April 18, 2001

EA-01-101

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 NUCLEAR POWER PLANT - NRC INSPECTION REPORT 50-315/01-07(DRP); 50-316/01-07(DRP)

Dear Mr. Powers:

On March 31, 2001, the NRC completed a baseline inspection at your D. C. Cook Units 1 and 2 reactor facility. The inspection results were discussed on April 3, 2001, with you 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 two findings. The first of these findings, a failure to demonstrate that the performance of the auxiliary feedwater system had been effectively controlled by the preventive maintenance program, was a violation of 10 CFR 50.65 (a)(2). This issue was evaluated through the significance determination process as a GREEN finding of very low safety significance. 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. The remaining finding involved several examples of ineffective corrective actions for previous violations of the Maintenance Rule. This issue was determined to be a NO COLOR finding related to the problem identification and resolution cross cutting performance area. If you deny the 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.

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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 50-315/01-07(DRP); 50-316/01-07(DRP)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/01-07(DRP); 50-316/01-07(DRP)
Licensee:	American Electric Power Company 1 Cook Place Bridgman, MI 49106
Facility:	D. C. Cook Nuclear Generating Plant
Location:	1 Cook Place Bridgman, MI 49106
Dates:	February 11, 2001 through March 31, 2001
Inspectors:	B. L. Bartlett, Senior Resident Inspector K. A. Coyne, Resident Inspector T. J. Madeda, Region III Inspector J. D. Maynen, Resident Inspector R. K. Walton, Region III Inspector
Approved by:	A. Vegel, Chief Reactor Projects Branch 6 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 05000315-01-07, IR 05000316-01-07, on 02/11-03/31/2001, Indiana Michigan Power Company, D. C. Cook Nuclear Plant Units 1 & 2. Equipment alignment, fire protection, licensed operator requalification, maintenance rule implementation, maintenance and emergent work, operability evaluations, operator workarounds, postmaintenance testing, surveillance testing, temporary modifications, security plan changes, performance indicator verification, identification and resolution of problems, event followup.

The inspection was conducted by the resident inspectors, a regional security inspector, and a regional reactor engineer. This inspection identified one green issue, which was a Non-Cited Violation, and one no-color finding associated with a problem identification and resolution cross cutting issue. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 “Significance Determination Process” (SDP). Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violation. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Mitigating Systems

GREEN. A non-cited violation was identified for the failure to demonstrate that the performance of the auxiliary feedwater (AFW) system had been effectively controlled by the preventive maintenance program. The inspectors concluded that this was a violation of 10 CFR 50.65 (a)(2). The licensee failed to identify and properly account for maintenance preventible functional failures (MPFFs) associated with the Unit 1 West motor drive auxiliary feedwater train and one repetitive MPFF associated with the turbine driven AFW pump train. The inspectors concluded that the performance and condition of Unit 1 AFW was not being effectively controlled through the performance of appropriate preventive maintenance.

The inspectors evaluated the risk significance of this issue using the Significance Determination Process. The AFW system was relied upon to support secondary heat removal following a loss of normal feedwater and therefore was within the mitigating systems cornerstone. The inspectors determined that the failure to recognize, monitor and correct ineffective maintenance practices could adversely impact the performance of a risk significant SSC. However, the inspectors concluded that this failure did not result in a total loss of the secondary heat removal safety function due to availability of redundant AFW trains. Additionally, reliability failures that could be attributable to ineffective maintenance activities occurred during conditions not requiring secondary heat removal or were repaired within the TS allowed outage time. Therefore, this issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. Although the risk significance of this issue was very low, the inspectors concluded that this was more than a minor concern because the failure to recognize and correct ineffective maintenance practices could result in decreased reliability and increased unavailability of the AFW system. (Section 1R12.1)

Cross-Cutting Issues: Problem Identification and Resolution

NO COLOR. The inspectors identified several examples of ineffective corrective actions for previous violations of the Maintenance Rule. Specifically, the licensee failed to properly evaluate and identify several maintenance preventable functional failures associated with the Unit 1 auxiliary feedwater system and set adequate goals for the ice condenser system. Previous violations involved the failure to properly evaluate and identify maintenance preventable functional failures of the Unit 2 250 Vdc system and failure to establish performance goals for the Unit 1 chemical and volume control system.

Ineffective corrective actions for previous Maintenance Rule violations are more than a minor concern in that ineffective corrective actions could impact the ability of the licensee to adequately maintain the reliability, availability and performance of risk-significant SSCs within the scope of the Maintenance Rule. Although the inspectors determined that this issue does not impact a specific reactor safety cornerstone and did not represent a violation of NRC requirements, this corrective action weakness provided substantive information relating to the problem identification and resolution cross-cutting issue and relates to previously identified findings. (Section 4OA2.1)

Report Details

Summary of Plant Status:

Unit 1 began the inspection period at full power. On February 15, 2001, the operators manually tripped Unit 1 following a trip of the Unit 1 East main feedwater pump. Unit 1 was placed in Mode 3 (Hot Standby) following the trip and returned to Mode 2 (Startup) on February 18, 2001. Unit 1 achieved full power on February 21, 2001. On March 4, 2001, Unit 1 commenced a power reduction to 30 percent to support a steam leak repair on a feedwater heater drain valve. Unit 1 was returned to full power on March 7, 2001. On March 30, 2001, Unit 1 commenced a power reduction to 55 percent to support cleaning of both Unit 1 main feedwater pump condenser waterboxes. At the end of the report period Unit 1 was ascending in power and was operating at 96 percent power.

Unit 2 operated at full power throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

a. Inspection Scope

The inspectors performed a complete mitigating system walkdown of the Unit 1 Emergency Core Cooling System (ECCS). The inspectors reviewed ongoing system maintenance, open job orders (JOs), and design issues for potential effects on the ability of the ECCS to perform its design functions. The inspectors ensured that the configuration of the ECCS was in accordance with applicable operating checklists. The inspectors also performed a complete system status check, which verified acceptable material condition of system components, availability of electrical power to system components, essential support systems availability, and that ancillary equipment or debris did not interfere with system performance. The Unit 1 ECCS was selected for this inspection based on its importance as a mitigating system used to prevent core damage. As part of this inspection, the inspectors reviewed the licensee's historical computerized issue tracking and JO database, as well as the following documents:

- 01-OHP [Operations Head Procedure] 4021.003.001, "Letdown, Charging, and Seal Water Operation," Revision 24b
- 01-OHP 4021.008.002, "Placing Emergency Core Cooling Systems in Standby Readiness," Revision 15a
- 01-OHP 4030.STP.030, "Daily and Shift Surveillance Checks," Revision
- Technical Specification 3.5.2, "ECCS [Emergency Core Cooling System] Subsystems - $T_{ave} > 350^oF$"
- Flow Diagram OP-1-5129, CVCS - Reactor Letdown and Charging, Unit 1
- Job Order (JO) C174976, U-bolt clamp on oil drain line missing nuts and lock nut
- Condition Report (CR) 01068061, NRC identified valve 1-SI-128 has boric acid buildup

- CR 01068062, NRC identified boric acid buildup on the cap at the end of 1-SI-130
- CR 01068064, NRC identified boric acid buildup on the vent of 1-ILA-250
- CR 01068065, NRC identified boric acid buildup on cap of 1-SI-144N

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: Unit 1 and Unit 2 Auxiliary Feedwater pump rooms (Fire Zone 17), Unit 1 elevation 596 Cable Tunnel (Fire Zones 7 through 11), Unit 2 elevation 596 Cable Tunnel (Fire Zones 23 through 27), Unit 1 HVAC Vestibule (Fire Zone 49), Unit 2 HVAC Vestibule (Fire Zone 50), and the spray additive tank room (Fire Zone 61). 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:

- Updated Final Safety Analysis Report (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"
- Plant Managers Procedure (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.

1R11 Licensed Operator Requalification

a. Inspection Scope

On March 20, 2001, the inspectors observed Operations Shift "C" during simulator training. The shift performed a scenario designed to exercise the use of Emergency Operating Procedure 02-OHP 4023.FR-S.1, "Response to Nuclear Power Generation/ATWS [Anticipated Transient Without Scram]." The shift also performed exercises involving instrument failures. The inspectors assessed communications and implementation of emergency operating procedures. In addition, the inspectors attended the licensee's critique following performance of the simulator scenarios.

b. Issues and Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

The inspectors evaluated the licensee's implementation of the Maintenance Rule, 10 CFR 50.65, for the following systems, structures and components (SSCs): the Unit 1 auxiliary feedwater (AFW) system, nuclear instrumentation, 600 VAC power distribution system, ice condenser, and main feedwater system. The inspectors reviewed a total of seven Maintenance Rule evaluations associated with the above SSCs. The inspectors assessed: (1) functional scoping in accordance with 10 CFR 50.65; (2) characterization of SSC failures; (3) SSC safety significance classification; (4) 10 CFR 50.65 (a)(1) or (a)(2) classification for the SSCs; and (5) performance criteria for SSCs classified as (a)(2) or goals and corrective actions for SSCs classified as (a)(1). The inspectors also interviewed the licensee's Maintenance Rule Coordinator and evaluated the licensee's monitoring and trending of performance data with the responsible system engineers.

.1 Maintenance Rule Evaluations Associated with the Unit 1 Auxiliary Feedwater System

a. Inspection Scope

The inspectors reviewed Maintenance Rule evaluations associated with three potential functional failures of the AFW system. These evaluations were associated with water contamination of the Unit 1 West motor driven AFW pump (MDAFP) bearing, failure to appropriately set flow retention limit switches during maintenance, and several failures of AFW pump discharge motor operated throttle valves. Because the licensee determined that appropriate preventive maintenance effectively controlled system performance, the AFW system was monitored under the requirements of 10 CFR 50.65(a)(2). The inspectors reviewed the following documents during this review:

- Auxiliary Feedwater Maintenance Rule Scoping Document
- Engineering Action Plan 00-516, Maintenance Rule
- UFSAR Section 10.5.2, "Auxiliary Feedwater System"
- UFSAR Section 14.1.9, "Loss of Normal Feedwater Flow"
- Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2
- NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2
- 12-EHP [Engineering Head Procedure] 5030.OIL.001, "Oil Analysis Program," Revision 1
- 12-EHP 5035.MRP.001, "Maintenance Rule Program Administration," Revision 0
- 01-OHP 4023.E-0, "Reactor Trip or Safety Injection," Revision 15
- Job Order (JO) 00348124, Adjust packing on Unit 1 West MDAFP

- CR 98-7665, Unit 1 East MDAFP outboard bearing is degraded
- CR 99-1907, NRC Inspection Report 50-315/96013; 50-316/96013 identified the need to initiate root cause level corrective action
- CR 99-21501, Evaluate moisture intrusion in the AFW pumps
- CR 00-8741, Unit 2 AFW motor operated discharge valves flow retention settings were unbalanced
- CR 00257078, Cook plant did not monitor SSC performance as required by 10 CFR 50.65
- CR 00286013, Unit 1 West MDAFP packing leakoff too low to support operation
- CR 00339035, 1-FMO-231 did not open from the control room switch
- CR 00339036, During performance of 12-EHP 4030.STP.218, 1-FMO-211 fully closed in response to a flow retention high flow signal instead of achieving an intermediate position
- CR 00341089, Flow retention limit switches may have been incorrectly set on valves 1-FMO-211 and 1-FMO-221
- CR 00342089, The condition originally reported in CR 00339035, that 1-FMO-231 does not operate when demanded from the control room, has not been corrected
- CR 00342090, Incorrect setting of the flow retention setting for the auxiliary feedwater motor operated valve actuators
- CR 00346053, 1-FMO-231 does not appear to open when the control switch is operated on the first attempt
- CR 00346065, 1-FMO-212 would not close remotely from the control room or hot shutdown panel
- CR 00348124, Unit 1 West MDAFP outboard pump packing is leaking
- CR 00355040, Tracking CR to capture the Maintenance Rule reconstitution report for AFW and Condensate Storage Tank systems
- CR 01032008, Water was found in the Unit 1 West MDAFP outboard bearing oil
- CR 01081007, Rework of Unit 1 West MDAFP recurring packing leak
- CR 01085014, During followup to NRC questions, identified multiple instances of misclassifications of functional failures and maintenance preventable functional failures since June 2000
- CR 01089007, NRC identified that Maintenance Rule evaluation for CR 00286013 is based on an incorrect assumption
- CR 01089012, NRC identified that Maintenance Rule evaluation for CR 00346065 was answered incorrectly
- CR 01094033, NRC identified four instances associated with the AFW where the Maintenance Rule evaluations were incorrect or inadequate

b. Observations and Findings

The licensee identified eight AFW system functions that were within the scope of the Maintenance Rule. The functions that were potentially impacted by the Maintenance Rule evaluations reviewed by the inspectors included: (1) AFW-01, “Provide makeup flow to the steam generators in the event of a loss of normal feedwater,” (2) AFW-03, “Provide flow from the turbine driven auxiliary feedwater pump to the steam generators during a station blackout,” and (3) AFW-09, “Provide runout protection for the AFW pumps via the flow retention circuitry.” The reliability performance criteria associated with these functions allowed no more than one maintenance preventable functional failure (MPFF) per MDAFP train per 24 months. During a recently completed review of historical AFW performance data, the licensee determined that there had been no MPFFs identified for the Unit 1 AFW system.

During the inspectors' review of the Maintenance Rule evaluations, the inspectors identified four inadequate Maintenance Rule evaluations associated with the AFW system. Three of the four evaluations were associated with the Unit 1 West MDAFP train while the remaining evaluation was associated with Unit 1 turbine driven AFW pump train. The inspectors determined that the conditions associated with these evaluations constituted two MPFFs, one potential MPFF, and one repetitive MPFF.

b.1 Inadequate Maintenance Rule Evaluations Affecting the Unit 1 West Motor Driven Auxiliary Feedwater Pump Train

The inspectors concluded that the licensee failed to properly evaluate and identify MPFFs that resulted in the Unit 1 West AFW train exceeding its performance criteria. The evaluations are discussed below:

- Unit 1 West MDAFP pump outboard pump bearing oil was contaminated with water.

  Condition Report 01032008 documented that on February 1, 2001, with Unit 1 in Mode 1 (Power Operation), an oil sample drawn from the Unit 1 MDAFP outboard pump bearing following quarterly inservice testing contained 75 percent water. The licensee determined that the source of the water contamination was water spray from the pump packing. The Maintenance Rule evaluation for this CR concluded that the water contamination did not constitute a functional failure of the Unit 1 West MDAFP because the as-found vibration and thermography analysis of the bearing obtained during the testing showed no adverse trends or abnormal data.

  The inspectors concluded that the licensee's evaluation, which was based on 48 minutes of testing run time, did not adequately consider pump capability over the entire 24 hour mission time for the MDAFP. The inspectors concluded that there was sufficient water in the bearing oil to question the pump's ability to perform its function. In response to the inspectors' questions, the licensee re-evaluated this condition and determined that the contaminated oil constituted an MPFF of the Unit 1 West AFW train.

- Motor operated valve 1-FMO-212, Unit 1 West MDAFP supply to Loop 1 Steam Generator, failed to close.

  Condition Report 00346065 documented that, on December 11, 2000, motor operated valve 1-FMO-212 failed to close in response to demand from either the control room or the hot shutdown panel. The control room operators identified this failure while attempting to control the level in the Loop 1 steam generator (SG) while the Unit was in Mode 4 (Hot Shutdown). After maintenance personnel cleaned the valve actuator torque switch contacts, the valve satisfactorily operated. The licensee's Maintenance Rule evaluation concluded that the condition was not MPFF based, in part, on the following: (1) the normal standby position of FMO-212 was fully open and remote closure of the valve was not necessary for accident mitigation; and (2) appropriate maintenance had been performed and the cause of the torque switch failure was essentially not preventable.

The inspectors determined that the licensee's Maintenance Rule evaluation conclusion was not adequately justified. For example, Emergency Operating Procedure 01-OHP 4023.E-0, "Reactor Trip or Safety Injection," required operators to control AFW flow in order to maintain SG water level within a specified band. Since operators would normally rely on remote operation of 1-FMO-212 to control level in the Loop 1 SG, maintaining the valve in a fully open standby position would not be sufficient to ensure that the all safety functions were met. The inspectors also questioned if valve maintenance activities could have caused the torque switch failure. The licensee concluded that the probable cause of the torque switch failure was a foreign material which prevented closure of the torque switch contacts. Although the licensee was unable to determine the source of the foreign material, the evaluation concluded that the torque switch contact failure was not maintenance preventable. The inspectors determined that the Maintenance Rule evaluation did not adequately support the conclusion that the failure was not maintenance preventable. In response to the inspectors questions, the licensee performed additional evaluation and determined that the 1-FMO-212 failure was an MPFF. The licensee initiated CR 01089012 to document the failure to perform an adequate evaluation of this condition.

- <u>Unit 1 West MDAFP outboard packing gland overheated due to low packing leak off</u>.

  Condition Report 00286013 documented that, during routine surveillance testing on October 12, 2000, the outboard pump packing gland for Unit 1 West MDAFP overheated. The control room operators stopped the pump after only 9 minutes of operation after receiving a report of smoke issuing from the packing area. The cause of the overheating was attributed to low packing gland leakage and consequent undercooling of the packing. At the time this condition was discovered, Unit 1 was defueled and TS 3.7.1.2 requirements for Unit 2 shutdown functions were met by the Unit 1 East MDAFP. The Maintenance Rule evaluation concluded that no functional or maintenance preventable failure had occurred because Unit 1 West MDAFP was not required to be available for service.

  The inspectors questioned the basis for this conclusion since reliability failures, including those occurring when a system was not required to be operable, could potentially indicate poor maintenance program effectiveness. During a review of control room logs and completed job order activities, the inspectors noted that Unit 1 West MDAFP packing had been adjusted during previous routine quarterly testing conducted in July 2000. The inspectors determined that the failure to appropriately adjust the West MDAFP packing during previous maintenance activities constituted a potential MPFF of Maintenance Rule function AFW-01.

The licensee initiated CR 01089007 to document the inadequate Maintenance Rule evaluation and stated that they would re-evaluate the Maintenance Rule impact of this potential failure.

b.2 Repetitive MPFF of the Unit 1 Turbine Driven AFW Train

Condition Report 00339036 documented that, on December 4, 2000, 1-FMO-211, the Unit 1 turbine driven AFW pump discharge to the Loop 1 SG, fully closed in response to a flow retention actuation signal rather than positioning to the required intermediate position. The flow retention position setting was intended to prevent pump runout while maintaining AFW flow to the associated SGs. This condition was discovered during routine surveillance testing conducted in accordance with 12-EHP-4030.STP.218, “Automatic Operation of Auxiliary Feedwater Pumps,” with Unit 1 in Mode 5 (Cold Shutdown). The licensee determined that incorrect limit switch setting guidance had been provided to maintenance personnel and, consequently, the flow retention limit switch for 1-FMO-211 was improperly set during refurbishment activities performed on or about October 30, 2000. Additional licensee investigation, documented in CR 01342090, determined that three of the four turbine driven AFW pump discharge valves were incorrectly set during actuator maintenance. The post maintenance testing associated with the actuator refurbishment maintenance activities failed to identify the incorrect flow retention switch settings. The licensee’s Maintenance Rule evaluation concluded that this occurrence was not a MPFF because the condition was identified and corrected prior to the AFW system being relied upon to perform its safety function.

The inspectors questioned the basis for the Maintenance Rule evaluation conclusion since a reliability failure occurring when a system was not required to be operable could potentially indicate poor maintenance program effectiveness, and the incorrect flow retention settings were identified during routine surveillance testing rather than during actuator refurbishment post maintenance testing. In response to the inspectors questions, the licensee re-evaluated the failure. The licensee’s evaluation, as documented in CR 00341090, concluded that this condition represented an MPFF of functions AFW-01 and AFW-03 for the turbine driven AFW pump train. Because the corrective actions from CR 00-8741, which identified a related failure to appropriately set Unit 2 AFW flow retention limit switches, failed to prevent recurrence of the condition, the licensee concluded that this condition constituted a repetitive MPFF.

b.3 Conclusions

Based upon the number of MPFFs identified by the inspectors, the inspectors concluded that the licensee could not demonstrate that the performance of the Unit 1 AFW system had been effectively controlled by the preventive maintenance program. 10 CFR 50.65 (a)(1) required, in part, that the performance or condition of structures, systems, or components (SSCs) within the scope of the rule as defined by 10 CFR 50.65 (b), be monitored against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, are capable of fulfilling their intended functions. 10 CFR 50.65 (a)(2) required, in part, that monitoring as specified in 10 CFR 50.65 (a)(1) was not required where it had been demonstrated that the performance of condition of an SSC had been effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. Contrary to the above, the licensee failed to demonstrate that the performance of Unit 1 West MDAFP and turbine driven AFW pump trains had been effectively controlled through the performance of appropriate preventive maintenance and did not monitor against licensee-established goals. Specifically, the licensee failed to identify and properly account for MPFFs associated with the Unit 1 West MDAFP pump train and one repetitive MPFF associated with the turbine driven AFW pump train which demonstrated that the performance or condition of this SSC was not being effectively controlled through the performance of appropriate preventive maintenance and, as a result, that goal setting and monitoring was required. The inspectors concluded that this failure constituted a **Non-Cited Violation (50-315/01-07-01; 316/01-07-01)** of 10 CFR 50.65(a)(2) consistent with Section VI.A. of the NRC Enforcement Policy. This violation is in the licensee's corrective action system as CR 01094033. This NCV is closed.

The inspectors evaluated the risk significance of this issue using the Significance Determination Process. The AFW system was relied upon to support secondary heat removal following a loss of normal feedwater and therefore was within the mitigating systems cornerstone. The inspectors determined that the failure to recognize, monitor and correct ineffective maintenance practices could adversely impact the performance of a risk significant SSC. However, the inspectors concluded that this failure did not result in a total loss of the secondary heat removal safety function due to availability of redundant AFW trains. Additionally, reliability failures that could be attributable to ineffective maintenance activities occurred during conditions not requiring secondary heat removal or were repaired within the TS allowed outage time. Therefore, this issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. Although the risk significance of this issue was very low, the inspectors concluded that this was more than a minor concern because the failure to recognize and correct ineffective maintenance practices could result in decreased reliability and increased unavailability of the AFW system.

.2 Review of Corrective Actions and Monitoring Goals for the Ice Condenser System

a. Inspection Scope

The licensee identified three functions for the ice condenser system: (1) absorb thermal energy from a loss of coolant accident or main steam line break, (2) assist in iodine removal from the containment atmosphere, and (3) provide a water inventory source. Due to previously identified problems, including poor maintenance practices, poor material condition, and ineffective corrective actions; the licensee monitored the ice condenser under the requirements of 10 CFR 50.65(a)(1). The NRC has previously evaluated the operability impact of these issues during Inspection Manual Chapter 0350 restart inspections associated with the following items:

- Unit 1 Restart Action Matrix (RAM) 1.1, "Unit 1 Ice Condenser Restoration," which was closed in NRC Inspection Report No. 50-315/00-22, 50-316/00-22, Section 4OA5.

- Unit 2 Case Specific Checklist Item 6, “Resolution of Ice Condenser Issues,” which was closed in NRC Inspection Report No. 50-315/99-26, 50-316/99-26.

The inspectors reviewed the ice condenser Maintenance Rule scoping and action plans, and discussed recent ice condenser performance problems with the system manger and Maintenance Rule coordinator. The inspectors reviewed the following documents:

- Ice Condenser Maintenance Rule Scoping Document
- Ice Condenser Maintenance Rule (a)(1) Action Plan
- Engineering Action Plan 01-606, Unit 1 Ice Condenser
- CR 00244059, Apparent inadequacies in Maintenance Rule evaluations associated with ice condenser door icing
- CR 00325009, Ice bed temperature recorder does not appear to have proper technical specification inputs to point 520
- CR 01002035, Unit 1 ice condenser intermediate deck doors inoperable
- CR 01073036, NRC identified inadequate goal setting established in the ice condenser (a)(1) action plan

b. Observations and Findings

No findings of significance were identified. During the review of the ice condenser Maintenance Rule (a)(1) action plan, the inspectors identified several weaknesses in the ice condenser goal setting and monitoring. Because development of (a)(1) action plans was associated with the licensee’s corrective actions for previous Maintenance Rule violations, this issue is discussed in Section 4OA2, “Identification and Resolution of Problems,” below.

.3 Failure to Reinstall Fuses in 600 VAC Breaker Following Maintenance Activity

a. Inspection Scope

On December 10, 2000, with Unit 1 in Mode 4 (Hot Shutdown), an operator identified that the control power fuses for the Technical Support Center (TSC) uninterruptible power supply breaker were not installed with the associated breaker closed. The absence of the control power fuses resulted in the unavailability of the engineered safety features diesel generator load conservation trip function for this breaker. The licensee’s evaluation concluded that the apparent cause of this condition was the failure of an auxiliary operator to reinstall the fuses following breaker refurbishment. The licensee determined that no Maintenance Rule functional failure had occurred because the available load margin on the associated diesel generator was sufficient to bound the expected additional load of the TSC. The inspectors reviewed the licensee’s Maintenance Rule evaluation and discussed the evaluation with the system manager. The inspectors reviewed the following documentation during this review:

- 4kV/600V AC Electrical Distribution Maintenance Rule Scoping Document
- Emergency Diesel Generator System Maintenance Rule Scoping Document
- Calculation 1-E-N-ELCP-4KV-001, “Unit 1 4kV/600V Load Control Calculation,” Revision 1

- Calculation 12-E-N-ELCP-EDG-005, “Computer Simulation Study to Evaluate Dynamic Performance of the Cook Nuclear Plant Emergency Diesel Generator,” Revision 0
- CR 00345038, Fuses for technical support center uninterruptible power supply breaker, 1-11C15, were not installed with breaker closed

b. Observations and Findings

No findings of significance were identified.

.4 Maintenance Rule Evaluations for Power Range and Gammametric Nuclear Instruments

a. Inspection Scope

The inspectors reviewed several recent Maintenance Rule evaluations associated with power range and Gammametric nuclear instruments. The inspectors also reviewed accumulated unavailability hours for the gammametric nuclear instruments and verified that equipment performance monitoring was consistent with the Maintenance Rule scoping document. The inspectors discussed nuclear instrumentation performance with the system manager, system engineering supervisor, and Maintenance Rule coordinator. The inspectors reviewed the following documents during this review:

- Nuclear Instrumentation Maintenance Rule Scoping Document
- Annunciator System Maintenance Rule Scoping Document
- Unit 2, Caution Tag Log, February 26, 2001
- CR 00338023, Power range nuclear instrument 2-N-43 is causing flux deviation alarms in the control room
- CR 00311063, Power range nuclear instrument 2-N-43 gives causing flux deviation alarms in the control room
- CR 00363044, Gammametrics channel 1-N-23 is indicating approximately 50 percent when reactor power is 78 percent
- CR 00363043, Gammametrics channel 1-N-21 is indicating approximately 10 percent when reactor power is 78 percent
- CR 00318038, Gammametrics channel 1-N-21 reads 30-40 counts per second with no fuel in the core
- CR 00-10896, Evaluate request for change to preventative maintenance program for Gammametrics channels 1-NRI-21, 1-NRI-23, 2-NRI-21, and 2-NRI-23
- CR 01008021, Evaluate change to preventative maintenance program for Gammametrics nuclear power channels
- CR 00355132, Declared power range N-43 inoperable due to no current indicated on the lower detector
- CR 00355025, Integrated results of Maintenance Rule recovery project for nuclear Instrumentation system
- CR 01087064, NRC identified that performance criteria for the power range nuclear instruments was inconsistent with reliability target

b. Observations and Findings

No findings of significance were identified.

.5 Unit 1 Manually Tripped Following High Back Pressure on East Main Feedwater Pump

a. Inspection Scope

On February 15, 2001, the operators manually tripped the Unit 1 reactor after the East Main Feedwater Pump (MFP) tripped on high back pressure. The inspectors reviewed the CRs and operation logs associated with the Unit 1 East MFP performance prior to the Unit 1 shutdown. The following documents were reviewed:

- 12-EHP [Engineering Head Procedure] 5035.MRP.001, “Maintenance Rule Program Administration,” Revision 0
- UFSAR Section 10.5.1, “Main Condensate and Feedwater System”
- UFSAR Section 14.1.9, “Loss of Normal Feedwater Flow”
- Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2
- NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 2
- Unit 1 Control Room logs for February 2001
- CR 00354027, Historical review of Maintenance Rule functional failures for the Main Feedwater system
- CR 00355129, Unit 1 West MFP trip on low vacuum
- CR 01012023, Steam generator water level control system does not respond adequately to small changes in blowdown flow
- CR 01032005, Unit 1 East MFP waterbox differential pressure is at the upper limit
- CR 01046024, Debris entering the Unit 1 East MFP condenser from the circulating water system is creating a differential pressure concern
- CR 01046054, A manual reactor trip was performed upon recognition that the Unit 1 East MFP had tripped on high backpressure
- CR 01065049, Plant level performance is not currently being documented in the Maintenance Rule database
- CR 01085013, NRC identified that no link exists between Maintenance Rule plant level criteria for unplanned capability loss and corrective action process that would cause a Maintenance Rule evaluation for a downpower event without an attendant equipment failure

b. Issues and Findings

No findings of significance were identified.

1R13 Maintenance and Emergent Work

.1 Repairs to Unit 2 Chemical and Volume Control System Valves in the Boration Flowpath

a. Inspection Scope

On February 21, 2001, the licensee removed the Unit 2 boration flowpath from the boric acid storage tanks from service to repair leakage from 2-CS-486, “2-QFC-421 Inlet Shutoff Valve,” and 2-CS-422S, “South Boric Acid Filter QC-12 to Middle Boric Acid Storage Tank Shutoff Valve.” Technical Specification 3.1.2.2, “Flow Paths - Operating,” required the boration flowpath associated with the boric acid storage tanks to an operable status within 72 hours. The inspectors reviewed the applicable maintenance job orders and clearance request, reviewed the on-line maintenance risk evaluation, and walked down portions of the maintenance area. The inspectors reviewed the following documents during this inspection:

- PMP 2291.OLR.001, “On-Line Risk Management,” Revision 0
- PMP-2291.OLR.001, On-Line Risk Management Work Schedule Review and Approval For Cycle 36, Week 2 Work Week Schedule
- Calculation MD-01-CVCS-057-N, “Unit 1 Boron Make-up System Minimum Required Flowrates,” Revision 0
- Calculation MD-01-CVCS-060-N, “Minimum BAT pump performance for emergency boration,” Revision 0
- Clearance 2003819, Isolate boric acid flowpaths for maintenance
- Flow Diagram OP-12-5131, “CVCS-Boron Make-Up Units No. 1 & 2"
- JO C0200822, Repair body to bonnet leak on 2-CS-486
- JO 00364032, Replace bonnet assembly on 2-CS-422S

b. Observations and Findings

No findings of significance were identified.

.2 Unit 1 East Main Feedwater Pump Condenser

a. Inspection Scope

On February 15, 2001, the Unit 1 East MFP tripped on high backpressure. In accordance with the off-normal operating procedure for a partial loss of main feedwater with reactor power above 80 percent, the operators tripped the Unit 1 reactor and performed an orderly shutdown of the reactor. The inspectors verified that the licensee was adequately controlling the cleaning of the MFP condensers in accordance with written instructions such that the licensee’s maintenance activities did not initiate a plant event. The inspectors considered the licensee’s response to the degrading circulating water system conditions to be significant in that secondary system degradation could lead to a primary plant transient. The inspectors reviewed the following condition reports, job orders, and procedures which documented recent maintenance activities on the Unit 1 East MFP condenser:

- 01-OHP 4021.055.003, “Placing a Main Feed Pump in Service,” Revision 13

- 01-OHP 4021.055.004, “Feed Pump Turbine Shutdown,” Revision 8
- 01-OHP 4021.057.001, “Circulating Water System Operation,” Revision 19
- 01-OHP 4022.055.001, “Loss on One Main Feed Pump,” Revision 4
- JO R0211262, “Clean East MFP condensers,” performed on January 12, 2001
- JO R0213136, “Clean East MFP condensers,” performed on February 11, 2001
- JO R0214021, “Clean East MFP condensers,” performed on February 18, 2001
- CR 01046054, A manual reactor trip was performed upon recognition that the Unit 1 East MFP had tripped on high back pressure

b. Issues and Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 Operability of Plant Computer Thermal Power Calculation

a. Inspection Scope

On February 8, 2001, the licensee wrote CR 01038038 to document that the NRC identified a non-conservative error in the plant process computer thermal power calculation. This issue was reviewed in Section 1R15.2 of NRC Inspection Report 50-315/01-02; 50-316/01-02. In order to avoid exceeding the maximum licensed power level, the licensee implemented a compensatory administrative power limit of 99.8 percent. On February 20, 2001, the licensee completed the evaluation of the condition and lifted the compensatory administrative limit on thermal power. The inspectors reviewed the operability determination and the licensee’s corrective actions to verify that the thermal power program measurement was capable of ensuring that maximum thermal power and nuclear instrumentation calibration requirements were met. The inspectors reviewed the following documents during this review:

- Unit 1 and Unit 2 Facility Operating License, DPR-58 and DPR-74, Section 2.C.(1), “Maximum Power Level”
- UFSAR Chapter 14.1, “Core and Coolant Boundary Protection Analysis”
- 10 CFR 50 Appendix K, “ECCS Evaluation Models”
- Shift Managers Logs for February 8 and 28, and March 10, 2001
- Performance Assurance Field Observation FO-01-C-002, “ODE Review of 01038038 for Error in the PPC Calorimetric Calculation”
- CR 01038038, NRC identified non-conservative error in the plant process computer calorimetric calculation
- CR 01072008, NRC identified that inadequate justification was used in CR 01038038 February 20, 2001 operability determination. An administrative thermal power limit should be implemented to prevent exceeding maximum licensed power level

b. Observations and Findings

No findings of significance were identified.

.2 Failure of Unit 1 Turbine Master Trip Solenoid

a. Inspection Scope

On January 28, 2001, the “A” Unit 1 main turbine master trip solenoid failed to reposition during routine turbine trip solenoid testing. The licensee determined that the “A” master trip solenoid was mechanically bound in the untripped position. The master trip solenoid required both the “A” and the “B” solenoids to move to the tripped position in order to initiate a turbine trip. During the time that the master trip solenoid was unavailable, the redundant mechanical trip solenoid was capable of providing main turbine trip protection. The inspectors reviewed the operability determination associated with the failure of the “A” master trip solenoid to verify TS and regulatory requirements were met. Following the reactor trip on February 15, 2001, the licensee replaced the failed “A” master trip solenoid. The inspectors reviewed the following documents during this inspection:

- UFSAR Section 14.1.10, “Excessive Heat Removal Due to Feedwater System Malfunctions”
- UFSAR 14.1.13, “Turbine Generator Safety Analysis”
- Drawing OP 1-98101, “Turbine Control Elementary Diagram”
- Drawing OP 1-98521, “AMSAC Control Elementary Diagram”
- Vendor Drawing 235R818, General Electric Turbine Control Diagram
- NRC Safety Evaluation Report associated with Unit 2 License Amendment 185, Deletion of TS Turbine Overspeed Protection Requirements, dated September 1, 1995
- CR 01028012, The “A” Main Turbine Master Trip solenoid valve failed during surveillance testing
- CR 01044029, NRC identified that prompt operability determination for failure of Unit 1 turbine trip master solenoid did not fully evaluate impact of failure

b. Observations and Findings

No findings of significance were identified.

.3 Foreign Material Intrusion Into Screenhouse Forebay

a. Inspection Scope

During cleaning of the Unit 1 MFP waterboxes on February 16, 2001, maintenance personnel dropped a Herculite cloth into the circulating water forebay. The licensee was unable to retrieve the Herculite material from the forebay and initiated CR 01048011 to document the loss of foreign material. Because the forebay was the suction source for the essential service water pumps, this loss of foreign material control could have potentially impacted the operation of a safety related system. The licensee performed an evaluation and determined that the loss of the Herculite material would not adversely impact either safety-related or TS equipment. The inspectors reviewed the foreign material evaluation, and discussed the conclusions of the evaluation with engineering, maintenance and operations personnel. The inspectors reviewed the following documentation during this inspection:

- PMP 2220.001.001, “Foreign Material Exclusion,” Revision 2
- Control Room Logs, February 18 - 19, 2001
- CR 01048011, An 8-foot piece of Herculite was dropped in the Unit 1 forebay while work was being performed in the vicinity of the #13 circulating water pump
- CR 01067031, Starting of #23 circulating water pump caused an anomaly in the circulating water system which caused flow degradation and vacuum loss to the 2E and 2W main feedwater pump water boxes
- CR 01093002, Herculite material found near inlet of Unit 2 East MFP condenser

b. Observations and Findings

No findings of significance were identified.

.4 Initial Operability Determinations

a. Inspection Scope

Plant Managers Procedure 7030.CAP.001, “Operability Determination,” Section 3.2, stated that an operational impact assessment should be completed within 24 hours of discovery of a condition adverse to quality. This impact assessment included an initial operability determination by an operations department senior reactor operator to determine if the adverse condition impacted licensing basis equipment, if reasonable assurance of operability existed, or if the condition warranted a more thorough evaluation. The inspectors reviewed selected recent initial operability determinations to assess the adequacy of operations shift initial operability determinations. The selected evaluations included adverse conditions identified on the post accident containment hydrogen monitoring system (PACHMS), the emergency diesel generator, main turbine trip control, and the nuclear instrument system. The inspectors discussed the conclusions of the operability review with operations personnel and reviewed the following documentation during this inspection:

- PMP 7030.OPR.001, “Operability Determination,” Revision 4
- PMP 7030.CAP.001, “Corrective Action Program Process Flow,” Revision 7
- CR 00348013, Jet assist isolation valve for 1CD diesel generator was found closed
- CR 01028012, The “A” Main Turbine Master Trip solenoid valve failed during surveillance testing
- CR 00363044, Gammametrics channel N-23 is indicating 50 percent when reactor power in 78 percent
- CR 01024057, Unit 2 Train “A” PACHMS backup air system was tested using a procedure with acceptance criteria less conservative than the calculation of record.
- CR 01044029, NRC identified that prompt operability determination for failure of Unit 1 turbine trip master solenoid did not fully evaluate impact of failure
- CR 01064014, NRC identified that prompt operability on diesel generator did not have adequate technical justification
- CR 01064027, Based on NRC questions, a potential adverse trend has been identified on the adequacy of initial operability determinations

b. Observations and Findings

No findings of significance were identified.

.5 Unit 1 AB Emergency Diesel Generator Starting Air Compressor Relief Valve Failure

a. Inspection Scope

On February 26, 2001, a licensee firewatch tour identified that the second stage relief valve on the Unit 1 AB emergency diesel generator (D/G) AB2 starting air compressor had lifted. In response, the licensee isolated the affected starting air compressor from the D/G starting air system. A failure in the Unit 1 AB D/G starting air system was potentially risk significant in that a failure of the starting air system could result in the D/G failing to start on an automatic actuation signal or failing to run for the entire mission time. The licensee evaluated the as-found condition and determined that, although the AB2 starting air compressor was inoperable, the Unit 1 AB D/G remained operable. The inspectors reviewed the following documents:

- UFSAR Section 8.4, Emergency Power System
- CR 01057022, Second stage relief valve on AB2 starting air compressor lifted and would not reseat

b. Issues and Findings

No findings of significance were identified.

1R16 Operator Workarounds

.1 Review of the Cumulative Effect of Operator Workarounds (Both Units)

a. Inspection Scope

The inspectors reviewed the cumulative effect of Operator Workarounds (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 interviewed the OWA Coordinator regarding the oversight and control of OWAs and reviewed the following licensee documents:

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

b. Issues and Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

During this inspection period, the inspectors reviewed the post maintenance testing associated with the following four maintenance activities: (1) repair of an oil leak on the Unit 2 West residual heat removal pump; (2) repair to the Unit 2 boric acid flow totalizer; (3) routine preventive maintenance on the Unit 1 CD emergency diesel generator (D/G) air compressor; and (4) repair of an ESW leak on the Unit 1 AB D/G south air aftercooler. Each of these maintenance activities involved systems important to safety. The inspectors reviewed selected test data to verify that components met design and licensing bases requirements. The inspectors also reviewed the following documents associated with the selected maintenance activities:

- Technical Specification 3.5.2, ECCS Subsystems, $T_{ave} \geq 350^{o}F$
- 02-IHP [Instrument Head Procedure] 6030.IMP.420, “Boric Acid Flow Control Calibration,” Revision 2
- 01-OHP 4030.STP.027AB, “AB Diesel Generator Operability Test (Train B),” Revision 16
- 01-OHP 4030.STP.027CD, “CD Diesel Generator Operability Test (Train A),” Revision 16
- 02-OHP 4030.STP.050W, “West Residual Heat Removal Train Operability Test Modes 1 - 4,” Revision 9
- PMP 2291.PMT.001, “Work Management Post Maintenance Testing Matrices,” Revision 2
- Contingency Equivalency Evaluation (EE) 01-0051
- JO C036452, Minor modification 404-317, replace valve 1-ESW-134-AB
- JO C204377, Calibrate and repair 2-QFC-421-FT, boric acid flow totalizer
- JO C321101, Install new style top hat on Unit 2 West RHR pump
- JO C341041, Correct ESW leak at pipe fitting on air aftercooler 1-ESW-134-AB
- JO R212448, Perform STP.027CD, isolate CD2 starting air system to test CD1 starting air system
- JO R212509, Perform preventive maintenance on Unit 1 CD D/G air compressor
- JO R214210, Unit 1 CD D/G monthly inspection per preventive maintenance task
- JO R215044, Perform STP.027CD, isolate CD2 starting air system to test CD1 starting air system
- CR 99-4361, There have been numerous problems with boric acid flow indicated with no actual flow taking place
- CR 00-9101, The boric acid flow totalizer, 2-QFC-421-FT, counts off with no flow through it
- CR 0157013, Job Order C0204377 was reported complete but continued to work without probabilistic risk assessment being performed
- CR 00341041, Unit 1 AB DG south aftercooler essential service water leak from vent trap

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed or reviewed portions of the following four surveillance tests: (1) Unit 1 reactor coolant system water inventory balances performed between February 17, 2001 and February 22, 2001; (2) rod control surveillances conducted during the Unit 1 reactor startup on February 18, 2001; (3) the monthly surveillance of the Unit 1 CD emergency diesel generator performed on March 14, 2001; and (4) the calibration of the Unit 2 North safety injection pump time overcurrent relay performed on March 15, 2001. The inspectors reviewed the test data and verified that the selected surveillance tests met the Technical Specifications and licensee procedure requirements. The inspectors discussed these surveillance tests with operations, engineering, and regulatory affairs personnel. In addition, the inspectors reviewed the following documents during this inspection:

- Unit 1 Technical Specification 3.1.3.1, “Movable Control Rod Assemblies”
- Unit 1 Technical Specification 3.8.1, “A. C. Sources - Operating”
- Unit 2 Technical Specification 3.4.6.2, “Reactor Coolant System - Operational Leakage”
- Unit 2 Technical Specification 3.5.2, “ECCS Subsystems - $T_{ave} \geq 350^oF$”
- Regulatory Guide 1.9, “Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants”
- NUREG-0452, Standard Technical Specifications for Westinghouse Pressurized Water Reactors, Revision 3
- Licensee response letter to Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” dated June 7, 1988
- Unit 1 Control Room Logs, February 15 - 22, 2001
- Prompt NRC Notification Event #37751, “Manual Reactor Trip Due to Loss of Feedwater Pump,” dated February 15, 2001
- 12-IHP 6030.RLY.001, “General Electric Single Contact Type 1AC Relay Without Instantaneous Overcurrent Device Calibration and Maintenance,” Revision 1A
- 01-OHP 4021.001.002, “Reactor Start-Up,” Revision 26
- 01-OHP 4030.STP.015, “Full Length Rod Operability Test,” Revision 10
- 01-OHP 4030.STP.016, “Reactor Coolant System Leak Test,” Revision 13
- 01-OHP 4030.STP.027CD, “CD Diesel Generator Operability Test (Train A),” Revision 16
- 02-OHP 4030.STP.051N, “North Safety Injection Pump System Test,” Revision 11
- 02-OHP 4030.STP.016, “Reactor Coolant System Leak Test,” Revision 9
- PMP 4010.TRP.001 Unit One Reactor Trip Report for trip occurring on February 15, 2001
- JO R097254, Calibrate protection relays for breaker 2-T21D5
- CR 01046035, Aggregate operability determination for Unit 1 startup following Unit trip on February 15, 2001
- CR 01046054, A manual reactor trip was performed due to low vacuum trip of east main feedwater pump

- CR 01051009, NRC identified that the compressor running light was illuminated for the Unit 1 East Motor Driven Auxiliary Feedwater Pump room cooler
- CR 01075001, Unit 2 North Safety Injection Pump breaker abnormal alarm did not clear after clearance was restored and control switch returned to neutral position

b. Issues and Findings

No findings of significance were identified.

1R23 Temporary Modifications

a. Inspection Scope

The inspectors performed a detailed review of the following four Temporary Modifications: (1) 12-TM-00-34, “Installation of Portable Cooling Equipment in the Nuclear Sampling Room;” (2) 1-TM-01-04-R1, “Installation of Nitrogen Blanket on Expansion Joint 1-XJ-51-1, Low Pressure Turbine ‘C’ North Hotwell Condensate Outlet Expansion Joint;” (3) 2-TM-00-36-R1, “On-Line Leak Repair of 2-BD-126;” and (4) 2-TM-01-02-R1, “Leak Sealing of Valve 2-FW-257.” The inspectors reviewed the temporary modification (TM) and associated 10 CFR 50.59 screening against the system design bases documentation, including the UFSAR and the TSs. In addition, the inspectors verified that the TM was installed in accordance with the required documentation and that configuration control was maintained. The inspectors reviewed the following documents

- 12-MHP 5021.001.051, “Control of Temporary On-Line Leak Sealing,” Revision 6
- 12-EHP 5040.MOD.001, “Temporary Modifications,” Revision 5
- PMP-5020.LCD.001, “Control of Leak Collection Devices,” Revision 1
- Temporary Modification 12-TM-00-34, “Installation of portable cooling equipment in the nuclear sampling room,” Revision 0
- Unit 1 Temporary Modification 1-TM-01-04-R1, “Installation of Nitrogen Blanket on Expansion Joint 1-XJ-51-1, Low Pressure Turbine ‘C’ North Hotwell Condensate Outlet Expansion Joint,” Revision 1
- Unit 2 Temporary Modification 2-TM-00-36-R1, “On-Line Leak Repair of 2-BD-126,” Revision 1
- Unit 2 Temporary Modification 2-TM-01-02-R1, “Leak Sealing of Valve 2-FW-257,” Revision 1
- Performance Assurance Field Observation FO-01-C-25, “Temporary Modification”
- System Description SD-12-NS-100, “Nuclear Sampling System,” Revision 0
- System Description SD-12-WASTED-100, “Waste Disposal System,” Revision 0
- CR 00292062, Temporary modification did not contain pump lubrication requirements and active TMs do not recommend preventive maintenance

b. Issues and Findings

No findings of significance were identified.

3. SAFEGUARDS

3PP4 Security Plan Changes

a. Inspection Scope

The inspector reviewed Revision 1 to the Donald C. Cook Nuclear Plant Physical Security Plan to verify that the changes did not decrease the effectiveness of the submitted document. The referenced revision was submitted in accordance with regulatory requirement 10 CFR 50.54 (p) by licensee letter dated February 20, 2001.

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 the following Unit 1, fourth quarter of 2000, information:

- 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 - High Pressure Injection System portion of the Mitigating Systems cornerstone;
- Safety System Unavailability - Residual Heat Removal System 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
- Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 0

- Regulatory Guide 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems”
- NRC Information Notice 94-46, Non-conservative Reactor Coolant System Leakage Calculation
- 01-OHP 4021.008.007, “Operation of the Safety Injection Pumps, Revision 3”
- 01-OHP 4030.STP.016, “Reactor Coolant System Leak Test,” Revision 13
- 02-OHP 4030.STP.016, “Reactor Coolant System Leak Test,” Revision 9
- Control Room logs for first quarter 2001
- TS 3.4.6.2, Reactor Coolant System Leakage
- Performance Assurance Field Observation FO-01-C-011, “ERO Performance Indicator Review”

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

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.

b.1 <u>Identified Leak Rate</u>

During the review of the reactor coolant system (RCS) Identified Leak Rate PI, the inspectors determined that the licensee’s leak rate procedures calculated total leak rate, which included both identified and unidentified RCS leakage. The licensee assumed that the entire calculated leak rate was unidentified and that identified leakage was 0 gpm. The inspectors discussed the issue with the licensee and determined that the leakage attributable to identified sources could not be precisely determined. Because Technical Specification 3.4.6.2 contained a limit on Identified Leak Rate, the industry guidance document on Performance Indicators, NEI [Nuclear Energy Institute] 99-02 required the use of Identified Leak Rate for the Performance Indicator. The inspectors questioned if the use of an assumed Identified Leak Rate of 0 gpm conformed with the NEI 99-02 guidance. Following the inspectors’ questions, the licensee planned to submit their position to the NEI for further evaluation. Because the total RCS leak rate was no more than approximately 0.1 gpm, the potential inaccuracy would not change the color of the PI.

<u>Unplanned Power Changes Per 7000 Critical Hours</u>

The inspectors determined that the licensee did not have a formal method of documenting whether a power change was planned or unplanned. The licensee stated that they would establish a formal method of determining and documenting whether power changes were planned or unplanned.

4OA2 Identification and Resolution of Problems

.1 Effectiveness of Corrective Actions for Maintenance Rule Weaknesses

a. Inspection Scope

As documented in NRC Inspection Report Nos. 50-315/00-20; 50-316/00-20 and 50-315/00-22; 50-316/00-22, issued in October and December 2000, respectively, the inspectors identified several licensee violations of the Maintenance Rule. These violations included:

- failure to place Unit 2 250 VDC in Maintenance Rule category (a)(1) after multiple MPFFs (NCV 50-315/00-20-03; 50-316/00-20-03), and

- failure to establish performance goals for a system monitored under paragraph (a)(1) of the Maintenance Rule (NCV 50-315/00-20-02).

In response to these violations, the licensee implemented a corrective action plan which included: Maintenance Rule training for personnel, a review of all Maintenance Rule SSC functions, historical reviews of reliability failures and SSC unavailability, and development of action plans for systems designated for monitoring under Maintenance Rule paragraph (a)(1). During routine inspections of the licensee's Maintenance Rule implementation, documented in Section 1R12 above, the inspectors assessed the effectiveness of these corrective actions in restoring compliance with the requirements of 10 CFR 50.65.

b. Observations and Findings

During routine reviews of the licensee Maintenance Rule implementation and corrective actions for previously identified violations, the inspectors identified several examples of ineffective corrective actions.

b.1 Repetitive Failure to Properly Evaluate and Identify MPFFs

In October 2000, the inspectors identified NCV 50-315/00-20-03; 316/00-20-03 as result of the licensee's failure to properly evaluate and identify MPFFs affecting the 250 VDC distribution system. This failure resulted in the licensee's inability to demonstrate that the performance of the 250 VDC system was being effectively controlled by the preventive maintenance program. In response to NCV 50-315, 316/00-20-03, the licensee implemented corrective actions to train system mangers on Maintenance Rule requirements, rescope SSC Maintenance Rule functions, and perform historical reviews to identify potential Maintenance Rule functional failures.

As documented in Section 1R12.1 above, the inspectors identified the licensee's failure to properly evaluate and identify several MPFFs associated with the Unit 1 AFW system. As a result of this failure, the licensee failed to identify that the preventive maintenance program was unable to maintain the reliability and performance of the Unit 1 AFW system. The inspectors determined that the circumstances and causes of the failures to properly identify and evaluate MPFFs on the Unit 1 AFW system were sufficiently similar to the earlier documented failure, NCV 50-315/00-20-03; 316/00-20-03, to conclude that the corrective actions for this previous violation had not been effective in preventing recurrence.

b.2 Inadequate Goal Setting in the Ice Condenser Maintenance Rule (a)(1) Action Plan

As discussed in NRC Inspection Report 50-315/00-20, Section 1R12.2, the inspectors identified NCV 50-315/00-20-02 for the failure to establish performance goals for Unit 1 chemical and volume control system in accordance with 10 CFR 50.65(a)(1). In response to this NCV, the licensee implemented actions to prepare, review and approve performance goals for systems monitored under 10 CFR 50.65(a)(1). As a result of these corrective actions, the Maintenance Rule expert panel revised the ice condenser system (a)(1) performance goals and monitoring plans on February 28, 2001.

Because of its risk significant function in mitigating radiological releases from the containment building during postulated accidents, the licensee determined that ice condenser functions were to be always available and that no functional failures could be tolerated. The inspectors concluded that the revised (a)(1) goals for the ice condenser were not commensurate with safety and failed to address the corrective actions for previous maintenance problems. For example:

- Several goals would not be exceeded until the ice condenser safety function was lost. Specifically goals relating to ice mal-distribution, door failures, and structural deficiencies would not be exceeded until the ice condenser exceeded its analytical safety limits.

- Specific goals had not been established to address previous maintenance problems associated ice basket damage or foreign material exclusion. As described in LER 50-315/98-017, “Debris was found in the ice condenser,” and LER 50-315/98-008, “Inadequate control of contractors resulted in ice basket damage,” these maintenance issues resulted in challenges to the ice condenser function.

The inspectors reviewed recent ice condenser performance problems and corrective actions and determined that the licensee’s failure to establish goals commensurate with safety did not result in the failure of the ice condenser to perform its safety function. However, 10 CFR 50.65, required, in part, that the licensee establish goals commensurate with safety for systems monitored under 10 CFR 50.65(a)(1). Contrary to this requirement, the ice condenser (a)(1) goals approved on February 28, 2001, were not commensurate with safety in that several goals would not be exceeded until the ice condenser safety function was lost. Although failure to establish goals commensurate with safety should be corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The licensee initiated CR 01073036 to document this issue.

The inspectors concluded that the licensee ineffectively implemented corrective actions for NCV 50-315/00-20-02; therefore, the Maintenance Rule expert panel approved inadequate performance goals for the ice condensers.

b.3 Conclusions

Ineffective corrective actions for previous Maintenance Rule violations are a problem identification and resolution cross-cutting issue of more than a minor concern in that ineffective corrective actions could impact the ability of the licensee to adequately maintain the reliability, availability and performance of risk significant SSCs within the scope of the Maintenance Rule. Although the inspectors determined that this issue does not impact a specific reactor safety cornerstone, these corrective action weaknesses provide substantive information relating to the problem identification and resolution cross-cutting issue and relate to previously identified findings. Because of these extenuating circumstances, the inspectors concluded that these corrective action program weaknesses constitute a **NO-COLOR Finding (FIN 50-315/01-07-02; 50-316/01-07-02)**.

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

(Closed) Licensee Event Report 50-316/96002-00,-01: Train “A” containment monitor causes two spurious isolation signals due to faulty computer board. On January 20, 1996 and February 10, 1996, spurious Train “A” containment ventilation isolation signals were received as a result of high radiation detected on radiation monitoring instrument 2-ERS-2300. The licensee’s investigation determined that a faulty detector input / output card located in 2-ERS-2300 was the root cause of the spurious containment ventilation isolation signals and that no high radiation condition actually existed on either occasion. The licensee replaced the faulty computer board on February 13, 1996, and there have been no recurrences of the problem. This LER is closed.

(Closed) Licensee Event Report 50-316/96004-00: Fire seal found to have been inoperable for extended period without compensatory action. On March 12, 1996, the licensee identified that a fire seal between the Unit 2 CD emergency diesel generator room and the adjacent corridor was damaged. The licensee established an hourly fire watch, and the seal was repaired and returned to service on March 15, 1996. The licensee determined that, even with the fire seal inoperable, sufficient fire protection measures were in place to ensure that a postulated fire on either side of the seal would have been effectively detected and extinguished. This LER is closed.

(Closed) Licensee Event Report 50-316/97007-01: Contract worker received exposure in excess of 10 CFR 20.2202 limits. On October 14, 1997, RP personnel identified that a contractor had arrived at the facility with several hot particles located in his shoe. The licensee's follow-up actions were evaluated by the NRC under Inspection Follow-up Item 50-315/97020-01; 50-316/97020-01, which was closed in NRC Inspection Report 50-315/98006; 50-316/98006. The Licensee Event Report documented an excessive dose which was found during the licensee's normal in-processing procedures. No violations of NRC requirements were identified; therefore this LER is closed.

(Closed) Licensee Event Report 50-316/97008-00: Engineered Safety Features actuation due to failure of Train "A" upper containment area radiation monitor. On October 24,1997, an unplanned actuation of the Unit 2 Train "A" containment ventilation isolation occurred. At the time of the event, Unit 2 was in Mode 6, Refueling, and a reactor head lift was in progress. The licensee's investigation could not duplicate the problem. However, none of the other redundant radiation monitors inside containment detected excessive radiation levels; therefore the licensee concluded that an actual high radiation condition did not exist. The suspect components of Unit 2 upper containment radiation monitor 2-VRS-2101 were replaced. This LER is closed.

(Closed) Licensee Event Report 50-315/98003-00: Missed procedure step results in engineered safety features and reactor protection system actuation. On January 7, 1998, while performing a routine surveillance, the licensee's maintenance workers missed a procedural step which resulted in an engineered safety features and reactor protection system actuation. This event was discussed in NRC Inspection Report 50-315/97025; 50-316/97025, and the inspectors determined that the missed procedure step constituted a non-cited violation. The NRC Inspection Report also concluded that the licensee had taken appropriate corrective action concerning this event. This LER is closed.

(Closed) Licensee Event Report 50-315/98027-00,-01,-02: Debris of unknown origin found in containment spray header. In 1998, the licensee found foreign material resembling sludge and numerous pieces of solid debris during a boroscopic examination of the Unit 1 West containment spray system (CTS) lower ring header. This issue was identified as Unit 2 Restart Action Matrix Item 2.10.1 which was closed in NRC Inspection Report 50-315/99022; 50-316/99022. The licensee issued a two supplemental LERs to add an evaluation of the Unit 2 CTS system and modify a inspection commitment. The inspectors concluded that the actions taken by the licensee to eliminate leakage into the CTS ring headers and to periodically check for leakage past the CTS heat exchangers during CTS surveillances appeared adequate to prevent build up of boric acid deposits in the CTS ring headers. This LER is closed.

(Closed) Licensee Event Report 50-315/98053-00,-01: Failure to calibrate the core exit thermocouples as required by Technical Specifications. This issue was identified as Unit 2 RAM Item R.2.1.19, which was closed in NRC Inspection Report 50-315/00-13; 50-316/00-13. The licensee wrote CR 98-6831 to document this issue and corrected the calibration procedure to account for the thermocouple non-linearity. In addition, the inspectors reviewed the supplemental LER which included the results of an evaluation of the significance and root cause of the event. The supplemental LER did not identify any new issues. Because the calibration error was conservative and the procedure was corrected, the inspectors concluded that this failure to use an adequate procedure to calibrate both units' core exit thermocouples constituted minor violations of Technical Specifications 4.3.3.8 (Unit 1) and 4.3.3.6 (Unit 2) that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

(Closed) Licensee Event Report 50-315/98060-00,-01: Retraction - Reactor system trip response time testing does not comply with Technical Specification requirement. The inspectors reviewed the licensee's use of a fixed constant to account for the unmeasured time from the opening of the reactor trip breakers to the loss of stationary gripper coil voltage. The inspectors concluded that the use of a fixed constant was acceptable; however, a minor violation was identified in NRC Inspection Report 50-316/00-01 for the failure to use a bounding constant in the reactor trip system response time testing procedures. The surveillance procedure contained a non-conservative figure to account for the stationary gripper voltage decay time; therefore, the TS requirement to demonstrate that the RTSRT was within its limit for each trip function was not met. The inspectors verified that the licensee had changed the surveillance procedure to use the bounding constant provided in the UFSAR. This LER is closed.

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

(Closed) Unresolved Item 50-315/98009-16; 50-316/98009-16: Generic Letter No. 89-13, "Service Water System Problems Affecting Safety-Related Equipment," 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 unresolved item is closed.

(Closed) Unresolved Item 50-315/98017-01; 50-316/98017-01: Auxiliary feedwater suction strainer special test result package review. On November 3, 1998, the differential pressure across the suction strainer for the Unit 1 West Motor Driven Auxiliary Feedwater Pump exceeded the maximum allowed approximately one minute into a special test procedure. The licensee wrote CR 98-5356 and Licensee Event Report (LER) 50-315/98046-00 to document the auxiliary feedwater pump suction strainer clogging. This issue was identified as Unit 2 Restart Action Matrix (RAM) Item R.1.22 and determined to have low risk significance. The Unit 2 RAM Item R.1.22 and LER 50-315/99046-00 were closed in NRC Inspection Report 50-315/99029; 50-316/99029. This unresolved item is closed.

(Closed) Inspection Follow-up Item 50-315/99009-02: Unreviewed change in the steam generator tube fouling factor. This inspection followup item was written to track the licensee's review of the effect of steam generator tube plugging on the tube fouling factor. The licensee replaced the steam generators affected by this IFI in 2000. This Inspection Follow-up Item is closed.

(Closed) Unresolved Item 50-315/00-22-02; 50-316/00-22-02: The unresolved item involved the identification of multiple examples of security plan language changes that required additional clarification by the licensee to ensure adherence to regulatory requirements. Those issues were closed based on NRC review of Revision 1 of the licensee's physical security plan (refer to Section 3PP4). No violation of NRC requirements occurred.

4OA6 Management Meetings

The inspectors presented the inspection results to licensee management listed below on April 3, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

R. Crane, Regulatory Affairs Supervisor
R. Ebright, Engineering Programs Manager
R. Gaston, Regulatory Affairs Manager
S. Greenlee, Engineering
M. Hoskins, System Engineering
S. Lacey, Director, Engineering
J. Mathis, Regulatory Affairs
R. Meister, Regulatory Affairs
J. Molden, Maintenance Department Director
D. Moul, Assistant Operations Superintendent
T. Noonan, Director, Performance Assurance
B. O'Rourke, Regulatory Affairs
J. Piazza, Chemistry Supervisor
J. Pollock, Plant Manager
R. Powers, Senior Vice President
A. Rodriguez, Manager, Security and Support Services
L. Thornsberry, Engineering Supervisor
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
71111-11	Licensed Operator Requalification	1R11
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
71111-23	Temporary Modifications	1R23
71130-04	Security Plan Changes	3PP4
71151	Performance Indicator Verification	4OA1
71152	Identification and Resolution of Problems	4OA2
71153	Event Followup	4OA3

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315/01-07-01	NCV	Failure to place Unit 1 West MDAFP and TDAFP trains in Maintenance Rule category (a)(1)
50-315/01-07-02	FIN	Failure to implement adequate corrective actions for previously identified Maintenance Rule violations

Closed

50-316/96002-01	LER	Train “A” containment monitor causes two spurious isolation signals due to faulty computer board
50-316/96004-00	LER	Fire seal found to have been inoperable for extended period without compensatory action
50-316/97007-01	LER	Contract worker received exposure in excess of 1020.2202 limits
50-316/97008-00	LER	Engineered Safety Features actuation due to failure of Train “A” upper containment area radiation monitor
50-315/98003-00	LER	Missed procedure step results in engineered safety features and reactor protection system actuation
50-315/98009-16 50-316/98009-16	URI	Generic Letter No. 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” testing of the emergency diesel generator heat exchangers
50-315/98017-01 50-316/98017-01	URI	Auxiliary feedwater suction strainer special test result package review
50-315/98027-00	LER	Debris of unknown origin found in containment spray header
50-315/98053-01	LER	Failure to calibrate the core exit thermocouples as required by Technical Specifications
50-315/98060-01	LER	Retraction - reactor trip system response time testing does not comply with Technical Specification requirement
50-315/99009-02	IFI	Unreviewed change in the steam generator tube fouling factor
50-315/00-22-02 50-316/00-22-02	URI	Security plan change

50-315/01-07-01	NCV	Failure to place Unit 1 West MDAFP and TDAFP trains in Maintenance Rule category (a)(1)
50-315/01-07-02	FIN	Failure to implement adequate corrective actions for previously identified Maintenance Rule violations

Discussed

None

LIST OF ABBREVIATIONS

AES	Engineered Safety Features Ventilation
AFW	Auxiliary Feedwater System
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
FIN	Finding
IMC	Inspection Manual Chapter
JO	Job Order
LER	Licensee Event Report
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
OWA	Operator Workaround
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
PRT	Pressurizer Relief Tank
RCS	Reactor Coolant System
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