October 30, 2003

Mr. M. Nazar
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000315/2003010;
05000316/2003010

Dear Mr. Nazar:

On September 30, 2003, the U. S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your D. C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on October 10, 2003, with you and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, five findings of very low safety significance (Green) were identified, two of which involved violations of NRC requirements. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III, 801 Warrenville Road, Lisle, IL 60532-4351; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the D. C. Cook Nuclear Power Plant.

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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Eric R. Duncan, Chief
Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000315/2003010; 05000316/2003010
w/Attachment: Supplemental Information

cc w/encl: J. Jensen, Site Vice President
M. Finissi, Plant Manager
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
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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:	05000315/2003010; 05000316/2003010
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	1 Cook Place Bridgman, MI 49106
Dates:	July 1, 2003, through September 30, 2003
Inspectors:	B. Kemker, Senior Resident Inspector I. Netzel, Resident Inspector B. Jorgensen, Operations Engineer E. Kleeh, Operations Engineer, NRR R. Matthew, Operations Engineer, NRR W. Slawinski, Senior Radiation Specialist
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000315/2003010, IR 05000316/2003010; 07/01/2003-09/30/2003; D. C. Cook Nuclear Power Plant, Units 1 and 2; Maintenance Effectiveness; Personnel Performance During Non-Routine Plant Evolutions; Post Maintenance Testing; Radiological Environmental Monitoring and Radioactive Material Control Programs

This report covers a 13-week period of inspection by resident, regional, and headquarters based inspectors. The report includes an announced baseline inspection in the area of radiation protection. Five Green findings were identified, two of which had an associated Non-Cited Violation (NCV). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be "Green" or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was self-revealed when maintenance craftsmen failed to accurately measure, machine and install a replacement coupling during a planned maintenance activity on the Unit 2 West motor driven auxiliary feedwater pump which resulted in the unavailability of the pump significantly beyond the original 18-hour planned maintenance period. The licensee was granted enforcement discretion for Technical Specification 3.7.2.1.a to preclude a plant shutdown. The licensee subsequently completed repairs to the motor driven auxiliary feedwater pump and returned the pump to service within the enforcement discretion period. The primary cause of this finding was related to the cross-cutting area of Human Performance.

  The finding was more than minor because the finding was associated with the Equipment Performance and Human Performance attributes of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the motor driven auxiliary feedwater pump was rendered unavailable for an extended period of time. The finding was of very low safety significance because the unavailability of the motor driven auxiliary feedwater pump on overall plant risk was not significant. No violation of regulatory requirements occurred. (Section 1R12.b.1)

- Green. A finding of very low safety significance was self-revealed when licensee personnel failed to control the sensing line configuration on the Control Room Air Conditioning (CRAC) chiller units in accordance with design documentation which resulted in spurious tripping of an idle CRAC chiller unit upon initial start following an extended shutdown period. The primary cause of this finding was related to the cross-cutting area of Human Performance. The licensee subsequently corrected the sensing line configuration and successfully tested the operation of all four chiller units.

The finding was more than minor because this finding was associated with the Design Control and Equipment Performance attributes of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences because the reliability of the CRAC chiller units was impacted. This finding was of very low safety significance because the design deficiency did not result in a loss of function of the CRAC chiller units per Generic Letter 91-18. No violation of regulatory requirements occurred. (Section 1R12.b.2)

- Green. A finding of very low safety significance was self-revealed when licensee personnel failed to accomplish testing of 345 kilovolt (kV) switchyard breaker "L" with an adequate procedure which resulted in the loss of the Class 1E reserve feed supply to Train “B” safety-related equipment for Unit 1 and Unit 2. The primary cause of this finding was related to the cross-cutting area of Human Performance. The licensee subsequently restored the switchyard Class 1E reserve feed supply and issued a standing order to control maintenance and testing in the switchyard.

  The finding was more than minor because this finding was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the reliability of the offsite Class 1E reserve feed supply to safety-related equipment for both units was affected. This finding was of very low safety significance since it did not result in the actual loss of the safety function of any safety-related equipment. No violation of regulatory requirements occurred. (Section 1R14.1)

- Green. The inspectors identified a Non-Cited Violation of Technical Specification 6.8.1.a. The licensee failed to correctly implement a design modification on the Unit 2 West residual heat removal (RHR) train in accordance with the approved work instructions and design documents. Specifically, the licensee failed to correctly install the first weld of the new high point vent assembly per the approved weld detail and returned the pump to service with the non-conforming condition. The inspectors identified this error after the weld had already been accepted by the licensee's quality control (i.e., performance verification) inspection process and the pump was returned to service. The licensee subsequently corrected the weld to meet the approved design.

  The inspectors concluded that this issue was associated with the mitigating systems cornerstone and adversely affected the cornerstone objective. Specifically, the inspectors determined that the installed weld would be more susceptible to vibration induced fatigue failure than the approved weld, and if this condition were not corrected it could lead to a premature failure of the weld, affecting the function and integrity of the RHR system. The inspectors concluded that this finding was a licensee performance deficiency of very low safety significance because it did not result in loss of safety function for the West RHR train for greater than its Technical Specification allowed outage time. (Section 1R19)

Cornerstone: Public Radiation Safety

- Green. A finding of very low safety significance was self-revealed when a second survey of a valve that was previously surveyed and unconditionally released from the radiologically controlled area identified that the valve was contaminated. The primary cause of this finding was related to the cross-cutting area of Human Performance.

  The finding was more than minor because this finding was associated with the Human Performance and Program and Process attributes of the Public Radiation Safety cornerstone and adversely impacted the cornerstone objective of ensuring adequate protection of the public health and safety from exposure to radioactive materials released or potentially released into the public domain. The finding was of very low safety significance because the public radiation exposure resulting from the problem was low and the finding was not repetitive. To address this issue, the licensee performed a thorough extent of condition evaluation to ensure that contaminated residue was identified which included radiation surveys in offsite areas and of personal items located outside the radiologically controlled area. One Non-Cited Violation of Technical Specification 6.8.1 regarding licensee procedures that govern the unconditional release of radioactive material was identified. (Section 2PS3.5)

B. <u>Licensee Identified Violations</u>

Violations of very low safety significance, which were identified by the license have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near full power during the inspection period.

Unit 2 operated at or near full power until August 13, 2003, when the licensee performed a reactor shutdown and cooled the plant to 205°F to isolate a steam leak from a feedwater system check valve due to a failed gasket. During plant heatup after repairing the check valve, the licensee identified another leak from a bolt hole on a different feedwater system check valve and again cooled the plant to 205°F to repair the valve. The unit was subsequently cooled to Mode 5 (Cold Shutdown) to perform additional forced outage maintenance activities. The licensee performed a reactor startup and synchronized the unit to the grid on August 29, 2003.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. The inspectors performed four partial system walkdowns of the following risk significant systems:

- Unit 1 Condensate System (risk significant for initiating events)
- Unit 2 West Containment Spray Train (risk significant with the Unit 2 East Containment Spray train out of service for maintenance)
- Unit 2 East Residual Heat Removal (RHR) Train (risk significant with the Unit 2 West RHR train out of service for maintenance)
- Unit 2 RHR System Restoration Alignment (risk significant following plant heat up from forced outage)

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones. The inspectors reviewed operating procedures, system diagrams, Technical Specification (TS) requirements, Administrative TSs, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components were aligned correctly.

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed one complete system walkdown of the following risk significant system:

- Unit 2 Chemical and Volume Control System

The inspectors reviewed ongoing system maintenance, open job orders, and design issues for potential effects on the ability of the system to perform its design functions. The inspectors reviewed operating procedures, system diagrams, TS requirements, applicable sections of the Updated Final Safety Analysis Report (UFSAR) and vendor manuals to ensure the correct system lineup. The inspectors verified acceptable material condition of system components, availability of electrical power to system components, and that ancillary equipment or debris did not interfere with system performance.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours

a. Inspection Scope

The inspectors performed 11 fire protection walkdowns of the following risk significant plant areas:

- Unit 1 4 kilovolt (kV) AB Switchgear Room (Zone 40A)
- Unit 1 4 kV CD Switchgear Room (Zone 40B)
- Unit 1 West Main Steam Valve Enclosure (Zone 108)
- Unit 1 AB Battery Room (Zone 42D)
  Unit 1 CD Battery Room (Zone 55)
- Unit 2 4 kV AB Switchgear Room (Zone 47A)
- Unit 2 4-kV CD Switchgear Room (Zone 47B)
- Unit 2 West Main Steam Valve Enclosure (Zone 109)
- Unit 2 Main Steam Access Way (Zone 111)
- Unit 2 AB Battery Room (Zone 46D)
- Unit 2 CD Battery Room (Zone 60)

The inspectors verified that fire zone conditions were consistent with assumptions in the licensee's Fire Hazards Analysis. The inspectors walked down fire detection and suppression equipment, assessed the material condition of fire fighting equipment, and evaluated the control of transient combustible materials.

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors performed two inspection activities related to the licensee's precautions to mitigate risk from internal and external flooding events. The following inspection activities were performed:

- reviewed the Unit 1 and Unit 2 Flooding Evaluation reports, the UFSAR, and other selected design basis documents to identify those areas susceptible to internal and external flooding;

- performed a walkdown of the 569 foot - 6 inch elevation of the Turbine Building and the Unit 1 and Unit 2 579 foot condenser pit areas; and

- reviewed selected operating procedures used to identify and mitigate flooding events.

In addition, the inspectors reviewed the issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for flood protection related issues documented in selected condition reports.

b. Findings

No findings of significance were identified. An observation related to the cross-cutting area of Problem Identification and Resolution is discussed in Section 4OA2.3.

1R11 Licensed Operator Requalification (71111.11)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

The inspectors assessed licensed operator performance and the training evaluators' critique during licensed operator re-qualification evaluations in the D. C. Cook Plant operations training simulator on August 27, 2003. The inspectors focused on alarm response, command and control of crew activities, communication practices, procedural adherence, and implementation of emergency plan requirements.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of selected degraded performance issues involving the following three risk-significant structures, systems, and components (SSCs):

- Unit 1 and Unit 2 Control Room Chiller Failures
- Unit 2 West Motor Driven Auxiliary Feedwater Pump Motor Replacement
- Unit 2 Feedwater Check Valve 2-FW-118-4 Non-Code Repair

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the SSCs. Specifically, the inspectors independently verified the licensee's evaluation of SSC performance or condition problems in terms of:

- appropriate work practices,
- identifying and addressing common cause failures,
- scoping of SSCs in accordance with 10 CFR 50.65(b),
- characterizing SSC reliability issues,
- tracking SSC unavailability,
- trending key parameters (condition monitoring),
- 10 CFR 50.65(a)(1) or (a)(2) classification and/or re-classification, and
- appropriate performance criteria for SSCs classified as (a)(2) and/or appropriate and adequate goals and corrective actions for SSCs classified as (a)(1).

b. Findings

b.1. Motor Driven Auxiliary Feedwater Pump (MDAFWP) Maintenance Errors

Introduction

The inspectors identified a finding of very low safety significance (Green) when maintenance craftsmen failed to accurately measure, machine, and install a replacement coupling for the Unit 2 West MDAFWP. The cumulative effect of these errors resulted in the unavailability of the pump significantly longer than the original 18-hour planned maintenance period and the TS 72-hour allowed outage time. No violation of regulatory requirements occurred.

Discussion

On March 5, 2003, the Unit 2 West MDAFWP was declared inoperable when it was taken out of service for planned maintenance activities. The expected duration of the planned work was 18 hours. Technical Specification 3.7.1.2.a. was entered, placing Unit 2 in a 72-hour allowed outage time.

The planned maintenance activities were completed and on March 5, 2003, during post-maintenance testing of the pump, an unusual noise was heard during the initial start of the pump motor. A problem solving team performed several activities to diagnose the source of this noise. On March 6, 2003, the problem solving team recommended that the motor be replaced. A replacement motor was located and was prepared for installation on March 7, 2003.

Licensee personnel also recognized that a motor coupling would be needed to couple the motor to the pump shaft. On March 6, 2003, an inaccurate measurement of the motor shaft was taken. This measurement was used to prepare a bill of materials (BOM) for an in-stock pre-bored coupling. In parallel with the BOM preparation, a blank coupling was procured and machining commenced on site for use as a spare coupling. On March 7, 2003, the BOM for the pre-bored coupling was completed and machining on the blank coupling was stopped due to the decision to install the pre-bored coupling. However, when the motor shaft measurements were re-taken, they were found to be smaller than previously measured which rendered the pre-bored coupling unusable. As a result, the machining on the spare coupling was resumed. In addition, a blank coupling was sent to a vendor for machining, however the dimensions specified for this coupling were also incorrect.

On March 8, 2003, the licensee requested and was granted enforcement discretion for the TS 3.7.2.1.a 72-hour allowed outage time for an additional 36 hours to preclude a required plant shutdown.

On March 8, 2003, licensee personnel discovered that the blank coupling that was machined on site was out of tolerance and unusable. As a result, the coupling from the originally installed motor was removed for re-use on the replacement motor. It was re-bored, measured, verified to be acceptable and installed. About 29 hours elapsed from the time it was recognized that a new coupling was needed to the time that an appropriately sized coupling was machined and ready for installation.

On March 8, 2003, the coupling that was installed on the replacement motor was found to have been installed backwards. The installed coupling was removed and the coupling that had been sent to the vendor for boring was re-bored and installed on the Unit 2 West MDAFWP. About 12 additional hours were expended replacing the incorrectly installed coupling.

On March 9, 2003, the pump was declared operable following successful post-maintenance testing.

Analysis

The inspectors determined that the multiple skill-based errors exhibited during this maintenance activity was a licensee performance deficiency warranting a significance evaluation. The Mitigating Systems cornerstone was impacted by this issue. The inspectors also concluded that this finding affected the cross-cutting area of Human Performance. The inspectors reviewed the samples of minor issues in Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues,” and determined that there were no examples similar to this issue. The inspectors concluded that the finding was of more than minor risk significance in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” since the finding was associated with the

Equipment Performance and Human Performance attributes of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the motor driven auxiliary feedwater pump was rendered unavailable for an extended period of time.

Utilizing IMC 0609, “Significance Determination Process,” Appendix A, “SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barrier Integrity] Cornerstones,” the inspectors determined that since Unit 2 was not shut down prior to exceeding the TS allowed outage time for the Unit 2 West MDAFWP, the finding represented an actual loss of safety function of a single train of safety-related equipment for greater than its TS allowed outage time and a Phase 2 SDP evaluation was required. The inspectors solved the Phase 2 SDP sequences that involved auxiliary feedwater with a duration of 0-3 days since the total unavailability of the Unit 2 West MDAFWP due to preventable maintenance errors was less than 3 days. All applicable Phase 2 Worksheets were solved, with the “Loss of Train B of 250 volts DC (LEDCB)” as the dominant accident sequence. It was assumed that at least two out of four steam generators were available, along with the redundant motor driven auxiliary feedwater train and the turbine driven auxiliary feedwater train. The cross-tie capability from the opposite unit was also credited. Based on the results, the inspectors determined that the finding was of very low safety significance.

Enforcement

No violations of regulatory requirements occurred. This issue was considered to be a finding (FIN 05000316/2003010-01). The licensee entered this finding into their corrective action program as condition report (CR) 03067008.

b.2 Control Room Air Conditioning Unit Trips Due to Incorrect Sensing Line Configuration

Introduction

The inspectors identified a finding of very low safety significance (Green) due to incorrect pressure sensing line configurations on the Unit 1 and Unit 2 North and South Control Room air conditioning (CRAC) chiller units. No violation of regulatory requirements occurred.

Discussion

Since at least 2002, frequent tripping of each of the four CRAC chillers (one chiller for each Unit 1 North and Unit 1 South CRAC unit, Unit 2 North and Unit 2 South CRAC unit) on high discharge pressure occurred. The normal control room configuration was one CRAC unit in service with the redundant unit in standby. The spurious tripping of a CRAC chiller occurred upon initial start of an idle CRAC unit that had been in standby for an extended period of time (i.e., greater than 4 hours), and only affected the unit being started. Once a CRAC unit was running successfully with its chiller in service, it was not susceptible to this intermittent failure mode.

On August 13, 2003, in the process of troubleshooting the most recent high discharge pressure trips on the Unit 2 South CRAC chiller unit, the licensee discovered that the pressure sensing element tubing on each train (North and South) for both Unit 1 and Unit 2 had been installed incorrectly (contrary to the vendor manual) from the outlet of its respective condenser unit rather than from the outlet of the compressor unit to the condenser water valve sensing element. The licensee also identified that the Unit 1 controlled drawing indicated the correct (vendor recommended) configuration, however the Unit 2 controlled drawing indicated the incorrect as-found condition. It was not known how long this condition had existed.

On August 16, 2003, the licensee completed the relocation of the pressure sensing element tubing on all four CRAC chillers under Corrective Minor Modification 12-CMM-30056. CR 03210005 documented the licensee's conclusion that the incorrect sensing line configuration was the cause of the chiller trips on high discharge pressure. Since this repair has been made, there has been no other occurrence of a CRAC chiller tripping on high discharge pressure.

Analysis

The inspectors determined that the licensee's failure to control the sensing line configuration of the four CRAC chillers in accordance with design and vendor documentation represented a performance deficiency warranting a significance determination. The Mitigating Systems cornerstone was impacted by this issue. The CRAC system is required to be operable per TS 3.7.5.2. The safety function of the CRAC system is to maintain Control Room temperature $\leq 102°F$ during accident conditions with the Control Room isolated to ensure vital Control Room equipment remains within recommended operating temperature ranges. The CRAC chillers provide the necessary cooling function. During a design basis event, the CRAC units are load shed from the safety-related power supply and operators restart the CRAC units from the Control Room as instructed in the emergency operating procedures. The sensing line configuration deficiency degraded the reliability of any CRAC unit that had been in a standby condition for an extended period of time such that the operator's ability to start and run a CRAC unit, with its associated chiller in service, from the Control Room was adversely impacted.

The inspectors concluded that the finding was of more than minor risk significance in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," since the finding was associated with the Design Control and Equipment Performance attributes of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences because the reliability of the CRAC chiller units was impacted. The inspectors also reviewed the examples of work in progress findings contained in IMC 0612, Appendix E, "Examples of Minor Issues," and concluded that this finding was of more than minor significance because the chillers had been in service for at least several years with the non-conforming condition.

Utilizing IMC 0609, "Significance Determination Process," Appendix A, "SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barrier

Integrity] Cornerstones," the inspectors determined that this finding was a design/qualification deficiency confirmed not to result in a loss of function per Generic Letter 91-18. This conclusion was based on the following information: 1) the CRAC units are each 100 percent capacity units, 2) the normal Control Room configuration is one unit in service and the redundant unit in standby, 3) spurious tripping only affected the standby unit, and 4) there would be sufficient time for operators to perform local compensatory measures to reset the high discharge pressure trip and restart the standby CRAC unit prior to any Control Room equipment failure. Therefore, the finding screened out as Green and was considered to be of very low safety significance.

Enforcement

The Control Room Air Conditioning chiller units were not safety-related. Consequently, no violation of regulatory requirements occurred. This issue was considered to be a finding (FIN 05000315/316/2003010-02). The licensee entered this finding into their corrective action program as CR 03210005. The licensee subsequently corrected the sensing line configuration and satisfactorily tested the operation of all four chiller units.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for seven maintenance and operational activities affecting the following equipment:

- Unit 1 East Charging Train Maintenance
- Unit 1 East RHR Train Maintenance
- Unit 1 Upper Ice Condenser Bay 4 Ice Formation
- Unit 2 South Safety Injection Pump Discharge Valve 2-SI-206 Repair
- Unit 2 West RHR Train Maintenance
- Unit 2 Feedwater Check Valve 2-FW-118-4 Repair and Unit 2 AB Emergency Diesel Generator (EDG) Governor Replacement
- Unit 1 and Unit 2 On-line Risk Changes due to Severe Weather (July 7, 2003)

These activities were selected based on their potential risk significance relative to the reactor safety cornerstones. The maintenance associated with the Unit 2 South safety injection pump discharge valve was emergent work to repair a body-to-bonnet leak. The feedwater check valve repair was emergent work to repair a through-wall leak in the valve's body and the EDG governor replacement was added to the scope of work for the forced outage period. There was also emergent work associated with the Unit 1 upper ice condenser to address ice formation in Bay 4.

As applicable for each of the above activities, the inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst and/or shift technical advisor, and verified that plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify that risk analysis assumptions were valid and applicable requirements were met.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions (71111.14)

.1 Partial Loss of Reserve Feed Power During Switchyard Breaker Testing

a. Inspection Scope

On February 18, 2003, while testing the trip controls for the newly installed 345 kV switchyard breaker "L," an unexpected trip of switchyard breakers L1, M1, N1, K1 and BE occurred. This resulted in the loss of the Class 1E reserve feed supply to Train “B” safety-related equipment for both units. The inspectors interviewed operations and engineering department personnel and reviewed the licensee's root cause evaluation, rapid event response report, applicable procedures, and the subject condition report to understand the details of the event.

b. Findings

Introduction

A finding of very low safety significance (Green) was self-revealed when licensee personnel failed to adequately accomplish testing of 345 kV switchyard breaker “L” which resulted in a partial loss of off-site power to both operating units. No violation of regulatory requirements occurred.

Discussion

In early 2002, American Electric Power's Energy Delivery (ED) Group began an upgrade project to replace all 345 kV breakers in the D.C. Cook plant switchyard. Prior to replacement, switchyard breaker “L” suffered a catastrophic failure on June 12, 2002. The breaker was subsequently replaced, however testing had not been completed before the breaker was returned to service. Final testing was in progress on February 18, 2003, when the L1, M1, N1, K1 and BE breakers unexpectedly opened, causing a partial loss of offsite reserve feed power to Train “B” safety-related equipment to both units. The field work activities, including testing, for this project was considered to be outside of the plant's controlled maintenance boundary specified in the Inter-Organizational Agreement between the ED Group and the D.C. Cook Nuclear Plant. Since the replacement of the breakers and controls was considered to be outside the specified responsibility of the plant, the licensee's design change and work control processes, including testing plans, were not used. Replacement and testing the new breaker and supporting control equipment was coordinated with the Operations Department to ensure availability of the necessary off-site power sources. While the Inter-Organizational Agreement required plant Engineering Department approval for design changes, it did not specify the controls for field work or post modification testing of the switchyard equipment. The ED Group did not use a formal test plan or work instructions comparable to what the licensee would otherwise require for work on plant equipment.

The licensee conducted an investigation to determine the cause for this event and determined that a standing trip signal was present in the fault current protection circuit for the breaker that had not been identified by the ED test engineers prior to the testing. There were three trip pathways for the “L” breaker. Two of the trip pathways had supervisory switches that were opened during the test to prevent an unwanted trip. The third trip pathway was a breaker failure trip circuit that required two inputs to initiate a trip. One input signal was generated during the breaker performance test and the second input required an indication of a system failure (high load) associated with another circuit. Previous testing of the other new switchyard breakers found that those circuits were lightly loaded and did not approach the system failure setpoint required for the second input to the breaker failure trip circuit. This third pathway was not considered by ED test engineers during testing of the “L” breaker because it was on a different drawing than the primary drawing that contained the two supervisory switches that was used while planning the test. The test engineers also noted that the third pathway had not been a concern during previous testing of the other switchyard breakers due to low system load. The setpoint for the breaker failure relay was 900 amperes. At the time of testing, the Collingwood line via the L1 breaker had approximately 900 amperes, which created the conditions to cause the unexpected breaker trips. When the line was restored to service, the load was about 1200 amperes. The partial loss of reserve feed power lasted about 2 hours.

The inspectors reviewed the licensee's root cause evaluation which identified several organization-to-program deficiencies associated with the control of switchyard maintenance activities. The inspectors noted that the licensee's root cause evaluation identified several other contributing causes for this event including: (1) the ED test engineers developed and implemented an inadequate test plan for testing the breaker trip logic; (2) the ED dispatch center permitted testing to proceed without identifying and discussing system conditions that would affect breaker testing with the test engineers, and; (3) the ED test engineers failed to use all available indications to verify system load conditions before conducting the testing.

In response to this event, the licensee issued a standing order to provide interim controls for authorizing work in the switchyard. The standing order allowed Priority 1 work to be performed under the Inter-Organizational Agreement with approval of the Shift Manager and required all other switchyard work to be controlled by an approved plant procedure. The standing order required work performed under a Job Order activity to be approved by the Operations Director and Plant Manager. As part of their long-term corrective actions, the licensee planned to revise the Inter-Organizational Agreement to: (1) require a plant approved Job Order and detailed work instructions for all maintenance, modifications, and testing activities in the switchyard; (2) clarify plant engineering responsibilities regarding switchyard design changes; and, (3) define when the plant's design change process is required to be used.

Analysis

The inspectors determined that the failure to provide an appropriate procedure for testing 345 kV switchyard breaker “L” was a performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting area of Human Performance. The Mitigating Systems cornerstone was impacted by this issue. The inspectors reviewed the samples of minor issues in IMC 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues,” and determined that there were no examples similar to this issue. The inspectors concluded that the finding was of more than minor risk significance in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” since the finding was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the failure to provide an appropriate procedure for testing 345 kV switchyard breakers affected the availability, reliability, and capability of the offsite power source to Train “B” safety-related equipment for both units.

Utilizing IMC 0609, “Significance Determination Process,” Appendix A, “SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barrier Integrity] Cornerstones,” the inspectors determined that this finding (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; (4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. Therefore, the finding screened out as Green and was considered to be of very low safety significance.

Enforcement

The switchyard breakers were not safety-related equipment. Consequently, no violation of regulatory requirements occurred. This issue was considered to be a finding (FIN 05000315/316/2003010-03). The licensee entered this finding into their corrective action program as CR 03049046. The licensee restored the switchyard Class 1E reserve feed supply and issued a standing order to control maintenance and testing in the switchyard.

.2 Unit 1 Reserve Feed Power Rendered Inoperable During Plant Startup

a. Inspection Scope

On February 5, 2003, after transferring power from the reserve auxiliary transformers to the unit auxiliary (normal power) transformers, operators discovered that the "no load" voltage setting of the load tap changers on the reserve auxiliary transformers was not correctly set. This rendered both trains of the Class 1E reserve feed supply to Unit 1 inoperable. This event was selected for review to evaluate the human performance errors that caused the event. The inspectors interviewed operations and engineering department personnel and reviewed the licensee's condition report evaluation and applicable procedures to understand the details of the event.

b. Findings

No findings of significance were identified.

.3 Unit 2 AB EDG Declared Inoperable Due to Incorrect Voltage Regulator Setting

a. Inspection Scope

On February 8, 2003, following a routine surveillance test of the Unit 2 AB EDG, operators identified during a control board walkdown that the automatic voltage regulator potentiometer was not correctly reset as directed in the surveillance test procedure. This resulted in operators declaring the engine inoperable and entering the applicable TS limiting condition for operation (LCO) action requirement until the potentiometer was reset to its correct position. The inspectors noted that similar events had occurred before and selected this event for review to evaluate the human performance errors that caused it. The inspectors interviewed Operations Department personnel and reviewed the licensee's Apparent Cause Evaluation, applicable procedures, and the subject condition report to understand the details of the event.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following three condition reports to ensure that either: (1) the condition did not render the involved equipment inoperable or result in an unrecognized increase in plant risk, or (2) the licensee appropriately applied TS limitations and appropriately returned the affected equipment to an operable status.

- CR 03163026, "Scheduled Replacement of 2-FFI-230 Has Passed Its Drop Dead Date Without Being Completed"
- CR 03209017,"1-HV-ACRA-1, North Control Room Air Conditioning Unit is Inoperable Due to Repeated Failures of the Compressor to Start Upon Receiving a Start Signal from the Control Room"
- CR 03240014, "Audible Count Rate Was Not Re-scaled as Count Rate Decayed after the Unit 2 Shutdown"

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed 10 post maintenance testing activities associated with the following scheduled maintenance:

- Unit 1CD EDG Maintenance
- Unit 1 AB EDG Maintenance

- Unit 1 East Centrifugal Charging Pump Maintenance
- Valve 2-IMO-361 Maintenance
- Unit 2 CD EDG Voltage Regulator Replacement
- Unit 2 West RHR Train Vent Valve Installation
- Valve 2-SI-121N Throttle Valve Repositioning Following 2-SI-206 Repair
- Unit 2 AB EDG Governor Replacement
- Unit 2 East Containment Spray Train Maintenance
- Unit 2 Feedwater System Check Valve Repairs

The inspectors reviewed the scope of the work performed and evaluated the adequacy of the specified post maintenance testing. The inspectors verified that the post maintenance testing was performed in accordance with approved procedures, that the procedures clearly stated acceptance criteria, and that the acceptance criteria were met. The inspectors interviewed operations, maintenance, and engineering department personnel and reviewed the completed post maintenance testing documentation.

In addition, the inspectors verified that post maintenance testing problems were entered into the corrective action program with the appropriate significance characterization.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) associated with the licensee's failure to correctly implement a design modification on the Unit 2 West RHR train in accordance with the approved work instructions and design documents. The inspectors determined that this issue constituted a violation of TS 6.8.1.a and therefore dispositioned this finding as a Non-Cited Violation.

Discussion

During review of completed modification work to install a new vent valve on the Unit 2 West RHR train recirculation line in the West RHR heat exchanger room, the inspectors identified that the weld attaching the new vent line was not performed per the approved weld detail. Step 4.8 of Job Order 02127061-03 required that the first weld off the elbowlet conform to Weld Detail 1 of Isometric Drawing INT-2-RH-25. The weld was supposed to be a "t x 2t" fatigue resistant weld, (where "t" is the nominal pipe wall thickness and the weld leg along the pipe side of the weld is equal to twice the weld leg dimension), as described in Electric Power Research Institute (EPRI) Report TR-111188, "Vibration Fatigue Testing of Socket Welds." However, the weld actually installed was a standard "t x 1.09t" fillet weld. The inspectors identified this error after the weld had already been accepted by the licensee's quality control (i.e., performance verification) inspection process and the pump was returned to service. The inspectors examined the weld with the licensee's welding engineer and verified that the weld was not performed correctly.

The inspectors noted that high cycle vibration induced fatigue of socket type welds to small bore piping is now one of the most dominant piping failure modes in the nuclear power industry. A substantial amount of industry operating experience and other documents have been written describing fatigue-related failures in piping systems. According to EPRI, vibration fatigue is the leading cause of piping failures at nuclear power plants in the United States, accounting for more than one-third of all piping failures. Many of these weld failures resulted in unisolable leaks from the reactor coolant system pressure boundary or adversely affected the availability of systems important to plant safety. In response to this concern, the design engineer for this modification specified the use of a "t x 2t" weld consistent with current industry practice. Although the standard fillet weld actually used for this installation met the applicable American Society of Mechanical Engineers (ASME) Code requirements, it was not per the approved design which evaluated and took credit for the fatigue resistant weld.

The inspectors determined that the licensee's failure to follow procedural instructions for installing this modification to the Unit 2 West RHR system is a licensee performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting issue of Human Performance. The inspectors reviewed the condition evaluation associated with this performance deficiency and noted that while it addressed the welder's failure to follow the job order instructions and to review the weld detail prior to performing the weld, as well as the failure of two maintenance supervisors to adequately review the work instructions with the welder, it did not address the failure of the performance verification process that accepted the non-conforming weld. In response to the inspectors' questions, the licensee modified the condition evaluation to include the performance verification error.

Analysis

The inspectors assessed this finding using the SDP and concluded that this issue was associated with the Mitigating Systems cornerstone and adversely affected the cornerstone objective. Specifically, the inspectors determined that the "t x 1.09t" weld would be more susceptible to vibration induced fatigue failure than a "t x 2t" weld, and if this condition were not corrected it could lead to a premature failure of the weld, affecting the function and integrity of the RHR system. This finding was associated with the Design Control and Human Performance attributes. The inspectors noted that the licensee planned to install this same modification on all four RHR pump trains. The inspectors also reviewed the examples of work in progress findings contained in NRC IMC 0612, Appendix E, "Examples of Minor Issues," and concluded that this finding was of more than minor significance because the pump was returned to service with the non-conforming condition. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was a licensee performance deficiency of very low safety significance because the finding: (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; (4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

Enforcement

Technical Specification 6.8.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 recommends, in part, procedures for the control of maintenance, repair, replacement, and modification work. Contrary to the above, on July 16, 2003, the licensee failed to correctly implement the requirements of Job Order 02127061-03, a maintenance procedure written to control a modification to the Unit 2 West RHR system. Specifically, the licensee failed to correctly install the first weld of the high point vent assembly on the Unit 2 West RHR train per Weld Detail 1 of Isometric Drawing INT-2-RH-25 as required by step 4.8 of the job order. This resulted in the installation of a modification to the system that did not conform to the approved design. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000316/2003010-04). The licensee entered this violation into its corrective action program as CR 03204050.

1R20 Refueling and Outage Activities (71111.20)

.1 Unit 2 Forced Outage

a. Inspection Scope

On August 13, 2003, the licensee entered a Unit 2 forced outage period in order to repair a steam leak from a feedwater system check valve due to a failed gasket. The licensee entered Mode 4 (Hot Shutdown) and maintained reactor coolant system temperature at approximately 205°F to inspect and repair affected equipment. The licensee subsequently entered Mode 5 (Cold Shutdown) when the 2AB EDG voltage regulator replacement work scope was added to the forced outage. The forced outage ended on August 29, 2003.

The inspectors evaluated the conduct of forced outage activities to assess the control of plant configuration and management of shutdown risk. The inspectors reviewed configuration management to verify that the licensee maintained defense-in-depth commensurate with the shutdown risk plan and reviewed outage work activities to ensure that correct system lineups were maintained for key mitigating systems. The inspectors interviewed operations, engineering, work control, radiological protection, and maintenance department personnel and reviewed selected procedures and documents.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed portions of the following six surveillance testing activities and/or reviewed the test results to determine whether risk significant systems and equipment were capable of performing their intended safety function and to verify that testing was conducted in accordance with applicable procedural and TS requirements.

- 01-OHP-4030-STP-050W, "West Residual Heat Removal Train Operability Test Modes 1-4"
- 01-OHP-4030-109-007W, "West Containment Spray System Operability Test"
- 01-OHP-4030-116-020E, "East Component Cooling Water Loop Surveillance Test"
- 02-OHP-4030-052W, "West Centrifugal Charging Pump Operability Test"
- 02-OHP-4030-017E, “East Motor Driven Auxiliary Feedwater System Test”
- 02-IHP-4030-STP-510, "Train 'A' RPS [Reactor Protection System] and ESF [Engineered Safety Features] Reactor Trip Breaker and SSPS [Solid State Protection System] Automatic Trip/Actuation Logic Functional Test"

The inspectors reviewed the test methodology and test results in order to verify that equipment performance was consistent with safety analysis and design basis assumptions. In addition, the inspectors verified that surveillance testing problems were being entered into the corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary modification 12-TM-03-65-R0, "Modify the Control Room Air Conditioner Condenser Essential Service Water (ESW) Flow Control Valve," and verified that the installation was consistent with design modification documents and that the modification did not adversely impact system operability or availability.

The inspectors verified that configuration control of the modification was correct by reviewing design modification documents and confirmed that appropriate post-installation testing was accomplished. The inspectors interviewed engineering and operations department personnel and reviewed the design modification documents and 10 CFR 50.59 evaluations against the applicable portions of the UFSAR.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs (71122.03)

.1 Reviews of Radiological Environmental Monitoring Reports and Data

a. Inspection Scope

The inspectors reviewed the Annual Radiological Environmental Operating Reports for calendar years 2001 and 2002, and the results of monthly radiological environmental monitoring analyses for 2003 thru June 2003. The inspectors also reviewed the results of the last two land use censuses, changes made to the Offsite Dose Calculation Manual (ODCM) in 2001 and 2002 relative to the radiological environmental monitoring program (REMP) and the results of the vendor laboratory inter-laboratory comparison program for 2001 and 2002. These reviews were conducted to verify that the REMP was implemented as required by TSs and the ODCM, and to verify that any changes to the program did not affect the licensee's ability to monitor the impact of radioactive effluents on the environment. Additionally, the inspectors evaluated the current locations of the environmental monitoring stations and the types of samples collected from each location to determine if they were consistent with the ODCM and NRC guidance in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Plants;" Regulatory Guide 4.8, "Environmental TSs for Nuclear Power Plants;" and an associated NRC Branch Technical Position.

These reviews represented three inspection samples; two samples related to the Radiological Environmental Operating Reports and ODCM, and one sample related to ODCM changes and land use censuses.

b. Findings

No findings of significance were identified.

.2 Radiological Environmental Monitoring Station and Meteorological Tower Walkdowns

a. Inspection Scope

The inspectors walked down all six onsite environmental air sample monitoring stations to determine whether they were located as described in the ODCM, to assess equipment material condition and operability, and to verify that monitoring station orientation relative to plant effluent release points, vegetation growth control, and equipment configuration allowed for the collection of representative samples. The primary and back-up meteorological towers were also walked down by the inspectors to verify that the towers were sited adequately and that instrumentation was installed consistent with Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants."

Meteorological data readouts and atmospheric stability information provided by the plant process computer were verified to be operable and data recording capabilities were discussed with the licensee's environmental staff to verify that meteorological data was sampled and compiled consistent with the aforementioned guide.

These reviews represented two inspection samples: One sample for air sampling station walkdowns; and one sample for meteorological tower equipment operability.

b. Findings

No findings of significance were identified.

.3 Reviews of Radiological Environmental Monitoring Equipment Maintenance and Testing

a. Inspection Scope

The inspectors reviewed environmental air sample station pump and meteorological tower equipment calibration and maintenance records for 2002 thru June 2003 and associated material and test equipment calibration records, to verify that the testing and maintenance programs for this equipment were implemented consistent with procedural requirements and industry standards. The most recent calibration records for the rotameters used by the licensee to calibrate air sample pumps were reviewed to verify that instrument certifications met industry standards and had traceability to the National Institute of Standards and Technology. The inspectors discussed air sample pump maintenance practices with the licensee's environmental staff and reviewed a newly developed procedure to assess the adequacy of the routine preventive maintenance program for environmental air sampling equipment.

These reviews represented two partial inspection samples; one sample for the calibration and maintenance of air samples, and one sample for the calibration and maintenance of meteorological equipment.

b. Findings

No findings of significance were identified.

.4 Reviews of REMP Sample Collection and Laboratory Analyses

a. Inspection Scope

The inspectors accompanied a REMP technician and observed the collection and replacement of air particulate filters and charcoal cartridges at each of the licensee's six onsite environmental air sampling stations to determine whether samples were collected in accordance with the sampling procedure and to determine if appropriate practices were used to ensure sample integrity and chain-of-custody. Sampling practices at one of two municipal drinking water treatment facilities were discussed with a water treatment facility worker to verify the adequacy of the sampling method and location. The inspectors also observed the REMP technician complete pump sampling train leak checks to verify that they were accomplished consistent with procedures and were adequate to ensure no in-leakage paths existed which could impact sample integrity. The inspectors reviewed the vendor inter-laboratory comparison and internal cross-check program results for 2001 and 2002, and reviewed lower limit of detection values achieved by the vendor laboratory for various sample media to assess the analytical detection capabilities of the contract laboratory used by the licensee to analyze its environmental samples. These activities were conducted to determine if the radiological environmental sample analysis and inter-laboratory comparison programs were implemented consistent with the ODCM and industry standards, and to verify that the vendor was capable of performing adequate radiological measurements. Additionally, the inspectors discussed with environmental department management its plans to revise the ODCM to better reflect the current inter-laboratory comparison program and plans to enhance its air sample pump leak check methods.

These reviews represented two inspection samples; one sample for the collection of environmental samples, and one sample related to laboratory analytical capabilities.

b. Findings

No findings of significance were identified.

.5 Unrestricted Release of Material From Radiologically Controlled Areas (RCAs)

a. Inspection Scope

The inspectors evaluated the licensee's procedures and practices for the unrestricted release of material from RCAs and for the survey of personnel leaving the RCA and the site. Specifically, the inspectors reviewed the licensee's personnel survey and unconditional release program to verify that: (1) radiation monitoring instrumentation used to perform surveys of personnel and for unrestricted release of materials and equipment were appropriate; (2) instrument sensitivities were consistent with NRC guidance contained in Inspection and Enforcement Circular 81-07, "Control of Radioactively Contaminated Material" and Health Physics Positions in NUREG/CR-5569, "Health Physics Positions Database" for both surface contaminated material and material in volumetric form; (3) criteria for survey and unconditional release conformed to NRC requirements; and (4) licensee procedures were technically sound and provided adequate guidance for conducting unconditional release surveys.

Additionally, the inspectors reviewed the circumstances associated with the inadvertent release of a contaminated valve outside the RCA on October 28, 2002, including the licensee's condition evaluation of the incident and radiation protection procedures governing the unconditional release program.

These reviews represented two inspection samples related to procedures, practices, and instrumentation for the unrestricted release of material.

b. Findings

Introduction

A finding of very low safety significance (Green) and an associated Non-Cited Violation (NCV) of TS 6.8.1.a were identified when a second survey of a valve that was previously surveyed and unconditionally released from the radiologically controlled area (RCA) identified that the valve was contaminated.

Description

On October 28, 2002, a small check-valve that was previously removed from the Unit 2 Volume Control Tank (VCT) system was surveyed by a radiation protection technician (RPT) and found to be free of contamination. The valve, which supplies hydrogen gas to the head space of the VCT to scavenge oxygen from the reactor coolant, was to be shipped offsite to a vendor for inspection. A small article monitor (gamma detector) used to survey the valve initially did not alarm, so the valve was deemed contamination free and unconditionally released from the RCA. The valve was not surveyed with a frisker to check for beta-gamma contamination nor was it dismantled and inaccessible surfaces surveyed for contamination even though the valve had internal surfaces shielded by more than 3/8-inch of steel.

After its unconditional release, the valve was temporarily stored in a maintenance shop outside the RCA, but within the protected area, where it remained for 4 days until retrieved by an engineer and brought to his office. The engineer partially disassembled the valve, visually inspected it, and used paper toweling to wipe down residue on the valve seat later discovered to be contaminated. The valve remained on the engineer's desk for 3 days until the engineer decided to contact the radiation protection staff and request a second survey before the valve was shipped offsite. Radiological surveys performed on November 4, 2002, using a Geiger-Mueller survey instrument identified low levels of both fixed and non-fixed (smearable) contamination ranging up to about 20,000 disintegrations per minute from residue buildup on the valve seat and piston. The contaminant was determined to be Carbon-14; a pure beta emitter which was not detectable using the small article monitor.

The licensee's condition evaluation determined that the initial radiological surveys performed on October 28, 2002, were inadequate because the valve was not disassembled for more thorough surveys with appropriate instrumentation. Additionally, the licensee failed to recognize that contaminants from pure beta emitters was possible on VCT system components although no contact with contaminated fluids occurred. The licensee's investigation later found that carbon-14 contaminated gas was prevalent in the head space of the VCT due to neutron interaction with oxygen in the reactor coolant system.

Analysis

The inspectors determined that the failure to perform an adequate survey of the VCT system check valve prior to an unconditional release from the radiologically controlled area (RCA) which was later found to be contaminated was a performance deficiency warranting a significance evaluation.

The Public Radiation Safety cornerstone was impacted by this issue. The inspectors also concluded that this finding affected the cross-cutting area of Human Performance. The inspectors reviewed the samples of minor issues in Inspection Manual Chapter (IMC) 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues,” and determined that there were no examples similar to this issue. The inspectors concluded that the finding was of more than minor risk significance in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” since the finding was associated with the Program and Process and Human Performance attributes of the Public Radiation Safety Cornerstone and adversely affected the cornerstone objective to ensure adequate protection of the public health and safety from exposure to radioactive materials released or potentially released into the public domain since the licensee failed to prevent the inadvertent release and loss of control of licensed material outside the RCA that could have potentially caused radiation dose to non-occupational workers had the engineer not decided to contact the radiation protection staff for another survey. The issue also involved an occurrence in the licensee's radioactive material control program that was contrary to the licensee's procedure that governs its unconditional release program.

Utilizing IMC 0609, Appendix D, "Public Radiation Safety SDP," the inspectors determined that this finding (1) involved radioactive material control but did not involve transportation, (2) public radiation exposure was not greater than 0.005 rem, and (3) the licensee did not have more than five radioactive material control occurrences in the previous 8 quarters. Consequently, the finding screened out as Green and was considered to be of very low safety significance.

Enforcement

Technical Specification 6.8.1.a requires, in part, that procedures be established, implemented and maintained that cover the activities recommended in Regulatory Guide 1.33, “Quality Assurance Program Requirements,” Revision 2, Appendix A, “Typical Procedures for Pressurized Water Reactors and Boiling Water Reactors,” which includes procedures for the control of radioactivity for limiting materials released to the environment and limiting personnel exposure such as radiation survey procedures and contamination controls procedures. Section 4.3.5 of licensee procedure 12-THP-6010-RPP-301, "Radiation Protection Actions for Restricted Area Material Control," Revision 0, requires that items not be unconditionally released from a restricted area unless the item contains no loose surface contamination and no readings above background count rates are detected. Section 4.3.1 and Figure 3 of procedure 12-THP-6010-RPP-301 exclude items with an internal surface that may be shielded by more than 3/8-inches of steel from being counted (surveyed) in a small article monitor. Section 4.3.2 requires that items not to be surveyed using a small article monitor be smeared for loose surface contamination and surveyed for beta-gamma contamination using a frisker, and Section 4.1.1 requires that such items be disassembled and surveyed if inaccessible surfaces exist. The failure to disassemble the valve and survey its inaccessible surfaces and to check for beta-gamma contamination using a frisker, which led to the unconditional release of the contaminated valve from the RCA on

October 28, 2002, was a violation of TS 6.8.1.a. However, because this violation was associated with a finding of very low safety significance and because the finding was entered into the licensee's corrective action program, this violation is being treated as a Non-Cited Violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000315/316/2003010-05). This violation was entered into the licensee's corrective action program as CR 02308023. To address this issue, the licensee performed a thorough extent of condition evaluation to ensure that contaminated residue and paper toweling were identified, including radiation surveys performed in offsite areas and of personal items located outside the RCA.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed condition reports generated in 2002 into July 2003 that related to REMP and radioactive material control issues. The results of a REMP self-assessment completed in February 2003 was also reviewed. These reviews were conducted to determine if the licensee adequately assessed the effectiveness of its REMP and to determine if the licensee, through the corrective action program, identified individual problems and trends, evaluated contributing causes and extent of condition, and developed adequate corrective actions. Additionally, several potential radioactive material control incidents that involved contaminated items identified in areas other than those intended for the material and which occurred during the 18 months preceding the inspection were reviewed to assess their significance, root and contributing causes, and the adequacy of the licensee's corrective actions.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Safety System Unavailability

a. Inspection Scope

The inspectors verified the following performance indicators for both units:

- Safety System Unavailability - Emergency AC Power System
- Safety System Unavailability - High Pressure Injection System
- Safety System Unavailability - Residual Heat Removal System
- Safety System Unavailability - Auxiliary Feedwater System

The inspectors reviewed operating logs, maintenance history, and surveillance test history for unavailability information for these systems from October 2002 to June 2003. The inspectors also verified the licensee's calculation of required hours for both units and evaluated applicable safety system equipment unavailability against the performance indicator definitions. The inspectors interviewed engineering and operations staff to determine whether the performance indicator data was collected and reported consistent with the guidance contained in NEI [Nuclear Energy Institute] 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2.

These reviews represented eight inspection samples; two samples for each unavailability performance indicator reviewed (one sample for each unit).

b. Findings

No findings of significance were identified.

.2 Radiological Effluent TS (RETS)/ODCM Radiological Effluent Occurrence

a. Inspection Scope

The inspector reviewed data associated with the RETS/ODCM performance indicator to determine if the indicator was accurately assessed and reported since last reviewed in September 2002. Specifically, the inspector reviewed the licensee's condition report database and selected condition reports generated between September 2002 and July 2003, to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspector also selectively reviewed gaseous and liquid effluent release data and the results of associated off-site dose calculations and quarterly performance indicator verification records generated between September 2002 and June 2003. Additionally, performance indicator data collection and analyses methods were discussed with the data steward for this performance indicator to determine if the process was implemented consistent with industry guidance in NEI 99-02.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Some minor issues entered into the licensee's corrective action system as a result of inspectors' observations are included in the list of documents reviewed which are attached to this report.

b. Findings

No finding of significance were identified.

.2 Annual Sample Review

a. Inspection Scope

The inspectors selected the following two issues for in-depth review:

- CR 03032004, "Discovered Knife Switches Pulled During Auxiliary Transformer Sudden Pressure Trip Switch Functional Testing Caused Unit 1 CD EDG to be Inoperable"

- CR 03009018, "2-WCR-912 Went to an Intermediate Position While Attempting to Isolate 2-WCR-913 for Maintenance"

On February 1, 2003, the Unit 1 CD EDG was declared inoperable when a defeated load shed circuit for the engine was discovered during deluge testing of the newly installed Unit 1 main transformer concurrently with sudden pressure relay testing. The sudden pressure relay trip switch functional test procedure was revised in June 2002 to allow testing of the Unit 1 main transformer sudden pressure relays in Modes 3 and 4. Previously, the testing was only allowed in Modes 5, 6, or defueled. This was the first time the option to perform this testing in Modes 3 or 4 was used.

On January 9, 2003, while attempting to place a clearance for replacement of a control air solenoid for the non-essential service water inner supply containment isolation valve to one of the lower containment ventilation units, operators noted that about 5 minutes after closing the valve, it would come off of its fully closed seat and go to an intermediate position. This was verified locally and was repeated three times with the same result in different combinations of valve closures.

The inspectors verified the following attributes during their review of the licensee's corrective actions for the above condition reports and other related condition reports:

- consideration of the extent of condition, generic implications, common cause and previous occurrences;
- classification and prioritization of the resolution of the problem, commensurate with safety significance;
- identification of the root and contributing causes of the problem; and
- identification of corrective actions which were appropriately focused to correct the problem.

The inspectors discussed the corrective actions and associated condition report evaluations with site personnel.

b. Findings and Observations

No findings of significance were identified. However, the inspectors had the following observations regarding the licensee's root cause evaluation and corrective actions for CR 03032004.

The event (i.e., Unit 1 CD EDG rendered inoperable) was discovered on February 1, 2003, and a Category 2 (i.e., significant condition adverse to quality) condition report was written on the same day. A due date for the root cause evaluation was set for March 19, 2003. The root cause evaluation was completed and approved on August 15, 2003. The inspectors noted that neither the scheduled due date nor the actual completion date met the licensee's expectations for timeliness. Per a memorandum from the Site Vice President dated August 9, 2002, it was expected that root cause evaluations be completed with quality within 30 days for Category 1 and 2 condition reports.

Four corrective actions to prevent recurrence (CATPR) were identified in the root cause evaluation. One of the CATPRs was completed on June 1, 2003. The scheduled due dates for the other three CATPRs were December 30, 2003; June 30, 2004; and June 5, 2005. The inspectors noted the neither the scheduled due dates nor the actual completion date for the one completed CATPR met the licensee's expectations for timeliness. Per the Site Vice President memorandum, it was expected that CATPRs associated with Category 1 and 2 condition reports be completed within 60 days.

.3 Cross-Reference to Problem Identification and Resolution Observations from Findings Documented Elsewhere in the Report

In reviewing documentation associated with the licensee's Flooding Evaluation in Section 1R06, the inspectors identified several condition reports dating back to 1999 and 2000 that had been inappropriately closed with no action, or closed to other condition reports which were then closed with inadequate justification. One condition report associated with the Turbine Building sump had not yet been screened for significance. The licensee initiated new condition reports to address these issues.

Section 1R12.b.1 described an event that occurred between March 5 and March 9, 2003, when the Unit 2 West MDAFWP motor was replaced. The inspectors reviewed the licensee's root cause evaluation of this event, and determined that this root cause evaluation incorrectly identified the cannibalization of the motor in 2002 as a root cause instead of a contributing factor, and incorrectly identified maintenance performance deficiencies as a contributing factor rather than a root cause. Analysis of the sequence of events indicated that the performance deficiencies associated with measuring, machining, and installing an appropriate coupling had the greatest impact on the return to service time, and that the bearing replacement due to cannibalization did not substantially delay the motor replacement. A new condition report was initiated to re-evaluate this root cause evaluation.

Section 1R14.1 described an event that occurred on February 18, 2003, and a Category 2 (i.e., significant condition adverse to quality) condition report was written on the same day. A due date for the root cause evaluation was set for March 21, 2003. The root cause evaluation was completed and approved on August 13, 2003. The inspectors noted that the actual completion date did not meet the licensee's expectations for timeliness. Per a memorandum from the Site Vice President dated August 9, 2002, it was expected that root cause evaluations would be completed with quality within 30 days for Category 1 and 2 condition reports.

4OA3 Event Follow-up (71153)

.1 (Closed) LER 50-316/2003-003-00: Unit 2 TS 3.7.1.2 Limiting Condition for Operation Exceeded for Auxiliary Feedwater System.

As discussed in Sections 1R12.b.1 of this report, a finding of very low safety significance was self-revealed when maintenance craftsmen failed to accurately measure, machine and install a replacement coupling during a planned maintenance activity on the Unit 2 West motor driven auxiliary feedwater pump which resulted in the unavailability of the pump significantly beyond the original 18-hour planned maintenance period. The licensee was granted enforcement discretion for TS 3.7.2.1.a to preclude a plant shutdown. The inspectors identified no other issues of significance during this review. This LER is closed.

.2 (Closed) LER 50-315/1999-030-00: Improper Use of Clarifications Results in Violations of Two TSs.

On January 12, 2000, the licensee reported two examples where incorrect TS clarifications resulted in violations of TS requirements. The first example related to a condition when the Unit 1 reactor vessel head was in place on the vessel flange, but was interpreted to be removed because it was not bolted down, permitting continuing operation of high pressure pumps which could discharge water into the reactor vessel. This condition occurred on one occasion on April 1, 1989, for a period of 31 minutes.

The second example involved de-energizing all three undervoltage relays for the 4 kV bus associated with the Unit 1 AB EDG at a time when the EDG was inoperable, but the plant operational mode required two of three relays to remain operable. The incorrect interpretation was made that, with the EDG not operable to receive a loss of voltage signal, the relays were not required to be operable. This condition occurred for a short duration on two occasions, one in 1991 and another in 1993. These issues were considered to be of minor significance and are not subject to formal enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The LER was reviewed by the inspectors and no findings of significance were identified. This LER is closed.

.3 (Closed) LER 50-315/2003-002-00: Technical Specification 3.3.3.1 Required Special Report for Inoperable Radiation Monitor 1-MRA-1701.

The licensee provided this Special Report as required by TS 3.3.3.1, Action 22B for a steam generator power operated relief valve outlet radiation monitor that was inoperable for greater than 7 days. The licensee corrected the cause for the monitor's failure. This event did not constitute a violation of NRC requirements. This LER is closed.

.4 (Closed) LER 50-315/316/2003-003-00: Dual Unit Manual Trip Due to the Failure of the Intake Traveling Screens and Failure to Comply with TS 3.8.1.1.

On April 24, 2003, operators manually tripped Unit 1 and Unit 2 in response to lowering condenser vacuum, degrading main feedwater system conditions, and indications of traveling screen fouling based on high differential pressures. The Emergency Plan was activated at the Alert level due to degraded Essential Service Water (ESW) flow to the EDGs. The licensee determined that an influx of fish had damaged the traveling screens and affected both the circulating water and ESW systems. The NRC reviewed the circumstances surrounding this event in NRC Inspection Report 50-315/316-03-08(DRP) and concluded that while the licensee effectively responded to the fish intrusion event within the significant limitations imposed by the available procedural guidance and Control Room indications, the three findings identified in the report clearly indicate that the licensee failed to act upon several previous opportunities to be prepared for and minimize the impact of this type of event. The inspectors determined that the information provided in LER 50-315/316-2003-003-00 did not raise any new issues or change the conclusions reached during the previous inspection. The inspectors identified that the licensee did not report the failure of operators to meet the requirement in TS 3.8.1.1.e to verify the availability of off-site power sources within 1 hour. However, because this issue did not impact the NRC's ability to perform its regulatory function. This issue was considered to be minor. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

.1 Section 1R12.b.1 of this report describes a finding in which maintenance craftsmen failed to appropriately measure, machine and install a coupling on the Unit 2 West MDAFWP motor. The inspectors concluded that this finding affected the cross-cutting area of human performance.

.2 Section 1R14.1 of this report describes a finding in which a partial loss of off-site power to both operating units resulted from the inadequate control of testing and procedures for testing of 345 kV switchyard breakers. The inspectors concluded that this finding affected the cross-cutting area of human performance.

.3 Section 1R19 of this report describes a finding in which maintenance craftsmen failed to follow procedural instructions for installing a modification to the Unit 2 West RHR system. The inspectors concluded that this finding affected the cross-cutting issue of human performance.

.4 Section 2PS3.5 of this report describes a finding in which radiation protection technicians failed to conduct an adequate radiological survey of a valve prior to its unconditional release outside the radiologically controlled area. The inspectors concluded that this finding affected the cross-cutting area of human performance.

4OA5 Other Activities

.1 Review of Institute of Nuclear Power Operations (INPO) Assessment Report

The inspectors completed a review of the INPO report for the D.C. Cook Nuclear Plant assessment conducted in July 2003. During this review, the inspectors did not identify any new safety significant issues.

4OA6 Meetings

.1 Resident Inspectors' Exit Meeting

The inspectors presented the inspection results to Mr. M. Nazar and other members of licensee management at the conclusion of the inspection on October 10, 2003. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was examined during this inspection, but is not specifically discussed in this report.

.2 Interim Exit Meetings

The results of the Radiological Environmental Monitoring Program Inspection were presented to Ms. S. Simpson and other members of licensee management at the conclusion of the inspection on August 1, 2003. The licensee acknowledged the findings presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following violation of very low significance was identified by the licensee and was a violation of NRC requirements which meet the criteria of Section IV of the NRC Enforcement Policy for being dispositioned as a Non-Cited Violation.

Cornerstone: Mitigating Systems

10 CFR 50, Appendix B, Criterion III, "Design Control," requires that the applicable regulatory requirements and design basis are correctly translated into specifications, drawings, procedures and instructions. 10 CFR 50.55a(g)(4) requires that pressurized water reactor components classified as American Society of Mechanical Engineers (ASME) Code Class 1, Class 2 and Class 3 meet the requirements of ASME Section XI. In temporary modification 2-TM-00-57-R1 installed on April 12, 2002, and temporary modification 2-TM-03-48-R0 installed on June 21, 2003, the licensee failed to incorporate the applicable regulatory requirements of the 1989 Edition of ASME Code Section XI, Article IWA-4000, associated with flaw evaluation, flaw removal, and component repair into applicable specifications, drawings, procedures, and instructions for repair to feedwater check valve 2-FW-118-4, an ASME Class 2 component. Furthermore, the licensee's repair method, a temporary leak seal enclosure, was not a

recognized repair method identified in Paragraph IWA-4130 of Section XI and relief had not been granted pursuant to 10 CFR 50.55a(g)(6)(i). The failure to incorporate the applicable regulatory requirements into these temporary modification packages was a violation of 10 CFR 50, Appendix B, Criterion III. The licensee entered this violation into their corrective action program as CR 03230032.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
J. Carlson, Environmental Manager
P. Cowan, System Engineering Manager
H. Etheridge, Regulatory Affairs Specialist
M. Finissi, Plant Manager
J. Giessner, Plant Engineering Director
J. Kobyra, Learning Organization Director
E. Larson, Work Management Director
M. Nazar, Senior Vice President
S. Simpson, Operations Director
L. Weber, Performance Assurance Director
D. Wood, Radiation Protection/Environmental Manager
J. Zowlinski, Engineering & Regulatory Affairs Director

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000316/2003010-01	FIN	Maintenance Errors Result in Delay of MDAFWP Replacement (Section 1R12.b.1)
05000315/316/2003010-02	FIN	Incorrect Sensing Line Configurations on Control Room Air Conditioning Units (Section 1R12.b.2)
05000315/316/2003010-03	FIN	Inappropriate Procedure for Testing Switchyard Breaker (Section 1R14.1)
05000316/2003010-04	NCV	Failure to Correctly Implement a Design Modification on the Unit 2 West Residual Heat Removal System Train (Section 1R19)
05000315/316/2003010-05	NCV	Failure to Conduct an Adequate Radiological Survey (Section 2PS3.5)

Closed

05000316/2003010-01	FIN	Maintenance Errors Result in Delay of MDAFWP Replacement (Section 1R12.b.1)
05000315/316/2003010-02	FIN	Incorrect Sensing Line Configurations on Control Room Air Conditioning Units (Section 1R12.b.2)

05000315/316/2003010-03	FIN	Inappropriate Procedure for Testing Switchyard Breaker (Section 1R14.1)
05000316/2003010-04	NCV	Failure to Correctly Implement a Design Modification on the Unit 2 West Residual Heat Removal System Train (Section 1R19)
05000315/316/2003010-05	NCV	Failure to Conduct an Adequate Radiological Survey (Section 2PS3.5)
50-316/2003-003-00	LER	Technical Specification LCO Exceeded for AFW System (Section 4OA3.1)
50-315/1999-030-00	LER	Improper Use of Clarifications (Section 4OA3.2)
50-315/2003-002-00	LER	Special Report for Inoperable Radiation Monitor (Section 4OA3.3)
50-315/316/2003-003-00	LER	Dual Unit Trip Due to the Failure of the Intake Traveling Screens (Section 4OA3.4)

<u>Discussed</u>

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection. Inclusion on this list does not imply the NRC inspectors reviewed the documents in their entirety but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R04 Equipment Alignment

OP-2-5129-45, "Flow Diagram CVCS-Reactor Letdown & Charging Unit No. 2," Revision 45
OP-2-5129A-29, "Flow Diagram CVCS-Reactor Letdown and Charging Unit No. 2," Revision 29
OP-2-5131-42, "Flow Diagram CVCS-Boron Make-up Units 1 and 2," Revision 42
OP-2-5143-53, Flow Diagram Emergency Core Cooling (RHR) [Residual Heat Removal] Unit 2," Revision 53
OP-2-5144-50, "Flow Diagram Containment Spray Unit No. 2," Revision 50
02-OHP-4030-205-002V, Boration System Valve Position Verification and Testing, Revision 0b
VTD-NUTT-0007, Nutall Gear Corp. (Formerly Westinghouse) Type SU high speed gear drives installation, operation, and maintenance instructions
Indiana and Michigan Power D. C. Cook Nuclear Plant Updated Final Safety Analysis Report, Chapter 9.2 Chemical and Volume Control System, Revision 18.1
Maintenance Rule Scoping Document, Chemical Volume Control System, Revision 2
Technical Specifications 3.1.1.1, Boration Control Shutdown Margin - Tavg Greater Than 200 Degrees Fahrenheit
Technical Specifications 3.1.1.3, Boration Dilution
Technical Specifications 3.1.2.2, Boration System Flow Paths - Operating
Technical Specifications 3.1.2.4, Charging Pumps - Operating
Technical Specifications 3.1.2.6, Boric Acid Transfer Pumps - Operating
Technical Specifications 3.1.2.8, Borated Water Sources - Operating
2-LDCP-5215, Add East and West Centrifugal Charging Pump Suction Line High Point Vents (Not Installed)
02-OHP-4025-001-001, Emergency Remote Shutdown, Revision 4
02-OHP-4025-LS-6, RCS Makeup, Seal Injection, and Boration with CVCS Crosstie, Revision 2
DIT No. DIT -B-01061-09, July 30, 2003, AEP Design Information Transmittal (DIT)
02-OHP-4023-ECA-0.0, "Loss of All AC Power," Revision 12
01-OHP-4021-054-001 Operation of Condensate System, Revision 12, Lineup Sheet 1 Condensate Initial Valve Lineup
OP-1-5107-69 Flow Diagram Condensate Unit No. 1, July 29, 2003
OP-1-5107A-29 Flow Diagram Condensate Unit No. 1, March 26, 2003
Procedure 02-OHP-4021-008-002, "Placing Emergency Core Cooling in Standby Readiness," Revision 12d, including the following lineup sheets
02-OHP-4021-008-002, Lineup Sheet 1, "Placing SI System in Standby Readiness (Manual Valves Outside Containment)"
02-OHP-4021-008-002, Lineup Sheet 3, "Placing SI System in Standby Readiness (Remote Operated Valves, Control Room)"
02-OHP-4021-008-002, Lineup Sheet 4, "Placing SI System in Standby Readiness (Remote Operated Valves, CAS Panel)"

02-OHP-4021-008-002, Lineup Sheet 5, “Placing RHR System in Standby Readiness (Manual Valves Outside Containment)”
02-OHP-4021-008-002, Lineup Sheet 7, “Placing RHR System in Standby Readiness (After RHR is Removed From Service)”
02-OHP-4021-009-001, "Placing the Containment Spray System in Standby Readiness," Revision 7a
Clearance Permit 2032849, "Spray Additive Tank to East Containment Spray Pump Eductor 2-OME-170E Inlet Isolation Valve," August 6, 2003
Clearance Permit 2032850, "Containment Spray Additive Tank," August 6, 2003
Condition Report 01064039, "Boric Acid Is Leaking from Plug," March 5, 2001
Condition Report 01345008, "Approximately ½ Cup of Dry Boric Acid on 2-IPI-220-V1 Indicates a Packing Leak Even After It Was Cleaned," December 11, 2001
Condition Report 03064051, "2-IPX-221-V1 Leakby and Pipe Plug Leakage," March 5, 2003
Condition Report 03251016 (1), "1-RH-163W Unit 1 West RHR Train High Point Vent Was Installed and Returned to Operations with No Label Installed," September 8, 2003

1R05 Fire Protection

D. C. Cook Nuclear Plant Fire Hazards Analysis, Units 1 and 2, Revision 10
D. C. Cook Nuclear Plant Updated Final Safety Analysis Report, Section 9.8.1, "Fire Protection System," Revision 18
D. C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook, February 1995
D. C. Cook Nuclear Plant Administrative Technical Requirements Manual, Revision 32
PMP-2270-CCM-001, Control of Combustibles, Revision 1
PMP-5020-RTM-001, Restraint of Transient Material, Revision 1
PMP-2270-WBG-001, Welding, Burning and Grinding Activities, Revision 0b
PMI-2270, Fire Protection, Revision 26
12-PPP-2270-066-001,Portable Fire Extinguisher Inspections, Revision 0b
12-PPP-4030-066-021, Inspection of Fire Dampers Protecting Safety-Related Areas, Revision 1c
Drawing No. 12-5974, Fire Hazards Analysis Mezzanine Floor, El. 609'-0," Revision 8
Drawing No. 2-4037, Auxiliary Building Unit 2 Elevations 609' - 6,” 625' - 0" Electrical, Switchgear Room and Reactor Containment Areas, Revision 31
Drawing No. 1-4034, Auxiliary Building Unit 1 Elevations 609' - 6," 625' - 0" Electrical Switchgear Room and Reactor Containment Areas, Revision 26
Drawing No. OP-2-5153G -7, Flow Diagram Fire Protection CO2 Lower 4Kv Areas Unit 2, Revision 7
Drawing No. OP-1-5153E, Flow Diagram Fire Protection CO2 Lower 4Kv Areas Unit 1, Revision 4

1R06 Flood Protection Measure

Condition Report 03231035 "The requirement of Commitment No. 399 are potentially not being satisfied," August 19, 2003
Condition Report 03234073 "CR 99-16669 closed without all required actions taken," August 22, 2003

Condition Report 03234067 "CR 99-12376 appears to have been closed without adequate justification," August 22, 2003
Condition Report 03234074 "CR 99-29555 is a back-log CAT. X CR that should potentially be considered a Condition Adverse to Quality," August 22, 2003
Condition Report 03234071 "CR 99-13655 closed without adequate justification," August 22, 2003
Condition Report 03234058 "CR 99-08207 closure lacks proper justification," August 22, 2003
Condition Report P-99-244260 "FSAR question response 2.24 disagrees with 02-OHP 4024.218 drop 81 relating to operator actions," November 30, 1999
Condition Report P-99-29291 "NESW Pump Area Hi-Level Alarm may be required by NRC Commitment, but not yet installed," December 17, 1999
Condition Report 02088011 "Tracking CR for development of a Design Basis Document for Flood Protection," March 29, 2002
Condition Report P-99-12376 "The design bases flood elevation of 595' does not include surface run-off contribution," May 18, 1999
Condition Report P-99-29255 "Flooding Evaluation Post-Restart Recommendations," December 16, 1999
Condition Report P-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," June 24, 1999
Condition Report P-99-08207 "Calculation Supporting UFSAR Chapter 14.4.2 Statement Could Not be Found," April 13, 1999
Condition Report P-99-13655 "Calculation which defines internal flooding in the Auxiliary Building is a steady state calculation in lieu of a transient calculation," May 26, 1999
Condition Report P-98-06706 "Calculation associated with ESW Pipe Tunnel Flood Protections needs to be revised. Calc will be added to the list of EN [Engineering] calcs to be revised prior to restart.," November 10, 1998
Condition Report P-99-08123 "Nonconservative Assumption in ESW Calculation," April 12, 1999
D. C. Cook Nuclear Plant Updated Final Safety Analysis Report Section 14.4.2.7: Flooding, Revision 18
Flooding Evaluation for AEP, DC Cook Unit #2, S&L Report No. SL-5369, Revision 0, AEP Report Number NED-2000-537-REP, May 19, 2000
HELB [High Energy Line Break] Program Flooding Evaluation Report, DC Cook Unit #1, AEP Report Number NED-2000-560-REP, October 2, 2000
Question 2.24 of Amendment No. 36, "Application for Construction Permits and Operating License for the Donald C. Cook Nuclear Plant Units 1 and 2," January 30, 1973
01-OHP 4024.118 Drop 74 "Condenser Pit Flooded Level HI-HI," Revision 8
01-OHP 4024.118 Drop 75 "Condenser Pit Flooded Level High," Revision 8
01-OHP 4024.118 Drop 82 "Condenser Pit Sump Level High," Revision 8
01-OHP 4024.118 Drop 83 "Heater Drain Pump Room Sump Level High," Revision 8
02-OHP 4024.218 Drop 81 "Condenser Pit Flooded Level HI-HI," Revision 8
02-OHP 4024.218 Drop 91 "Condenser Pit Flooded Level High," Revision 8
02-OHP 4024.218 Drop 82 "Condenser Pit Sump Level High," Revision 8
02-OHP 4024.218 Drop 83 "Heater Drain Pump Room Sump Level High," Revision 8
OP-12-5125-50 Flow Diagram Station Drainage - Turbine Room Units No 1 and Unit 2, May 13, 2003

1R11 Licensed Operator Requalification

02-OHP-4023-E-0, "Reactor Trip or Safety Injection," Revision 20
02-OHP-4023-E-3, "Steam Generator Tube Rupture," Revision 9b
02-OHP-4023.ECA-3.3, "SGTR Without Pressurizer Pressure Control," Revision 6c
02-OHP-4023.FR-P.1, "Response To Imminent Pressurized Thermal Shock Condition," Revision 6d
Condition Report 03245045, “Evaluation scenario used for E Shift Period 2804 took an unexpected path”

1R12 Maintenance Implementation

System Health Report, Ventilation Control Room, 1Q03
Updated Safety Analysis Report, Section 9.10, Revision 18.1
PMI-5035, Maintenance Rule Program, Revision 9
PMP-5035-MRP-001, Maintenance Rule Program Administration, Revision 4
Letter from G. Grant, NRC to C. Bakken, AEP, Notice of Enforcement Discretion for Indiana Michigan Power company Regarding D.C. Cook, Unit 2 (NOED 03-3-003) dated March 13, 2003
OP-1-5149-42 Flow Diagram Control Room Ventilation Unit No. 1, April 24, 2002
OP-2-5149-48 Flow Diagram Control Room Ventilation Unit No. 2, April 24, 2002
Unit 1 Control Room Logs, July 4 - September 1, 2003
Unit 2 Control Room Logs, March 7, 2003
12-MHP-5021-056-001 “Motor Driven Auxiliary Feed Pump Maintenance,” Revision 10
FO-02-H-009 Performance Assurance Field Observation “2-PP-3E-MTR (East Motor Driven Auxiliary Feedwater Pump PP-3E Motor) Bearing Replacement,” August 15, 2002
Unit 2 Control Room Logs, July 4 - August 15, 2003
Condition Report 00317008, Unit 1 Control Room Ventilation, dated November 12, 2000
Condition Report 01043003, Unit 2 Control Room Ventilation, dated February 12, 2001
Condition Report 02083012, Unit 1 Control Room Ventilation, dated March 24, 2002
Condition Report 02085014, Unit 1 Control Room Ventilation, dated March 26, 2002
Condition Report 0210641, Unit 2 Control Room Ventilation, dated April 16, 2002
Condition Report 02163036, Unit 2 Control Room Ventilation, dated June 12, 2002
Condition Report 02260027, Unit 1 Control Room Ventilation, dated September 17, 2002
Condition Report 02261034, Unit 1 Control Room Ventilation, dated September 18, 2002
Condition Report 00317008, Unit 1 Control Room Ventilation, dated November 12, 2000
Condition Report 02362007, Unit 1 Control Room Ventilation, dated December 28, 2002
Condition Report 03185003, Unit 2 Control Room Ventilation, dated July 4, 2003
Condition Report 03196035, Unit 1 Control Room Ventilation, dated July 15, 2003
Condition Report 03203042, Unit 2 Control Room Ventilation, dated July 22, 2003
Condition Report 03203048, Unit 2 Control Room Ventilation, dated July 22, 2003
Condition Report 03209017, Unit 1 Control Room Ventilation, dated July 28, 2003
Condition Report 03067008 "Multiple Maintenance Errors Result in Delay for Return to Service of Unit 2 West Motor Driven Auxiliary Feed Pump,” August 12, 2003
Condition Report 03065020 "When the Unit 2 West MDAFP, 2-PP-3W was started the pump inboard oil bubbler oil level went to 0 percent level,” March 6, 2003
Condition Report 03066066 "The corrective actions to resolve CR 0212006 were ineffective,” March 7, 2003.

Condition Report 03273031 Submit an addendum to CR 03067008 - Cat 2 Root Cause for WMDAFP revising root cause #1 to reflect inconsistent Maintenance Work
Condition Report 03223052, Unit 2 South Control Room Air Conditioning Unit (2-HV-ACRA-2) Practices as the second root cause, September 30, 2003
Job Order 03171066-03, "2-FW-118-4 Temporary Leak Seal Bearing Cover Post Maintenance Leak Inspection," August 29, 2003
Job Order 03171066-17, "Section XI Repair/Replacement Plan - 2-FW-118-4 Weld Repair Excavated Areas, Reassemble Feedwater to Steam Generator Number 4 Containment Isolation Check Valve," August 21, 2003
Plant Operations Review Committee Presentation, "Unit 2 Feedwater Check Valve Through-Wall Leakage (2-FW-118-4)," August 26, 2003
NRC Generic Letter 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME [American Society of Mechanical Engineers] Code Class 1, 2, and 3 Piping," June 15, 1990
American National Standard Institute (ANSI) B16.5, "Pipe Flanges and Flanged Fittings," 1981
ASME B16.34, "Valves-Flanged, Threaded, and Welding End," 1996
ASME Boiler and Pressure Vessel Code, Section III, Division I, Article NC-3500, "Valve Design," 1998
ASME Boiler and Pressure Vessel Code, Section XI, Article IWA-3000, "Standards for Examination Evaluation," 1989 and 1992
ASME Boiler and Pressure Vessel Code, Section XI, Article IWA-4000, "Repair and Replacement," 1989 and 1992
Design Information Transmittal (DIT) B-02765-00, "To Provide Minimum Required Valve Wall Thickness Value for the 14-Inch 900 Pound Main Feed Valve 2-FW-118-4 that Is Shown on Isometric 2-FW-71, Sheet 2," August 20, 2003
Condition Report 03223132, "Repetitive Failure/Leakage on Feedwater Check Valve 2-FW-118-4 Requires Engineering Resolution to Crush the Leakage Issue," August 19, 2003
Condition Report 03230031, "Through-wall Leakage Identified on 2-FW-118-4 in 12 O'clock Stud Location on South Bearing Cover," August 18, 2003
Condition Report 03230032, "Two Furmanite Leak Seal Temporary Modification Repairs Were Unknowingly Used to Seal a Through-wall Pressure Boundary Leak on 2-FW-118-4, an ASME Class 2 Component," August 18, 2003

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

Unit 1 Control Room Logs dated July 7, 2003 from 06:00 to 18:00
Unit 2 Control Room Logs dated July 7, 2003 from 06:00 to 18:00
Unit 1 Control Room Logs dated July 7, 2003 from 18:00 to July 8, 2003 06:00
Unit 2 Control Room Logs dated July 7, 2003 from 18:00 to July 8, 2003 06:00
Unit 1 Control Room Logs dated August 4 through-7, 2003
Unit 1 Control Room Logs, August 17 - September 9, 2003
Unit 1 Control Room Logs, September 15, 2003
Unit 1 Instantaneous Core Damage Frequency Safety Monitor Online Risk Profile for July 5, 2003 at 03:59 to July 12, 2003 at 03:59
Unit 2 Instantaneous Core Damage Frequency Safety Monitor Online Risk Profile for July 5, 2003 at 03:59 to July 12, 2003 at 03:59
Unit 1 Instantaneous Large Early Release Frequency Safety Monitor Online Risk Profile for July 5, 2003 at 03:59 to July 12, 2003 at 03:59

Unit 2 Instantaneous Large Early Release Frequency Safety Monitor Online Risk Profile for July 5, 2003 at 03:59 to July 12, 2003 at 03:59
PMP-2291-OLR-001 On-Line Risk Management, Revision 4
Data Sheet 1, On-Line Risk Management Work Schedule Review and Approval Form, June 25, 2003
D. C. Cook Nuclear Plant Units 1 and 2 TS
PMP-2291-OLR-001, Work Schedule Review and Approval Form Work Week Cycle 10 (August 3 through August 9, 2003)
PMP-2291-OLR-001, On-Line Risk Management , Revision 4
August 12 - September 15, 2003
12-IHP-4030-046-227 "Unit 1 and Unit 2 Personnel Airlock Door Seal Leak Rate Surveillance," Revision 0a, September 9, 2003
12-IHP-4030-046-227 "Unit 1 and Unit 2 Personnel Airlock Door Seal Leak Rate Surveillance," Revision 0a, September 2, 2003
12-EHP-5030-001-008 "Recirculation Loop Total Leak Rate," Revision 03
Pre-Job Brief: Operation of the Unit 1 Upper Airlock Outer Door Compensatory Action, September 9, 2003
On-Line Risk Monitor Profiles for August 4 through August 8, 2003
T-1 Lookahead Production Schedule, August 31 to September 6, 2003, Cycle 47 Week 02, Train A
Limited Design Change Package 1-LDCP-5203, Installation of Socklet/Vent for U1 East Residual Heat Removal Pump 1-PP-35E
Job Order Number R0250933 Activity 01, "Perform 12IHP-4030-046-227 for 1-AIRLOCK-C612," September 9, 2003
Job Order Number R0250936 Activity 01, "Perform 12IHP-4030-046-227 for 1-AIRLOCK-C650," September 9, 2003
Job Order Number R0250634 Activity 01, "Perform 12IHP-4030-046-227 for 1-AIRLOCK-C612," September 2, 2003
Job Order Number R0250636 Activity 01, "Perform 12IHP-4030-046-227 for 1-AIRLOCK-C650," September 2, 2003
Condition Report 03257001, "During the weekly ice condenser tour it was noted that Bay 4 had a thin sheet of ice build up over 5 of the 8 intermediate deck doors," September 14, 2003
Condition Report 03243008, "U-1 Ice Condenser has ice buildup between the outer Intermediate Deck Door and the second Intermediate Deck Door of the doors for Bay 4," August 31, 2003
Condition Report 03258003, "Clarification and documentation of failed airlock door seal surveillance on September 9, 2003 for 1-AIRLOCK-C650 outer door seal," September 15, 2003
Condition Report 03262002, "After Performing work in upper containment, several workers attempted to leave upper containment, only to find that the inner airlock door would not open," September 19, 2003
Condition Report 03175042 "Increased Leakrate at 2-SI-206, 2-ICM-265 Outlet Shutoff Valve," June 24, 2003
NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Section 11, "Assessment of Risk Resulting From Performance of Maintenance Activities," Revision 2
Shift Manager's Logs, July 16, 2003 through July 17, 2003

PMP-2291-OLR-001, Data Sheet 1, "On-Line Risk Management Work Schedule Review and Approval Form Cycle 46, Week 7," July 13, 2003 through July 19, 2003
Condition Report 02335012, "On-line Risk During a Severe Weather Condition Was Not Appropriately Documented and Updated in the Safety Monitor Program," December 1, 2002
Condition Report 03023012, "Tracking Condition Report Is Requested for Implementation of Two Regulatory Commitments Cook Plant Response to NRC Conditions and Limitations on
Reactor Protection System/Engineered Safety Features Actuation System License Amendment Request," January 23, 2003
PMP 4100-SDR-001, “Plant Shutdown Safety and Risk Management,” Revision 6

1R14 Personnel Performance During Non-routine Plant Evolutions

01-OHP-4021-001-001, "Plant Heatup from Cold Shutdown to Hot Standby," Revision 30
01-OHP-4021-001-001, "Plant Heatup from Cold Shutdown to Hot Standby," Revision 31
01-OHP-4021-082-001, "4kV Busses Power Source Transfer and De-energizing and Re-energizing a Safeguards Bus," Revision 11
01-OHP-082-026, "Operation of the Load Tap Changer," Revision 1
02-OHP-4030-STP-027AB, "AB Diesel Generator Operability Test (Train B),” Revision19
Standing Order SO-2003-0001, "Authorized Switchyard Maintenance/Work Activities," Revision 0
Rapid Event Response Report, "Partial Loss of Off-site Power CR 03049046," March 4, 2003
Job Order 03049046-01, "Troubleshooting Plan for Unexpected Trip of Switchyard Breakers L1, M1, N1, K1 and BE," February 21, 2003
Root Cause Evaluation, "Partial Loss of Off-site Power When Testing 'L' Breaker in the Switchyard," July 25, 2003
Condition Report 01343015, "Discovered Emergency Diesel Generator 1AB Voltage Potentiometer Settings Incorrect," December 9, 2001
Condition Report 03039025, "The Unit 2 AB EDG Auto Voltage Regulator Pot Was Improperly Set Following a Surveillance Run," February 8, 2003
Condition Report 03049046, "Circuit Breaker BE Tripped Open Unexpectedly Resulting in a Partial Loss of Off-site Power," February 18, 2003
Condition Report 03036006, "Transformers 101AB and 101CD Were Made Inoperable After Transferring Auxiliaries," February 5, 2003

1R15 Operability Evaluations

Generic Letter 91-18, “Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions,” Revision 1
OHI-4016, “Conduct of Operations Guidelines,” Revision 8
PMP-7030-OPR-001, “Operability Determinations,” Revision 7
Condition Report 03163026 Scheduled EQ replacement of 2-FFI-230 has passed is drop dead date without being completed
Condition Report 03017039 JOA R203485-02 Failed
Condition Report 03240014 Audible Count Rate was not re-scaled as count rate decayed after the Unit 2 Shutdown of August 13, 2003
EQ Evaluation 2003-002, “EQ Evaluation for 2-FFI-230 Foxboro transmitter”
NRC NUREG-0933 Inadvertent Boron Dilution Events (Rev. 2)
NRC Generic Letter 85-05 Inadvertent Boron Dilution Events, January 31, 1995

NRC Information Notice 93-32 Nonconservative Inputs for Boron Dilution Event Analysis, April 21, 1993
NRC Information Notice 96-69 Operator Actions Affecting Reactivity, December 20, 1996
NRC NUREG-0800 Standard Review Plan Section 15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN BORON CONCENTRATION IN THE REACTOR COOLANT (PWR), Draft Rev. 2, April 1996
D. C. Cook Nuclear Plant Updated Final Safety Analysis Report, Chapter 14.1.5.2.4, Dilution During Shutdown, Revision 18.1
Letter from AEP to NRC dated March 27, 1986 regarding "Additional Information Required for Cycle 6 Reload"
NS-TMA-2273 Letter from Westinghouse Electric Corporation to Mr. Victor Stello, USNRC regarding "Boron Dilution Concerns at Cold and Hot Shutdown" dated July 8, 1980
90AE*-G-0028 W/AEP2-0061 Letter from Westinghouse Electric Corporation to Mr. Thomas A. Georgantis, AEP regarding "Revised Boron Dilution Procedure for Donald C. Cook Unit 2" dated February 12, 1990
02-OHP-4021-013-005 Visual Audio Count Rate Channel (NIS), Revision 6
02-OHP-4021-001-003 Power Reduction, Revision 17
02-OHP-4021-001-004 Plant Cooldown from Hot Standby to Cold Shutdown, Revision 28
12-OHP-4023-ES-0.1 ERG Step #10 Background Document, Revision 4
02-OHP-4024.210 Drop 3 Annunciator Panel Number 210 Flux Rod Annunciator "Source Range Hi Flux at Shutdown," Revision 8
Condition Report 02083012, "1-QH-407N (North CRAC Unit) Failed to Start," March 24, 2002
Condition Report 02163036, "An Attempt Was made to Swap Unit 2 CRAC from the North Unit (2-ACRA-1) to the South Unit (2-ACRA-2). Upon Starting the South CRAC Unit, the compressor ran for Approximately 10 Seconds and then Stopped," June 12, 2002
Condition Report 02261034, "1-HV-ACR-2 (Unit 1 South CRAC Liquid Chiller) Tripped on High Pressure Upon Startup," September 18, 2002
Condition Report 03209017, "1-HV-ACRA-1, North Control Room Air Conditioning (CRAC) Unit is Inoperable Due to Repeated Failures of the AC Unit Compressor to Start After Receiving a Start Signal from the Control Room," July 28, 2003
Condition Report 03210005, "While Swapping from North CRAC Unit to the South CRAC Unit, the South CRAC Unit Tripped," July 29, 2003
Condition Report 03212061, "1-QH-407N and 1-QH-407S Print OP-1-5149-42 Shows Compressor Condenser Water Temperature Control Valve Is Piped Different than the Print Shows," July 31, 2003

1R19 Post Maintenance Testing

Unit 2 Control Room Logs from 03:00 on July 10, 2003 to 06:10 on July 12, 2003.
Unit 1 Tech Data Book Figure 1-15.1 "Safety Related Pump Inservice Test Hydraulic Reference," Revision 80
Unit 2 Tech Data Book Figure 2-15.1 "Safety Related Pump Inservice Test Hydraulic Reference," Revision 66
Unit 1 Tech Data Book Figure 1-15.2 "Safety Related Pump Inservice Test Vibration Reference," Revision 73
Unit 2 Tech Data Book Figure 2-15.2 "Safety Related Pump Inservice Test Vibration Reference," Revision 56.
Unit 1 Tech Data Book Figure 1-19.1 "Power Operated Valve Stroke Time Limits," Revision 68

Unit 2 Tech Data Book Figure 2-19.1 "Power Operated Valve Stroke Time Limits," Revision 59
Unit 1 Tech Data Book Figure 1-19.9 "Diesel Generator Pot Settings," Revision 21
Unit 2 Tech Data Book Figure 2-19.9 "Diesel Generator Pot Settings," Revision 41
PMI-5071, Inservice Testing Attachment 1 Operability After Maintenance, Revision 1
PMP-2291-PMT-001, Work Management Post Maintenance Testing Matrices, Revision 4
Job Order R0226142 Activity 01, 2-IMO-361 / Grease Valve Stem Generic Letter 89-10 / External PM, July 1, 2003
Job Order R0230628, Activity Numbers 2 and 3, "1-PP-50E-DC, Drain & Refill Speed Increaser Reservoir"
Job Order R0230717, Activity Numbers 2, "1-PP-50E, (48wk) Lube Oil/Couplings/Cuno Filter"
Job Order Number R0236942 Activity 03 "AB DG South Air Aftercooler ESW Temp Control Valve," completed September 12, 2003
Job Order Number R0236968 Activity 03 "AB DG North Air Aftercooler ESW Temp Control Valve," completed September 12, 2003
Job Order Number R0230781 Activity 02 "East ESW Header AB DG Alternate Supply Valve," completed September 11, 2003
Job Order Number R0229214 Activity 03 "AB DG N Combustion Air Aftercooler Auto Vent Trap," completed September 12, 2003
Job Order Number R0229214 Activity 06 "AB DG S Combustion Air Aftercooler Auto Vent Trap," completed September 12, 2003
Job Order Number R0083161 Activity 02 "AB DG Phase 2 Remote Ammeter (Control Room)," completed September 11, 2003
Job Order Number R0083160 Activity 02 "AB DG Phase 3 Remote Ammeter," completed September 12, 2003
Job Order Number R0095148 Activity 03 "AB DG Full Flow Lube Oil Strainers High D/P Alarm," completed September 12, 2003
Job Order Number R0230088 Activity 06 "AB DG Control Module Air Filters Number 1 and Number 2," completed September 12, 2003
Job Order Number 00318006 Activity 02 "1-DG-121A: Replace Threaded Valve," completed September 11, 2003
Job Order Number R0229214 Activity 02 "1-T-131-5 & 6 Open/Inspect/Repair as Needed," completed September 11, 2003
Job Order Number R0229214 Activity 05 "1-T-131-5 & 6 Open/Inspect/Repair as Needed," completed September 11, 2003
Job Order Number R0073153 Activity 01 "Perform Calibration and PMT for 1-XPI-219," completed July 30, 2003
Job Order Number R0076345 Activity 11 "Calibrate Time Delay Relays for Unit 1 CD Diesel," completed July 31, 2003
Job Order Number 02156042 Activity 02 "1-SV-78-CD2 Return Tail Piece to Design Config," completed July 30, 2003
Job Order Number R0091156 Activity 03 "Calibrate Pressure Indicator," completed July 31, 2003
Job Order Number R0091155 Activity 03 "Calibrate Pressure Indicator," completed July 31, 2003
Job Order Number R0091185 Activity 03 "Calibrate Pressure Indicator," completed July 31, 2003
Job Order Number R0091184 Activity 03 "Calibrate Pressure Indicator," completed July 31, 2003

Job Order Number R0076345 Activity 10 "Calibrate Time Delay Relays for U-1 CD Diesel," completed July 31, 2003
Job Order Number R0247144 Activity 05 "Calibrate Time Delay Relays for U-1 CD Diesel," completed July 31, 2003
Job Order Number R0247144 Activity 06 "Calibrate Time Delay Relays for U-1 CD Diesel," completed July 31, 2003
Job Order Number R0209254 Activity 04 "1-QT-502-CD, Change/Sample Oil in Turbo-Charger," completed July 31, 2003
Job Order Number 03192001 Activity 03 "Repair 2-DGCD-VRCKT, DG Voltage Regulator," completed July 12, 2003
Job Order Number R0244962 Activity 01 "CD Emergency Diesel Generator," completed July 12, 2003
Job Order Number 03192001 Activity 13 "Repair 2-DGCD-VRCKT, DG Voltage Regulator," completed July 12, 2003
01-OHP-4030-STP-052E, "East Centrifugal Charging Pump Operability Test," Revision 12
02-OHP-4030-208-053A, ECCS Valve Operability Test - Train A, Attachment 1, Rev. 0a
01-OHP-4030-STP-027AB Attachment 1 "DG1AB Slow Speed Start," Revision 20, completed September 12, 2003
01-OHP-4030-STP-027AB Attachment 4 "Removing Accumulated Water From AB Diesel Generator Fuel Oil Day Tank," Revision 20, completed September 12, 2003
01-OHP-4030-STP-027AB Data Sheet 1 "DG1AB Operating Data," Revision 20, completed September 12, 2003
01-OHP-4030-STP-027AB Data Sheet 1 "DG1AB Local Operating Data," Revision 20, completed September 12, 2003
01-OHP-4021-032-008CD Attachment 6, "Operating DG1CD Subsystems" Revision 3, completed July 31, 2003
01-OHP-4030-STP-027CD Attachments 1, 2, 4 and 6, Data Sheets 1 and 2 "CD Diesel generator Operability Test (Train A)" Revision 19b, completed July 31, 2003
01-OHP-4021-032-001CD "DG1CD Operations," Revision 5a, completed July 31, 2003
01-OHP-4021-032-001CD Attachment 1, Data Sheets 1 and 2 "DG1CD Operations," Revision 5a, completed July 31, 2003
02-OHP-4030-STP-027CD Attachment 2 "CD Diesel Generator Operability Test (Train A) DG2CD Fast Speed Start," Revision 20b, completed July 12, 2003
02-OHP-4030-STP-027CD Attachment 4 "CD Diesel Generator Operability Test (Train A) Removing Accumulated Water From CD Diesel Generator Fuel Oil Day Tank," Revision 20b, completed July 12, 2003
02-OHP-4030-STP-027CD Attachment 7 "CD Diesel Generator Operability Test (Train A) Fuel Oil Transfer Pumps 2-QT-106-CD2 Quarterly and 2-QT-106-CD1 Monthly Checks," Revision 20b, completed July 12, 2003
02-OHP-4030-STP-027CD Attachment 14 "CD Diesel Generator Operability Test (Train A) Alternate ESW Supply Valve Testing," Revision 20b, completed July 12, 2003
02-OHP-4030-STP-027CD Data Sheet 1 "CD Diesel Generator Operability Test (Train A) DG2CD Operating Data," Revision 20b, completed July 12, 2003
02-OHP-4030-STP-027CD Data Sheet 2 "CD Diesel Generator Operability Test (Train A) DG2CD Local Operating Data," Revision 20b, completed July 12, 2003
02-OHP-4021-032-001CD "DG2CD Operation," Revision 6a, completed July 11, 2003
02-OHP-4021-032-001CD "DG2CD Operation," Revision 6a, completed July 12, 2003

02-OHP-4021-032-001CD Data Sheet 1 "DG2CD Operation, DG2CD Operating Data - Control Room," Revision 6a, completed July 12, 2003
02-OHP-4021-032-001CD Data Sheet 2 "DG2CD Operation, DG2CD Operating Data - Local," Revision 6a, completed July 12, 2003
12-IHP-6030-032-001 "Emergency Diesel Generator Automatic Voltage Regulatory Tuning and Adjustment," Revision 0a, completed July 12, 2003
12-EHP-6040-PER-106 "Emergency Diesel Generator Control Panel Tests," Revision 0, completed July 12, 2003
OP-2-98033-31 "Diesel Generator 2CD Excitation & Regulation and Miscellaneous Elementary Diagram," March 26, 2000
OP-2-98033-34 "Diesel Generator 2CD Excitation & Regulation and Miscellaneous Elementary Diagram," June 11, 2003
PMI-5071, Inservice Testing Attachment 1 Operability After Maintenance, Revision 1
PMP-2291-PMT-001, Work Management Post Maintenance Testing Matrices, Revision 4
Electric Power Research Institute Report TR-111188, "Vibration Fatigue Testing of Socket Welds," December 1998
Electric Power Research Institute Report TR-113890, "Vibration Fatigue Testing of Socket Welds (PWRMRP-07)," PWR Materials Reliability Project Final Report, December 1999
Institute of Nuclear Power Operations - Operations and Maintenance Reminder 424, "Small Bore Piping Connection Failures," January 7, 1998
Limited Design Change Procedure (LDCP) 2-LDCP-5205, "Add East and West RHR Pump Minimum Flow Line High Point Vents," May 23, 2002
Design Information Transmittal B-02755-00, "2-SI-121S ECCS [Emergency Core Cooling System] Flow Balance Position Impact," July 18, 2003
02-OHP-4030-STP-050W, "West Residual Heat Removal Train Operability Test Modes 1-4," Revision 9
02-OHP-4030-209-007E, "East Containment Spray System Operability Test," Revision 3
12-MHP-5021-005-003, "Chesterton Packing Removal, Installation and Adjustment," Revision 6
12-MHP-5021-001-014, "Crane-Aloyco Manual and Motor Operated Gate Valve Maintenance," Revision 5
Job Order 02127061-03, "2-LDCP-5205 Install High Point Vent in Minimum Flow Line for West RHR Pump 2-PP-35W," July 16, 2003
Job Order 02127061-04, "2-LDCP-52505 Testing for West RHR Pump 2-PP-35W," July 17, 2003
Job Order 03167019-09, "2-SI-121S - Support Operations to Obtain Open Dimension," July 21, 2003
Job Order 03216002-01, "2-CTS-119E, Disassemble, Repair to Stop Binding," August 6, 2003
Job Order C0047173-01, "Replace Air Vents in Accordance with EE-2003-0116," August 7, 2003
Isometric Drawing INT-2-RH-25, Revision 25
Isometric Drawing INT-2-RH-26, Revision 26
CR 03204050 [(1)], "The Fillet Weld Attaching the 3/4" Schedule 40 Stainless Steel Pipe to the Stainless Steel Socket Weld Elbowlet Does Not Conform to Weld Detail 1 of Drawing INT-2-RH-25," July 23, 2003
Condition Report 03261027 "1AB EDG observations during operation," September 18, 2003.

1R20 Refueling Activities

DC Cook Nuclear Plant Unit 2 TSs
D.C. Cook Nuclear Plant UFSAR, Revision 18
PMP 4100-SDR-001, “Plant Shutdown Safety and Risk Management,” Revision 6
Plant Operations Review Committee Presentation, "Unit 2 Feedwater Check Valve Through-Wall Leakage (2-FW-118-4)," August 26, 2003
Plant Operations Review Committee Presentation, "Unit 2 Feedwater Check Valve Leakage and Shutdown (2-FW-118-2)," August 13, 2003
Condition Report 03225008, "Unit 2 Completed a Controlled Power Reduction and Manual Reactor Trip Due to 2-FW-118-2 Leak," August 13, 2003
Condition Report 03227037, "2-FW-118-4 Discrepant Conditions Identified During 86-03 Examination," August 15, 2003
Condition Report 03227049, "Forced Outage Designated Work Packages Were Not Ready to Support U2F03D on August 13, 2003 When the Forced Outage Team Was Called Out for 2-FW-118-4 Feedwater Leak," August 15, 2003
Condition Report 03227074, "2-FW-118-2 Feedwater to Steam Generator Number 2 Containment Isolation Check Valve Discrepant Conditions Identified During 86-03 Examinations," August 15, 2003
Condition Report 03228004, "2-FW-118-3 Feedwater to Steam Generator Number 3 Containment Isolation Check Valve Discrepant Conditions Identified During As-found 86-03 Examinations," August 16, 2003

1R22 Surveillance Testing

Condition Report 031900061, 2-FW-160 leaks by while performing 02-OHP-4030-STP-017E, July 9, 2003
Condition Report 02136014, 2-FW-160 West MDAFW Pump Emergency Leakoff Check Valve leaked by during the performance of 2-OHP-4030-017E, May 16, 2002
Unit 1 Tech Data Book, Safety Related Pump Inservice Test Hydraulic Reference, Figure 1-15.1, Revision 80
Unit 1 Tech Data Book, Safety Related Pump Inservice Test Vibration Reference, Figure 1-15.2, Revision 73
Unit 2 Tech Data Book, Safety Related Pump Inservice Test Hydraulic Reference, Figure 2-15.1, Revision 66
Unit 2 Tech Data Book, Safety Related Pump Inservice Test Vibration Reference, Figure 2-15.2, Revision 56
Unit 2 Control Room Logs dated July 9, 2003 from 18:00 to July 10, 2003 06:00
OP-2-5106A-49, Flow Diagram Auxiliary Feedwater Unit No 2, Revision 49
01-OHP 4030-STP-052W Attachment 2 "West Centrifugal Charging Pump Operability Test," Revision 11, completed September 18, 2003
02-OHP-4030-017E, East Motor Driven Auxiliary Feedwater System Test, Revision 10
02-OHP-4030-STP-052W Attachment 2 "West Centrifugal Charging Pump Operability Test," Revision 11, completed September 18, 2003
02-IHP-4030-STP-510, “Train ‘A’ RPS and ESF Reactor Trip Breaker and SSPS Automatic Trip/Actuation Logic Functional Test,” Revision 5, Change 0
12-OHP-4021-082-018, "Racking In and Out Reactor Trip, Reactor Trip Bypass, and MG Set Output Breakers," Revision 6a

01-OHP-4030-116-020E, East Component Cooling Water Loop Surveillance Test, Revision 1a
Power Operated Valve Stroke Time Limits dated July 16, 2003
Updated Safety Analysis Report, Section 9.5, Revision 18.1
Technical Specifications Sections 3.7.3.1, 4.05, 4.7.3.1.a, and 4.7.3.1.c, Amendment 275
Flow Diagram, OP-5135, Flow Diagram CCW Pumps and CCW Heat Exchangers, Revision 41
Flow Diagram, OP-5135A, Flow Diagram CCW Safety related Loads, Revision 42
Flow Diagram, OP-5135B, Flow Diagram CCW Misc. Services Auxiliary Building, Revision 21
Flow Diagram, OP-5135D, Flow Diagram CCW Misc. Services Containment Loads, Revision 4
01-OHP-4030-STP-050W, "West Residual Heat Removal Train Operability Test Modes 1-4," Revision 9
01-OHP-4030-STP-050W, "West Residual Heat Removal Train Operability Test Modes 1-4," Revision 10
01-OHP-4030-109-007W, "West Containment Spray System Operability Test," Revision 2
01-OHP-4030-109-007W, "West Containment Spray System Operability Test," Revision 3
Shift Manager's Logs, July 17, 2003
OP-1-5143-59, Flow Diagram Emergency Core Cooling (RHR) [Residual Heat Removal] Unit 1, Revision 59
OP-1-5144-37, Flow Diagram Containment Spray Unit 1, Revision 37
Condition Report 03070029, "Procedure Enhancement Is needed for OHI-4016, 'Conduct of Operations: Guidelines, Attachment 3, In-service Test Criteria' to Provide Guidance on Preventing Preconditioning," March 11, 2003

1R23 Temporary Modifications

Temporary Modification 12-TM-03-65-R0, "Modify the CRAC Condenser ESW Flow Control Valve," July 31, 2003
Condition Report 02083012, "1-QH-407N (North CRAC Unit) Failed to Start," March 24, 2002
Condition Report 02163036, "An Attempt Was made to Swap Unit 2 CRAC from the North Unit (2-ACRA-1) to the South Unit (2-ACRA-2). Upon Starting the South CRAC Unit, the compressor ran for Approximately 10 Seconds and then Stopped," June 12, 2002
Condition Report 02261034, "1-HV-ACR-2 (Unit 1 South CRAC Liquid Chiller) Tripped on High Pressure Upon Startup," September 18, 2002
Condition Report 03209017, "1-HV-ACRA-1, North Control Room Air Conditioning (CRAC) Unit is Inoperable Due to Repeated Failures of the AC Unit Compressor to Start After Receiving a Start Signal from the Control Room," July 28, 2003
Condition Report 03210005, "While Swapping from North CRAC Unit to the South CRAC Unit, the South CRAC Unit Tripped," July 29, 2003
Condition Report 03212061, "1-QH-407N and 1-QH-407S Print OP-1-5149-42 Shows Compressor Condenser Water Temperature Control Valve Is Piped Different than the Print Shows," July 31, 2003

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs

PMP-6010-OSD-001; Offsite Dose Calculation Manual; Revisions 17 and 17a
Annual Radiological Environmental Operating Reports for 2001 and 2002
12-THP-6010-RPP-630; Collection of REMP Surface Water Samples; Revision 2b
12-THP-6010-RPP-632; Collection of Environmental Air Samples; Revision 4a
12-THP-6010-RPP-633; Collection of Environmental Radiation Dosimeters; Revision 5

12-THP-6010-RPP-634; Collection of REMP Groundwater Samples; Revision 4a
12-THP-6010-RPP-635; Collection of Milk Samples; Revision 1
12-THP-6010-RPP-637; Collection of Lake Sediment Samples; Revision 1a
12-THP-6010-RPP-638; Collection of Grape and Broadleaf Samples; Revision 3
12-THP-6010-RPP-642; Collection of Drinking Water Samples; Revision 2
SA-2003-REA-001; Self-Assessment Report of REMP; dated March 5, 2003
Condition Report 03175025; Contaminated Material Found in Clean Area; dated June 24, 2003
Condition Report 02308023; Valve Released from Restricted Area with Beta Contamination; dated November 4, 2002
Condition Report 03129059; Radioactive Material Found Outside Restricted Area on Yellow Stanchions; dated May 9, 2003
Condition Report 02133045; Radioactive Material Found Outside Restricted Area Boundary; dated May 13, 2002
Condition Report 02233051; Problem with Power Supply to Environmental Air Sampling Station; dated August 20, 2002
Condition Report 03022033; Drinking Water Samples Not Collected; dated January 22, 2003
12-THP-6010-RPP-301; Radiation Protection Actions for Restricted Area Material Control; Revisions 0 and 1
RPM-1; Guidelines for REMP Equipment Preventive Maintenance and Performance Monitoring; Revision 0
12-THP-6010-RPC-514; Calibration of the AVS-28A and AVT-100 Air Volume Totalizer; Revision 3a
Data Sheet 1 of 12-THP-6010-RPC-514 for selected air sample pumps; January 2002 - June 2003
Wyle Laboratories Calibration Certificate for Torque Watches Model 366.1M and Model 366.3M (SN 4443 and 4444); dated November 27, 2002 and October 1, 2002
12-THP-6030-IMP-333; Records of Meteorological Instrumentation Calibration; dated March 14, 2002, August 27, 2002, February 14, 2003 and April 17, 2003
Laboratory Services Certificates of Calibration for Rotameter Model 1110-08H2GIA (SN 8404H07767 and 8504H37933); dated November 21, 2002 and April 5, 2002
PMP-6010-RPP-301; Control of Material in a Restricted Area; Revision 15
RP-TB-001; Evaluation of the Use of the Bicron Small Article Monitor for Unconditional Release of Material from a Restricted Area; Revision 0

4OA1 Performance Indicator Verification

Monthly and Quarterly Dose Projections from Liquid and Gaseous Effluents; dated September 2002 - June 2003
Monthly Radiological Effluent Occurrence Summary Data; dated October 2002 - June 2003
Procedure PMP-7110-PIP-001, “Regulatory Oversight Program - Performance Indicators,” Revision 1
Cook Unit 1 and 2 - 4Q2002 - PI Data Elements (QR and CR), January 21, 2003
Cook Unit 1 and 2 - 1Q2003 - PI Data Elements (QR), April 21, 2003
Cook Unit 1 and 2 - 2Q2003 - PI Data Elements (QR), July 23, 2003
PI Data Summary Reports: Q4/2002. Q1/2003, Q2/2003
Input PI Data Reports: Q4/2002, Q1/2003, Q2/2003

PMP-7110-PIP-001 Input Data Sheets (all "Cornerstones"): Q4/2002, Q1/2003, Q2/2003
Monthly Operating Reports: October 2002 through June 2003 (AEP:NRC:2691-24 through (AEP:NRC:2691-32)
Control Room Logs:
U1 Night Shift A, December 23, 2002
U2 Day Shift A, May 25, 2002
U2 Night Shift B, May 25, 2002
U1 Night Shift B, January 15, 2003
U1 Night Shift D, February 3, 2003
U2 Night Shift D, February 5, 2003
U2 Night Shift D, February 13, 2003
U2 Night Shift A, January 25, 2003
U2 Day Shift C, January 29, 2003
U1 Night Shift A, April, 23, 2003
U1 Night Shift B, May 27, 2003
U2 Night Shift A, April 23, 2003
U2 Day Shift A, June 19, 2003
Condition Report 02281052 - Add dampers to list of components affecting DG ventilation operability
Condition Report 02340005 - Errors in NRC Drill/Exercise Performance PI Data
Condition Report 02357020 - W MFP Discharge Pressure Instrument Weld Leak
Condition Report 02146035 - U2 Scrammed 5/25/2002 - MSR Steam Leak
Condition Report 03016007 - U1 Reactor Trip due to fire in Main Xfrmer 1/15/2003
Condition Report 03026001 - Track U2 Shutdown and Reportability
Condition Report 03114044 - Fish Intrusion Causing Unit Trips and Degraded Cooling
Condition Report 03125096 - Evaluation of Underweight of Ice in Basket 11-7-1
Condition Report 03065001 - U2 Train B (W) MDAFP Motor Replacement at Power; NOED
Condition Report 03151047 (1), "Potential for Mis-reporting NRC Performance Indicator Data Identified by NRC Inspector," September 8, 2003

4OA2 Identification and Resolution of Problems

D. C. Cook Nuclear Plant Units 1 and 2 TSs
D. C. Cook Nuclear Plant Units 1 and 2 Updated Final Safety Analysis Report, Section 5.4, "Containment Isolation System"
Rapid Event Response Investigation Report: "CD Diesel Inadvertent Inoperability Due to Testing the Unit 1 Main Transformer Sudden Pressure Relays," February 2, 2003
Memo from Joseph E. Pollock, Site Vice President, Subject: "Expectations on Corrective Action Program," August 9, 2002
Condition Report P-00-06696, "Containment Isolation Valves Such as Non-essential Service Water Can Hydraulic Lock Open When Required to Close if the Outer Most Containment Isolation Valve Closes First," May 9, 2000
Condition Report 00279011, "The Evaluation for CR 00-6696 Improperly Evaluated the Possibility of Hydraulic Locking in Non-essential Service Water Containment Isolation Valves," October 5, 2000
Condition Report 03009018, "2-WCR-912 Went to an Intermediate Position While Attempting to Isolate 2-WCR-913 for Maintenance," January 9, 2003

Condition Report 03032001, "During Unit 1 Main Transformer Deluge Testing the Main Transformer Cooling Fans and the 1-TR-101AB Transformer Cooling Fans Did Not Trip Off As Expected," February 1, 2003
Condition Report 03032004, "Discovered Knife Switches Pulled During Auxiliary Transformer Sudden Pressure Trip Switch Functional Testing Caused Unit 1 CD EDG to Be Inoperable," February 1, 2003

4OA3 Event Follow-up

Licensee Event Report 05000315/99-030-00, Improper Use of Clarifications Results in Violation of Two TSs, January 12, 2000
Condition Reports P-99-22345, Several TS Clarifications (currently effective) contain statements that potentially contradicts (sic) TSs
Condition Report 99-02928, Sampling and analysis of the RCS to meet TS 4.4.7 requirements was not done from October 30, 1997 to November 23, 1997 for Unit One
Condition Report P-99-29313, Reportable Events not processed in accordance with NUREG-1022
Technical Specification Clarification No. 42, Revision 1, "Low Temperature Overpressure Protection"
Unit 1 TS 3.1.2.3, Amendment 230
Technical Specification Clarification No. 58, Revision 0, "4 KV Bus Undervoltage Relays"
Unit 1 TS 3/4.3, Table 3.3-3, Amendment 153
LER 50-315-2003-003-00, "TS 3.3.3.1 Required Special Report for Inoperable Radiation Monitor 1-MRA-1701," March, 28, 2003
LER 50-315/316-2003-003-00, "Dual Unit Manual Trip Due to the Failure of the Intake Traveling Screens and Failure to Comply with TS 3.8.1.1," June 23, 2003
NUREG 1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Revision 2
Letter from A. C. Bakken, III, Indiana Michigan Power Company to J. E. Dyer, U. S. Nuclear Regulatory Commission, Region III, Subject: "Donald C. Cook Nuclear Plant Units 1 and 2 Response to April 24, 2003, Fish Intrusion Event," April 28, 2003
PMP-2291-EXE-001, "Work Control Activity Execution Process," Revision 7
12-OHP-5030-057-001, "Screen House Vulnerability Determination," Revision 1
12-OHP-4022-057-001, "Screen House Forebay Degraded Condition," Revision 1
01-OHP-4024-123, "Annunciator #123 Response: Circulating Water," Revision 10
02-OHP-4024-123, "Annunciator #223 Response: Circulating Water," Revision 8
Shift Manager’s Logs, April 23, 2003 through April 24, 2003
Training Plan RQ-S-2811, "Screenhouse [sic] Forebay Degraded Condition," Revision 0
Training Plan RO-C-05700, "Circulating Water and Screen Wash System," Revision 2
Condition Report 03075003, "1-MRA-1701 Went Into High Alarm After Routine Source Check," March 16, 2003
Condition Report 03082008, "At 1119 on March 23, 2003, 7 Days Has Passed and MRA-1701 Is Still Inoperable Requiring Submitting a Special Report to the Commission," March 23, 2003
Condition Report 03084004, "Leads at 1-MRA-1701 Splice Box Not in Accordance with Print," March 25, 2003
Condition Report 03114044, "Large Intrusion of Lake Fish Into the Plant Circulating Water Intake Caused Both Units to Be Removed from Service and Resulted in Degraded Cooling Flows to Several Safety Related Components," April 24, 2003

Condition Report 03122099, "During Fish Intrusion Event, the Circulating Water Pumps Were Secured in and Order that Was Not the Prescribed Order in the Operating Procedure," May 2, 2003

Condition Report 03118030, "Tracking CR for Assessing Emergency Response from Alert Declaration," April 28, 2003

Condition Report 03121045, "Unit 2 Reactor Coolant System Average Temperature Was Slowly Lowering After the Trip So the Main Steam Stop Valves Were Closed to Stop the Cooldown (50 Minutes After the Trip," May 1, 2003

Condition Report 03120060, "Unit 1 Average Temperature Dropped Lower Than Expected After the Reactor Trip," April 30, 2003

Condition Report 03114018, "Both Unit Two EDGs Were Declared Inoperable at 0348 Due to Inadequate ESW Flow," April 24, 2003

Condition Report 03114035, "Unit One Did Not Comply with TS 3.8.1.1.e for Verifying Power Sources Within One Hour of Declaring Both Diesel Generators Inoperable," April 24, 2003

Condition Report 03269028 [(1)], "Two Inadequate Reportability Evaluations for Two April 24, 2003, Non-compliance Events Associated with CRs 03114018 and 03114035, Failure to Submit LER," September 26, 2003

LER 50-316/2003-003-00, "Unit 2 TS 3.7.1.2 Limiting Condition for Operation Exceeded for Auxiliary Feedwater System." May 5, 2003

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
AEP	American Electric Power
ASME	American Society of Mechanical Engineers
BOM	Bill of Materials
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRAC	Control Room Air Conditioning
CVCS	Chemical Volume Control System
DIT	Design Information Transmittal
DG	Diesel Generator
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EHP	Electrical Maintenance Head Procedure
EPRI	Electrical Power Research Institute
ERG	Emergency Response Guideline
ESW	Essential Service Water
EQ	Equipment Qualification
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
IHP	Instrument Maintenance Head Procedure
IMC	Inspection Manual Chapter
kV	Kilovolts
LER	Licensee Event Report
LCO	Limiting Condition For Operation
MG	Motor Generator
MHP	Maintenance Head Procedure
MDAFWP	Motor Driven Auxiliary Feedwater Pump
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NESW	Non-Essential Service Water
NOED	Notice of Enforcement Discretion
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OA	Other Activities
ODCM	Offsite Dose Calculation Manual
OHP	Operations Head Procedure
PARS	Publically Available Records
PI	Performance Indicator
PMI	Plant Manager’s Instruction
PMP	Plant Manager’s Procedure
PMT	Post Maintenance Test
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program

RHR	Residual Heat Removal
RPS	Reactor Protection System
SDP	Significance Determination Process
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SSCs	Structures, Systems, and Components
SSPS	Solid State Protection System
STP	Surveillance Test Procedure
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VCT	Volume Control Tank