October 25, 2000

EA-00-219

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 50-315-00-20(DRP); 50-316-00-20(DRP)

Dear Mr. Powers:

On September 30, 2000, the NRC completed a baseline inspection at your D. C. Cook Units 1 and 2 reactor facility. The inspection results were discussed on September 28, 2000 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. Specifically, this inspection focused on resident inspection activities.

Based on the results of this inspection, the NRC has determined that three violations of NRC requirements occurred involving implementation of the Maintenance Rule program. The three maintenance rule violations involved the failure to establish goals for system, structures, and components monitored under 10 CFR 50.65(a)(1); failure to monitor system unavailability during the recent extended shutdown; and failure to identify that your preventative maintenance program was not effectively controlling the performance of 250 VDC control power fuse holders. These violations are being treated as Non-Cited Violations (NCV) because of their very low risk significance, consistent with Section VI.A of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violations or severity level of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region III; and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001.

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).

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

/RA/

Geoff Grant, Director
Division of Reactor Projects

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

Enclosures: Inspection Report 50-315-00-20(DRP);
50-316-00-20(DRP)

cc w/encl: A. C. Bakken III, Site Vice President
J. Pollock, Plant Manager
M. Rencheck, Vice President, Nuclear Engineering
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

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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:	50-315-00-20(DRP); 50-316-00-20(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:	August 27, 2000 through September 30, 2000
Inspectors:	B. L. Bartlett, Senior Resident Inspector K. A. Coyne, Resident Inspector J. D. Maynen, Resident Inspector K. S. Green-Bates, Region Based 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

NRC Inspection Report 50-315/00-20, 50-316/00-20, American Electric Power Company, Unit 1 and 2, conducted between August 26 and September 30, 2000. The inspection was conducted on the following activities:

Mitigating Systems

- NO COLOR. A non-cited violation was identified for the failure to demonstrate that the preventative maintenance program effectively controlled the performance of systems required to be functional during shutdown conditions. The licensee suspended monitoring of unavailability for structures, systems, and components (SSCs) within the scope of the Maintenance Rule during the September 1997 through June 2000 dual unit extended outage. The failure to monitor shutdown system unavailability impacted the Maintenance Rule monitoring of 46 systems required during shutdown.

  Since the licensee failed to monitor the performance of SSCs, the reliability of systems and effectiveness of the licensee’s maintenance program could not be demonstrated. (Section 1R12.1)

- NO COLOR. A non-cited violation was identified for the failure to establish performance goals for the Unit 1 Chemical and Volume Control System (CVCS) monitored per the Maintenance Rule under 10 CFR 50.65(a)(1).

  The failure to establish performance goals for the Unit 1 CVCS was significant in that without established goals the licensee could not demonstrate the effectiveness of maintenance to assure reliability of the system. (Section 1R12.2)

- GREEN. A non-cited violation was identified for the failure to demonstrate that the performance of the 250 VDC system had been effectively controlled through the performance of appropriate preventive maintenance. Specifically, the licensee failed to identify and properly account for maintenance preventable functional failures of 250 VDC control power fuse blocks. The inspectors identified that the licensee had mis-classified a maintenance related failure of a 250 VDC fuse block on a diesel generator output breaker. The inspectors identified that three additional maintenance preventable functional failures associated with fuse blocks have occurred since May 1999. The performance criteria allowed no maintenance preventable functional failures over a 24-month period.

  The inspectors determined that these fuse block failures could have had a credible impact on safety and could become a more significant safety concern if left uncorrected. This issue was screened as GREEN (very low risk significance) after a Phase 1 Safety Significance Determination review. (Section 1R12.3)

Report Details

Summary of Plant Status:

Unit 1 remained defueled throughout the inspection period. Licensee restart efforts were concentrated on the completion of modifications and corrective maintenance required to support Unit 1 restart. Major accomplishments during this inspection period included commencement of ice condenser loading and ice basket weighing, continuation of electrical bus outage work, and continued progress on ventilation and high energy line break modifications.

Unit 2 operated at or near full power during the inspection period with short duration power level changes for surveillance testing or other operational needs.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

.1 Partial Walkdown of the Unit 2 West Essential Service Water Train

a. Inspection Scope

The inspectors performed a partial walkdown of the Unit 2 West Essential Service Water (ESW) train. At the time of the walkdown, the Unit 2 East ESW Pump was being replaced, under Job Order (JO) C206691, due to degrading performance identified during periodic inservice testing. The inspectors verified that the alignment and condition of the Unit 2 West ESW train was consistent with Technical Specification (TS) and procedural requirements. The inspectors reviewed normal operating procedures, flow diagrams, and operator log sheets to determine the normal system configuration. The following documents were reviewed during this inspection:

- 02-OHP [Operations Head Procedure] 5030.001.001, “Operations Plant Tours,” Revision 15
- 12-OHP 4021.019.001, “Operation of the Essential Service Water System,” Revision 19
- Flow Diagram OP-2-5113, “Essential Service Water”
- Flow Diagram OP-2-5113A, “Essential Service Water”
- Unit 2 Technical Data Book (TDB), Figure 19.8, “Safety Related Throttled Valves,” Revision 16
- 02-OHP 4030.STP.030, “Daily and Shiftly Surveillance Checks,” Revision 31
- Plant Managers Procedure (PMP) 4043.SLV.001, “Sealed/Locked Valves,” Revision 2
- Job Order (JO) C206691, Disassemble, Inspect, and Repair ESW Pump

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Partial Walkdown of the Unit 2 West Residual Heat Removal Train

a. Inspection Scope

The inspectors performed a partial walkdown of the Unit 2 West Residual Heat Removal (RHR) train. At the time of the walkdown, the Unit 2 East ESW pump was being replaced (see Section 1R04.1). Because the Unit 2 East ESW train provided the ultimate heat sink for the Unit 2 East RHR train, the inspectors verified the alignment of the Unit 2 West RHR train to identify any conditions that could preclude proper functioning of the emergency core cooling system. The inspectors verified that the RHR system alignment was consistent with procedural and TS requirements. The inspectors reviewed the following documents during this review:

- 02-OHP 5030.001.001, "Operations Plant Tours," Revision 15
- 02-OHP 4021.008.002, "Placing ECCS System in Standby Readiness," Revision 12
- OP 2-5143, "Flow Diagram Emergency Core Cooling (RHR) Unit No. 2"
- 02-OHP 4021.017.003, "Removing Residual Heat Removal Loop From Service," Revision 6
- 02-OHP 4030.STP.030, "Daily and Shiftly Surveillance Checks," Revision 31
- PMP-4043.SLV.001, "Sealed/Locked Valves," Revision 2
- JO C206691, Disassemble, Inspect, and Repair ESW Pump

b. Issues and Findings

There were no findings identified and documented during this inspection.

.3 Partial Walkdown of Spent Fuel Pool Cooling System

a. Inspection Scope

The inspectors conducted a partial walkdown of the spent fuel pool cooling system. As discussed in Updated Final Safety Analysis Report (UFSAR) Section 14.2.1, "Fuel Handling Accident," the spent fuel pool cooling system was provided to remove fuel assembly decay heat and prevent high temperature conditions in the spent fuel pool that could potentially reduce the effectiveness of the pool as a barrier to fission product release. The inspectors reviewed the normal operating, abnormal, and annunciator response procedures for the spent fuel pool cooling system. The inspectors also reviewed system flow diagrams and applicable UFSAR sections. The inspectors verified that system alignment was consistent with procedural requirements. The inspectors reviewed the following documentation during this review:

- UFSAR Section 9.4, "Spent Fuel Pool Cooling System"
- UFSAR Section 14.2.1, "Fuel Handling Accident"
- 12-OHP 4021.018.002, "Placing in Service and Operating the Spent Fuel Pit Cooling and Cleanup System," Revision 11
- 12-OHP 4022.018.001, "Loss of Spent Fuel Pit Cooling," Revision 5
- 1[2]-OHP 4024.105[205], "Annunciator #105[205] Response: Containment Spray," Revision 8[6]

- 01-OHP 4024.134, “Annunciator #134 Response: Spent Fuel Pit,” Revision 3
- Flow Diagram OP 12-5136, “Spent Fuel Pit Cooling and Cleanup Unit 1 and 2,” Revision 21
- 02-OHP 5030.001.001, “Operations Plant Tours,” Revision 16

b. Issues and Findings

There were no findings identified and documented during this inspection.

.4 Partial Walkdown of Unit 2 Safety-Related Electrical Distribution

a. Inspection Scope

The inspectors performed a partial walkdown of the Unit 2 safety-related 4160 VAC and 600 VAC breaker alignment. As discussed in Section 14.1.12 of the UFSAR, the loss of all AC power to the plant auxiliaries was analyzed as a potential consequence of a loss of off-site power. The inspectors verified that the Unit 2 safety-related electrical distribution system was aligned in accordance with plant procedures to minimize the in-plant effects of a loss of off-site power or other off-site electrical grid disturbance. In addition to the walkdown, the inspectors reviewed the following documents:

- 02-OHP 4022.001.005, “Loss of Offsite Power With Reactor Shutdown,” Revision 4a
- 02-OHP 4022.082.004, “Degraded Offsite AC Voltage Response,” Revision 0
- 02-OHP 4023.ECA-0.0, “Loss of All AC Power,” Revision 10
- 02-OHP 4030.STP.030, “Daily and Shiftly Surveillance Checks,” Revision 31
- 02-OHP 4030.STP.031, “Operation Weekly Surveillance Checks,” Revision 11
- TS 3.8.1.1a, AC Sources Operating
- TS 3.8.2.1, AC Distribution Operating
- UFSAR Section 14.1.12, “Loss of All AC Power to the Plant Auxiliaries”
- Design Information Transmittal (DIT) B-01099, “Minimum and Maximum Acceptable Voltages at the 4160V and 600V Safety Buses for Modes 1 Through 6 and Defueled Condition,” Revisions 3 and 4
- Condition Report (CR) 00252078, NRC located deficiency tag 44517 still hanging in field with a note that it was a Unit 2 restart, Mode 4 constraint

b. Issues and Findings

There were no findings identified and documented during this inspection.

.5 Partial Walkdown of the Unit 2 Auxiliary Feedwater System

a. Inspection Scope

The inspectors performed a partial walkdown of the Unit 2, Turbine Driven Auxiliary Feedwater Pump (TDAFWP). The inspectors compared the operating status and configuration of the TDAFWP to the applicable operating procedures, the system valve lineup, and applicable flow diagrams. As part of this inspection, the inspectors reviewed the following documents:

- 02-OHP 4021.056.001, “Filling and Venting Auxiliary Feedwater System,” Revision 11a
- 02-OHP 4024.213, “Annunciator #213 Response: Steam Generator 1 and 2,” Revision 5
- 02-OHP 4030.STP.031, “Operation Weekly Surveillance Checks,” Revision 11
- Flow Diagram OP 2-5106A, “Auxiliary Feedwater System”
- Flow Diagram OP 2-5105E, “Main Steam”

b. Issues and Findings

There were no findings identified and documented during this inspection.

.6 Partial Walkdown of Unit 2 Containment Spray System

The inspectors performed a partial walkdown of the Unit 2 Containment Spray System (CTS). The inspectors compared the operating status and configuration of the CTS to the applicable operating procedures, the system valve lineup, and applicable flow diagrams. As part of this inspection, the inspectors reviewed the following documents:

- 02-OHP 4021.009.001, “Placing the Containment Spray System in Standby Readiness,” Revision 6
- 02-OHP 4024.205, “Annunciator 205 Response: Containment Spray,” Revision 6
- Flow Diagram OP 2-5143, “Emergency Core Cooling”
- Flow Diagram OP 2-5144, “Containment Spray”

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed fire protection walkdowns of three risk-significant plant areas: Fire Zone 18, the Unit 2 CD Emergency Diesel Generator (D/G) room; Fire Zone 19, the Unit 2 AB D/G room; and Fire Zone 44S, the south auxiliary building 609’ elevation near the Component Cooling Water (CCW) pumps. The Donald C. Cook Nuclear Plant Fire Analysis Notebook stated that a fire in any of these areas could increase the probability of a loss of both of the Unit 2 trains of CCW. The following plant procedures were reviewed during this inspection:

- 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
- UFSAR Section 9.8.1, “Fire Protection System”

- Donald C. Cook Nuclear Plant Fire Hazards Analysis, Units No. 1 and 2, Revision 8
- Donald C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed and assessed flood protection measures for internal and external flooding events. The inspectors reviewed the licensee's probabilistic risk analysis and associated flood protection reports to identify risk significant flood areas and protective features. The inspectors performed walkdowns of risk significant flood areas, including the turbine building sub-basement, the auxiliary feedwater (AFW) pump area, the D/G rooms, the screenhouse, and the auxiliary building. The inspectors reviewed abnormal and alarm response procedures associated with the diagnosis and mitigation of flooding events. The inspectors also discussed flood protection features and recently identified problems with the turbine and auxiliary building system manager and design engineering. The inspectors reviewed the following documents during this inspection:

- Donald C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Internal Flooding Analysis Notebook, April 1992.
- Calculation MD-12-SCRN-001-N, "Screen House Internal Flood Levels," Revision 0
- Calculation DCC-PV-12-MC17-N, "Flood Protection Features"
- Calculation DCC-PV-12-MC22-N, "Flood Protection ESW Pipe Tunnel"
- Flooding Evaluation for AEP [American Electric Power], D.C. Cook Unit #2, Report Number NED-2000-537-REP, Revision 0
- Final Expanded System Readiness Report, Essential Service Water System (Unit 2)
- UFSAR, Section 14.4.2.6.2, "Flooding"
- 12-OHP 4022.001.009, "Seiche," Revision 0
- 01(2)-OHP 4024.118(218), "Annunciator #118(218) Response: Main & FPT [Feed Pump Turbine]," Revision 7(7)
- 01-OHP 4024.124, "Annunciator #124 Response: Containment," Revision 3
- 02-OHP 4024.205, "Annunciator #205 Response: Containment Spray," Revision 6
- 02-OHP 4024.206, "Annunciator #206 Response: Residual Heat Removal," Revision 6
- 02-OHP 4024.216, "Annunciator #216 Response: Condensate," Revision 7
- CR 99-16625, Physical condition of condenser pit walls not consistent with assumptions in internal flood analysis

- CR 99-15588, Expanded System Readiness Review indicated that flood protection calculations were unacceptable and did not include the screenhouse in the review
- CR 00-8792, Local flooding of the plant access road and north gatehouse
- CR 99-12376, The design basis flood elevation of 595' does not include surface run-off contribution
- CR 99-16669, Sufficient information to demonstrate reasonable assurance that the auxiliary building and turbine building are protected from design basis flooding events is not available
- CR 99-13299, The turbine building has two unprotected roll-up doors located below the design basis flood elevation of 595'
- CR 00257090, Failure to evaluate flooding impact of shutting roll up door for high energy line break protection
- CR 00264091, Configuration control of turbine building drain system valves not established by plant procedures
- CR 00262020, Auxiliary Building flooding safety analysis report description of operator response for flooding requires clarification
- CR 00271026, Potential inappropriate use of NRC individual plant examination safety evaluation report regarding the acceptability of flood protection

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R07 Heat Sink Performance

a. Inspection Scope

Generic Letter (GL) 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” identified a number of problems and events with service water systems at nuclear power plants. In response to Generic Letter 89-13, the licensee reported that they had established a routine inspection and maintenance program for open-cycle service water system piping and components. The inspectors discussed the most recent inspection results for the following safety-related heat exchangers with the licensee’s GL 89-13 program owner: the Unit 2 East CCW heat exchanger (JO R80406) and the Unit 2 West CCW heat exchanger (JO R88985). In addition, the inspectors reviewed the following documents:

- Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”
- NRC Inspection Report 50-315/95003; 50-316/95003
- Engineering Head Instruction (EHI) 89-13, “Program for Implementing Generic Letter 89-13 (Service Water Reliability),” Revision 0
- 12-EHP [Engineering Head Procedure] 6040.PER.002CCW, “Component Cooling Water Heat Exchanger Functional Test,” Revision 1
- JO R80406, Open and inspect both ends of the Unit 2 East CCW heat exchanger
- JO R88985, Open and inspect both ends of the Unit 2 West CCW heat exchanger

- CR 99-23346, The divider plate between the tube side of the CCW heat exchanger was bowed
- CR 99-24126, A Service Water System Operational Performance Inspection self assessment audit was performed in 1995 on the GL 89-13 program and ten recommendations were made. This CR was written to track resolution of each of those recommendations
- CR 99-24195, A pit in the wall of the end bell of the Unit 2 East CCW heat exchanger was discovered while craft were scraping painted surface to remove loose coatings
- CR 99-25793, During performance of heat exchanger inspection on Unit 2 West CCW heat exchanger, a small amount (less than 1 cup) of foreign debris was found

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R12 Maintenance Rule Implementation

The inspectors evaluated the licensee's implementation of the Maintenance Rule, 10 CFR Part 50.65, for the Unit 1 Chemical and Volume Control System (CVCS) and Spent Fuel Pool Cooling systems. The inspectors reviewed recent Maintenance Rule evaluations for these systems to assess: (1) scoping in accordance with 10 CFR 50.65; (2) characterization of systems, structures, and components (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 engineer.

.1 Maintenance Rule Implementation for the Spent Fuel Pool Cooling System

a. Inspection Scope

The inspectors evaluated the licensee's implementation of the Maintenance Rule, 10 CFR Part 50.65, for the spent fuel pool cooling system. As discussed in UFSAR Section 14.2.1, "Fuel Handling Accident," the spent fuel pool cooling system provided a barrier function to offsite releases during postulated fuel handling accidents by providing adequate water inventory above irradiated fuel assemblies. The spent fuel pool cooling system was scoped in the Maintenance Rule as a low risk significant system. The spent fuel pool cooling system was categorized as a 10 CFR 50.65(a)(2) system with performance criteria established for both train unavailability and spent fuel pool level. The inspectors reviewed the system risk ranking, function scoping, and performance criteria. The inspectors reviewed a sampling of recent system performance problems to verify that issues were appropriately evaluated within the licensee's Maintenance Rule program. The inspectors also discussed system performance with the Maintenance Rule coordinator and the system manager. The inspectors reviewed the following documents during this review:

- UFSAR Section 9.4, “Spent Fuel Pool Cooling System”
- PMI 5035, “Maintenance Rule Program,” Revision 4
- NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
- Regulatory Guide 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”
- PMP 5035.MRP.001, “Maintenance Rule Program Administration,” Revision 0
- Maintenance Rule Scoping Matrix for Spent Fuel Pool Cooling
- 02-OHP 4030.STP.030, “Daily and Shiftly Surveillance Checks,” Revision 31
- PMP 4030.EIS.001, “Event Initiated Surveillance Testing,” Revision 1
- Calculation MD-12-SFP-002-N, “Spent Fuel Pool Water Level,” Revision 0
- CR 99-15908, The discharge check valve on the north spent fuel cooling pump stuck open while switching pumps
- CR 00251103, A Maintenance Rule functional failure evaluation was not performed for a spent fuel pool cooling pump discharge check valve failure
- CR 99-0181, It was determined that the spent fuel pool cooling system should have been classified as risk-significant
- CR 00245087, Maintenance Rule unavailability data collection could not be demonstrated for the spent fuel pool cooling system prior to April 2000
- CR 00253031, The functional failure definition of the spent fuel pool cooling system differs from the system design bases as discussed in UFSAR Section 9.4.1
- CR 00259037, Control Room logkeeping guidelines do not include all Maintenance Rule logging activities

b. Issues and Findings

During a review of Maintenance Rule unavailability data for the spent fuel pool system, the inspectors identified that cooling train unavailability time had not been accumulated prior to April 2000. The licensee’s Maintenance Rule expert panel identified several functions for the spent fuel pool cooling system, including heat removal capability and spent fuel pool inventory control. Because the licensee believed that the performance of the spent fuel pool cooling system was effectively controlled through the performance of appropriate preventive maintenance, these functions were monitored under the requirements of 10 CFR 50.65(a)(2). Based on independent evaluations performed by the inspectors, a significant number of spent fuel pool cooling train unavailability hours had been accumulated within the 24-month performance criteria window prior to April 2000. The inspectors discussed this issue with the Maintenance Rule group and were informed that monitoring of unavailability for all maintenance rule systems had been suspended during the September 1997 through June 2000 dual unit extended outage. Because the licensee was not monitoring unavailability during the extended shutdown, the effectiveness of the preventative maintenance program for systems required to be available during shutdown conditions, including the spent fuel pool cooling system, was not demonstrated.

On September 1, 2000, the licensee initiated CR 00245087 to document that Maintenance Rule unavailability had not been collected during the extended shutdown prior to April 2000. During the follow-up investigation for this issue, the licensee identified that the failure to monitor system unavailability during the shutdown impacted the Maintenance Rule monitoring of 46 systems. The systems affected included essential service water, component cooling water, chemical and volume control, spent fuel pool cooling, residual heat removal, nuclear instrumentation, radiation monitoring, reactor coolant system, 250 VDC, offsite power, and 120 VAC vital power. At an expert panel meeting convened on September 6, 2000, the licensee placed these 46 systems in (a)(1) status because sufficient data was not available to demonstrate compliance with unavailability performance criteria.

10 CFR 50.65 (a)(1), required, in part, that the license shall monitor the performance or condition of structures, systems, or components (SSCs) within the scope of the rule as defined by 10 CFR 50.65 (b), 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 is being 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 or condition of systems required to be available during shutdown conditions and within the scope of the rule had been effectively controlled through the performance of appropriate preventive maintenance and did not monitor against licensee-established goals. Specifically, the licensee failed to monitor the unavailability of 46 systems required during shutdown mode operation. This violation is being treated as a Non-Cited Violation (50-315/316-00-20-01) consistent with Section VI.A. of the NRC Enforcement Policy. This violation is in the licensee's corrective action system as CR 00245087. Because of the number of systems the licensee failed to monitor, the inspectors concluded that this issue was not an isolated case, and, if left uncorrected, could have become a more significant safety concern. Since the licensee failed to monitor the performance of SSCs, the reliability of systems and effectiveness of the licensee's maintenance program could not be demonstrated. This NCV is closed.

The inspectors also identified that the licensee had not performed an apparent cause determination for several condition reports, contrary to Maintenance Rule program administrative requirements. During follow-up to the inspectors' questions, the licensee identified a total of 35 functional failures that had not received an apparent cause determination. The inspectors and the licensee identified other program deficiencies, including: inadequate performance criteria, inadequate Maintenance Rule expert panel reviews, failures to identify and evaluate functional failures, and inadequate scoping of emergency operating procedure functions. As a result of these identified problems, the licensee initiated Engineering Action Plans 00-511 and 00-516 to determine the extent of condition of program deficiencies, evaluate the root causes of these deficiencies, and initiate corrective actions.

.2 Unit 1 Chemical and Volume Control System

a. Inspection Scope

The inspectors evaluated the licensee's implementation of the Maintenance Rule, 10 CFR Part 50.65, for the Unit 1 Chemical and Volume and Control System (CVCS). The inspectors reviewed the system risk ranking, function scoping, and performance criteria. The CVCS was originally scoped into the Maintenance Rule as a low risk significant system, but the licensee's Maintenance Rule expert panel reclassified the system as risk significant in early 1999. The inspectors reviewed a sampling of recent system performance problems to verify that issues were appropriately evaluated within the licensee's Maintenance Rule program. The inspectors also discussed system performance with the Maintenance Rule Coordinator. The inspectors reviewed the following documents during this review:

- PMI 5035, "Maintenance Rule Program," Revision 4
- PMP 5035.MRP.001, "Maintenance Rule Program Administration," Revision 0
- NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"
- Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"
- UFSAR Section 9.2, Chemical and Volume Control System
- Maintenance Rule Expert Panel Meeting Minutes for meetings conducted January 6, 1999; March 18, 1999; March 25, 1999; March 31, 1999; and November 18, 1999
- CR 98-3495, Significant weaknesses have been identified in the Maintenance Rule program
- CR 99-0346, Maintenance Rule Expert Panel classified the CVCS as risk-significant
- CR 99-5572, Recurring failures of the limit switches for QRV-61 and QRV-62
- CR 00245090, Emergency Operating Procedure function of QRV-61 and QRV-62 was not scoped into the Maintenance Rule
- CR 00277037, NRC identified failure to develop and implement monitoring goals for systems classified as "Administrative (a)(1)"
- Flow Diagram OP-1-5129, "Flow Diagram - CVCS, Reactor Letdown and Charging, Unit No. 1"
- Unit 1 Emergency Operating Procedure 01-OHP 4023.E-3, "Steam Generator Tube Rupture," Revision 6

b. Issues and Findings

On January 6, 1999, the licensee's Maintenance Rule expert panel placed the Unit 1 CVCS in "administrative (a)(1)" following the CVCS reclassification as a risk-significant system. The licensee used the term, "administrative (a)(1)," to identify systems which needed a review of their performance history data to demonstrate adequate system performance. Because the Unit 1 CVCS had not been previously monitored for both reliability and unavailability, the licensee could not initially demonstrate that the performance of the system was being effectively controlled through preventive maintenance. The licensee initiated Condition Report 99-0346 to identify the actions needed to appropriately classify and monitor the Unit 1 CVCS. In accordance with the corrective actions identified in the CR, the Unit 1 CVCS performance history data was collected. However, the historical data was never formally reviewed by the expert panel, and the Unit 1 CVCS remained in “administrative (a)(1).” The licensee added Action Item 41 to CR 98-3495 to develop performance criteria for the Unit 1 CVCS. At the time of this inspection; however, CR 98-3495 Action Item 41 was still open.

The inspectors discussed the lack of performance criteria and performance goals for the Unit 1 CVCS with the licensee’s Maintenance Rule Coordinator. The Maintenance Rule Coordinator stated that performance criteria for the Unit 1 CVCS were being developed, but that performance goals sufficient to assure capability to perform intended functions were not routinely established for systems placed in “administrative (a)(1).” The licensee added a section to Engineering Action Plan 00-516 to complete a review of the performance of the systems in “administrative (a)(1)” and develop appropriate goals for those systems.

10 CFR 50.65 (a)(1) required, in part, that each holder of a license to operate a nuclear power plant shall monitor the performance or condition of structures, systems, or components, 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. Contrary to the above, on September 13, 2000, the inspectors identified that the licensee had not established performance goals for the Unit 1 CVCS system and the licensee had not demonstrated that the performance of the Unit 1 CVCS was being effectively controlled through the performance of appropriate preventive maintenance. The inspectors determined that the failure to establish performance goals for the Unit 1 CVCS has a credible impact on safety and constituted a violation of 10 CFR 50.65 (a)(1). This violation is being treated as a Non-Cited Violation (50-315-00-20-02) consistent with Section VI.A. of the NRC Enforcement Policy. This issue was entered into the licensee’s corrective action program as CR 00277037. The failure to establish performance goals for the Unit 1 CVCS was significant in that without established goals the licensee could not demonstrate the effectiveness of maintenance to assure reliability of the system. This NCV is closed.

.3 (Closed) Unresolved Item (URI) 50-316/2000019-01: Potential mis-classification of a Maintenance Rule maintenance preventable functional failure.

During the a review of Maintenance Rule functional failures, the inspectors identified a potential mis-classification of a maintenance preventable functional failure (MPFF). Condition Report 99-20499 described a loss of 250 VDC control power to a Unit 2 CD D/G output breaker, that occurred on August 7, 1999, due to loose fuse clips. The licensee’s investigation determined that the probable cause for the loose fuse clips was repeated mechanical cycling of the fuse clip assembly. The CR stated that the fuse clip assembly degraded due to repeated tagging out of the D/G output breaker to support a high number of maintenance activities. The associated Maintenance Rule evaluation concluded that, although the fuse clip failure was a functional failure, the failure was not maintenance preventable. However, the inspectors concluded that tagging to support maintenance was a maintenance support activity, and therefore maintenance preventable. After the inspectors questioned the initial categorization of the failure, the licensee determined that the failure was maintenance preventable and therefore represented an MPFF. The licensee initiated CR 00238033 to document this issue.

Control power from the 250 VDC system was required to operate certain circuit breaker protection devices. The licensee monitored performance of the 250 VDC system under the requirements of 10 CFR 50.65(a)(2). The performance criteria for 250 VDC system function associated with providing electrical circuit protection allowed no MPFFs over a 24-month period. The identification of the Unit 2 CD diesel generator output breaker MPFF resulted in the 250 VDC system exceeding its established performance criteria. The inspectors reviewed the licensee's corrective action and work management systems to determine if other 250 VDC control power failures had occurred due to fuse block problems. The inspectors identified three additional control power failures associated with degraded fuse block clips:

- Condition Report 99-27422 documented a failure of the Unit 2 West ESW pump breaker to remotely operate from the control room on November 16, 1999. The CR evaluation concluded the failure was due to a high fuse block contact resistance caused by bending and misalignment of one of the fuse block stab connections. The condition evaluation further stated that: (1) the fuse block may have been damaged during improper storage or the damage may have occurred during installation, and (2) the procedure providing guidance for a previous breaker refurbishment included instructions to verify that fuse clips had proper clamping force. The evaluation stated that it would have been reasonable to assume that inspection of the stabs would have been included in fuse clip clamping verification.

  Although CR 99-27422 documented a potential functional failure, the CR had not been evaluated for impact to system monitoring under the Maintenance Rule. The engineers performing the CR evaluation failed to identify any Maintenance Rule impact because they incorrectly believed that the ESW pumps were outside the scope of the Maintenance Rule. The inspectors discussed this CR with the Maintenance Rule Coordinator and confirmed that; (1) the ESW pumps are included within the Maintenance Rule database, and (2) the condition described in CR 99-27422 should have received a Maintenance Rule evaluation. The licensee initiated CR 00243033 to document the failure to evaluate the functional failure described in CR 99-27422 for impact on Maintenance Rule monitoring of the 250 VDC system.

- Condition Report 99-10158, initiated on May 1, 1999, documented that control power was lost to the 2 East CCW pump. At the time of the discovery, the unit was in Mode 5 and the affected CCW pump breaker was open. The licensee determined that the loss of power was caused by a loose stab on the 250 VDC control power fuse block. The licensee determined that this occurrence was an MPFF; however, this MPFF was never recorded against the 250 VDC system. During discussions with Maintenance Rule group program personnel, the inspectors were informed that this failure was improperly categorized as a failure of the CCW system rather than a 250 VDC failure. The licensee initiated CR 00266067 to document and evaluate this issue.

- Condition Report 00249082, initiated on September 5, 2000, documented that control power was lost for breaker 1-11C1 (the feeder breaker to Unit 1 600 VAC Train "A" bus 11C) with the breaker shut. The licensees' initial investigation determined corrosion product build up on fuse block electrical connections resulted in inadequate circuit continuity.

Based on the 250 VDC control power failures that have occurred due to degraded fuse blocks, the inspectors determined that the licensees' preventative maintenance program had not adequately maintained control power reliability. On September 22, 2000, the licensee initiated CR 00266029 to document that the allowable functional failure goal for the 250 VDC system had been exceeded.

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) requires, 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 250 VDC system 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 of 250 VDC control power fuse blocks 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. This violation is being treated as a Non-Cited Violation (50-315/316-00-20-03) consistent with Section VI.A. of the NRC Enforcement Policy. This violation is in the licensee's corrective action system as CR 00266029. The inspectors determined that these fuse block failures could have had a credible impact on safety and could become a more significant safety concern if left uncorrected. This issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. This NCV is closed.

1R13 Maintenance Risk Assessments and Emergent Work Control

The inspectors reviewed the licensee's evaluation of plant risk, scheduling, configuration control, and performance of maintenance associated with planned and emergent work activities.

.1 Temporary Modification on Unit 2 East Containment Spray Pump

a. Inspection Scope

On September 8, 2000, during a routine surveillance test of the Unit 2 East CTS pump, the licensee measured high motor vibrations on the pump motor. The vibration levels exceeded the action levels which the licensee had established. This surveillance test is discussed in Section 1R22.1.

After reviewing the vibration data, the licensee identified that the high vibrations were due to an interaction between the natural resonance frequency of the motor and the vane pass frequency of the pump impeller. The licensee also determined that the Unit 2 West CTS pump and both Unit 1 CTS pumps were susceptible to a similar frequency interaction. The licensee determined that a possible long-term solution would be to change the pump impellers from a four-vane design to a five-vane design to eliminate the frequency interaction. However, due to the long lead time required to design and acquire the new pump impellers, the licensee developed a Temporary Modifications (TM) which was designed to reduce the high vibrations by adding weight to the motors. The inspectors compared the licensee's preparations for installing the TMs to work control process procedures and verified compliance with TS requirements during the TM installation. The inspectors reviewed the following documents during this review:

- Engineering Action Plan CTS-00-490, Containment Spray
- TS 3.6.2.1, "Containment Spray System"
- CR 00252092, Unit 2 CTS pump failed its surveillance due to high vibration
- TM 02-TM-00-49-R0, "Reduce Vibration of CTS Pump 2-PP-9E," Revision 0
- TM 02-TM-00-50-R0, "Reduce Vibration of CTS Pump 2-PP-9W," Revision 0
- Calculation EVAL-SD-000913-002, Evaluation of Structural Attachment to Resolve Vibration Problem in CTS Pumps 2-PP-9E and 2-PP-9W
- 12-EHP 5040.MOD.001, "Temporary Modifications," Revision 5
- Job Order 00253026, Install additional mass and/or dynamic absorber on Unit 2 East CTS pump
- Job Order 00253027, Install additional mass and/or dynamic absorber on Unit 2 West CTS pump
- CR 00265001, Calculation number EVAL-SD-000913-002 did not include the weight of two corner angles in the structural evaluation of Temporary Modification 2-TM-00-49-R0
- CR 00265013, Administrative error identified on Temporary Modification 2-TM-00-50-R0 and 2-TM-00-49-R0 request forms

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Risk Assessment for Removal of Unit 2 West Component Cooling Water Pump from Service

a. Inspection Scope

The inspectors reviewed the risk assessment performed to support preventative maintenance on the Unit 2 West CCW pump. The maintenance was performed in accordance with Job Orders R0098717 and R0098718. The inspectors verified that this maintenance activity was reflected in the risk analysis. The inspectors assessed compliance with procedural requirements for on-line maintenance, and reviewed active clearances in effect at the time of the maintenance. Additionally, the inspectors walked down safety-significant areas of the plant to identify any additional issues that could impact the risk associated with removal of the Unit 2 West CCW from service. These walkdowns included the control room, the CCW pumps, emergency core cooling system pump areas, and electrical switchboards. The inspectors reviewed the following documents during this inspection:

- PMP 2291.OLR.001, "On-Line Risk Management," Revision 0
- 02-OHP 4022.016.004, "Loss of Component Cooling Water," Revision 6
- CR 00246002, Unavailability of Unit 1 charging pumps adversely affected Unit 2 risk
- CR 00255080, The Unit 2 West CCW pump was removed from service without performing a PMP 2291.OLR.001 contingency briefing
- CR 00254022, The method of determining on-line risk conditions (or unit risk color) is not well defined

b. Issues and Findings

There were no findings identified and documented during this inspection.

.3 Risk Assessment for Unit 2 September 24 through 30, 2000, Work Week Schedule

a. Inspection Scope

The inspectors reviewed the risk assessment performed to support Unit 2 maintenance activities during the week of September 24, 2000. The scheduled maintenance activities included removal from service of a condensate booster pump, a control air compressor, the north safety injection pump, and loss of cross-plant charging and auxiliary feedwater flow capability from Unit 1. The inspectors verified that major equipment outages were reflected in the risk analysis and assessed compliance with procedural requirements for on-line maintenance. Additionally, the inspectors walked down safety-significant areas of the plant to identify any additional issues that could impact the risk assessment, including active clearances and unscheduled maintenance activities. These walkdowns included the control room, emergency core cooling system pump areas, and electrical switchboards. The inspectors reviewed the following documents during this inspection:

- PMP 2291.OLR.001, "On-Line Risk Management," Revision 0

- Work Week Cycle 34, Week 5 On-Line Risk Assessment, Unit 2, dated September 19, 2000
- Unit 2, T-1 Look Ahead Schedule, run date September 18, 2000
- Unit 2 Surveillance Testing Schedule, September 1 through September 30, 2000

b. Issues and Findings

There were no findings identified and documented during this inspection.

.4 Risk Assessment for Unit 1 Second Train "A" Outage

a. Inspection Scope

The inspectors reviewed the licensee's shutdown risk assessment for the second Unit 1 Train "A" outage work schedule. The licensee planned a second Train "A" outage work window to support Unit 1 restart work and electrical bus refurbishment activities. The licensee determined that the work scheduled during the second Train "A" outage would result in several yellow risk path entries over the course of the work window. The inspectors reviewed the licensee's schedule, the licensee's outage risk assessment, and the following document:

- PMP 4100.SDR.001, "Plant Shutdown Safety and Management," Revision 4

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R15 Operability Evaluations

.1 Review of Operability Determination Associated with Impact of Minimum Auxiliary Building Temperature on Boration System

a. Inspection Scope

The inspectors reviewed the operability determination associated with CR 00-6906, which was initiated on May 13, 2000, to document a potential discrepancy between the design basis of the auxiliary building ventilation system and the minimum required temperature for the CVCS boration flowpath piping. Specifically, TS 3.2.1, "Boration System Flow Paths - Shutdown," and TS 3.1.2.2, "Reactivity Control Systems Flow Paths - Operating," required that the temperature of the areas containing certain boration flow path components in the auxiliary building be verified to be greater than or equal to 63°F. Updated Final Safety Analysis Report Section 9.9.2, "Design Basis - Auxiliary Building Ventilation System," stated that the auxiliary building ventilation system heating was designed to maintain a minimum auxiliary building temperature of 60°F. The condition report was initiated to determine if the auxiliary building ventilation lower design temperature of 60°F adversely impacted the operability of the boration flow path. The inspectors reviewed the condition report evaluation and operability determination. The inspectors reviewed applicable normal operating procedures and operator logging procedures to verify consistency with the operability determination. The inspectors reviewed the following documents:

- CR 00-6906, Auxiliary building system design may not protect operability requirements of chemical volume control boration system
- CR 00-5752, Daily and shiftly surveillance checks have no contingency actions if boron injection flow path area temperature acceptance criteria is not met
- CR 00-5440, Design Change Package 642 reduced the required boron concentration from 12 percent to 4 percent and required a minimum temperature of 63°F. The design minimum temperature for the auxiliary building ventilation system is 60°F.
- Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 200 to Facility Operating License No. DPR-74
- 02-OHP 4030.STP.030, "Daily and Shiftly Surveillance Checks," Revision 31
- 12-OHP 4021.028.011, "Auxiliary Building Ventilation," Revision 10

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Operability of Spent Fuel Pool Cooling System Following Complete Core Offload

a. Inspection Scope

On January 27, 2000, the licensee initiated CR 00-1543 to document an operability determination for the spent fuel pool cooling system. The operability determination evaluated the licensee's normal practice of performing full core offloads during refueling outages. Updated Final Safety Analysis Report Section 9.4.1, "Spent Fuel Pool Cooling System," stated that one train of spent fuel cooling was capable of maintaining adequate spent fuel pool cooling during a partial core offload, consisting of the discharge of 80 fuel assemblies from the reactor. The UFSAR further stated that during a full core offload, consisting of a 193 fuel assembly discharge, two trains of spent fuel pool were required to maintain adequate pool cooling. The NRC safety evaluation report for licensee amendment 169 (Unit 1) and 152 (Unit 2), dated January 14, 1993, described the partial core offload as the normal fuel assembly discharge to the pool during refueling outages. However, in the past the licensee had performed full core offloads as the normal practice. The operability determination evaluated the capability of the spent fuel cooling system following a full core offload with only a single cooling train in service. The evaluation concluded that the system was operable with non-conformances relating to the system description contained in the UFSAR.

The inspectors reviewed the condition report evaluation and operability determination, including applicable compensatory actions for operable but non-conforming equipment status. The inspectors reviewed applicable normal operating procedures and emergency operating procedures to verify consistency with the operability determination. Additionally, the inspectors reviewed the engineering calculations supporting the operability determination conclusions. The inspectors reviewed the following documents:

- Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 169 to Facility Operating License No. DPR-58
- AEP Letter to NRC dated July 26, 1991, “Spent Fuel Pool Reracking Technical Specification Changes”
- PMP 4030.EIS.001, “Event Initiated Surveillance Testing,” Revision 1
- PMP 4100.SDR.001, “Plant Shutdown Safety and Risk Management,” Revision 4
- 01-OHP 4030.STP.021, “Event Initiated Surveillances,” Revision 9
- PMP 2291.OLR.001, “On-Line Risk Management,” Revision 0
- 01[02]-OHP 4024.105[205], “Annunciator #105[205] Response: Containment Spray,” Revision 8[6]
- 12-OHP 4024.134, “Annunciator #134 Response: Spent Fuel Pit,” Revision 3
- D. C. Cook NRC Commitment 7790, Spent Fuel Pool Cooling System
- Calculation MD-12-SFP-006-N, “Design Basis and Cycle-Specific Spent Fuel Pool Thermal Performance,” Revision 0
- CR 99-14632, Technical Specification basis for TS 3.9.3 should be clarified to reflect that the fuel handling accident analysis assumes a decay time of 100 hours and the thermal analysis assumes a decay of 168 hours
- CR 99-14492, Spent fuel pool cooling and makeup does not meet Standard Review Plan Section 9.1.3 requirements for seismic category I makeup source
- CR 00265057, Condition Report 00-7202 stated that it was acceptable for the spent fuel pool to boil
- CR 00-1543, The performance of a full core offload does not appear to be consistent with the normal core offload terminology used in previous AEP and NRC reports, analysis, and guidance documentation
- CR 99-14654, UFSAR assumptions regarding spent fuel pool cooling train availability during full core offload are invalid

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and observed post-maintenance testing following routine and emergent maintenance activities. During post-maintenance testing observations, the inspectors verified that: (1) the test was adequate for the scope of the maintenance work which had been performed, (2) the testing acceptance criteria were clear and demonstrated operational readiness consistent with the design and licensing basis documents, (3) the impact of the testing had been properly characterized during the pre-job briefing, (4) the test was performed as written and all testing prerequisites were satisfied, and (5) the test data was complete, appropriately verified, and met the requirements of the testing procedure. Following the completion of the test, the inspectors verified that the test equipment was removed and that the equipment was returned to a condition in which it could perform its safety function. The inspectors reviewed the following post-maintenance testing activities and verified that the post-maintenance testing met the requirements of PMP 2291.PMT.001, “Work Management

Post Maintenance Testing Matrices," Revision 2 and PMI 2294, "Post-Maintenance Testing Program," Revision 0:

- On September 6, 2000, the inspectors observed the as-left leak rate testing on 1-ICM-251, the Unit 1 Train "B" boron injection tank outlet valve. The valve was tested in accordance with 01-EHP (Engineering Head Procedure) 4030.STP.203, "Unit 1 Type B and C Leak Rate," Revision 4 under Job Order R49475.

- On September 22, 2000, the inspectors observed the post-maintenance test for installing TM 02-TM-00-50 on the Unit 2 West CTS pump per JO 00253027. The TM installation was tested as part of Surveillance Procedure 02-OHP 4030.STP.007W. The TM is discussed in more detail in Section 1R13.1, above.

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R22 Surveillance Testing

.1 Surveillance Test on Unit 2 East Containment Spray Pump

a. Inspection Scope

The inspectors observed the performance of an increased frequency surveillance test on the Unit 2 East CTS pump. This pump was tested at a shorter testing interval due to motor vibration measurements which exceeded the alert values recorded in the Unit 2 Technical Data Book (TDB) Figure 2-15.2, "Safety Related Pump Inservice Test Vibration Data," Revision 48. The inspectors reviewed the surveillance results and the following documents:

- 02-OHP 4030.STP.007E, "East Containment Spray System Operability Test," Revision 14
- Safety Screening 2000-1824-00, "Revision to Technical Data Book Inservice Test Vibration Data for Unit 2 Containment Spray Pump, Figure 2-15.2, Safety Related Pump Inservice Test Vibration Data"
- TDB Figure 2-15.2, "Safety Related Pump Inservice Test Vibration Data," Revision 48
- Licensee Event Report (LER) 50-316/00012-00, Failure to Perform Increased Frequency Surveillance on Unit 2 East Containment Spray Pump
- PMI 5071, "Inservice Testing," Revision 0
- Engineering Head Instruction 5071, "Inservice Testing Program Implementation," Revision 0
- CR 00-11227, Surveillance Test 02-OHP 4030.STP.007E shows Unit 2 East CTS pump in alert with high vibrations
- CR 00-11271, On June 2, 2000, the Unit 2 West CTS pump vibration test was in the alert range and the test frequency was not increased
- CR 00252092, Unit 2 East CTS pump failed its surveillance test due to high vibrations above the action limit

- CR 00259088, Data collector CNP-812 collected inaccurate data on the CTS pump during post-modification testing

b. Issues and Findings

There were no findings identified and documented during this inspection.

.2 Inservice Testing of the Unit 2 East Residual Heat Removal Pump

a. Inspection Scope

On August 31, 2000, the inspectors observed quarterly inservice testing of the Unit 2 East RHR pump. The inspectors assessed procedural compliance, communication, operator performance, and work control associated with the test performance. The following documents were reviewed during this inspection:

- 02-OHP 4030.STP.050E, “East Residual Heat Removal Train Operability Test Modes 1 - 4,” Revision 8
- Unit 2 TDB, Figure 2-15.1, “Safety Related Pump Inservice Test Hydraulic Reference,” Revision 49
- 12-EHP 5070.ISI.017R, “Section XI Centrifugal Pump Performance Verification,” Revision 6
- DIT-B-00757-00, Design Basis Performance Parameters for the Safety Injection, Centrifugal Charging, and Residual Heat Removal Pumps
- OHI-4032, “Leakage Monitoring Program,” Revision 0
- ASME OMa-1988, Part 6, “Inservice Testing of Pumps in Light Water Reactor Power Plants”
- CR 00244020, East RHR Pump Surveillance Test requires a reference flow of 460 gpm. The designated flowmeter cannot be read that accurately.
- CR 00-8494, East RHR pump failed to meet recirculation flow as specified in the technical data book
- Engineering Action Plan RHR 00-467, The RHR pump surveillance test, 02-OHP 4030.STP.050E(W), requires flow to be throttled by declutching and manually throttling a Generic Letter 89-10 program motor operated valve

b. Issues and Findings

There were no findings identified and documented during this inspection.

1R23 Temporary Modifications

a. Inspection Scope

The inspectors screened active temporary modifications on systems which were ranked high in risk in order to assess their effect on the safety functions of important safety systems. As part of this effort, the inspectors reviewed the following documentation:

- Open Temporary Modification Log - Unit 2
- Open Temporary Modification Log - Unit 1 and Common

b. Issues and Findings

There were no findings identified and documented during this inspection.

4. OTHER ACTIVITIES (OA)

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 50-315/95006-01: Containment Type B and C leakrate exceeds LCO value due to leakage of postaccident sample line check valve and ice condense glycol header isolation valve pressure relief check valve. On August 25, 1995, the licensee determined that the accumulated leakage of the type B and C leak rate testing exceeded the TS 3.6.1.2. The licensee identified that two valves, 1-NS-357 and 1-R-137, had exceeded their allowable leakage rates. Technical Specification 3.6.1.2 required, in part, that containment leakage rates shall be limited to a combined leakage rate of less than or equal to 60 percent of the maximum allowable leakage rate, 0.60 L_a, for all penetrations and valves subject to Types B and C tests when pressurized to peak accident pressure, P_a. The licensee tested the isolation valves located in series with the identified leaking valves and found that no significant leakage pathway existed. In accordance with the TS 3.6.1.2 action statement, both of the leaking valves were replaced, and the combined leakage rate for all penetrations subject to Type B and C testing was verified to be less than 0.60 L_a prior to raising RCS temperature above 200°F.

Based on the licensee's actions following the identification of the leaking isolation valves, the inspectors concluded that this event did not constitute a violation of NRC requirements. The inspectors also concluded that the failure of a containment isolation valve could have a credible impact on safety. However, the isolation valves located in series with the identified leaking valves prevented an actual open pathway in the physical integrity of the reactor containment. Therefore, this issue was screened as GREEN (very low risk significance) after a Phase 1 Significance Determination Process review. This LER is closed.

b.2 (Closed) Licensee Event Report 50-315/00006-00: Emergency diesel fuel oil Technical Specification surveillances not met. This LER was a minor issue and was closed.

b.3 (Closed) Licensee Event Report 50-316/00012-00: Failure to perform increased frequency surveillance on Unit 2 East CTS pump. This LER was a minor issue and was closed.

4OA4 Cross-Cutting Issues

.1 Licensee Identified Human Performance Issues

In inspection report 50-315/316-2000019, licensee- and NRC-identified human performance issues and corrective actions were discussed. During this inspection period the licensee continued to identify human performance issues. Examples included:

- On September 5, 2000, a boric acid transfer pump was operated for approximately 30 minutes without a suction flow path.

- On September 7, 2000, a wrong unit error resulted in a Unit 2 local shutdown indication panel being de-energized. A senior reactor operator de-energized the Unit 2 panel while intending to de-energize a Unit 1 panel.

- On September 11, 2000, a clearance order was released prior to all the work being completed.

- On September 14, 2000, an inadequate clearance order was placed on the Unit 1 "B" train diesel generator air compressors. Craft personnel were working on the air compressor discharge piping while the compressors were capable of automatically starting.

- On September 18, 2000, a contract radiation protection technician left the Unit 2 upper containment outer airlock door open and unattended. The configuration problem was unnoticed for approximately 7 hours, even though an annunciator was lit in the control room.

The human performance root cause team investigating previous events added the new events to their evaluation as appropriate. Additional focus by the licensee on human performance issues continued.

4OA5 Other

.1 Inspection Manual Chapter 0350 Activities for Unit 1 Restart

(Open) Restart Action Matrix Item 2.1, Evaluate Modifications to the Unit 1 Recirculation Sump; and Staff Guidelines for Restart Approval (SGRA) Item C.4.a, Operability of Technical Specification Systems: The inspectors reviewed and observed portions of the licensee's installation of Design Change Package (DCP) 1-DCP-0728, Revision 0, "Modification to Containment Flood Up Overflow Wall." Modification 1-DCP-0728 installed five holes in the containment flood up overflow wall in order to resolve one of the dead ended compartment issues that were identified by the NRC Architect Engineering Team in August of 1997.

Confirmatory Action Letter Item 1, Recirculation Sump Inventory/Containment Dead Ended Compartments Issue; and Unit 2 Restart Action Matrix Item R.2.2.4, URI 50-315/316/97017-05, "The As-Found Condition of the Containment Recirculation

Sump Relative to Technical Specification Operability During Modes 1, 2, and 3;" and Unit 2 RAM Item R.2.3.26, EEI 50-315/316/98009-06, “Apparent Failure to Demonstrate, Using Design Basis Documentation, That There Was Adequate Containment Recirculation Sump Water Volume Following a LOCA,” addressed the regulatory and safety aspects of this issue.

The inspectors determined that the modification was performed in accordance with the design change package and field change notices. This RAM Item and SGRA Item remain open pending additional inspection efforts.

(Open) SGRA Item C.4.f, Significant Hardware Issues Resolved: The licensee identified that, after several surveillance intervals, the vibrations measured on the Unit 2 East CTS pump motor exceeded the action levels recorded in TDB Figure 2-15.2. To correct the high vibrations, the licensee developed temporary modifications (TMs) for both Unit 2 CTS pumps which were designed to dampen the pump vibrations. After installing the TM on the Unit 2 West CTS pump, the measured vibrations had been reduced to below the alert level.

The inspectors reviewed the Unit 2 West CTS pump TM as it related to resolution of significant hardware issues. The licensee identified that the Unit 1 CTS pumps were also exhibiting high vibrations similar to the Unit 2 CTS pumps. Based on the pump vibration results following the TM installation on Unit 2, the licensee decided to install a similar TM on the Unit 1 CTS pumps. Item C.4.f remains open pending the inspectors' review of other Unit 1 hardware issues.

(Open) SGRA Item C.4.h, Effectiveness of the Plant Maintenance Program: The inspectors reviewed the results of Maintenance Rule implementation baseline inspections for the spent fuel pool cooling system and the Unit 1 CVCS as they related to effectiveness of the plant maintenance program. The inspectors identified three non-cited violations of the requirements of 10 CFR 50.65. As discussed in section 1R12, these violations involved the failure to establish goals for systems monitored under paragraph (a)(1); the failure to place the 250 VDC under monitoring per the requirements of paragraph (a)(1) when the preventative maintenance program failed to effectively control system performance; and the failure to monitor unavailability of structures, systems, and components required to be functional during the extended dual unit outage. With the exception of the violation associated with the 250 VDC system, these violations do not directly impact the quality of the licensee's maintenance program, but indicate that Maintenance Rule may not have provided timely feedback on the effectiveness of plant maintenance. The failure of preventative maintenance program to control 250 VDC system performance was evaluated using the significance determination process and determined to be of very low risk significance. Item C.4.h remains open pending the inspectors' review of other maintenance program issues.

.2 Draindown During Shutdown and Common-Mode Failure (Temporary Instruction 2515/142)

a. Inspection Scope

In 1998, the NRC issued Generic Letter (GL) 98-02, "Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition." In their response to the GL, the licensee documented that the plant was susceptible to a loss of reactor coolant inventory while shut down. However, the licensee's response also documented that the plant processes and procedures would prevent such an event. The inspectors discussed the issue with members of the licensee's operating, engineering, and licensing departments. In addition, the following documents were reviewed:

- Generic Letter 98-02, "Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition."
- Indiana Michigan Power Company to NRC letter C0700-01, July 28, 2000, "Response to Generic Letter 98-02, 'Loss of Reactor Coolant Inventory and Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition.'"
- 02-OHP 4030.STP.035, "Controlled Valve Position Logging," Revision 17
- 02-OHP 4030.STP.050W, "West Residual Heat Removal Train Operability Test Modes 1-4," Revision 8
- PMI 4090, "Criteria for Conducting Infrequently Performed Tests or Evolutions," Revision 6
- PMP 2291.PLN.001, "Work Management Activity Planning Process," Revision 3
- PMP 4100.SDR.001, "Plant Shutdown Safety and Risk Management," Revision 4
- PMP 4043.SLV.001, "Sealed/Locked Valves," Revision 1a
- CR 00-9786, NRC issued supplement 2 to Information Notice 95-03
- CR 99-8319, Failure to assess potential loss of emergency mitigation function in accordance with GL 98-02
- OP 2-5143, "Flow Diagram, Emergency Core Cooling (RHR)"
- OP 2-5144, "Flow Diagram, Containment Spray"

b. Issues and Findings

There were no findings identified and documented during this inspection. This Temporary Instruction is closed.

.3 (Closed) Inspection Follow-up Item 50-315/316/94020-02: Review of Reactor Vessel Head Penetration Inspections.

In 1994, in response to an industry concern, the licensee performed an eddy current inspection of the Unit 2 reactor head penetrations used for control rod drive mechanism (CRDM) and in-core thermocouple (TC) insertion. Of the 71 penetrations inspected, the licensee identified several small cracks in in-core thermocouple penetration number 75. The licensee initiated a Westinghouse evaluation and Westinghouse Report WCAP-14118 "Structural Integrity Evaluation of Reactor Vessel Upper Head

Penetrations to Support Continued Operation," documented calculations which supported continued operation with cracks for 6 years. The evaluation concluded that within the 6 year time period, the reactor coolant pressure boundary would not be breached and no reactor coolant system (RCS) leakage would result from the identified cracks. The licensee committed to reinspect the penetration during the next refueling outage. NRC headquarters personnel reviewed the licensee's evaluation and did not have any further concerns. NRC review of future reactor vessel head penetration inspections was documented as an inspection follow-up item.

Inspector follow-up found that in 1995, a Westinghouse probabilistic assessment was performed for the 78 Cook reactor vessel head penetrations. The results identified five outer reactor vessel head penetrations as the most susceptible to cracking, and in 1996, the five outer reactor vessel head penetrations including penetration 75 were reinspected. No additional indications were identified in the five outer penetrations and reinspection of penetration 75 identified no significant flaw growth. The cracks in Penetration 75 were subsequently repaired after the reinspection as a long term solution to the problem.

Since 1996, the licensee has monitored the reactor pressure head penetrations via the following:

- Reactor head penetration leakage inspection (VT-2) set forth in procedure 12-QXP-5070-NDE.001, "RPS System Leakage Test," as required by ISI and ASME Section XI requirements.

- Liquid Penetrant inspection of the reactor pressure vessel head penetrations' bi-metallic welds as required by the ISI requirements of ASME Code Section XI table IWB-2500-1 examination category B-O.

Visual examinations conducted as part of RCS leakage tests have not identified any RPV head penetration leakage, and the liquid penetrant inspections of the penetrations' bi-metallic welds during the first and second ten year ISI intervals did not identify any degradation.

In 1997 and 1999, the licensee submitted information to the NRC concerning Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," and stated that Cook was monitoring the Westinghouse Owners Group\Industry resolution associated with this active corrosion mechanism problem and that subsequent inspections would be based on evaluation of those findings. However, the inspectors noted that there did not appear to be a method to track and implement the long range solution at Cook. The licensee subsequently initiated CR 00-7527 to track industry resolution of the long range activities associated with the reactor vessel CDRM tube cracking issue, and to evaluate the need and schedule for subsequent inspections based on those results. This item is considered closed.

4OA6 Management Meetings

The inspectors presented the inspection results to Mr. Rencheck and other members of licensee management listed below at the conclusion of the inspection on September 28, 2000. The licensee acknowledged the findings presented. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

M. Barfelz, Regulatory Affairs
T. Foster, Performance Assurance
R. Gaston, Regulatory Affairs
J. Gebbie, Plant Engineering
M. Hoskins, System Engineering
W. Kropp, Regulatory Affairs
R. Meister, Regulatory Affairs
J. Molden, Director, Maintenance
D. Naughton, System Engineering
S. Partin, Assistant Operations Manager
M. Rencheck, Vice-President Engineering
L. Weber, Manager, Operations
D. Wood, Manager, Radiological Protection, Chemistry, and Environmental

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
71111-06	Flood Protection	1R06
71111-07	Heat Sink Performance	1R07
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Risk Assessment and Emergent Work Evaluation	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
71153	Event Followup	4OA3
TI 2515/142	Draindown During Shutdown and Common-Mode Failure	4OA3

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315-00-20-01 50-316-00-20-01	NCV	Failure to monitor the unavailability of 46 systems required during shutdown mode operation (Section 1R12.1)
50-315-00-20-02	NCV	Failure to establish performance goals for Maintenance Rule (a)(1) system (Section 1R12.2)
50-315-00-20-03 50-316-00-20-03	NCV	Failure to place Unit 2 250 VDC in Maintenance Rule category (a)(1) after multiple maintenance preventible functional failures (Section 1R12.3)

Closed

50-315/94020-02 50-316/94020-02	IFI	Review of Reactor Vessel Head Penetration Inspections
50-315/95006-01	LER	Containment type B and C leakrate exceeds LCO value due to leakage of post accident sample line check valve and ice condense glycol header isolation valve pressure relief check valve (Section 4OA3.1)
50-315/00006-00	LER	Emergency diesel fuel oil Technical Specification surveillances not met verbatim (Section 4OA3.1)
50-316/0012-00	LER	Failure to perform increased frequency surveillance on Unit 2 east containment spray pump (Section 4OA3.1)
50-316/00-19-01	URI	Potential mis-classification of a Maintenance Rule maintenance preventable functional failure (Section 1R12.3)
50-315-00-20-01 50-316-00-20-01	NCV	Failure to monitor the unavailability of 46 systems required during shutdown mode operation (Section 1R12.1)
50-315-00-20-02	NCV	Failure to establish performance goals for Maintenance Rule (a)(1) system (Section 1R12.2)
50-315-00-20-03 50-316-00-20-03	NCV	Failure to place Unit 2 250 VDC in Maintenance Rule category (a)(1) after multiple maintenance preventible functional failures (Section 1R12.3)

Discussed

None

LIST OF ABBREVIATIONS

AES	Engineered Safety Features Ventilation
AFW	Auxiliary Feedwater System
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CTS	Containment Spray
CVCS	Chemical and Volume Control System
DCP	Design Change Package
D/G	Diesel Generator
DIT	Design Information Transmittal
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EHI	Engineering Head Instruction
EHP	Engineering Head Procedure
ESF	Engineered Safety Features
ESW	Essential Service Water
FMEZ	Foreign Material Exclusion Zone
GL	Generic Letter
HELB	High Energy Line Break
JO	Job Order
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
MC	Manual Chapter
MHP	Maintenance Head Procedure
MOV	Motor Operated Valve
MPFF	Maintenance Preventable Functional Failure
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OHI	Operations Head Instruction
OHP	Operations Head Procedure
OSO	Operations Standing Order
PDR	Public Document Room
PMI	Plant Manager’s Instruction
PMP	Plant Manager’s Procedure
PMT	Post-maintenance Testing
PORV	Power Operated Relief Valve
PPC	Plant Process Computer
RAM	Restart Action Matrix
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SFP	Spent Fuel Pool
SGRA	Staff Guidelines for Restart Approval
SRO	Senior Reactor Operator

SSC	Structures, Systems, and Components
STP	Surveillance Test Procedure
TBD	To Be Determined
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TDB	Technical Data Book
TM	Temporary Modification
TS	Technical Specification
URI	Unresolved Item
UFSAR	Updated Final Safety Analysis
VIO	Violation