January 12, 2001

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

SUBJECT: D. C. COOK INSPECTION REPORT 50-315/00-25(DRP); 50-316/00-25(DRP)

Dear Mr. Powers:

On December 31, 2000, the NRC completed a baseline inspection at your D. C. Cook Units 1 and 2 reactor facility. The inspection results were discussed on January 3, 2001, with the Plant Manager and other members of your staff.

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

No findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

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

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 50-315/00-25(DRP); 50-316/00-25(DRP)

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REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/00-25(DRP); 50-316/00-25(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:	November 12, 2000, through December 31, 2000
Inspectors:	B. L. Bartlett, Senior Resident Inspector K. A. Coyne, Resident Inspector J. A. Lennartz, Senior Resident Inspector, Palisades J. D. Maynen, Resident Inspector D. Passehl, Project Engineer, Region III
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 05000315-00-25, IR 05000316-00-25, on 11/12-12/31/2000, Indiana Michigan Power Company, D. C. Cook Nuclear Plant Units 1 & 2. Resident inspector report.

The report covered a 6 week period of resident inspection.

No findings of significance were identified.

Report Details

Summary of Plant Status:

At the beginning of this inspection period, Unit 1 was defueled. On November 17, 2000, the licensee began refueling Unit 1. At 1:55 a.m. on December 18, 2000, the licensee entered Mode 2 (Startup), and the reactor was made critical at 5:00 a.m. on December 18, 2000. At the end of the inspection period, Unit 1 was operating at approximately 86 percent power and continuing with power ascension activities.

Unit 2 operated at full power throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

.1 Partial System Walkdown of the 250 VDC System (Unit 1)

a. Inspection Scope

The inspectors conducted a partial system walkdown of the Unit 1 250 VDC distribution system. At the time of the walkdown, Unit 1 was in Mode 5 (Cold Shutdown) and one train of 250 VDC distribution was required to be operable in accordance with Technical Specification (TS) 3.8.2.4, "D.C. Distribution - Shutdown." The 250 VDC system provided indication and control power to enable operators to maintain the plant in a safe shutdown condition and therefore was considered to be a mitigating system. The areas inspected included the 4kV switchgear room, the control room instrumentation distribution inverter room, and the Train "A" and "B" battery rooms. The inspectors assessed general equipment material condition, equipment alignment, and housekeeping conditions. The inspectors also verified that transient material was controlled and appropriately stored in the vicinity of system components. The inspectors reviewed the following documents:

- 01-OHP [Operations Head Procedure] 4021.082.006, "Operation of 1AB & 1CD Battery Chargers," Revision 8
- 01-OHP 4030.STP.030, "Daily and Shiftly Surveillance Checks," Revision 32A
- 01-OHP 5030.001.001, "Operations Plant Tours," Revision 17
- Drawing OP-1-98055, "250 VDC Battery 'AB' Distribution Schematic," Revision 19
- Drawing OP-1-98057, "250 VDC Battery 'CD' Distribution Schematic," Revision 14
- Drawing OP-1-12003, "250 VDC Main One Line Diagram Engineered Safety System," Revision 27

b. Issues and Findings

No findings of significance were identified.

.2 Partial System Walkdown of the Chemical and Volume Control System (Unit 1)

a. Inspection Scope

The inspectors conducted a partial system walkdown of the Unit 1 boration and chemical and volume control system (CVCS) makeup charging path. At the time of the walkdown, Unit 1 was in Mode 5 and required one operable boration flowpath in accordance with TS 3.1.2.1, "Boration Systems Flow Paths - Shutdown," and TS 3.1.2.3, "Charging Pump - Shutdown." Because the CVCS system was relied upon to provide reactor coolant system (RCS) inventory makeup and boration, the inspectors considered the CVCS to be a mitigating system. The inspectors compared the system alignment against procedural requirements, assessed material condition, and observed housekeeping practices. The inspectors reviewed the following documents:

- 01-OHP 4021.003.001, "Letdown, Charging, And Seal Water Operation," Revision 24B
- 01-OHP 4021.005.007,"Operation of Emergency Boration Flow Paths," Revision 3
- 12-OHP 4021.005.001, "Boron Makeup System Operation," Revision 22
- Unit 1 Caution Tag Log
- Unit 1 Abnormal Position Log
- Drawing OP-1-5129, "Flow Diagram CVCS-Reactor Letdown & Charging Unit No 1," Revision 41
- Drawing OP-1-5129A, "Flow Diagram CVCS-Reactor Letdown & Charging," Revision 28
- Drawing OP-1-2-5130, "Flow Diagram CVCS-Boration Make-Up Units No. 1 & 2," Revision 40
- Condition Report (CR) 00333095, NRC questioned the lack of specific guidance in positioning manual valves in the boration flowpath valve lineup procedure

b. Issues and Findings

No findings of significance were identified.

.3 Partial System Walkdown of Control Air and Condensate Systems (Unit 1)

a. Inspection Scope

The inspectors performed partial walkdowns of the Unit 1 control air system and the Unit 1 condensate systems. The walkdowns were conducted in order to support an assessment of secondary system readiness for restart in accordance with Unit 1 Restart Action Matrix Item C.4.b, "Operability of Required Secondary Support Systems." Closeout of Item C.4.b was documented in NRC Inspection Report 50-315/00-23; 50-316/00-23.

Because the condensate system was required to support decay heat removal in Mode 4 (Hot Shutdown) and Mode 3 (Hot Standby), and the loss of the condensate or control air systems at power could result in a plant transient, the inspectors considered that these systems were within the initiating events cornerstone. The inspectors compared the systems' alignment against procedural requirements, assessed material condition, and observed housekeeping practices. The inspectors reviewed the following documents:

- 01-OHP 4021.054.001, "Operation of the Condensate System," Revision 10
- 01-OHP 5030.001.001, "Operations Plant Tours," Revision 16a
- Drawing No. OP-1-5120, Flow Diagram Compressed Air System (Key Plan)
- Drawing No. OP-1-5120A, Flow Diagram Compressed Air System Plant Air Turbine Room
- Drawing No. OP-1-5120B, Flow Diagram Compressed Air System Plant Air Auxiliary Building and Control Air for Containment
- Drawing No. OP-1-5120C, Flow Diagram Compressed Air System (Arrangement of Control Air Equipment Unit No. 1)
- Drawing OP-1-5107, Flow Diagram Condensate Unit No. 1
- Drawing OP-1-5107A, Flow Diagram Condensate Unit No. 1
- Job Order (JO) 00242134, Pre-startup flush of Condensate, Feedwater, Turbine Auxiliary Cooling, Main Feed Pump Seal Water, and Heater Drains and Vents Systems
- Expanded System Readiness Review System Test Plan, Control Air and Containment Control Air Header, Revision 1, dated October 27, 2000
- CR 00334027, NRC identified that Unit 1 Plant Air Compressor Non-Essential Service Water Test Line Isolation valves were open, contrary to normal operating procedure valve lineup
- CR 00335043, During followup to NRC questions, identified that test gages installed on Unit 1 Plant Air Compressor to monitor Non-Essential Service Water pressure were not in compliance with temporary modification requirements

b. Issues and Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed fire protection walkdowns of the following risk-significant plant areas: the Unit 1 CD Diesel Generator Room (Fire Zone 15), the Unit 1 AB Diesel Generator Room (Fire Zone 16), the Unit 2 "A" Train and "B" Train 4kV switchgear rooms (Fire Zones 47A and 47B), and the Unit 2 600V switchgear mezzanine area (Fire Zone 45). 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:

- 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
- 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
- CR 01002010, Fire protection procedures should be clarified to ensure conformance with Fire Hazards Analysis

b. Issues and Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors reviewed the operator training developed and conducted to support the restart of Unit 1. The inspectors noted that in addition to the normal training cycle topics, the training department developed additional “just-in-time” training which was specific to Unit 1 operation. The purpose of the additional training was to make the operators aware of the differences between Unit 1 and Unit 2. The inspectors reviewed the training documents to verify that specific differences between Unit 1 and Unit 2 operation, particularly with regard to differences in the Emergency Operating Procedures, were covered prior to Unit 1 restart.

This inspection also supported the closure of NRC Restart Action Matrix Item C.3.3.d, “Effectiveness of Restart Simulator/Required Training Necessary to Re-Familiarize Personnel With Operating.” Closure of Item C.3.3.d was documented in NRC Inspection Report 50-315/00-23; 50-316/00-23. The inspectors discussed the Unit 1 startup training with members of Operations management and several licensed operators. The inspectors reviewed the following training documents:

- Requalification Training (RQ)-F-2561, “Unit 1 Cycle 17 Core Design Review,” Revision 0
- RQ-R-2550, “Operational & TS Review,” Revision 0
- RQ-R-2551, “Design Change Read-It,” Revision 0
- RQ-R-2561, “Period 2506 Read-It Package,” Revision 0
- RQ-R-2562, “Period 2506 Procedure and Issues Review,” Revision 0
- RQ-R-2563, “Unit Differences Read-It,” Revision 0
- RQ-R-2564, “Unit 1 Steam Generators,” Revision 0
- RQ-R-2565, “Unit 1 Main Turbine Controls,” Revision 0
- Performance Assurance Field Observation FO-00-K-111, “Unit 1 Startup Just-In-Time Training and Simulator Observation”

b. Issues and Findings

No findings of significance were identified.

1R13 Maintenance and Emergent Work Control

.1 Unit 2 Containment Spray System Valve Leakby

a. Inspection Scope

On December 14, 2000, the licensee entered the 72 hour action statements of TS 3.6.2.1 and 3.6.2.2 on the containment spray (CTS) system and the containment spray additive system to support routine surveillance testing. During the test, the licensee concluded that the CTS additive tank to the west CTS pump suction check valve, 2-CTS-120W, failed to meet the back leakage acceptance criteria. Back leakage through this valve was significant in that, during certain postulated accidents, higher dose rates could occur near these valves which would hinder access to the Post Accident Sampling System. The licensee wrote Condition Report 00350036 to document the leakby and develop a plan to repair the valve.

Check valve 2-CTS-120W was disassembled and inspected, but no cause for the back leakage could be determined. Based on the spray additive system design and the test methodology, the licensee determined that back leakage through the east train check valve, 2-CTS-120E, could appear as back leakage through the west train. The licensee then developed a plan to inspect and repair the east train check valve. Throughout the completion of both work plans, the licensee remained in the TS 72 hour action statements for the CTS system and the spray additive system. The inspectors reviewed the licensee's work planning and risk analysis for the emergent work on both CTS check valves. The following documents were reviewed as part of this inspection:

- Unit 2 TS 3.6.2.1, Containment Spray System
- Unit 2 TS 3.6.2.2, Spray Additive System
- 12-MHP [Maintenance Head Procedure] 5021.001.040, "Aloyco Self Actuated Swing Check Valve Maintenance," Revision 4a
- 01-OHP 4030.STP.007W, "West Containment Spray System Operability Test," Revision 12
- PMP 2291.OLR.001, "On-Line Risk Management," Revision 0
- PMP 2291.OLR.001, Data Sheet 1 for week ending December 16, 2000
- PMP 2291.OLR.001, Data Sheet 1 for week ending December 23, 2000
- Drawing OP-2-5144, " Flow Diagram: Containment Spray, Unit 2"
- JO C350036, Repair leakby on 2-CTS-120W
- JO C351011, Disassemble and repair or replace 2-CTS-120E
- JO R208583, Perform 01-OHP 4030.STP.007W, West CTS operability test
- CR 00350036, 2-CTS-120W failed leakage test
- CR 00351007, 2-CTS-119E, spray additive tank isolation valve to 2W CTS pump, reach rod needs adjustment
- CR 00351008, 2-CTS-120E is leaking by
- CR 00351011, Leakby of 2-CTS-120E may have been cause of 2-CTS-120W test failure

b. Issue and Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 Operability Review for Steam Generator Overfill During Postulated Steam Generator Tube Rupture

a. Inspection Scope

Following the replacement of the Unit 1 steam generators at D. C. Cook, the licensee determined that operational and design changes were required to the auxiliary feedwater (AFW) system to ensure that design basis requirements were met. In particular, because the replacement steam generators had a smaller secondary side volume than the original steam generators, the maximum available AFW flow delivered to the generators during a postulated steam generator tube rupture event needed to be reduced in order to prevent steam generator overfill. Consequently, the licensee changed the normal standby position of the turbine driven auxiliary feedwater pump (TDAFWP) discharge valves to each steam generator (1-FMO-211, -221, -231, and -241) from fully open to an intermediate throttled position. On December 13, 2000, while in Mode 3, the licensee performed AFW system flow testing and determined that the flow rate supplied to three steam generators was less than the required amount. The licensee performed an operability determination and determined that the flow rates were sufficient to support AFW system operability up to a maximum power level of 83 percent rated thermal power. The inspectors reviewed this operability determination, documented in CR 00300052, and a related operability determination contained in CR 99-7284 associated with steam generator overfill during a steam generator tube rupture. The inspectors reviewed the following documents:

- 01-OHP 4030.STP.017CS, “Main and Auxiliary Feedwater System Shutdown Testing,” Revision 8
- Calculation TH-00-06, D. C. Cook Unit 1 Steam Generator Tube Rupture with Operator Actions
- Calculation MD 12-AFW-001-N, AFW System Analysis for Loss of AC and Main Steam Line Break
- Calculation MD 01-AFW-041-N, “Turbine Driven Auxiliary Feedwater Pump FMO Valve Position Determination”
- Calculation MD 01-AFW-042-N, “Determination of Acceptance Criteria for Functional Test to Verify TDAFW FMO Valve Position”
- Calculation MD 01-AFW-043-N, “Calibration of D. C. Cook Unit 1 AFW System FMO Valve Curves”
- Calculation MD 01-AFW-044-N, “Turbine Driven Auxiliary Feedwater Pump Discharge Valves 1-FMO-211, -221, -231, and -241 Position Settings”
- DIT B-01872, Accuracy of AFW flow as read at the output of 1-FFI-210, -220, -230, and -240
- DIT B-01905, Expanded “Correct” Valve Positions for FMO-211, -221, -231, and -241 for Procedure 01 OHP 4030.STP.017CS
- CR 99-7284, Steam Generator Overfill following Tube Rupture

- CR 00300052, Operability determination evaluation (ODE) for the Unit 1 steam generator tube rupture overfill issue
- CR 00336086, Operability determination is needed to support TDAFW discharge valve test for 1-DCP-4894
- CR 00351014, NRC identified that engineering design information supporting operability determination in CR 00300052 contained non-conservative errors

b. Issues and Findings

In order to support ODE documented in CR 00300052, Design Engineering personnel provided an expanded AFW flow rate acceptance criteria in Design Input Transmittal (DIT) B-01905-00. The purpose of this DIT was to extrapolate the results from the formal engineering calculation, 01-MD-AFW-044-N, which assumed nominal Mode 1 conditions, to the actual Mode 3 test conditions. Because the measured AFW flows were greater than the minimum flows specified in DIT B-01905, the licensee concluded that the AFW system could be considered operable up to a maximum thermal power level of 83 percent.

The inspectors reviewed the information and identified errors in the minimum AFW flow rate acceptance criteria documented in the DIT B-01905. Licensee personnel acknowledged the inspectors finding, and initiated CR 00351014. The licensee reissued DIT B-01905 with corrected AFW flow rate acceptance criteria, and revised the CR 00300052 AFW operability determination. Because sufficient margin was available, the error did not affect the original operability conclusion of CR 00300052. The inspectors determined that the errors should have been identified during the DIT review and approval process.

.2 Review of Reactor Coolant System Leakage Detection Capability of Lower Containment Radiation Monitoring System Particulate Detectors

a. Inspection Scope

The inspectors reviewed the engineering evaluation performed to verify that the Radiation Monitoring System (RMS) lower containment particulate detectors were capable of detecting an 0.8 gpm RCS system leak within one hour. On November 8, 2000, the NRC staff approved the licensee’s use of a leak-before-break evaluation to remove consideration of the dynamic effects associated with the postulated rupture of the pressurizer surge line piping. The NRC staff’s approval for this methodology was based, in part, upon the licensee’s demonstration that the leakage detection system was capable of detecting 0.8 gpm of RCS system leakage within one hour. The inspectors reviewed the licensee’s leakage detection engineering evaluation. During this review, the inspectors verified analysis input assumptions and methodology, discussed operation of the RMS particulate detectors with radiation protection and operations personnel, and walked down portions of the containment lower compartment particulate radiation monitors. The inspectors reviewed the following documents during this review:

- 12-OHP 4021.013.006, “Operation of the Eberline Radiation Monitoring System Control Terminal,” Revision 4a

- 12-OHP 4024.139, “Annunciator #139 Response: Eberline Radiation,” Revision 8A
- 12-THP [Technical Head Procedure] 6010.RPI.805, “Radiation Monitoring System Setpoints,” Revision 10A
- UFSAR Chapter 5, Reactor Coolant System
- UFSAR Chapter 5, Containment System
- Administrative Technical Requirement (ATR) 1-RCS-2, “Reactor Coolant System - Leakage Detection Systems”
- ATR 1-RCS-3, “Reactor Coolant System - Operational Leakage”
- EVAL-RD-00-004, Evaluation of Radiation Monitoring System Leak Detection Capability
- Calculation RD-00-12, Determination of Radiation Monitoring System Particulate Channel Leak Detection Capability
- NRC Letter to Mr. Robert Powers, “Donald C. Cook Nuclear Plant, Units 1 and 2 - Review of Leak-Before-Break for the Pressurizer Surge Line Piping as Provided by 10 CFR 50, Appendix A, GDC 4,” dated November 8, 2000
- CR 98-5131, Potential discrepancies in statements in UFSAR Section 4.2, System Design and Operation
- CR 00-2911, Calculation RS-C-0032 unable to accurately verify conclusion statement that RCS leakage of 1 gpm over 4 hour period can be detected by lower containment radiation monitor
- CR 00350124, NRC identified that assumptions in the leak before break detection analysis, including the particulate radiation monitor setpoint, did not reflect actual plant conditions. The non-conservatively established radiation monitor setpoint would have resulted in a longer time to detect 0.8 gpm RCS leakage.

b. Issue and Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the licensee implementation of permanent plant modifications for the emergency diesel generators and AFW pump rooms. These modifications were developed to address operability and design basis issues. The diesel generator modification replaced the diesel generator air system with safety-related and seismically qualified air compressors and piping. The AFW pump rooms were sealed to provide the AFW pumps high energy line break protection during postulated steam line break accidents and room coolers were also installed in each AFW pump room. Because of the accident mitigation functions provided by the emergency diesel generators during loss of offsite power events and the auxiliary feedwater system in providing steam generator secondary water makeup, the inspectors determined that these modifications were associated with the mitigating systems cornerstone. The inspectors reviewed the associated design change, supporting drawings, walked down portions of the modifications to verify consistency of the installed configuration with design, and reviewed completed test data. During this inspection, the inspector reviewed the following documents:

- 1-DCP-548, Unit 1 CD Diesel Generator Starting Air Compressor Replacement
- 01-DCP-548-TP.1, “DG1CD Starting Air Compressor Test - 1-QT-142-CD1,” Revision 0
- 01-DCP-548-TP.2, “DG1CD Starting Air Compressor Test - 1-QT-142-CD2,” Revision 0
- 1-DCP-4595, Modification of Auxiliary Feedwater Pump Rooms Ventilation System
- 01-DCP-4595-TP.1, “East Motor Driven Aux Feedwater Pump Room Cooler,” Revision 0
- 01-DCP-4595-TP.4, “Turbine Driven Aux Feedwater Pump Room Cooler T2AC,” Revision 0
- 01-OHP 4022.019.001, “ESW System Loss/Rupture,” Revision 2
- 01-OHP 4024.114, “Annunciator #114 Response: Steam Generator 3 and 4,” Revision 6
- 01-OHP 4024.113, “Annunciator #113 Response: Steam Generator 1 and 2,” Revision 6
- 12-PMP 4030.001.001, “Impact of Safety Related Ventilation on the Operability of TS Equipment,” Revision 3
- Letter from Indiana and Michigan Power to NRC, “Donald C. Cook Nuclear Plant Units 1 and 2, License Amendment Request, Modifications to Auxiliary Feedwater Pump Room Cooling,” dated February 18, 2000 (ADAMS Ascension No. ML003685926)
- Letter from John Stang, NRC Senior Project Manager, to Robert P. Powers, Issuance of Amendment 244 to DPR-58 and Amendment 225 to DPR 74, dated April 25, 2000 (ADAMS Ascension No. ML003710132)

b. Issues and Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

.1 Flow Testing of Unit 1 Turbine Driven Auxiliary Feedwater Pump Discharge Valves

a. Inspection Scope

During the extended outage, the licensee replaced all of the Unit 1 steam generators with new steam generators. Due to the lower secondary side internal volume of the replacement steam generators, the available margin to steam generator overfill during a postulated steam generator tube rupture was decreased. In order to limit the possibility of a steam generator overfill, the licensee requested, and was issued, an amendment to TS 3.7.4.1 to change the required position of the TDAFWP discharge valves from fully open to a throttled position. To implement the TS amendment, the licensee installed 1-DCP-4894, which removed the signal to automatically open the TDAFWP discharge valves and positioned these valves from fully open to throttled. The inspectors reviewed the design change package, supporting job orders, walked down the installed plant configuration changes to verify consistency with design requirements, and reviewed supporting calculations and analyses. The inspectors reviewed the following documents:

- 01-DCP-4894-TP.1, “TDAFWP Discharge Flow Control Valve Position Test,” Revision 0a
- 01-DCP-4894-TP.2, “TDAFWP Discharge Flow Control Valve Position Test,” Revision 0
- Calculation MD 12-AFW-001-N, AFW System Analysis for Loss of AC and Main Steam Line Break
- Calculation MD 01-AFW-041-N, “Turbine Driven Auxiliary Feedwater Pump FMO Valve Position Determination”
- Calculation MD 01-AFW-042-N, “Determination of Acceptance Criteria for Functional Test to Verify TDAFW FMO Valve Position”
- Calculation MD 01-AFW-043-N, “Calibration of D. C. Cook Unit 1 AFW System FMO Valve Curves”
- Calculation MD 01-AFW-044-N, “Turbine Driven Auxiliary Feedwater Pump Discharge Valves 1-FMO-211, -221, -231, and -241 Position Settings”
- Letter from John Stang, NRC Senior Project Manager, to Robert P. Powers, Issuance of Amendment No. 250 to DPR-58 and Amendment 231 to DPR-74, dated November 30, 2000 (ADAMS Ascension No. ML003773534)
- JO 00315067, DCP-4894 Changes “Standby Readiness” of 1-FMO-241
- JO 00315059, DCP-4894 Changes “Standby Readiness” of 1-FMO-211
- JO 00315065, DCP-4894 Changes “Standby Readiness” of 1-FMO-231
- JO 00315064, DCP-4894 Changes “Standby Readiness” of 1-FMO-221
- CR 00359015, As found TDAFWP flow through 1-FMO-211,221,231, and 241 below acceptance criteria with 1-FMO-221 being downscale
- CR 00359023, Wrong oil was added to the inboard turbine bearing of the Unit 1 TDAFWP
- CR 00360003, Procedural issue with shiftly surveillance on the Unit 1 TDAFWP discharge valves
- CR 00360004, Unit 1 TDAFWP discharge pressure gauge pegged high after installation of 1-DCP-4894
- CR 00336086, NRC and licensee identified that an operability determination is required to establish AFW system operability in Modes 1, 2, and 3 prior to performing full flow AFW testing in Mode 1.
- CR 00337023, NRC identified that auxiliary feedwater system hydraulic model calibration was not adequately documented, consequently, independent review of calculated AFW discharge valve positions was not possible
- CR 00348060, NRC identified that post modification test 01-DCP-4894-TP.2 did not have adequate acceptance criteria for valve throttle position

b. Issues and Findings

No findings of significance were identified.

1R20 Refueling Outage

.1 Unit 1 Core Reload

a. Inspection Scope

On November 17, 2000, the licensee began reloading the Unit 1 reactor vessel. The inspectors reviewed the fuel handling operations and other ongoing activities to verify that they were performed in accordance with TSs and approved procedures.

This inspection also supported the closure of NRC Restart Action Matrix Item C.4.i, “Maintenance Backlog Managed and Impact on Operation Assessed.” Item C.4.i was closed in NRC Inspection Report 50-315/00-23; 50-316/00-23. The inspectors reviewed the following documents for this inspection:

- 12-OHP 4050.FHP.001, “Refueling Procedure Guidelines,” Revision 2
- 12-OHP 4050.FHP.005, “Core Unload/Reload and Incore Shuffle,” Revision 2
- CR 00319046, Aggregate operability determination for Mode 6 (Refueling)
- CR 00322054, Foreign material found in refueling cavity
- CR 00322088, Upender trips overboard light on conveyer cart on the SFP side
- CR 00324018, Debris found on new fuel assembly (lower grid strap of assembly JJ24)
- Performance Assurance Field Observation FO-00-K-105, Observation of Fuel Handling Activities in the Control Room

b. Observations and Findings

No findings of significance were identified.

.2 Unit 1 Ice Condenser and Containment Closeout

a. Inspection Scope

Prior to Unit 1 entry into Mode 4, the inspectors walked down Unit 1 upper and lower containment and the Unit 1 ice condenser to identify conditions which may have prevented these systems from performing their design basis functions. The inspectors specifically verified that transient or loose material which could block the recirculation sump was removed, and that the general material condition and housekeeping of the Unit 1 containment and ice condenser were adequate to support Unit 1 mode ascension into Mode 4. The inspectors were accompanied by members of the licensee’s radiation protection staff who documented the identified deficiencies in several condition reports. The inspectors reviewed the following procedures and documents:

- 12-MHP 4030.046.001, “Inspection of Access Doors Separating Containment Upper and Lower Volumes,” Revision 0
- 01-OHP 4030.001.002, “Containment Inspection Tours,” Revision 17
- PMP 4010.CAC.001, “Containment Access and Cleanliness,” Revision 0
- CR 00340073, During the NRC closeout inspection of lower ice condenser, several debris issues were identified

- CR 00341072, NRC inspection of lower containment volume identified several items
- CR 00341090, During the NRC walkdown of the Unit 1 upper containment, several items were found and either removed or have no direct impact on containment operability
- CR 00341091, During the NRC inspection of the Unit 1 upper ice condenser, several items were found

After reviewing the condition reports, the inspectors determined that the identified deficiencies were either corrected or evaluated to show that these items would not impact containment or ice condenser operability.

b. Issues and Findings

No findings of significance were identified.

.3 Unit 1 Restart Observations

a. Inspection Scope

On December 18, 2000, the licensee entered Mode 2 (Startup) on Unit 1 and the reactor was made critical at 5:00 a.m. on December 18, 2000. The inspectors observed the approach to criticality, low power physics testing, and main generator synchronization to the grid. Low power physics testing is discussed in Section 1R22, below. During these activities, the inspectors maintained continuous control room observations. The inspectors sampled various TSs, license conditions, and other requirements, commitments, and administrative procedure prerequisites for mode changes to verify that the requirements, commitments, and prerequisites were met prior to changing modes or plant configurations. The inspectors reviewed the following documents associated with the Unit 1 reactor startup:

- 12-EHP [Engineering Head Procedure] 6040.PER.352, "Rod Worth Verification Test Utilizing RCC Bank Interchange," Revision 4a
- 12-EHP 6040.PER.370, "Estimation of Critical Position," Revision 2
- Operations Head Instruction (OHI) 4000, "Conduct of Operations: Standards," Revision 2
- 01-OHP 4021.001.001, "Plant Heatup From Cold Shutdown to Hot Standby," Revision 28
- 01-OHP 4021.001.002, "Reactor Start-up," Revision 26
- 01-OHP 4021.050.001, "Turbine Generator Normal Startup and Operation," Revision 17
- 01-OHP 4030.STP.015, "Full Length Control Rod Operability Test," Revision 9
- 01-OHP 4030.STP.030, "Daily and Shiftly Surveillance Checks," Revision 32a
- PMP 4010.CRC.001, "Control Room Conduct," Revision 0
- PMP 4015.RMP.001, "Reactivity Management Program," Revision 0
- PMP 7200.RST.002, "Startup and Power Ascension," Revision 1
- CR 00352025, Unit 1 Steam Generator #1 Main Steam Stop Valve, 1-MRV-210, drifting off its open seat

- CR 00353005, Individual rod position indicator (IRPI) for control rod N-11 is low from demand by 14 steps
- CR 00353006, IRPI for control rod F-10 is low from demand by 10 steps
- CR 00353007, IRPI for control rod F-2 is erratic by +/- 10 steps
- CR 00352030, Wrong control rod reference bank was used to calculate estimated critical rod position for Unit 1
- CR 00353023, Operations surveillance procedure 01-OHP 4030.STP.015 was not performed during reactor startup and provisions were not immediately apparent that it was not required
- CR 00354003, IRPI for control rod P-10 is low from demand by 13 steps
- CR 00354002, IRPI for control rod P-6 is low from demand by 13 steps
- CR 00354099, IRPIs for multiple rods are drifting low as a result of rod worth testing in accordance with procedure 12-EHP 6040.PER.352
- CR 00355130, Unit 1 power range nuclear instrument N-43 lower detector not reading properly
- CR 00355129, Unit 1 West Main Feed Pump tripped on low vacuum
- CR 00355132, Declared Unit 1 power range nuclear instrument inoperable due to no current indicated on the lower detector
- CR 00357004, Nuclear instrumentation power trips for 20 percent actuated at greater than 20 percent

b. Issues and Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Unit 1 Diesel Generator Load Sequence Testing

a. Inspection Scope

The inspectors reviewed the surveillance tests associated with load sequence testing and engineered safety features (ESF) actuation for both Unit 1 emergency diesel generators. Load sequence testing was conducted in order to satisfy the portions of the surveillance requirements of TS 4.8.1.1.2. The inspectors reviewed the associated surveillance test procedure, acceptance criteria, supporting documentation, and completed test data. The inspectors verified that the test methodology and acceptance criteria were consistent with TS and design basis requirements. The inspectors reviewed the following documents:

- Unit 1 T Ss 4.8.1.1 and 4.8.1.2; AC Distribution, Operating; and AC Distribution, Shutdown
- Updated Final Safety Analysis Report Section 8.4, “Emergency Power System”
- 01-OHP 4030.132.217A, “DG1CD Load Sequencing & ESF Testing,“ Revision 0
- 01-OHP 4030.132.217B, “DG1AB Load Sequencing & ESF Testing,” Revision 0
- Attachment 3 to 01-OHP 4030.132.217A, “Equipment Re-Test Sheet” for deferred Train “A” equipment
- Attachment 3 to 01-OHP 4030.132.217B, “Equipment Re-Test Sheet” for deferred Train “B” equipment

- Performance Assurance Field Observation FO-00-L-031, “Review LOOP/LOCA Tests Deferred Equipment Log Re-Tests”

The licensee identified about thirty separate plant components which could not be tested during the initial load sequence test procedure. As allowed by the surveillance procedure, these components were identified as needing a retest at a later time. The inspectors verified that all of the deferred test components were satisfactorily tested before Unit 1 was placed in Mode 4.

b. Issues and Findings

No findings of significance were identified.

.2 Measurement of Unit 1 Isothermal Temperature Coefficient

a. Inspection Scope

Following the Unit 1 ascension into Mode 2, the licensee performed low power physics testing to verify that core parameters were consistent with the core operating limits report and safety analyses assumptions. During low power physics testing, the licensee measured the isothermal temperature coefficient (ITC) of reactivity. The ITC was used to calculate the moderator temperature coefficient (MTC). Technical Specification 3.1.1.4, “Moderator Temperature Coefficient” placed limitations on the magnitude of the MTC. Technical Specification 4.1.1.4.a required measurement of the MTC prior to operation above 5 percent rated thermal power. Because the MTC impacts core temperature reactivity feedback assumed in safety analyses, including the analyses for steam line breaks and positive reactivity excursions, the inspectors considered that this surveillance test was related to the mitigating system cornerstone. The inspectors observed the measurement of the ITC, reviewed completed test data, assessed reactivity control and procedural compliance, and discussed the results with the reactor engineering supervisor. The inspectors reviewed the following documents during this inspection:

- 12-EHP 4030 STP.350, “Isothermal Temperature Coefficient (ITC) Measurement and Moderator Temperature Coefficient Calculation
- 12-EHP 6040 PER.359, “Zero Power and Power Ascension Tests for Post-Refueling Startups,” Revision 6
- 01-OHP 4021.001.006, “Power Escalation,” Revision 23a
- OHI-4000, “Conduct of Operations: Standards,” Revision 2
- PMP 4015.RMP.001, “Reactivity Management Program,” Revision 0

b. Issues and Findings

No findings of significance were identified.

.3 Unit 1 Bank Rod Worth Measurement

a. Inspection Scope

Prior to entry into Mode 1, the licensee conducted low power physics testing to measure the fundamental nuclear characteristics of the reactor core. During this physics testing, the licensee measured the reactivity of the control rod banks to verify that the actual control rod worths were consistent with the Core Operating Limits Report and safety analysis assumptions. Because control rod reactivity worth can affect the accident mitigation capability of the control rods during postulated accidents, the inspectors determined that this testing was associated with the mitigating systems cornerstone. The inspectors observed group rod worth testing, assessed reactivity control and procedural compliance, reviewed completed test data, and discussed the results of the testing with the reactor engineering supervisor. The inspectors verified that the requirements of TS special test exception 3.10.3, “Physics Tests,” were met during bank rod worth testing. The inspectors reviewed the documents:

- Core Operating Limits Report, Donald C. Cook Nuclear Plant, Unit 1 Cycle 17, Revision 0
- UFSAR Section 13.3.2, “Low Power Testing”
- 12-EHP 6040 PER.357, “Initial Criticality, All Rods Out Boron Concentration and Nuclear Heating Level,” Revision 7d
- 12-EHP 6040 PER.352, “Rod Worth Verification Utilizing RCC Bank Interchange,” Revision 4a
- 12-EHP 6040 PER.359, “Zero Power and Power Ascension Tests for Post-Refueling Startups,” Revision 6
- 01-OHP 4021.001.006, “Power Escalation,” Revision 23a
- OHI-4000, “Conduct of Operations: Standards,” Revision 2
- PMP 4015.RMP.001, “Reactivity Management Program,” Revision 0

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 Unit 2 Residual Heat Removal portion of the Mitigating Systems cornerstone. The inspectors utilized the following documents during this review:

- PMP 7110.PIP.001, “Regulatory Oversight Program Performance Indicators,” Revision 0

- PMI [Plant Managers Instruction] 7110, “Regulatory Oversight Program,” Revision 0

b. Issues and Findings

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

No findings of significance were identified.

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 50-315/98045-01: Insufficient deliverable volume in containment spray (CTS) system chemical additive tank. On October 9, 1998, the licensee identified that during certain postulated Loss of Coolant Accident or Main Steam Line Break (MSLB) scenarios, the sodium hydroxide contained in the CTS system chemical additive tank may be exhausted prior to the end of the injection phase. Consequently, continued spraying of refueling water storage tank water would result in a CTS spray pH lower than the range evaluated for containment equipment qualification. After this issue was identified, the licensee evaluated the containment equipment qualification for the expected CTS spray pH following a MSLB. Based on the short period of time that the containment equipment would be subject to an acidic spray following a MSLB, the licensee concluded that the containment equipment qualification would not be challenged. The inspectors concluded that the failure to evaluate the containment equipment qualification for the expected spray pH following a MSLB in accordance with 10 CFR 50, Appendix B, Criterion III, constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC’s Enforcement Policy. This issue was entered into the licensee’s corrective action program as Condition Report 98-5605. This LER is closed.

b.2 (Closed) Licensee Event Report 50-316/2000-015-00: Containment airlock door seals not tested at frequency required by TSs. On October 19, 2000, the licensee’s Performance Assurance organization identified that the Unit 2 containment airlock door seals were not tested within 7 days after a containment entry. Technical Specification 4.6.1.3.a required, in part, that whenever containment integrity is required, the containment airlocks shall be tested within 7 days after each containment access. Further, the TS allowed that, for periods where the airlock doors are routinely used for access more frequently than once every 7 days, door seals may be tested once per 30 days rather than after each containment entry during this time period. On

July 27, 2000, an operator entered the Unit 2 containment at the lower airlock for a surveillance test line-up. The next lower airlock access occurred on August 10, 2000, more than 7 days after the previous entry; however, the lower airlock was not tested until August 23, 2000. The Unit 2 lower airlock door seals were tested satisfactorily on August 23, 2000; therefore, this event had minimal safety significance. The inspectors concluded that the failure to test the Unit 2 containment lower airlock within 7 days after access in accordance with TS 4.6.1.3.a constituted a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This issue was entered into the licensee's corrective action program as CR 00256023. This LER is closed.

4OA6 Management Meetings

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

PARTIAL LIST OF PERSONS CONTACTED

R. Gaston, Regulatory Affairs
J. Gebbie, Plant Engineering
S. Greenlee, Engineering
M. Hoskins, System Engineering
S. Lacey, Director, Engineering
J. Mathis, Regulatory Affairs
R. Meister, Regulatory Affairs
D. Moll, Assistant Operations Superintendent
T. Noonan, Director, Performance Assurance
J. Pollock, Plant Manager
R. Powers, Senior Vice President
T. Quaka, Engineering
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
71111-13	Maintenance and Emergent Work Control	1R13
71111-17	Permanent Plant Modifications	1R17
71111-15	Operability Evaluations	1R15
71111-19	Post-Maintenance Testing	1R19
71111-20	Refueling Outage	1R20
71111-22	Surveillance Testing	1R22
71151	Performance Indicator Verification	4OA1
71153	Event Followup	4OA3

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

50-315/98045-01	LER	Insufficient deliverable volume in containment spray system chemical additive tank
50-316/2000-015-00	LER	Containment airlock door seals not tested at frequency required by TSs

Discussed

None

LIST OF ABBREVIATIONS

AES	Engineered Safety Features Ventilation
AFW	Auxiliary Feedwater System
ASME	American Society of Mechanical Engineers
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CTS	Containment Spray System
CVCS	Chemical and Volume Control System
D/G	Diesel Generator
DIT	Design Input Transmittal
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Features
ESW	Essential Service Water
JO	Job Order
LER	Licensee Event Report
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
ODE	Operability Determination Evaluation
OHI	Operations Head Instruction
OHP	Operations Head Procedure
OSO	Operations Standing Order
PDR	Public Document Room
PI	Performance Indicator
PMI	Plant Manager’s Instruction
PMP	Plant Manager’s Procedure
PMT	Post-maintenance Testing
PORV	Power Operated Relief Valve
PPC	Plant Process Computer
RCS	Reactor Coolant System
RWST	Refueling Water Storage Tank
SFP	Spent Fuel Pool
SRO	Senior Reactor Operator
SSC	Structures, Systems, and Components
STP	Surveillance Test Procedure
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TDB	Technical Data Book
TS	Technical Specification
URI	Unresolved Item
UFSAR	Updated Final Safety Analysis
VAC	Volts, Alternating Current

VDC	Volts, Direct Current
VIO	Violation