dosimeter as survey instrument. Inspection Procedure 71121, “Occupational Radiation Safety,” was implemented to provide periodic assessments of radiological worker performance and ALARA goals; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96007-01; 50-316/96007-01: Work control process. Inspection Procedure 71111.13, “Maintenance Risk Assessments and Emergent Work Control,” was implemented to provide periodic assessments to verify that troubleshooting evolutions and maintenance activities are adequately controlled at the job site to minimize risk to the system/component being worked and that all activities are within the approved work control boundary; therefore, this inspection follow-up item is closed.

- (Closed) Inspection Follow-up Item 50-315/96014-02; 50-316/96014-02: Licensee evaluation of Shift Supervisor work authorization. Inspection Procedure 71111.13, “Maintenance Risk Assessments and Emergent Work Control,” was implemented to provide periodic assessments to verify that troubleshooting evolutions and maintenance activities are adequately controlled at the job site to minimize risk to the system/component being worked and that all activities are within the approved work control boundary; therefore, this inspection follow-up item is closed.

- <u>(Closed) Inspection Follow-up Item 50-315/99021-03; 50-316/99021-03</u>: Verify adequacy of long term corrective actions to resolve Generic Letter 89-13 programmatic weaknesses for performance testing of the emergency diesel generator heat exchangers. Inspection Procedure 71111.07, "Heat Sink Performance," was implemented to provide periodic assessments to verify that the licensee has adequately identified and resolved heat sink performance problems that could result in initiating events or affect multiple heat exchangers in mitigating systems and thereby increase risk; therefore, this inspection follow-up item is closed.

.2 <u>Inspectors' Review of Low Risk Significant Open Items</u>

<u>(Closed) Inspection Follow-up Item 50-316/94020-01</u>: Review of licensee's fastener control process. In 1994, the licensee quality control inspectors identified that the studs used for the Unit 2 pressurizer safety valves were two different types. This issue was documented in CR 94-1951. As a corrective action for this CR, Procedure 12-MHP [Maintenance Head Procedure] 5021.001.092, "Pressurizer Safety Valves Removal and Installation," Revision 4, was changed to require independent verification and documentation that all of the inlet and outlet studs were of ASTM SA-453 Grade 660 stainless steel. This inspection follow-up item is closed.

<u>(Closed) Inspection Follow-up Item 50-315/95012-02; 50-316/95012-02</u>: Review of licensee's large bore piping reconstitution program. The licensee's large bore piping reconstitution program was reviewed as part of Unit 2 RAM Item 1.37. The closure of this RAM Item was documented in NRC Inspection Report 50-315/00-13; 50-316/00-13. The inspection report stated that, "To correct the discrepancies, the licensee issued design change package 2-DCP-647. The licensee also added administrative controls to prevent changing Unit 2 operational Modes until the required modifications for each Mode were completed. The inspectors reviewed the licensee's evaluation and proposed corrective actions and concluded that the licensee's actions adequately address the identified deficiencies." This inspection follow-up item is closed.

<u>(Closed) Inspection Follow-up Item 50-315/96005-01; 50-316/96005-01</u>: Auxiliary feedwater flow retention. In response to the spurious actuation of the auxiliary feedwater flow retention circuit, the licensee installed design change 12-DCP-0817, "Revise Auxiliary Feedwater Flow Retention Circuit." The installation of this design change was discussed in NRC Inspection Report 50-315/97009; 50-316/97009. This inspection follow-up item is closed.

<u>(Closed) Inspection Follow-up Item 50-3105/96006-16; 50-316/96006-16</u>: Backlog of modification packages. Prior to the restart of either unit, the NRC developed two unit-specific Restart Action Matrices. Each unit specific Restart Action Matrix included an evaluation of Inspection Manual Chapter 0350, Staff Guidelines for Restart Approval Item C.4.i, "Maintenance Backlog Managed and Impact on Operation Assessed." For Unit 2, this evaluation was documented in Inspection Report 50-315/00-04; 50-316/00-04. The Unit 1 maintenance backlog evaluation was documented in Inspection Report 50-315/00-23; 50-316/00-23. These inspection reports concluded that, for each unit, an evaluation of the backlogged items assured the inspectors that the restart scoping process was satisfactory and deferred actions did not individually or

collectively have a risk-significant impact on restart, containment performance or fire suppression. Based on the results of the restart inspections, this inspection follow-up item is closed.

(Closed) Inspection Follow-up Item 50-315/98004-18; 50-316/98004-18: NRC review of the quality assurance audit methodology. In a follow-up inspection to the NRC Architect Engineer (AE) team inspection, the inspectors concluded the licensee's Quality Assurance organization did not identify the extent of the problems identified by the AE team. The licensee wrote CR 98-3332 to address this issue. Prior to the restart of Unit 2, quality assurance in engineering was reassessed as part of the Unit 2 Restart Readiness Team Inspection documented in NRC Inspection Report 50-315/00-03; 50-316/00-03. This inspection report concluded that, "The team considered the Performance Assurance department audits of engineering adequate and the findings appropriately resolved." This inspection follow-up item is closed.

(Closed) Unresolved Item 50-315/99012-01, 50-316/99012-01: Commitments Related to NRC Bulletin 88-10 inappropriately closed. NRC Bulletin 88-10, Nonconforming Molded-Case Circuit Breakers, was issued on November 11, 1988. D. C. Cook submitted a response to the bulletin on May 16, 1989, which included nine commitments. During the System Readiness Review it was discovered that seven of the nine commitments were closed with insufficient documentation to determine if the commitments were satisfied.

The inspectors reviewed the licensee's corrective and preventive actions which were documented in Condition Report 99-04344. The inspectors determined that this issue constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC Enforcement Policy. This Item is Closed

(Closed) Notice of Deviation 50-315/97002-08: The licensee moved new fuel shipping containers, containing new fuel assemblies, and weighing approximately 4 tons, higher than 7 feet above the floor when moving the containers to a location to be unloaded. The NRC inspectors had identified multiple problems with the licensee's control and movement of new fuel. These issues were documented in the several violations issued in Inspection Report 50-315/97002. The closure of these violations was documented in Inspection Report 50-315/2000-019. Since this Notice of Deviation was issued the inspectors have observed the licensee during new fuel receipt and observed that the containers are only moved higher than 7 feet above the floor when allowed by procedure and license basis. This Deviation is closed.

(Closed) Notice of Deviation 50-315/97004-05: Three of four recorder pens inoperable for the power range channels that were capable of recording overpower excursions up to 200 percent of full power. The licensee has implemented a program to improve operator human performance. As part of the improvement process, operator observation of the control panels has greatly improved as documented in various NRC inspection reports such as 50-315/2000-019 and 50-315/2000-025. This Deviation is closed.

4OA6 Management Meetings

The inspectors presented the inspection results to licensee management listed below on February, 14, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

## PARTIAL LIST OF PERSONS CONTACTED

R. Crane, Regulatory Affairs
R. Gaston, Regulatory Affairs
J. Gebbie, Plant Engineering
J. Giesnner, Assistant Operations Manager
S. Greenlee, Director, Nuclear Engineering & Regulatory Affairs
M. Hoskins, System Engineering
S. Lacey, Director, Engineering
J. Mathis, Regulatory Affairs
R. Meister, Regulatory Affairs
D. Moul, Assistant Operations Manager
J. Nadeau, Corrective Action Department Supervisor
T. Noonan, Director, Performance Assurance
J. Pollock, Plant Manager
T. Quaka, Engineering
J. St. Amand, Engineering Programs Supervisor
L. Thornsberry, Engineering Programs Manager
L. Weber, Manager, Operations

## LIST OF INSPECTIONS PERFORMED

The following inspectable-area procedures were used to perform inspections during the report period. Documented findings are contained in the body of the report.

| Inspection Procedure | | Report Section |
|---|---|---|
| Number | Title | |
| 71111-04 | Equipment Alignments | 1R04 |
| 71111-05 | Fire Protection | 1R05 |
| 71111-12 | Maintenance Rule Implementation | 1R12 |
| 71111-13 | Maintenance and Emergent Work Control | 1R13 |
| 71111-15 | Operability Evaluations | 1R15 |
| 71111-16 | Operator Workarounds | 1R16 |
| 71111-19 | Post-Maintenance Testing | 1R19 |
| 71111-22 | Surveillance Testing | 1R22 |
| 71151 | Performance Indicator Verification | 4OA1 |
| 71153 | Event Followup | 4OA3 |

## ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

| | | |
|---|---|---|
| 50-316/2001-01 | NCV | Non-conservative test acceptance criteria in PostAccident Hydrogen Monitoring System (PACHMS) backup air system test procedure |

Closed

| | | |
|---|---|---|
| 50-316/2001-01 | NCV | Non-conservative test acceptance criteria in PostAccident Hydrogen Monitoring System (PACHMS) backup air system test procedure |
| 50-315/94004-00 | LER | Failure of three pressurizer safety valves to lift within tolerance |
| 50-315/94006-01 | LER | Seismic gaps found filled with untreated Styrofoam behind fire seal |
| 50-315/94010-01<br>50-316/94010-01 | IFI | Fire watch lesson plan had not been approved or validated |
| 50-315/94015-00<br>50-315/94015-01 | LER | The motor for the East motor driven auxiliary feedwater pump was not capable of extended operation to support postaccident conditions |
| 50-316/94020-01 | IFI | Review of licensee's fastener control process |
| 50-316/95003-00 | LER | Failure of two pressurizer safety valves to meet Technical Specification requirements |
| 50-316/95007-00 | LER | Fire watch tour not conducted due to personnel error |
| 50-315/95009-00 | LER | Fire protection compensatory actions incorrectly established due to personnel error |
| 50-315/95012-02<br>50-316/95012-02 | IFI | Review of licensee's large bore piping reconstitution program |
| 50-315/96004-04<br>50-316/96004-04 | IFI | Review of training for communications PARs |
| 50-315/96005-01<br>50-316/96005-01 | IFI | Auxiliary feedwater flow retention |
| 50-315/96006-08<br>50-316/96006-08 | IFI | Operating crews did not function the same |

| | | |
|---|---|---|
| 50-315/96006-09<br>50-316/96006-09 | IFI | Administrative activities were distracting the Shift Supervisor from oversight responsibilities |
| 50-315/96006-16<br>50-316/96006-16 | IFI | Backlog of modification packages |
| 50-315/96006-18<br>50-316/96006-18 | IFI | Weaknesses in licensee's process for identifying rework |
| 50-315/96006-19<br>50-316/96006-19 | IFI | Foreign material exclusion practices considered weakness |
| 50-315/96006-20<br>50-316/96006-20 | IFI | Licensee failed to replace the emergency diesel generator quick exhaust valve diaphragms at the scheduled interval |
| 50-315/96006-21<br>50-316/96006-21 | IFI | Procedure adherence issues or inadequate maintenance procedures were identified |
| 50-315/96006-22<br>50-316/96006-22 | IFI | Reliance on electronic dosimeter as survey instrument |
| 50-315/96007-01<br>50-316/96007-01 | IFI | Work control process |
| 50-316/96008-00 | LER | Two cam follower springs found broken on Unit 2 CD emergency diesel generator due to manufacturing flaw, 10 CFR 21 follow-up report |
| 50-315/96014-02<br>50-316/96014-02 | IFI | Licensee evaluation of Shift Supervisor work authorization |
| 50-315/97002-08 | DEV | Lifting of Heavy Loads higher than 7 feet |
| 50-315/97004-01<br>50-316/97004-01 | VIO | Failure to follow procedures results in inadvertent ESF actuation |
| 50-315/97004-05 | DEV | Overpower Chart Recorder Pen Inoperability |
| 50-315/97024-02<br>50-316/97024-02 | VIO | Failure to implement quality assurance commensurate with safety function |
| 50-315/97025-01<br>50-316/97025-01 | VIO | Failure to report an event outside the design basis in a timely manner in accordance with 10 CFR 50.72 |
| 50-315/97025-02<br>50-316/97025-02 | VIO | Failure to report an event outside the design basis in a timely manner in accordance with 10 CFR 50.73 |
| 50-316/98003-00 | LER | Failure of two pressurizer safety valves to meet Technical Specification requirements |

| | | |
|---|---|---|
| 50-315/98004-18<br>50-316/98004-18 | IFI | NRC review of the quality assurance audit methodology |
| 50-316/98008-03 | VIO | Failure to follow a procedure when using a continuous use procedure to operate in Mode 5 |
| 50-315/98021-01 | LER | Oil Drip Pans Not Installed on Reactor Coolant Pump Motors Results in Appendix R Noncompliance |
| 50-315/99012-01<br>50-316/99012-01 | URI | Traceability of Non-conforming MCCBs |
| 50-316/99017-01 | VIO | Failure to restore compliance from a previous identified violation involving inadequate 10 CFR 50.59 evaluations |
| 50-315/99021-03<br>50-316/99021-03 | IFI | Verify adequacy of long term corrective actions to resolve Generic Letter 89-13 programmatic weaknesses for performance testing of the D/G heat exchangers |

## LIST OF ABBREVIATIONS

| | |
|---|---|
| AES | Engineered Safety Features Ventilation |
| AFW | Auxiliary Feedwater System |
| ASME | American Society of Mechanical Engineers |
| CCW | Component Cooling Water |
| CFR | Code of Federal Regulations |
| CR | Condition Report |
| CTS | Containment Spray System |
| CVCS | Chemical and Volume Control System |
| D/G | Diesel Generator |
| DRP | Division of Reactor Projects |
| ECCS | Emergency Core Cooling System |
| ESF | Engineered Safety Features |
| ESW | Essential Service Water |
| JO | Job Order |
| LER | Licensee Event Report |
| MC | Manual Chapter |
| MHP | Maintenance Head Procedure |
| MOV | Motor Operated Valve |
| NCV | Non-Cited Violation |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| NRR | Nuclear Reactor Regulation |
| ODE | Operability Determination Evaluation |
| OHI | Operations Head Instruction |
| OHP | Operations Head Procedure |
| OSO | Operations Standing Order |
| PDR | Public Document Room |
| PI | Performance Indicator |
| PMI | Plant Manager's Instruction |
| PMP | Plant Manager's Procedure |
| PMT | Post-maintenance Testing |
| PORV | Power Operated Relief Valve |
| PPC | Plant Process Computer |
| RCS | Reactor Coolant System |
| RWST | Refueling Water Storage Tank |
| SSC | Structures, Systems, and Components |
| STP | Surveillance Test Procedure |
| TS | Technical Specification |
| URI | Unresolved Item |
| UFSAR | Updated Final Safety Analysis |
| VAC | Volts, Alternating Current |
| VDC | Volts, Direct Current |
| VIO | Violation |