July 27, 2004

Mr. M. Nazar
Senior Vice President and
Chief Nuclear Officer
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 05000315/2004006;
05000316/2004006

Dear Mr. Nazar:

On June 30, 2004, 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 July 1, 2004, with Mr. M. Finissi 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, three findings of very low safety significance (Green) were identified, one of which involved a violation of NRC requirements. However, because of the very low safety significance and because the issue was entered into your corrective action program, the NRC is treating the violation as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; 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 2.390 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

cc w/encl: J. Jensen, Site Vice President
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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:	05000315/2004006; 05000316/2004006
Licensee:	Indiana Michigan Power Company
Facility:	Donald C. Cook Nuclear Power Plant, Units 1 and 2
Location:	1 Cook Place Bridgman, MI 49106
Dates:	April 1, 2004, through June 30, 2004
Inspectors:	B. Kemker, Senior Resident Inspector I. Netzel, Resident Inspector R. Ruiz, Reactor Engineer P. Lougheed, Senior Engineering Inspector C. Acosta, Nuclear Safety Intern W. Slawinski, Senior Radiation Specialist R. Jickling, Emergency Preparedness Analyst
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000315/2004006, IR 05000316/2004006; 04/01/2004-06/30/2004; D. C. Cook Nuclear Power Plant, Units 1 and 2; Personnel Performance During Non-Routine Plant Evolutions; Other Activities

This report covers a 13-week period of inspection by resident and regional based inspectors. The report includes announced baseline inspections in the areas of radiation protection and emergency preparedness. Three Green findings were identified, one 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. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance associated with an error by an auxiliary equipment operator while racking a DB-50 reactor trip bypass breaker that resulted in a Unit 2 reactor trip. The finding was more than minor because the finding was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations since a human performance error caused a reactor trip. The finding was of very low safety significance because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. No violation of regulatory requirements was identified.

  Corrective actions to address this issue included performing an evaluation of auxiliary equipment operators in DB-50 breaker racking, permitting individuals to perform DB-50 racking operations after demonstrating competency, implementing new peer checking requirements for DB-50 breaker racking activities, reviewing operational activities that have a significant potential for adversely impacting plant safety or operation to determine if peer checking beyond the existing requirements was needed, and a continued emphasis on operations standards. (Section 1R14.1)

- Green. The inspectors identified a finding of very low safety significance associated with an error by a control room reactor operator during main feedwater system flow adjustments that resulted in a Unit 2 reactor trip. The finding was more than minor because the finding was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations since a human performance error caused a reactor trip. The finding was of very low safety significance because it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. No violation of regulatory requirements was identified.

Corrective actions to address this issue included revising the operating procedure to require the in-service main feedwater pump controller to be maintained in automatic control; revising the conduct of operations procedure to require that anytime a controller is operated in manual and the controlled parameter deviates outside the normal band, the reactor operator shall notify the Unit Supervisor; the requirement to make a control room announcement anytime a controller is placed in manual; and the implementation of the Human Performance Scorecard for the evaluation of operator performance during simulator evaluations. (Section 1R14.2)

Cornerstone: Mitigating Systems

C Green. A finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control," was identified for the failure to include adequate acceptance limits in the procedure for inspecting and cleaning the component cooling water system heat exchangers. This finding was more than minor because, if left uncorrected, the issue could become a more significant safety concern. Specifically, the testing acceptance limit deficiencies could have designated a component cooling water heat exchanger as acceptable, when the heat exchanger heat removal capability had actually degraded below its design requirements. The issue was of very low safety significance since the licensee had recently cleaned all four component cooling water system heat exchangers and operability limits were not challenged.

Corrective actions to address this issue included revising testing acceptance limits to adequately define what constituted a blocked heat exchanger tube. (Section 4OA5.2)

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

One violation of very low safety significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violation and the licensee's corrective action tracking number is listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 was shutdown in Mode 5 (Cold Shutdown) at the beginning of the inspection period to determine the location of an unidentified reactor coolant system (RCS) leak. The licensee identified a leak from the pressurizer manway cover due to a failed gasket. Following repairs, the licensee synchronized the unit to the grid on April 7, 2004 and operated at or near full power for the remainder of the inspection period.

Unit 2 was shutdown in Mode 3 (Hot Standby) at the beginning of the inspection period following a reactor trip on March 29, 2004. The unit was operated at or near full power during the inspection period with the following exceptions:

- On March 29, 2004, Unit 2 experienced an automatic reactor trip when the Train 'B' reactor trip bypass breaker was manipulated during surveillance testing. An electrical fault developed during the manipulation and shorted out one phase of power to the control rod drive mechanisms. Several control rods fell into the core and generated a negative rate reactor trip signal. Following replacement of several power supplies in the control rod drive system, the unit was re-started and synchronized to the grid on April 2, 2004.

- On April 8, 2004, Unit 2 experienced an automatic reactor trip due to a high-high water level in the #24 steam generator during a planned power reduction. The high-high water level was due to inadequate feedwater system control with the feedwater pump control and the steam generator level control systems in manual. Following an investigation into the cause of the event, the unit was re-started and synchronized to the grid on April 13, 2004.

- On April 29, 2004, the licensee notified the NRC of a condition which met the criteria for declaring an Unusual Event as a result of a leak of approximately 65 gallons-per-minute (gpm) caused by the inadvertent lifting of the regenerative heat exchanger letdown outlet safety valve. The relief valve reseated after about 5 minutes, terminating the condition. Unit 2 remained at full power during the event.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's procedures and preparations for high temperature and high wind conditions. The inspectors reviewed severe weather and plant de-winterization procedures and performed general area walkdowns. During walkdowns of the plant and switchyard conducted the first 2 weeks of June 2004, the inspectors observed housekeeping conditions and verified that material capable of becoming an airborne missile hazard during high wind conditions or severe weather was appropriately restrained. Additionally, the inspectors reviewed condition reports (CRs) and the identification and resolution of equipment deficiencies associated with adverse weather mitigation. This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Partial System Walkdowns

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

- Unit 1 Auxiliary Feedwater (AFW) System (risk significant during turbine driven auxiliary feedwater pump lube oil and governor oil cooler valve replacement) performed on May 20, 2004.

- Unit 1 AB Emergency Diesel Generator (EDG) (risk significant during Unit 1 CD EDG planned maintenance activity) performed on June 1, 2004

- Unit 1 and 2 East Essential Service Water (ESW) System trains (risk significant with the Unit 2 West ESW train out of service from maintenance) performed on June 15, 2004

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.

In addition, the inspectors verified that equipment alignment problems were entered into the corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours

a. Inspection Scope

The inspectors performed 13 fire protection walkdowns of the following risk significant plant areas:

C Units 1 and 2 Auxiliary Building Pipe Tunnel Elevation 601' (Zone 6A)
C Units 1 and 2 Auxiliary Building Drumming/Drum Storage Area Elevation 587' (Zone 3)
C Units 1 and 2 Auxiliary Building Sampling Room Elevation 587' (Zone 4)
C Unit 1 CD EDG Room (Zone 15)
C Unit 1 AB EDG Room (Zone 16)
C Unit 2 CD EDG Room (Zone 18)
C Unit 2 AB EDG Room (Zone 19)
C Unit 1 Turbine Driven AFW Pump Room (Zone 17E)
C Unit 2 Turbine Driven AFW Pump Room (Zone 17F)
C Unit 1 Turbine Building Elevation 609' (Zones 90-93)
C Unit 2 Turbine Building Elevation 609' (Zones 96-99)
C Units 1 and 2 Turbine Driven AFW Pump Battery Rooms (Zones 106 & 107)
C Unit 2 Control Room Cable Vault (Zone 58)

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. In addition, the inspectors verified that fire protection related problems were entered into the corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

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

C The inspectors reviewed the Unit 1 and Unit 2 Flooding Evaluation reports, the Updated Final Safety Analysis Report (UFSAR) and other selected design basis documents to identify those areas susceptible to external flooding;

- C The inspectors interviewed plant engineering staff to understand which plant areas were susceptible to external flooding and what actions the licensee had taken to assure that the impact to plant equipment was minimized;

- C The inspectors reviewed the status of underground manholes subject to flooding which contained risk-significant cables; and

- C The inspectors reviewed selected operating procedures used to identify and mitigate external flooding events and reviewed preparations for possible flooding of susceptible plant areas due to heavy Spring rainfalls.

In addition, the inspectors reviewed the issues that the licensee entered into the corrective action program and verified that identified problems were 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 Problem Identification and Resolution cross-cutting area 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 operations training simulator on May 18, 2004. The inspectors focused on alarm response, command and control of crew activities, communication practices, procedural adherence, and the implementation of emergency plan requirements.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following two risk-significant structures, systems, and components (SSCs):

- C Unit 2 Control Rod Drive Mechanism (CRDM) K-10 Malfunction

- Unit 1 and 2 Component Cooling Water (CCW) Check Valves Found in Degraded Condition

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 actions to address SSC performance or condition problems in terms of the following:

- 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).

In addition, the inspectors verified that maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance were identified.

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 the following seven maintenance and operational activities affecting safety-related equipment:

- Unit 1 East CCW Pump Planned Maintenance for Routine Coupling Adjustment and Motor Lubrication
- Unit 1 Main Generator Automatic Voltage Regulator Card Replacement Emergent Activity
- Unit 1 RCS Leak Evaluation Emergent Activity
- Unit 1 Plant Air Compressor Annual Inspection
- Unit 1 Main Generator Neutral Ground Connection Enclosure Vibration Emergent Activity
- Unit 2 West ESW Pump Coupling Adjustment Emergent Activity
- Unit 2 East ESW Pump Replacement Emergent Activity

These activities were selected based on their potential risk significance relative to the reactor safety cornerstones.

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 March 29, 2004, Unit 2 Reactor Trip Due to Human Performance Error

a. Inspection Scope

On March 29, 2004, an automatic reactor trip of Unit 2 occurred when performing reactor trip switchgear breaker racking due to the inadvertent grounding of one phase of the rod control motor-generator set output with the racking bar. The inspectors reviewed the circumstances associated with this event, including the root cause evaluation and corrective actions.

b. Findings

(Closed) Licensee Event Report (LER) 50-316/2004-001-00: "Automatic Reactor Trip Due to Reactor Protection System Actuation While Manipulating Reactor Trip Bypass Breaker."

Introduction

The inspectors identified a finding of very low safety significance (Green) when an auxiliary equipment operator performing reactor trip breaker racking operations made physical contact between the racking bar and energized equipment inside the switchgear cubicle resulting in a Unit 2 reactor trip. No violation of regulatory requirements was identified.

Description

On March 29, 2004, an automatic reactor trip of Unit 2 occurred when performing reactor trip switchgear breaker racking due to the inadvertent grounding of one phase of the rod control motor-generator set output with the racking bar. In the process of racking out the Unit 2 'B' bypass breaker, an auxiliary equipment operator placed the racking bar into the breaker cubicle incorrectly and contacted an energized component. The grounding caused multiple control rods to drop into the core, producing a negative rate flux trip signal to trip the reactor. The plant response to the reactor trip was normal; however, several equipment problems were identified including two main steam stop valves drifting off their open detents requiring operator action to manually reset the valves in the open direction, and steam dump valve 2-URV-110 indicating mid-position with no demand from its controller. The inspectors reviewed the circumstances associated with this event, including the root cause evaluation and corrective actions.

The inspectors thoroughly examined the licensee's root cause evaluation and concluded the licensee had not neglected any likely factors. Two root causes were identified:

(1) The first root cause was determined to be human error during the placement of the racking bar. The auxiliary equipment operator's attention was focused on the lever positioning pin and therefore incorrectly identified the pin as the support for the racking bar. During racking operations, the auxiliary equipment operator located what was thought to be the racking bar alignment pin on the left side of the cubicle. The auxiliary equipment operator then inserted the racking bar into the cubicle near the pin. When the left side of the bar was near the pin, the operator began looking for the corresponding pin for placement of the bar on the right side of the cubicle. The auxiliary equipment operator inadvertently moved the bar toward the right, contacting an energized component and grounding one phase of the rod control motor-generator set 260 Volt Alternating Current (AC) output to the rod control cabinet. This de-energized the stationary gripper coils for numerous control rods, causing them to drop into the core and initiating the reactor trip on negative rate flux.

(2) The second root cause was determined to be inadequate management of the evolution, specifically with respect to deficiencies in peer checking, pre-job briefs, scheduling, and performing activities with inadequate guidance.

The root cause evaluation noted two contributing factors:

(1) Training on reactor trip switchgear (style DB-50 breakers) was inadequate. Auxiliary equipment operators received no training on the energized breaker components and the risk to unit operation during racking activities.

(2) The design of the DB-50 reactor trip switchgear was such that it required insertion of a cumbersome metal racking tool into the breaker cubicle in order to rack the breaker in and out of the cubicle. No guides or guards were provided to prevent the contact of the tool with energized components.

The inspectors concluded that the root cause evaluation was thorough and that corresponding corrective actions appropriately addressed the root and contributing causes.

<u>Analysis</u>

The inspectors determined that the human performance error which resulted in the Unit 2 reactor trip was a licensee performance deficiency warranting a significance evaluation. This finding was associated with the Initiating Events cornerstone. The cross-cutting area of Human Performance was also impacted by this finding. The inspectors assessed this finding using the Significance Determination Process (SDP). 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 related to this issue. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," the inspectors determined that the finding was of more than minor significance because this issue was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations since a human performance error caused a reactor trip. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in 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 (Green) because the finding: (1) did not contribute to the likelihood of a primary or secondary system loss-of-coolant-accident initiator, (2) did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available, and (3) did not increase the likelihood of a fire or internal/external flooding event.

Enforcement

No violation of regulatory requirements was identified. This issue was considered to be a finding (FIN 05000316/2004006-01). The licensee entered this finding into their corrective action program as CR 04089034.

Corrective actions planned or implemented to address this issue included the following:

(1) a performance evaluation of all auxiliary equipment operators in DB-50 breaker racking, permitting individuals to perform DB-50 racking operations only after demonstrating competency;

(2) a revision to the peer checking requirements for DB-50 breaker racking activities to require a peer check by a qualified operator of the breaker racking actions;

(3) a review of operational activities that have a significant potential for adversely impacting plant safety or operation to determine if peer checking beyond the existing requirements was needed; and

(4) continued reinforcement of operations standards in response to this issue and previously identified human performance issues.

.2 April 8, 2004 Unit 2 Reactor Trip Due to Human Performance Error

a. Inspection Scope

On April 8, 2004, a steam generator #24 high-high water level condition caused a main turbine trip which resulted in a Unit 2 reactor trip. The inspectors reviewed the circumstances associated with this event, including the root cause evaluation and corrective actions.

b. Findings

(Closed) Licensee Event Report (LER) 50-316/2004-002-00: "Unplanned Automatic Reactor Protection System Actuation Due to Feedwater Transient During a Power Reduction."

Introduction

The inspectors identified a finding of very low safety significance (Green) when a reactor operator did not appropriately control the #24 steam generator feedwater regulating valve and the in-service main feedwater pump, both of which were being controlled in manual, resulting in a high-high narrow range water level condition of the #24 steam generator. When steam generator water level reached 67 percent narrow range in the #24 steam generator, a turbine trip signal was generated and the main turbine tripped causing a reactor trip. No violation of regulatory requirements was identified.

Description

On April 8, 2004, a steam generator #24 high-high water level condition caused a main turbine trip which resulted in a Unit 2 reactor trip. This occurred during a planned shutdown to address an identified equipment malfunction of the Unit 2 main turbine control fluid system. At approximately 50 percent reactor power, following the removal of the East main feedwater pump from service, a reactor operator performing manual feedwater regulating valve control and manual main feedwater pump control overfilled the #24 steam generator. When water level reached 67 percent narrow range level in the #24 steam generator, a turbine trip signal was generated and the main turbine tripped causing a reactor trip.

As a result of the trip transient, a leak in the Unit 2 'C' condenser hotwell occurred at the condensate booster pump emergency leak off penetration. Also, an electrical fault occurred at the left outer main steam stop valve 'D', resulting in the loss of the Unit 2 main turbine control bus 2-CRD-5 and motoring of the main generator for approximately 9 minutes. The inspectors reviewed the circumstances associated with this event, including the root cause evaluation and corrective actions.

The inspectors thoroughly examined the licensee's root cause evaluation and concluded the licensee had not neglected any likely factors. There were two human performance related root causes identified by the licensee:

(1) the failure of the operator to manually control main feedwater differential pressure, and

(2) inadequate communication among the control room crew members, specifically between the operators at the controls and the Unit Supervisor.

Two contributing causes were identified:

(1) inadequate task/crew briefings that did not meet management's expectations and appeared to be cursory, and

(2) an abnormal secondary plant configuration, including a degraded main turbine control fluid system that required the power reduction.

It was also noted in the root cause evaluation that the Unit Supervisor did not exercise appropriate command and control during the evolution. This item was identified by the licensee as a failed barrier in the root cause evaluation, but was determined by the inspectors to be more appropriately categorized as an additional root cause. Although this represented a weakness in the root cause evaluation, the licensee had initiated corrective actions to improve Unit Supervisor performance in the area of command and control as a part of the Operations Department Improvement Plan.

Apart from the one weakness noted above, the inspectors concluded that the root cause evaluation was thorough and that corresponding corrective actions appropriately addressed the root and contributing causes.

Analysis

The failure to properly control steam generator level was a performance deficiency warranting a significance evaluation. This finding was associated with the Initiating Events cornerstone. The cross-cutting area of Human Performance was also impacted by this finding. The inspectors assessed this finding using the SDP. 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 related to this issue. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," the inspectors determined that the finding was of more than minor significance because this issue was associated with the Human Performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations since a human performance error caused a reactor trip. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding was of very low safety significance (Green) because the finding: (1) did not contribute to the likelihood of a primary or secondary system loss-of-coolant-accident initiator, (2) did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available, and (3) did not increase the likelihood of a fire or internal/external flooding event.

Enforcement

No violation of regulatory requirements was identified. This issue was considered to be a finding (FIN 05000316/2004006-02). The licensee entered this finding into their corrective action program as CR 04100009.

Corrective actions to address this issue included the following:

(1) revising the operating procedure to require the remaining in-service main feedwater pump controller to be maintained in automatic, rather than manual control;

(2) revising the conduct of operations procedure to require that anytime a controller is operated in manual and the controlled parameter deviates outside the normal band, the reactor operator shall notify the unit supervisor;

(3) requiring that any time a controller is placed in manual, a control room announcement is to be made notifying all team members;

(4) implementing the Human Performance Scorecard for the evaluation of individual performance during simulator evaluations. The Human Performance Scorecard is a method to systematically observe, record, and communicate operator performance in key job skills and behaviors.

.3 Unit 2 Unusual Event Due to Letdown Outlet Relief Valve Lifting

a. Inspection Scope

On April 29, 2004, operators were realigning the Unit 2 letdown system to perform maintenance on letdown orifice valve 2-QRV-162. Orifice valve 2-QRV-162 (75 gallons per minute (gpm)) had been taken out of service, letdown orifice valve 2-QRV-161 (75 gpm) was being placed in service, and letdown orifice valve 2-QRV-160 (45 gpm) was already open. When 2-QRV-161 was opened, operators observed indication of low letdown flow on the control board indications. The letdown flow indicator decreased from 120 gpm to approximately 70 gpm; annunciator 209 drop 10, "Letdown Relief Valve Discharge Temperature High" was received; and a rising level in the pressurizer relief tank was observed.

Operators identified that regenerative heat exchanger letdown outlet safety valve 2-SV-51 was lifting. Operators took action to isolate normal letdown and place excess letdown in service. During the 5 minutes that the letdown relief valve was lifting, approximately 323 gallons of water was diverted from the volume control tank to the pressurizer relief tank.

For the 5 minute period, the licensee concluded that D. C. Cook Unit 2 met the conditions for an Unusual Event for identified leakage in excess of 25 gpm. The licensee determined that the apparent cause of this event was inadequate procedural guidance.

Following the event, the inspectors reviewed control room logs, computer data, operator statements, plant drawings, procedures, condition reports, the D. C. Cook UFSAR, the D. C. Cook Emergency Plan and associated procedures, TSs, and other plant documents to determine the adequacy of the event response.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- CR 04128006, "Pressure Read Too High During Turbine Driven Auxiliary Feedwater Pump Surveillance"
- CR 04086051, "Discovered Some High Energy Line Break Dampers in East Main Steam Enclosure Closed"
- CR 04111017, "Valves Relied Upon to Isolate the Emergency Core Cooling System and Containment Spray Pump Drains From Recirculation Piping Network Are Not Leak Tested"
- CR 03200013, "West Essential Service Water Pump Strainer Backwash Valves Emergency Air Tank Pressure Less Than Minimum"

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Review of Selected Operator Workarounds

a. Inspection Scope

The inspectors evaluated the issue listed below as a potential operator work-around (OWA) to identify any potential impact on the functionality of mitigating systems or on the operators' response to initiating events:

- DRV-407 Caused Cooldown Following a Reactor Trip

The inspectors selected this issue to review as a potential OWA in order to understand the conditions causing additional post-trip RCS cool down attributed to the main steam system drains and the potential impact on plant operations. The inspectors interviewed operating and engineering department personnel and reviewed selected procedures and documents.

b. Findings

No findings of significance were identified.

.2 Semiannual Review of the Cumulative Effect of Operator Workarounds

a. Inspection Scope

The inspectors reviewed the cumulative effect of OWAs, control room deficiencies, and degraded conditions on equipment availability, initiating event frequency, and the ability of the operators to implement abnormal or emergency operating procedures. During this review, the inspectors considered the cumulative effects of OWAs on the following:

- the reliability, availability and potential for mis-operation of a system;
- the ability of operators to respond to plant transients or accidents in a correct and timely manner; and
- the potential to increase an initiating event frequency or affect multiple mitigating systems.

In addition, the inspectors reviewed the issues that the licensee entered into their 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 issues potentially affecting the functionality of mitigating systems or operator response to initiating events that were documented in selected condition reports.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

- Unit 1 Main Generator Voltage Regulator
- Unit 1 CD EDG Fuel Injector Replacement
- Unit 2 South Safety Injection Pump Planned Maintenance
- Unit 1 Pressurizer Manway Repair
- Unit 2 West ESW Pump Coupling Adjustment
- Unit 2 East ESW Pump Replacement

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

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Unit 1 Forced Outage

a. Inspection Scope

On March 30, 2004, the licensee entered a forced outage on Unit 1 to determine the location of an unidentified RCS leak. The licensee identified a leak from the pressurizer manway cover due to a failed gasket. The licensee entered Mode 5 (Cold Shutdown) to make repairs. Following repairs, the licensee performed a reactor startup and synchronized the unit to the grid on April 7, 2004.

The inspectors evaluated the licensee's 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. Other major outage activities evaluated included the licensee's control of the following:

- structures, systems, and components (SSCs) which could cause unexpected reactivity changes;
- switchyard activities and the configuration of electrical power systems in accordance with the TSs and shutdown risk plan; and
- SSCs required for decay heat removal.

The inspectors observed portions of the plant cooldown, including the transition to shutdown cooling, to verify that the licensee controlled the plant cooldown in accordance with the TSs. The inspectors also observed portions of the restart activities to verify that TS requirements and administrative procedure requirements were met prior to changing operational modes or plant configurations. Major restart inspection activities performed included:

- verification that RCS boundary leakage requirements were met prior to entry into Mode 4 (Hot Shutdown) and subsequent operational mode changes;
- inspection of the Containment Building to assess material condition and search for loose debris, which if present could be transported to the containment recirculation sumps and cause restriction of flow to the emergency core cooling system pump suctions during loss-of-coolant-accident conditions.

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.

.2 Unit 2 Forced Outage

a. Inspection Scope

On March 29, 2004, the licensee entered a forced outage on Unit 2 following an automatic reactor trip caused by an error when manipulating the Train 'B' reactor trip bypass breaker during surveillance testing. An electrical fault shorted out one phase of power to the control rod drive mechanisms. Several control rods fell into the core and caused a negative rate trip signal. The licensee maintained the unit in Mode 3 (Hot Standby) to replace the breaker and perform additional maintenance work. The licensee performed a reactor startup and synchronized the unit to the grid on April 2, 2004.

The inspectors evaluated the conduct of forced outage activities to assess the control of plant configuration and management of risk. The inspectors reviewed configuration management to verify that the licensee maintained defense-in-depth commensurate with the 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, and maintenance department personnel and reviewed selected procedures and documents.

b. Findings

A finding of very low safety significance (Green) associated with this event is discussed in Section 1R14.1 of this report.

.3 Unit 2 Forced Outage

a. Inspection Scope

On April 8, 2004, the licensee entered a forced outage on Unit 2 following an automatic reactor trip caused by a high-high water level in the #24 steam generator during a planned power reduction. The high-high water level was due to inadequate feedwater system control with the feedwater pump control and the steam generator level control systems in manual. The licensee maintained the unit in Mode 3 (Hot Standby) to verify proper operation of the feedwater pump and steam generator level control systems. The licensee performed a reactor startup and synchronized the unit to the grid on April 13, 2004.

The inspectors evaluated the conduct of forced outage activities to assess the control of plant configuration and management of risk. The inspectors reviewed configuration management to verify that the licensee maintained defense-in-depth commensurate with the 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, and maintenance department personnel and reviewed selected procedures and documents.

b. Findings

A finding of very low safety significance (Green) associated with this event is discussed in Section 1R14.2 of this report.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed portions of the following eight 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.

- Unit 2 CD EDG Operability Test
- Unit 1 and Unit 2 Accumulator Level Shiftly Surveillance
- Unit 2 West Motor Driven AFW Pump Surveillance
- Unit 1 CD EDG Operability Test
- Unit 2 Turbine Drive AFW Pump Surveillance
- Unit 2 West ESW Pump Surveillance
- Unit 2 West Residual Heat Removal Train
- Unit 2 Distributed Ignition System Surveillance and Baseline Testing

The inspectors reviewed the test methodology and test results 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.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System (ANS) Testing (71114.02)

a. Inspection Scope

The inspectors discussed with D. C. Cook Emergency Preparedness (EP) staff the operation, maintenance, and periodic testing of the ANS in the D. C. Cook Plant's Emergency Planning Zone to determine whether the ANS equipment was adequately maintained and tested in accordance with Emergency Plan commitments and procedures. The inspectors reviewed records of test and maintenance activities for the period from March 2003 through March 2004.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization (ERO) Augmentation Testing (71114.03)

a. Inspection Scope

The inspectors reviewed and discussed with EP staff the procedures that included the primary and alternate methods of activating the ERO to augment the onshift ERO, plus provisions for maintaining the ERO call-out roster. The inspectors also reviewed critiques and a sample of corrective action program records of unannounced, off-hours augmentation drills, which were conducted between February 2003 and March 2004, to determine the adequacy of the drills' critiques and associated corrective actions.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors reviewed Revisions 17, 18, and 19 of the D. C. Cook Nuclear Power Plant Emergency Plan (Emergency Plan) to determine if changes identified in these revisions reduced the Plan's effectiveness, pending further review by the NRC.

b. Findings

The inspectors identified one revised Emergency Action Level (EAL) in Revision 17 of the Emergency Plan that deviated from the version approved in 1996 via a Safety Evaluation Report. Also, Revision 17 of the Emergency Plan underwent a major reformatting which included reductions of standards from the previous revision. These issues were identified as an Unresolved Item (URI) pending further review by NRC personnel.

Specifically, the Fission Product Barrier Reference Table, Table 12-4, Containment Barrier, under the Loss column, 3.4L, "Steam Generator (SG) Secondary Side Release," was revised beginning in Revision 17 through Revision 19 of the Emergency Plan. These changes were implemented without prior NRC approval. The wording of this EAL from the approved 1996 revision and Revisions 17 through 19 were as follows:

Approved 1996 EAL:

3.4L SG Secondary Side Release

Primary to secondary leakage rate greater than TS limit.

AND

Release of secondary coolant from the associated steam generator to the environment is occurring.[1]

[1] Does not include a release through the condenser air ejectors or the gland steam condenser vents.

Revision 17 EAL:

3.3 SG Secondary Side Release

Primary to secondary leak rate > TS limit.

AND

Release of secondary coolant from the associated SG to the environment is occurring with alert alarm on any SG PORV [Power Operated Relief Valve] radiation monitor.

Revisions 18 and 19 EAL:

3.3 SG Secondary Side Release

1a. Primary to secondary leak rate > TS limit.

AND

b. Secondary line Break OUTSIDE Containment results in release (>30 minutes) to the environment.

OR

2. Release of secondary coolant from the affected SG to the environment with alert on any SG to the environment with alert alarm on any pressure operated relief valve rad monitor.[1]

[1] Does not include a release through the condenser air ejectors or the gland steam condenser vents for the purpose of declaration of a SITE AREA EMERGENCY.

The inspectors identified that the criteria added to the above EAL in Table 12-4, Fission Product Barrier, appeared to result in a reduction of the number of classifiable events. Also, the licensee added criteria (which did not appear in the NUMARC examples) in its original EAL submittal for approval in 1994 where the NRC indicated that specific additional criteria (including a non-conservative 30 minutes criteria which also did not appear in the NUMARC examples) for the SG Secondary Side Release Fission Product Barrier table was unacceptable.

Additionally, the licensee identified that Revision 17 of the Emergency Plan was a departure from the previous revision's format and classified the changes as: (1) plan enhancements; (2) administrative changes; and (3) potential decreases in the effectiveness of the Emergency Plan that "reduce the standards of the previous revision but is not a reduction of the Federal standards in 10 CFR 50.47(b)(1 through 16), 10 CFR 50, Appendix E, and NUREG-0654 FEMA-REP-1 (NUREG 0654)."

The inspectors identified that Revisions 17 (as well as 18 and 19) of the Emergency Plan were a major rewrite of the plan which appeared to include reductions in commitments from Revisions 15 and 16 of the plan.

The potential decrease in effectiveness of the Steam Generator Secondary Side Release EAL and the apparent reductions in commitments in Emergency Plan Revisions 17 through 19 were considered an Unresolved Item (URI 05000316/2004006-03) pending additional review by NRC personnel.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

a. Inspection Scope

The inspectors reviewed Performance Assurance staff's 2003 audits of the D. C. Cook Emergency Preparedness program to verify that these independent assessments satisfied the requirements of 10 CFR 50.54(t) and that the licensee adequately identified and corrected deficiencies. The inspectors also reviewed site self-assessments of the EP program conducted in 2003 and 2004, and critiques to evaluate the EP staff's efforts to identify and correct weaknesses and deficiencies. Additionally, the inspectors reviewed a sample of EP items, condition reports, and corrective actions related to the facility's EP program to determine whether corrective actions were acceptably completed.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed activities in the plant simulator and Technical Support Center during the licensee's annual emergency preparedness exercise conducted on June 15, 2004. The inspectors verified that the emergency classifications and notifications to offsite agencies were completed in an accurate and timely manner as required by the emergency plan implementing procedures. The inspectors also verified that the drill was conducted in accordance with the prescribed sequence of events, drill objectives were satisfied and that the required prompts from the licensee drill controllers were appropriately communicated to the drill participants.

The inspectors observed the post-drill critique in the Technical Support Center and reviewed documented post-drill critique comments by licensee evaluators to verify that licensee personnel and licensee drill evaluators adequately self-identified drill performance problems of significance. The inspectors also verified that condition reports were generated for drill performance problems of significance and were entered into the corrective action program with the appropriate significance characterization.

b. Findings

No findings of significance identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System Description and Waste Generation

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system descriptions in the UFSAR and the 2002 and 2003 Annual Radioactive Effluent Release Reports for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors evaluated the scope of the licensee's audit/self-assessment activities with regard to radioactive material processing and transportation programs to determine if those activities satisfied the requirements of 10 CFR 20.1101(c), and the quality assurance audit requirements of Appendix G to 10 CFR 20 and of 10 CFR 71.137, as applicable. Opportunities to enhance audits of the licensee's quality assurance program required by Subpart H of 10 CFR 71 were discussed with Performance Assurance department management and staff.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors performed walkdowns of the liquid and solid radwaste processing systems to verify that the configuration of these systems was consistent with the descriptions in the UFSAR and the Process Control Program, and to assess the material condition and operability of those systems. The inspectors reviewed the status of radioactive waste process equipment that had either never been used or had not been operated for more than 10 years, but not declared abandoned in place. These systems included the radwaste evaporator system, the radwaste concentrates system, and the waste solidification/drumming equipment. The inspectors discussed with licensee management concerns regarding the lack of adequate administrative and/or physical controls preventing the inadvertent use of this laid-up radwaste processing equipment to ensure its use would not contribute to an unmonitored release path or be a source of unnecessary personnel exposure if not adequately isolated from other systems. The inspectors concerns regarding material condition issues in the radwaste drumming room were also discussed with licensee management.

The inspectors reviewed the licensee's processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and sampling was completed to obtain representative waste stream samples for analysis. The inspectors also reviewed how and in what locations area smear surveys were collected to represent the dry active waste (DAW) stream and the method used for determining the radionuclide mix of filter media to ensure they were representative of the intended radwaste stream. Additionally, the inspectors reviewed the methodologies for quantifying gamma emitting radionuclide waste stream content, for determining moisture content including waste stream tritium concentrations and for waste concentration averaging to ensure that representative samples of the waste products were provided for the purposes of waste classification pursuant to 10 CFR 61.55.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's methods and procedures for determining the classification of radioactive waste shipments including the use of scaling factors to quantify difficult-to-measure radionuclides (e.g., pure alpha or beta emitting radionuclides). The inspectors reviewed the licensee's radiochemical sample analysis results for each of the licensee's current waste streams which consisted of primary system resins, radwaste demineralizer resins, process filters and Dry Active Waste (DAW). The reviews were conducted to verify that the licensee's program was in compliance with 10 CFR 61.55 and 10 CFR 61.56, as required by Appendix G of 10 CFR Part 20. The inspectors also reviewed the licensee's waste characterization and classification program to ensure that reactor coolant chemistry data was periodically evaluated to account for changing operational parameters that could potentially affect waste stream classification and thus validate the continued use of scaling factors between annual sample analysis updates.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation and Shipping Records

a. Inspection Scope

The inspectors reviewed the documentation for shipment packaging, surveying, package labeling and marking, vehicle checks and placarding, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness for seven non-excepted low specific activity (LSA) and surface contaminated object (SCO) and for one excepted (limited quantity) radioactive material and radwaste shipments completed in 2002 and 2003. These shipments included:

- Reactor Coolant Pump Motor to Vendor as SCO - II;
- High Activity DAW to Waste Processor as LSA - II;
- Radwaste System Resins in High Integrity Container to Waste Processor as LSA-II (two shipments);
- Primary System Resins in High Integrity Container to Waste Processor as LSA - II;
- FRAC Tank to Vendor as SCO - II;
- Contaminated Valves to Vendor as Limited Quantity (excepted shipment); and
- High Activity DAW to Waste Processor as LSA - II.

The inspectors selectively verified that the requirements of 10 CFR Parts 20 and 61 and those of the Department of Transportation (DOT) in 49 CFR 170-189 were met for each shipment. Specifically, records were reviewed and those staff involved in shipment activities were interviewed to verify that packages were labeled and marked properly, that package and transport vehicle surveys were performed with appropriate instrumentation and survey results satisfied DOT requirements, and that the quantity and type of radionuclides in each shipment were determined accurately including the proper application of scaling factors. The inspectors also verified that shipment manifests were completed in accordance with the regulations, included the required emergency response information, that the receiving licensee was authorized to receive the shipment, and that shipments were tracked as required by 10 CFR 20.

The inspectors observed technicians perform surveys of old Unit 1 steam generator lower assemblies which were being prepared for shipment to a burial site as unpackaged surface contaminated objects, consistent with a DOT granted exemption. Additionally, these technicians and the licensee’s primary and alternate shippers were questioned to verify that they had adequate skills to accomplish shipment related tasks, to determine if the shippers were knowledgeable of the shipping regulations and whether shipping personnel demonstrated adequate skills to accomplish package preparation requirements for public transport with respect to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR 172 Subpart H.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed Corrective Action Program documents, audit and self-assessment reports, and field observation records that addressed the radioactive waste and radioactive materials shipping program since the last inspection to verify that the licensee had effectively implemented the corrective action program and that problems were identified, characterized, prioritized, and corrected. The inspectors also verified that the licensee's oversight mechanisms collectively were capable of identifying repetitive deficiencies or significant individual deficiencies in problem identification and resolution.

The inspectors also selectively reviewed CRs generated since the previous inspection that dealt with the radioactive material shipping program, and interviewed staff and reviewed documents to determine if the following activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions; and
- Implementation/consideration of risk significant operational experience feedback.

These reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Mitigating Systems

.1 Safety System Unavailability

a. Inspection Scope

The inspectors verified the following performance indicators for both units:

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

The inspectors reviewed operating logs, maintenance history and surveillance test history for unavailability information for these systems from July 2003 to March 2004. 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 definition. The inspectors interviewed engineering staff to determine whether the performance indicator data was being collected and reported consistent with the guidance contained in NEI [Nuclear Energy Institute] 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2.

b. Findings

No findings of significance were identified.

.2 Emergency Preparedness

a. Inspection Scope

The inspectors reviewed the licensee's records associated with the three EP performance indicators (PIs) listed below. The inspectors verified that the licensee accurately reported these indicators in accordance with relevant procedures and Nuclear Energy Institute guidance endorsed by NRC. Specifically, the inspector reviewed licensee records associated with PI data reported to the NRC for the period July through December 2003. Reviewed records included procedural guidance on assessing opportunities for the three performance indicators, assessments of performance indicator opportunities during pre-designated Control Room Simulator training sessions and drills, revisions of the roster for personnel assigned to key Emergency Response Organization positions, and results of periodic Alert and Notification System (ANS) operability tests. The following performance indicators were reviewed:

- Alert and Notification System;
- Emergency Response Organization Drill Participation; and
- Drill and Exercise Performance.

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 findings of significance were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors completed a review of repetitive or closely related issues documented in the licensee's corrective action program and other processes/programs utilized by the licensee to track the status of plant issues. This review included, but was not limited to, system health reports, self-assessment reports, maintenance rule program reports, operator workaround lists, equipment reliability lists, corrective and elective maintenance backlogs, and various plant performance indicators. The purpose of this review was to identify trends not previously identified or adequately addressed by the licensee that might indicate the existence of more safety significant issues.

b. Findings

No findings of significance were identified.

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

During the Flood Protection inspection activity discussed in Section 1R06, the inspectors reviewed CR 02086005 associated with the licensee's review of NRC Information
Notice 2002-12, "Submerged Safety-related Electrical Cables." The inspectors noted that the licensee's condition report evaluation was limited to addressing only safety-related cables routed in manholes and embedded conduits. The evaluation concluded that Cook Plant was not susceptible to the loss of safety-related equipment powered by submerged cables since the plant had no safety-related cables routed in manholes.

There was some concern for safety-related cables routed in embedded conduits and the licensee implemented actions to perform characterization testing of all of the potentially affected cables as well as actions to replace some cables. The inspectors noted, however, that there were numerous risk-significant cables that were routed through manholes, including cables credited in the licensee's design basis to provide off-site power to the plant's safety-related equipment following an accident. However, the inspectors noted that the licensee had not opened and visually inspected any of the manholes for water build-up or other degraded conditions described in the Information Notice and therefore was not aware of the material condition of the cables in the manholes. The licensee did not have a recurring activity to open and visually inspect any of the manholes. The licensee also concluded characterization testing of the cables would not be beneficial. The licensee stated that it intended to open and visually inspect each of the manholes as part of its license renewal activities, but had not planned to do so in the near future. The inspectors concluded that this was a missed opportunity to identify and prevent possible problems comparable to those described in the Information Notice before they result in equipment failures.

4OA3 Event Followup (71153)

.1 (Closed) LER 50-316/2002-005-01: "Unit 2 Trip Due to Instrument Rack 24-Volt DC [Direct Current] Power Supply Failure," Supplement 1.

The licensee submitted Supplement 1 to LER 50-316/2002-005-00 to correct a statement regarding actions completed in the time frame of the event. The original LER incorrectly stated that all remaining 24-Volt DC control group power supplies in Unit 2 were verified to be no older than 2 years old. Although all of the Unit 2 control group power supplies were inspected and one power supply was replaced, many of the remaining power supplies were older than 2 years. This reporting discrepancy was considered to be of minor significance and is not subject to formal enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

.2 Failure to Report Non-compliance with TS 3.8.1.1.e During the April 2003 Fish Intrusion Event as Required by 10 CFR 50.73

a. Inspection Scope

The inspectors reviewed the following CRs for compliance with the reporting requirements of 10 CFR 50.73:

- CR 03114018, "Both Unit Two EDGs Were Declared Inoperable at 0348 Due to Inadequate ESW Flow"

- CR 03114035, "Unit One Did Not Comply with TS 3.8.1.1.e for Verifying Power Sources Within 1 Hour of Declaring Both Diesel Generators Inoperable"

b. Findings

The inspectors identified a minor violation of 10 CFR 50.73(a) because the licensee failed to make a required report for a condition prohibited by the plant's TS in accordance with 10 CFR 50.73(a)(2)(i)(B).

Discussion

(Closed) LER 50-315/2003-003-01: "Supplemental LER for Dual Unit Manual Trip Due to the Failure of the Intake Traveling Screens and Failure to Comply with TS 3.8.1.1."

The licensee submitted Supplement 1 to LER 50-315/2003-003-00 to report the failure to comply with the actions required by TS 3.8.1.1.e during the event.

During their review of LER 50-315/316/2003-003-00, 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. Because of degraded ESW system flow, both EDGs on both Unit 1 and Unit 2 were declared inoperable on April 24, 2003. Technical Specification 3.8.1.1.e required, in part, that with two EDGs inoperable, the licensee demonstrate the operability of two off-site circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Operators completed the 1-hour verification requirement 24 minutes late for Unit 1 and 9 minutes late for Unit 2. When the verification was completed, all of the off-site equipment was found to be available to perform its safety function. The inspectors therefore concluded that this failure to meet TS 3.8.1.1.e was of minor significance.

The inspectors reviewed the CR evaluations and discussed this issue with the licensee to understand why this TS non-compliance was not originally reported as a condition prohibited by the plant's TS in accordance with 10 CFR 50.73(a)(2)(i)(B). The inspectors determined that the licensee had incorrectly believed that this non-compliance was not reportable because the TS Limiting Condition for Operation (LCO) action requirement that was not met referred to a surveillance test requirement. The inspectors concluded that while there is an exception for not reporting an event that consists solely of a late surveillance test, this event was required to be reported because the non-compliance involved the failure to meet a TS LCO action requirement rather than a regularly scheduled surveillance test. The purpose of demonstrating operability of two off-site power sources was, in this case, in response to a degraded plant condition that relied on the operability of the equipment for which the surveillance test was being performed.

Enforcement

This failure to report a condition prohibited by the plant's TS in accordance with 10 CFR 50.73(a)(2)(i)(B) was considered to be of minor significance and is not subject to formal enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

.3 Response to Unit 2 Regenerative Heat Exchanger Letdown Outlet Safety Valve Lifting

a. Inspection Scope

On April 29, 2004, the licensee notified the NRC of a condition which met the criteria for declaring an Unusual Event on Unit 2 as a result of a leak of approximately 65 gallons-per-minute caused by the inadvertent lifting of the regenerative heat exchanger letdown outlet safety valve. The relief valve was reseated after about 5 minutes, terminating the condition. Unit 2 remained stable at full power during the event.

The inspectors assessed control room operator performance during the event. The inspectors evaluated the plant conditions and the licensee's actions to mitigate the affect on plant systems and recover from the event to determine the need for a special inspection. The inspectors also confirmed that the licensee made a timely notification to the NRC after identifying that the condition met the criteria for declaring an Unusual Event.

b. Findings

No findings of significance were identified.

.4 Unit 2 Reactor Trip Response

a. Inspection Scope

On March 29, 2004, Unit 2 experienced an automatic reactor trip when the Train 'B' reactor trip bypass breaker was manipulated during surveillance testing. An electrical fault developed during the manipulation and shorted out one phase of power to the control rod drive mechanisms. Several control rods fell into the core and caused a negative rate trip signal. Following replacement of several power supplies in the control rod drive system, the unit was synchronized to the grid on April 2, 2004. The inspectors assessed control room operator performance immediately following the reactor trip and reviewed the post trip report.

b. Findings

A finding of very low safety significance (Green) associated with this event is discussed in Section 1R14.1 of this report.

.5 Unit 2 Reactor Trip Response

a. Inspection Scope

On April 8, 2004, Unit 2 experienced an automatic reactor trip due to a high-high water level in the #24 steam generator during a planned power reduction. The high-high water level was due to inadequate feedwater system control with the feedwater pump control and the steam generator level control systems in manual. Following an investigation into the cause of the event, the unit was synchronized to the grid on April

13, 2004. The inspectors assessed control room operator performance immediately following the reactor trip and reviewed the post trip report.

b. Findings

A finding of very low safety significance (Green) associated with this event is discussed in Section 1R14.2 of this report.

4OA4 Cross-Cutting Aspects of Findings

.1 Cross-Reference to Human Performance Findings Documented Elsewhere in the Report

Section 1R14.1 of this report describes a finding where human performance error by an auxiliary equipment operator resulted in a Unit 2 reactor trip.

Section 1R14.2 of this report describes a finding where a human performance error by a control room reactor operator resulted in a Unit 2 reactor trip.

Section 4OA3.2 of the report describes a finding where the licensee failed to properly recognize a condition involving the failure to meet a TS LCO action requirement for inoperable EDGs. This resulted in a failure to report a condition prohibited by the plant's TS in accordance with 10 CFR 50.73(a)(2)(i)(B).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000315/316/2003002-01: "ESW System Water Hammer Load Calculation Concern."

Unresolved Item 05000315/316/2003002-01 was opened to document a number of concerns with the licensee's evaluation of a hydraulic transient which occurred in April 2000 following a dual unit loss of offsite power. In order to evaluate the issue, and the licensee's actions, the inspectors reviewed licensee documents, interviewed personnel, and performed a walkdown of the ESW system, specifically looking for signs of piping movement. The licensee had previously determined that the ESW system would be subject to column-rejoining hydraulic transients under certain conditions which were within the design basis. Following the 2000 event, the licensee had walked down the system and had performed an operability evaluation, which concluded that although some stresses appeared to be above the B31.1 Code allowables, the calculation contained sufficient conservatisms to conclude that the piping was not over-stressed. The licensee had a corrective action in place to perform a design basis calculation of the system. During the walkdown, the inspectors identified some indications of previous pipe movement, such as damaged insulation and unpainted segments of pipe. Based on the relative flexibility of the system, a licensee examination of the weld surface of one hanger, and the licensee's planned action to perform a design basis calculation, the inspectors concluded there was no operability concerns associated with these indications. Because this issue was identified by the licensee and was captured in

CR P-00-10960 and the corrective actions planned or taken were appropriate, the inspectors determined that the licensee should be given credit for identifying the violation (see Section 4OA7). This unresolved item is closed.

.2 (Closed) Unresolved Item 05000315/316/2003002-02: "Estimation of Tube Blockage in the Component Cooling Water Heat Exchangers."

Introduction

A finding of very low safety significance (Green) and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XI, “Test Control,” was identified by the inspectors for the failure to include adequate acceptance limits in the procedure for inspecting and cleaning the component cooling water system heat exchangers.

Description

Unresolved Item (URI) 05000315/316/2003002-02 was previously opened to document a concern regarding the as-found acceptability of heat exchangers. Specifically, the licensee’s test method to demonstrate that safety-related heat exchangers will perform satisfactorily consisted of a visual inspection of the heat exchanger tubes for blockage. Prior to 2001, the licensee's procedures accepted tubes as not being blocked if they were capable of being cleaned with 105 pounds per square inch gauge (psig) air. In 2001, during an NRC Safety System and Performance Capability inspection, the inspectors questioned the basis for this value. Subsequently, the licensee lowered the acceptance limit to 65 psig. In 2003, during an NRC Heat Sink inspection, the inspectors again questioned the basis for this acceptance limit. At that time, the licensee obtained design engineering support and determined the acceptance limit was only 5 psig. During this inspection, the inspectors reviewed the licensee's actions and performed independent calculations which determined that blowing 5 psig air through the tubes was less than the pressure drop experienced during normal operation. Therefore, the inspectors concluded the use of 5 psig air to identify a blocked tube was acceptable.

Analysis

The inspectors determined that the use of a non-conservative acceptance criteria was a performance deficiency warranting a significance evaluation. This finding was associated with the Mitigating Systems cornerstone. 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 related to this issue. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," the inspectors determined that the finding was of more than minor significance because if left uncorrected, the non-conservative acceptance criteria could result in the licensee incorrectly concluding that a heat exchanger was capable of performing its system function. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was of very low safety significance because the finding was a qualification deficiency which was confirmed to not result in a loss of function per Generic Letter 91-18 since all potentially affected heat exchangers had been cleaned and operability limits were not actually challenged.

Enforcement

10 CFR 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program be established to demonstrate that components will perform satisfactorily in service and that the tests be performed in accordance with written procedures which incorporate acceptance limits contained in applicable design documents. The use of a non-conservative value for determining heat exchanger tube blockage as of 2001 up until June 2003 was a violation of 10 CFR 50, Appendix B, Criterion XI. The licensee entered this issue into their corrective action program as CRs 01282046 and 03083036. However, because this violation was of very low safety significance and because it was entered into the licensee's corrective action program, this violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000315/316/2004006-03). To address this issue, licensee personnel revised testing acceptance criteria to establish a limit that would adequately identify whether heat exchanger tube blockage existed.

.3 (Closed) Unresolved Item 05000315/316/2003002-03: "Questionable Data Regarding Component Cooling Water Heat Exchanger As-Built Specification Sheet."

This unresolved item was opened to document a concern regarding the reasonableness of a licensee calculation involving the outside heat transfer coefficient correction factor. There was also a concern that the original manufacturer's specification sheet might have overestimated the CCW heat exchangers' heat transfer capabilities and; therefore, the impact on their ability to perform their safety function. During the inspection, the inspectors performed independent calculations of these heat transfer parameters and agreed that the outside heat transfer coefficient correction factor was lower than the one calculated by the licensee; however, the inspectors concluded that the original manufacturer's specification sheet was accurate. The issue was minor because the licensee did not use the calculation to determine heat exchanger acceptability. Instead, the licensee opened, inspected, and cleaned the heat exchangers each refueling outage. The inspectors determined that the heat exchangers had been cleaned in 2003 and that the as-left condition of the heat exchangers was acceptable. No violation of NRC requirements was identified. This unresolved item is closed.

.4 (Closed) TI 2515/156: Offsite Power System Operational Readiness.

a. Scope

The inspectors collected data from licensee maintenance records, event reports, corrective action documents and procedures, and through interviews of station engineering, maintenance, and operations staff, as required by TI 2515/156. The data was gathered to assess the operational readiness of the offsite power systems in accordance with NRC requirements such as Appendix A to 10 CFR 50, General Design Criteria (GDC) 17; Criterion XVI of Appendix B to 10 CFR 50; TSs for offsite power systems; 10 CFR 50.63; 10 CFR 50.65(a)(4), and licensee procedures. Documents reviewed for this TI is listed in the attachment.

b. Findings

No findings of significance were identified. Based on the results of the inspection, no immediate operability issues were identified. In accordance with TI 2515/156 reporting requirements, the inspectors provided the required data to the headquarters staff for further analysis.

4OA6 Meetings

.1 Resident Inspectors' Exit Meeting

The inspectors presented the inspection results to Mr. M. Finissi and other members of licensee management at the conclusion of the inspection on July 1, 2004. 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

- Emergency Preparedness program and performance indicators inspection meeting with Mr. J. Jensen on April 16, 2004.

- Emergency Preparedness program telephone exit with Mr. L. Weber on April 29, 2004.

- Public Radiation Safety Radioactive Waste Processing and Transportation program inspection meeting with Mr. J. Jensen on May 20, 2004.

- Heat Sink Unresolved Items Review with Mr. J. Jensen on April 14, 2004.

4OA7 Licensee-Identified Violation

The following violation of very low safety significance was identified by the licensee and was a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600, for being dispositioned as a Non-Cited Violation.

Cornerstone: Mitigating Systems

10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that applicable regulatory requirements and the design basis are correctly translated into

specifications, drawings, procedures, and instructions. The failure to correctly translate the design loads on the ESW piping system into specifications which demonstrated that ASME Code allowables were not exceeded was a violation of 10 CFR 50, Appendix B, Criterion III, "Design Control." However, this violation was of very low safety significance because there was sufficient conservatism in the operability evaluation to demonstrate that the system remained operable. This issue was entered into the licensee's corrective action program as CR P-00-10960. Corrective actions to address this issue included a verification that loads were within the ASME Code allowables.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Carlson, Environmental Manager
D. Fadel, Engineering Vice President
M. Finissi, Plant Manager
R. Gillespie, Operations Director
C. Graffenius, Emergency Preparedness Coordinator
J. Jensen, Site Vice President
M. Nazar, Senior Vice President, Chief Nuclear Officer
S. Partin, Site Protective Services/Emergency Preparedness Manager
R. Serocke, Radiation Protection Manager
J. Zwolinski, Design Engineering & Regulatory Affairs Director

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000316/2004006-01	FIN	Breaker Manipulation Error Resulted in Unit 2 Trip (Section 1R14.1)
05000316/2004006-02	FIN	Feedwater Control Error Resulted in Unit 2 Trip (Section 1R14.2)
05000316/2004006-04	URI	Potential Decrease in Effectiveness of the Steam Generator Secondary Side Release EAL (Section 1EP4)
05000315/316/2004006-03	NCV	Inadequate Acceptance Criteria for Heat Exchanger Tube Blockage (Section 4OA5.2)

Closed

05000316/2004006-01	FIN	Breaker Manipulation Error Resulted in Unit 2 Trip (Section 1R14.1)
50-316/2004-001-00	LER	Automatic Reactor Trip Due to Reactor Protection System Actuation While Manipulating Reactor Trip Bypass Breaker (Section 1R14.1)
05000316/2004006-02	FIN	Feedwater Control Error Resulted in Unit 2 Trip (Section 1R14.2)
50-316/2004-002-00	LER	Unplanned Automatic Reactor Protection System Actuation Due to Feedwater Transient During a Power Reduction (Section 1R14.2)
50-316/2002-005-01	LER	Unit 2 Trip Due to Instrument Rack 24-Volt DC [Direct Current] Power Supply Failure (Section 4OA3.1)
50-315/2003-003-01	LER	Supplemental LER for Dual Unit Manual Trip Due to the Failure of the Intake Traveling Screens and Failure to Comply with TS 3.8.1.1 (Section 4OA3.2)
05000315/316/2003002-01	URI	Essential Service Water System Water Hammer Load Calculation Concern (Section 4OA5.1)
05000315/316/2003002-02	URI	Estimation of Tube Blockage in the Component Cooling Water Heat Exchangers (Section 4OA5.2)
05000315/316/2004006-03	NCV	Inadequate Acceptance Criteria for Heat Exchanger Tube Blockage (Section 4OA5.2)

05000315/316/2003002-03	URI	Questionable Data Regarding Component Cooling Water Heat Exchanger As-Built Specification Sheet (Section 4OA5.3)

<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.

1R01 Adverse Weather Protection

- PMI-5055, "Winterization/Summarization," Revision 1
- PMP-5055-001-001, "Winterization/Summarization," Revision 0
- PMP-2291-SCH-002, "Work Control Seasonal Readiness Process," Revision 0A
- 12-IHP-5040-EMP-004, "Plant Winterization and De-Winterization," Revision 5
- PMP 2080-SWM-001, "Severe Weather Guidelines," Revision 0
- 12-OHP 4022.001.010, "Severe Weather," Revision 1
- Significant Operating Experience Report 02-1, "Severe Weather," December 3, 2002
- CR 04167035, "Temporary Modification Tags Were Found Still Hanging in the Plant After the Proceduralized Temporary Modification Was Closed," June 15, 2004
- CR 04163050, "During an NRC Walk Through There Were Some Noted Comments That Needed to Be Taken Care of," June 11, 2004

1R04 Equipment Alignment

- D. C. Cook Unit 2 TSs and Bases
- D. C. Cook Updated Final Safety Analysis Report, Revision 18.1
- 01-OHP-4021.056.001, "Filling and Venting Auxiliary Feedwater System," Revision 20
- 01-OHP-4021-056-002, "Auxiliary Pump Operation," Revision 21a
- 01-OHP-4021-032-001A, "Operating DG1AB Subsystems," Revision 4
- 01-OHP-4030-119-022E, "East Essential Service Water System Test," Revision 2b
- 02-OHP-4030-219-022E, "East Essential Service Water System Test," Revision 2b
- 12-OHP-4021-019-001, "Operation of the Essential Service Water System," Revision 27a
- OP-1-5106A-56, "Flow Diagram Auxiliary-Feedwater," Revision 56
- OP-1-5113-82, "Flow Diagram Essential Service Water," Revision 82
- OP-1-5113A-6, "Flow Diagram, Essential Service Water," Revision 6
- OP-2-5113-74, "Flow Diagram, Essential Service Water," Revision 74
- OP-2-5113A-8, "Flow Diagram, Essential Service Water," Revision 8
- OP-1-5151A-44, "Flow Diagram Emergency Diesel Generator 'AB', Unit No. 1," Revision 44
- OP-1-5151B-58, "Flow Diagram Emergency Diesel Generator 'AB', Unit No. 1," Revision 58

1R05 Fire Protection

- D. C. Cook Fire Hazards Analysis, Units 1 and 2, Revision 10
- D. C. Cook UFSAR, Section 9.8.1, "Fire Protection System," Revision 18

- D. C. Cook Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook, February 1995
- D. C. Cook 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
- Drawing 12-5975-4, "Fire Hazard Analysis Plan, El. 601'-0", 609'-0", 620'-6" to 625'-10", Units 1 & 2," Revision 4
- Drawing No. 12-5973, "Fire Hazards Analysis Basement Plan, El. 591'-0" and 587’-0", Revision 9
- Drawing No. 12-5974, "Fire Hazards Analysis Mezzanine Floor, El. 609'-0" Units 1 and 2," Revision 8
- CR 04113078, "ESAT Written at the Request of Operations to Document Oversite Driven Question Concerning the Float Voltage Indicator Meters Installed in Emergency Battery Light Units," April 22, 2004

1R06 Flood Protection Measures

- 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
- NRC Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," March 21, 2002
- CR 03162017, "Documenting a Systematic Approach and Identification of the Fundamental Building Blocks for the Cable Aging Management Program," June 11, 2003
- CR 01162003, "Plant Flooding During Heavy Rainstorm," June 11, 2001
- CR 01323022, "Program Controls for Protection Against Plant Flooding Need to Be Reviewed for Adequacy and Understanding by Plant Personnel," November 19, 2001
- CR 02088011, "Development of a Design Basis Document for Flood Protection," March 29, 2002
- CR 04151017, "2-VRS-2500 Unit 2 Vent Effluent Radiation Monitor is Flooded and Requires Draining," May 30, 2004
- CR 02086005, "Industry Operating Experience - NRC Information Notice 2002-12, 'Submerged Safety-related Electrical Cables'," March 27, 2002
- CR 03173016, "Water Seepage Into Areas Where It Should Not Be and Its Impact on Foundations, Buried Conduit and Embedded Commodities," June 22, 2003

1R12 Maintenance Effectiveness

- PMI-5035, "Maintenance Rule Program," Revision 11
- PMP-5035-MRP-001, "Maintenance Rule Program Administration, Revision 4
- "Pump and Valve Inservice Test Program for Donald C. Cook Nuclear Plant Third Ten Year Interval," Revision 3, December 20, 2001
- ASME/ANSI OMa-1988, "Inservice Testing of Valves in Light-Water Reactor Power

Plants," 1988

- Maintenance Rule Scoping Document for Containment Isolation Valve System, Revision 1
- CR P-00-03302, "2-CCW-135 Leaks Through or Is Stuck Open," February 26, 2000
- CR P-00-07699, "Containment Isolation Check Valve CCW-135 Leaked Excessively During Local Leak Rate Testing," May 27, 2000
- CR P-00-07823, "2-CCW-135 Valve Was Installed Incorrectly in Line," May 29, 2000
- CR 03127014, "2-CCW-135 Leaked at Greater Than the Administrative Limit During Local Leak Rate Testing," May 7, 2003
- CR 03137008, "Valve 2-CCW-135 Was Found Out of the Expected Configuration During Inspection," May 16, 2003
- CR 03155004, "Valve 2-CCW-135 Failed the As-left Local Leak Rate Test," June 4, 2003
- CR 02128037, "1-CCW-135 is Leaking by During Local Leak Rate Testing at Greater than La," May 8, 2002
- CR 03296019, "Local Leak Rate Test of Valve 1-CCW-135 Resulted in Leakage Greater than the Administrative Limit," October 23, 2003
- CR 03308067, "1-CCW-135 As-left Local Leak Rate Test Resulted in 4400 Cubic Centimeters Which Is Greater than the Administrative Limit of 1125 Cubic Centimeters," November 4, 2003
- CR 03254001, "2-CCW-176E Check Valve Slam During Performance of 02-OHP-4030-216-020W," September 10, 2003
- CR 03139078, "The Internals of 2-CCW-176W Are Degraded," May 19, 2003
- CR 03308072, "Check Valve 1-CCW-176E Failed the 'As-found' Inspection," November 4, 2003
- CR 00268020, "1-CCW-176E Excessive Wear on Various Parts," September 24, 2000
- CR 04001018, "Investigate and Take Long Term Action for Unit 2 Repeated Rod K-10 Failures," January 1, 2004
- CR 04033042, "Visually Inspect Control Rod Drive Shaft for K-10," February 2, 2004
- CR 04101006, "CRDM Cable Resistance Checks Indicate High Resistance on CRDM K-8," April 10, 2004
- CR 04101007, "CRDM Cable Resistance Checks Indicate High Resistance on CRDM H-10," April 10, 2004
- CR 04001006, "Control Bank "C" Rod K-10 Indicates that it has Possibly Dropped Partially During Reactor Startup," January 1, 2004
- CR 04033039, "Disconnect/Open, Clean and Inspect All CRDM Connectors on the Head During U2C15. Replace Any Degraded Connectors," February 2, 2004
- CR 04003014, "While Performing Post Cable Integrity Checks for Rod Control CRDM Cables, Found High Resistance on Lift and Movable Coils for Rod N-7, as well as, High Resistance on Movable Coil for Rod G-13," January 3, 2004
- CR 01025001, "Rod K-10 Does Not Appear to be Withdrawing," January 24, 2001
- CR 01029009, "Multiple Electrical Connection Problems in the Rod Control System Have Resulted in an Extended Forced Outage for Unit 2," January 26, 2001
- VTD-WEST-0490, "Westinghouse Instruction and Operating Book for Model L-106A Magnetic Control Rod Drive Mechanism," Revision 1
- Unit 2 Control Room Logs, January 1, 2004 through January 4, 2004

- Job Order C0017951-01, "Determine the Reason That 2-CRDM-K10 Indicated 25 Steps Lower," September 10, 1993
- Job Order 01025001-01, "Took Resistance Readings On All 3 Coils," March 1, 2001
- Job Order 01064022-01, "Several Deficiencies Were Noted During Westinghouse Clean and Inspect of RPI Connectors and Subsequent Testing," February 26, 2002
- Job Order R0211278- 09, "Install/Clean/Test U-2 CRDM and Messenger Cables," February 18, 2002
- Job Order 01064022-02, "U2, CRDM, Clean/Inspect All Connectors," February 20, 2002
- Job Order R0226551-09, "Install/Clean/Test U-2 CRDM and Messenger Cables," June 11, 2003
- Job Order 04001006-01, "Test Unit 2 CRDM and Messenger Connections," January 1, 2004
- Job Order 04001006-02, "2-CRDM-K10: Inspect/Clean Connections," January 1, 2004
- Job Order 04001006-04, "Inspect/Clean Connector at Reactor Head/Patch Board," January 1, 2004
- Job Order 04001002-07, "Replace Connector 2-CRDM-K10 at Reactor Head," January 2, 2004
- Job Order 04001006-06, "2-CRDM-K10: Perform TRS-001 On Rods F6 and K10," January 2, 2004
- Job Order 04002031-04, "2-CRDM-H14, Perform Resistance Checks on All Rods," January 3, 2004
- Job Order 04002031-10, "2-CRDM-H-14, Post Resistance Checks on All Rods," January 3, 2004
- Job Order R0246455-01, "Perform 12-IHP-6030-IMP-024 'As Found' and 'As Left'," January 3, 2004
- Job Order 04002031-05, "2-CRDM-H14, Repair or Replace Connector," January 3, 2004
- Job Order 01064022-03, "Fabricate Replacement Analog Rod Position Indicator Cables (Pre-Outage)," January 11, 2002
- Job Order 04002031-08, "2-CRDM-H14, Perform Characterization Testing on Cables to Coil," January 3, 2004
- Job Order 01087033-01, "Train 'A' Inspect SSPS Termi-Point Connections," February 6, 2002
- Job Order 01087033-03, "Repair/Replace Termi-Points as Required," February 12, 2002
- Job Order 010870330-02, "Train 'B' Inspect SSPS Termi-Point Connections," February 4, 2002
- Job Order 01087033-04, "Repair/Replace Termi-Points as Required," February 19, 2002
- Job Order R0102537-01, "Perform 12IHP6030.IMP.024," February 19, 2002
- Job Order R0204154-01, "Perform 2IHP4030.STP.518," February 24, 2002
- Job Order R0232898-01, "Characterize Unit 2 CRDM Coils/Cabinet 2-RCS-2D," May 6, 2003
- Job Order R0232898-02, "Characterize Unit 2 CRDM Coils/Cabinet," May 6, 2003

- Job Order R0226658- 01, "Perform 12-IHP-6030-IMP-024," June 11, 2003
- Job Order R0226952-01, "2-IHP-4030-STP-518, Rod Control Coil Current Test," June 18, 2003
- Job Order R0232899-01, "Characterize Unit 2 Analog Rod Position Indication," June 11, 2003

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

- D. C. Cook TSs and Bases
- D. C. Cook Updated Final Safety Analysis Report, Revision 18.1
- PMP-2291-OLR-001, "On-Line Risk Management," Revision 5
- PMP-2291-OLR-001, "On-Line Risk Management," Data Sheet 1, "Work Schedule Review and Approval Form," Cycle 49, Week 10, April 11, 2004 through April 17, 2004
- PMP-2291-OLR-001, "On-Line Risk Management," Data Sheet 1, "Work Schedule Review and Approval Form," Cycle 50, Week 3, May 9, 2004 through May 15, 2004
- Shift Manager's Logs, May 11, 2004
- PMP-2291-OLR-001, "On-Line Risk Management," Data Sheet 1, "Work Schedule Review and Approval Form," Cycle 50, Week 10, June 27, 2004 through July 3, 2004
- Shift Manager's Logs, June 27, 2004 through June 28, 2004
- Unit 1 Control Room Logs, April 13, 2004 through April 15, 2004
- 01-OHP-4030-116-020E, "East Component Cooling Water Loop Surveillance Test," Revision 1b
- PMP-2291-OLR-001, "On-Line Risk Management," Data Sheet 1, "Work Schedule Review and Approval Form," Cycle 49, Week 12, April 25, 2004 through May 1, 2004
- Unit 1 and Unit 2 Control Room Logs, April 25, 2004 through April 30, 2004
- 01-OHP-4022-064-001, "Control Air Malfunction," Revision 5
- 02-OHP-4022-064-001, "Control Air Malfunction," Revision 4
- PMP-2291-OLR-001 Attachment #1, "Contingency Plan for Removal of a CCW Pump or an Air Compressor," Revision 5
- Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May, 2000
- NUMARC 93-01, "Nuclear Energy Institute: Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3
- PS-1-95001-9, "Turbine and Generator Bus Wiring Diagram," Revision 9
- OP-1-98021-40, "Generator and Transformer Differential Elementary Diagram," Revision 40
- OP-1-12001-71, "Main Auxiliary One-Line Diagram Bus "A" and "B" Engineered Safety System," Revision 71
- General Electric Drawing 0202A7343, "Generator Neutral Connection," September 23, 1969
- General Electric Drawing 0153B8507, "Copper Neutral Connection," October 13, 1962
- General Electric Drawing 0104D5746, "Generator Neutral Enclosure," September 9, 1972
- CR 04120001, "The Unit One Main Generator (1-OME-81) Has a Loud Metallic Clanking, Rattling Noise Coming From the Area Near the Output Bushings," April 29, 2004
- ALTERREX-1, "Alterrex Voltage Regulator On-Line Calibration," Revision 2

- DTG-OPS-015, "Operational Decision Making," Data Sheet 1, "Operational Decision Making Checklist," Revision 2 for Unit 1 Main Generator Voltage Regulator Not Responding in Manual
- JO 04105056-11, "Post Maintenance Testing for Automatic Voltage Regulator Board," April 23, 2004
- Unit 1 Control Room Logs, April 20, 2004 through April 23, 2004
- CR 04111004, "Operations Reported that the Unit 1 Main Generator Voltage Could Not Be Adjusted Using the Manual Adjust Switch," April 20, 2004
- CR 04118049, "Based on Observation of the Approval and Use of a Contractor Procedure to Install and Calibrate the Unit 1 Main Generator Auto Voltage Regulator, Enhanced Guidance is Needed for the Overall Approval, Revision and Use of Contractor Procedures at D. C. Cook," April 27, 2004
- CR 04114035, "Bad Solder Joint Found on Newly Installed Automatic Voltage Regulator Card," April 23, 2004

<u>1R14</u> <u>Personnel Performance During Non-routine Plant Evolutions</u>

- CR 04089034, "An Automatic Reactor Trip of Unit 2 Occurred During Testing," March 29, 2004
- CR 04091072, "Low Resistance Circuit Between Neutral Bus of Control Drive Power Supply and Station Ground Will Block Operation of the Unit 2 Rod Control System Grounded Annunciator," March 31, 2004
- CR 04093002, "Over-voltage Units on DC Output of Rod Control Power Cabinet 24 Volt Power Supplies Not Tested as Part of Failed Power Supply Replacement," April 2, 2004
- CR 04091006, "Unexpected Reactor Trip During Reactor Trip Breaker Testing," March 31, 2004
- Licensee Event Report 50-316/2004-001-00, "Automatic Reactor Trip Due to RPS Actuation, While Manipulating Reactor Trip Bypass Breaker," May 26, 2004
- PMP-7030-001-001 Data Sheet 1, "Prompt NRC Notification, Event Notification Worksheet," March 29, 2004
- CR 04092007, "Urgent and Non-Urgent Failure Alarms Did Not Clear When the Full Length Power Alarm Reset Was Pushed While Placing the Control Rod Drive Motor Generator in Service," April 1, 2004
- Job Order 04092007-02, "2-RC-LC, Investigate Urgent and Non-Urgent Failure," April 2, 2004
- Job Order 04092007-03, "2-RC-LC, Investigate Urgent and Non-Urgent Failure," April 2, 2004
- Job Order 04092007-05, "2-RC-LC, Investigate Urgent and Non-Urgent Failure," April 2, 2004
- Job Order 04092007-07, "2-RC-LC, Investigate Urgent and Non-Urgent Failure," April 2, 2004
- Job Order 04092007-06, "2-RC-LC, Investigate Urgent and Non-Urgent Failure," April 2, 2004
- Job Order 04092007-08, "2-RC-LC, Investigate Urgent and Non-Urgent Failure," April 2, 2004
- Job Order 04092007-11, "2-RC-LC, Investigate Urgent and Non-

Urgent Failure," April 2, 2004
- Job Order 04092007-12, "2-RC-LC, Investigate Urgent and Non-Urgent Failure," April 2, 2004
- CR 04092018, "Cannibalize 1-RCS-1BD-PS1 For Use in Unit 2, To Be Used For 2-RCS-1BD-PS1 Replacement," April 1, 2004
- Job Order 04092018-01, "Cannibalize 1-RCS-1BD-PS1 For Use in 2-RCS-1BD-PS1," April 1, 2004
- Job Order 04092018-02, "Cannibalize 1-RCS-1BD-PS1 For Use in 2-RCS-1BD-PS1," April 3, 2004
- CR 04092020, "Cannibalize 1-RCS-2AC-PS1 For Use in Unit 2, To Be Used For 2-RCS-2AC-PS1 Replacement," April 1, 2004
- Job Order 04092020-01, "Cannibalize 1-RCS-2AC-PS1 For Use in Unit 2, To Be Used For 2-RCS-2AC-PS1 Replacement," April 1, 2004
- Job Order 04092020-02, "Cannibalize 1-RCS-2AC-PS1 For Use in Unit 2, To Be Used For 2-RCS-2AC-PS1 Replacement," April 3, 2004
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- Job Order 04089033-03, "2-52-BYB Investigate/Repair Breaker," April 3, 2004
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- OP-2-98236-2, "Rod Control Power Cabinet 1AC Auxiliaries Elementary Diagram," Revision 2
- OP-2-98240-3, "Rod Control Power Cabinet 1BD Auxiliaries Elementary Diagram," Revision 3
- OP-2-98244-3, "Rod Control Power Cabinet SCD Auxiliaries Elementary Diagram," Revision 3
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- CR 04103015, "While Starting Up Unit 2, Annunciator 210 Drop 22 'Rod Dropped or Rod Bottom' Alarmed While Withdrawing Control Bank "C" to 22.5 Steps," April 12, 2004
- CR 04103016, "While Starting Up Unit 2, Annunciator 210 Drop 22 'Rod Dropped or

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- CR 04100008, "2-CRCD-5, Main Turbine Control Bus Supply Breaker, Tripped During the Unit 2 Trip," April 8, 2004
- CR 04102013, "Rod Control Non-Urgent Failure Did Not Clear As Expected When Starting the Initial Control Rod Drive MG Set," April 11, 2004
- CR 04100002, "After Unit 2 Reactor Trip, Water was Discovered Coming from the C-S Condenser Hotwell Adjacent to the Condensate Booster Pump Emergency Leakoff Line Penetration," April 8, 2004
- CR 04101017, "East-West Movement of Condensate Booster Emergency Leakoff Line Caused Crack in Unit 2 South Condenser Penetration," April 10, 2004
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- OP-2-5129-41, "Flow Diagram Chemical and Volume Control System-Reactor Letdown and Charging," Revision 41

<u>1R15</u> <u>Operability Evaluations</u>

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C CR 04128006, "Pressure Read Too High During Turbine Driven Auxiliary Feedwater Pump Surveillance," May 7, 2004

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<u>1R16</u> <u>Operator Workarounds</u>

C PMP 4010-OWA-001, "Oversight and Control of Operator Workarounds," Revision 1

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C CR 02124003, "Unit 1 Main Turbine High Vibration After Manual Reactor Trip Required Partial Condenser Vacuum Breaking," May 4, 2002

C CR 01048019, "The Unit 1 Main Turbine Was Deliberately Slowed Due to High Vibration
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C CR 02195007, "1-OME-114 Lube Oil Purifier (Centrifuge) Caught Fire," July 14, 2002

C CR 02195008, "Unit 2 Centrifuge Has No Handbrake," July 14, 2002

C CR 02212063, "2-RH-128E Requires an Abnormal Amount of Torque to Adequately Seat," July 31, 2002

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- CR 03176003, "Essential Service Water Pump Room Temperature Sensors are Improperly Located Causing False 'High Temperature' Alarms and Creating Operator Workarounds to Monitor Room Conditions in Response to These False Alarms," June 25, 2003
- CR 03301073, "Unable to Maintain a Pressurized Gas Space in the Expansion Tanks for Both Unit's Control Room Air Conditioning Chill Water Systems," October 28, 2003
- CR 03326020, "High Differential Pressure Condition on the North Screen Wash Pump Discharge Strainer Occurred with Both Screen Wash Pumps in Service," November 22, 2003
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- CR 04173036, "2-KRV-792 Condensate Makeup to the Condensate Storage Tank Locked Up and Would Not Open Until Manually Pried Open With a 100% Demand Signal to Open," June 21, 2004

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- Unit 1 Technical Data Book Figure 1-19.1, "Power Operated Valve Stroke Time," Revision 70
- Unit 1 Technical Data Book Figure 1-15.1, "Safety Related Pump Inservice Test Hydraulic Reference," Revision 81
- Unit 1 Technical Data Book Figure 1-19.8, "Safety Related Throttled Valves," Revision 23
- Unit 1 Technical Data Book Figure 1-15.2, "Safety Related Pump Inservice Test Vibration Reference," Revision 75
- Unit 2 Technical Data Book Figure 2-19.1, "Power Operated Valve Stroke Time," Revision 61
- Unit 2 Technical Data Book Figure 2-15.1, "Safety Related Pump Inservice Test Hydraulic Reference," Revision 66
- Unit 2 Technical Data Book Figure 2-19.8, "Safety Related Throttled Valves," Revision 29
- Unit 2 Technical Data Book Figure 2-15.2, "Safety Related Pump Inservice Test Vibration Reference," Revision 56
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- Job Order 04105056-11, "Post Maintenance Testing for Automatic Voltage Regulator Board,"April 23, 2004
- Unit 1 Control Room Logs, April 20, 2004 through April 23, 2004
- CR 04111004, "Operations Reported that the Unit 1 Main Generator Voltage Could Not Be Adjusted Using the Manual Adjust Switch," April 20, 2004
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Enhanced Guidance is Needed for the Overall Approval, Revision and Use of Contractor Procedures at D. C. Cook," April 27, 2004

C CR 04114035, "Bad Solder Joint Found on Newly Installed Automatic Voltage Regulator Card," April 23, 2004

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C CR 04180002, "Unit 2 East ESW Pump Failed 02-OHP-5030-019-002E Surveillance and was Declared Inoperable," June 28, 2004

C CR 04181003, "Work Control Process for Support of Job Order 04180002-01 was not Ready to Support Work," June 29, 2004

C CR 04181058, "Unit 2 East ESW Strainer Inlet Slide Gate Rubber Seal Is Delaminated From the Slide Gate," June 29, 2004

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C Job Order 03346018-11, "1-OME-150-CD-EN, Perform Leak Inspection (Post Maintenance Test)," May 5, 2004

C CR 04127003, "Unit 1 CD Emergency Diesel Generator Full Load Exhaust Gas Temperature Band is 60 Degrees Fahrenheit Versus the Desired Band Width of 50 Degrees Fahrenheit," May 6, 2004

C CR 04126068, "Three High Pressure Injection Pumps Did Not Have the Correct Metering Rod Length," May 5, 2004

C CR 04126001, "Unit 1 CD Emergency Diesel #2 Front Cylinder Fuel Injector Has a Minor Fuel Leak," May 5, 2004

• CR 03313040, "During Valve Strokes on 1-QMO-201, Operations Observed Questionable Indication," November 9, 2003

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C Job Order 04075005-02, "2-SI-145S, Replace Drain Valve," May 18, 2004

C MHI-5075 Attachment 6, "ASME Section XI Repair/Replacement Checklist," Revision 5

C 12-MHP-5050-MWP-001 Data Sheet 4, "Maintenance Welding Process: Safety Related Weld Data Block," Revision 0

C 12-QHP-5050-NDE-001 Attachment 1, "Liquid Penetrant Examination Report," Revision 4

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C Job Order 04075005-03, "2-SI-146S, Replace Drain Valve," May 18, 2004

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- Job Order 040900012-03, "1-OME-4 Perform VT-2 System Leakage Test," April 6, 2004
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- Technical Data Book Figure 2-15.2, "Safety Related Pump Inservice Test Vibration Reference," Revision 56
- CR 04132003, "Unit 2 West Essential Service Water Pump 2-PP-7W Is Inoperable Due to Failed Surveillance Low Pump Differential Pressure at 63.6 Psid," May 11, 2004

<u>1R20</u> <u>Refueling Activities</u>

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- D. C. Cook Nuclear Plant UFSAR, Revision 18
- Shift Manager's Logs, March 28, 2004 through April 13, 2004

- 01-OHP-4021-001-001, "Plant Heatup From Cold Shutdown to Hot Standby," Revision 33
- 01-OHP-4021-001-004, "Plant Cooldown From Hot Standby to Cold Shutdown," Revision 40A
- 01 OHP 4021-017-002, "Placing In Service the Residual Heat Removal System," Revision 16B
- 01-OHP-4030-114-030, "Daily and Shiftly Surveillance Checks," Revision 2
- 01-OHP-4021-001-002, "Reactor Startup," Revision 30
- PMP 4100-SDR-001, "Plant Shutdown Safety and Risk Management," Revision 6
- CR 04073015, "A Peak of Increased Activity Was Found on History Plot of ERS-1301 and ERS-1401 (Unit 1 Lower Containment Particulate Airborne Monitors)," March 13, 2004
- CR 04089044, "1-ERS-1301 and 1-ERS-1401 (Unit 1 Lower Containment Particulate Airborne Monitors) Went Into Alert Alarm," March 29, 2004
- CR 04090012, "Upper Pressurizer Manway Is Leaking at Normal Operating Pressure and Temperature," March 30, 2004
- CR 04090033, "Number 13 Steam Generator Upper Manway Has a Steam Leak in Containment," March 30, 2004

<u>1R22</u> <u>Surveillance Testing</u>

- Unit 1 Technical Data Book Figure 1-19.1, "Power Operated Valve Stroke Time," Revision 70

C Unit 1 Technical Data Book Figure 1-15.1, "Safety Related Pump Inservice Test Hydraulic Reference," Revision 81
C Unit 1 Technical Data Book Figure 1-19.8, "Safety Related Throttled Valves," Revision 23
C Unit 1 Technical Data Book Figure 1-15.2, "Safety Related Pump Inservice Test Vibration Reference," Revision 75
• Unit 2 Technical Data Book Figure 2-19.1, "Power Operated Valve Stroke Time," Revision 61
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C CR 03244012, "Diesel Generator 2CD Started Too Fast When Aligned for a Slow Speed Start During Scheduled Surveillance," September 9, 2003
C CR 04107050, "2 CD Emergency Diesel Generator Conduit Loose at Coupling Connection," April 16, 2004
C CR 04107051, "2CD Emergency Diesel Generator Tubing Track is Loose," April 16, 2004
• CR 04094031, "While Isolating Non-Essential Service to 1-HV-CLV-2 Due to Coil Leaks 1-WCR-905 Did Not Get a Closed Indicating Light Lit," April 3, 2004
C D. C. Cook Unit 1 and Unit 2 TSs
C American Electric Power Design Information Transmittal B-2872-00, "Use of Plant Process Computer Indications for Accumulator Level Surveillances," April 16, 2004
C Unit 1 and Unit 2 Control Room Logs, April 17, 2004 through April 24, 2004
C CR 04105037, "Create Activity to Perform Ultrasonic Check of Level Instrument 1-ILA-141, #4 Accumulator Narrow Range Level Instrument Until Understanding of Other Narrow Range Level Instruments for Accumulators #1, 2, and 3 is Resolved," April 14, 2004
C CR 04108012, "While Performing Calibration of 1-ILA-140, a Leak Was Identified on the Transmitter," April 17, 2004
C CR 04108011, "While Performing Corrective Calibration, Found 1-ILA-110 to be Out-of-Specification," April 17, 2004
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C OP-2-5106A-51, "Flow Diagram Auxiliary Feedwater," Revision 51
C 01-OHP-4030-STP-027CD, "CD Diesel Generator Operability Test (Train A),"

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- Unit 2 Control Room Logs, May 14, 2004 through May 16, 2004
- CR 04136031, "2-SV-169W (Unit 2 West Motor Driven Auxiliary Feedwater Pump Suction Safety) Lifted During Performance of 02-OHP-4030-STP-017T," May 15, 2004
- CR 04136001, "2-OME-33, the Unit 2 Turbine Driven Auxiliary Feed Pump Suction Duplex Strainer Will Not Manually Shift Baskets," May 15, 2004
- CR 04136038, "2-OME-33 Strainer Chain Sprockets are Loose From the Screw Shafts," May 15, 2004
- CR 04137013, "Perform Lubrication of Auxiliary Feedwater Pump Strainer Shafts as Extent of Condition for the Unit 2 Turbine Driven Auxiliary Feedwater Pump Strainer Binding," May 16, 2004
- CR 04137024, "Strainer Petcock 1-FW-240-1 Could Not be Opened by Hand," May 16, 2004
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- CR 04145107, "Following Surveillance Testing on May 11, 2004 Could Not Find a Condition Report Written by Operations on Vibration Above the Alert Limit," May 24, 2004
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- CR 04142060, "As-found Voltage Readings Were Outside the Acceptable Band During the Performance of Quarterly Surveillance," May 21, 2004

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- Berrien County Early Warning Siren System Operation Manual; December 1, 2003
- Berrien County Emergency Preparedness Brochure
- Donald C. Cook Nuclear Power Plant Emergency Plan; Sections E.6, E.7, F.1, and G.1; Revision 19
- Donald C. Cook Nuclear Power Plant Site-Specific Offsite Radiological Emergency Preparedness
- Early Warning System Quality Assurance Verification Final Report; June 18, 1999
- 2004 Emergency Information Calendar for Berrien County

1EP3 Emergency Response Organization (ERO) Augmentation Testing

- D. C. Cook Nuclear Power Plant Emergency Plan; Sections B, E.2, and N; Revision 19
- PMP-2080-EPP-107; Notification; Revision 18
- Emergency Plan Administrative Manual, Attachment-5A; Drill/Exercise Objectives; Revision 0
- Post Order SPO.203; Emergency Response Organization (ERO) Pager Activation (Dialogic System); Revision 2
- D. C. Cook Nuclear Plant Emergency Planning Emergency Response Organization Phone Directory; January 29, 2004
- August 27, 2003 ERO Callout Unannounced Drill Records
- CR 04089018/SA-2004-SPS-012-QH; March 24, 2004 Off-Hours Unannounced Drill/Self-Assessment; April 31, 2004
- CR 04043033; ERO Callout Test Failure at Prairie Island; February 12, 2004
- CR 03055030; Unannounced Off-Hours Emergency Plan Drill; February 24, 2003

1EP4 Emergency Action Level and Emergency Plan Changes

- D. C. Cook Nuclear Plant Emergency Plan; Revisions 15, 16, 17, and 18

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

- PMP-7030-CAP-001; Corrective Action Program Process Flow; Revision 16
- ESAT 03118030; Site Protective Services/Emergency Planning Assessment of Emergency Response to the April 24, 2003, Alert Declaration; May 6, 2003
- PA-03-14; Emergency Planning Performance Assurance Audit; October 24, 2003
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- CR 04075060; Evaluation of the RP Aspects Relative to the Seal Water Injection Filter Leak That Occurred on December 19, 2003; March 15, 2004
- CR 04044093; Perform Aggregate Evaluation of Issues Related to the Unusual Event Declaration on December 19, 2003; February 13, 2004
- CR 03363005/SA-2004-SPS-003-QH; ESAT to Assess the Unusual Event Made on December 19, 2003/Self-Assessment; December 29, 2003
- CR 03358021/SA-2004-OPS-006-QH; Results of a Quick Hit Self Assessment to be Performed in January 2004 for Operations Emergency Plan Response; December 24, 2003
- CR 03282038; NRC Regulatory Issue Summary 2003-18; Use of NEI 99-01, "Methodology For Development of Emergency Action Levels," Revision 4, January 2003; October 19, 2003
- CR 03261019; During EP Audit PA-03-014, Discovered the Eplan Respiratory Protection
  Inventory Had Not Been Conducted on a Quarterly Basis; September 18, 2003
- CR 03245042; During EP Audit PA-03-014, Discovered the Daily Checks of Communications Links Between the Plant and the Sheriff’s Department, State Police and the NRC Were Not Always Performed and/or Documented; September 2, 2003
- CR 03129012; Operating Experience 16094 Failure to Complete Accountability Within 30 Minutes at Another Facility; May 9, 2003
- CR 03118028; During the Alert Declared on April 24, 2003, There Was an Approximately 22 Minute Delay Between the Declaration of the Alert and Activation of the ERO Pagers; April 24, 2003
- CR 03052010; Change the Eplan Regarding the Expectation for Called in Personnel to Report Immediately to Their Facility After Being Notified; February 21, 2003

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- PMP-2080-EPP-107, "Notification," Revision 18
- RMT-2080-TSC-001, "Activation and Operation of the Technical Support Center," Revision 4
- Timeline With Initial Actions, Emergency Response Drill, June 15, 2004
- Emergency Response Drill Exercise Messages, June 15, 2004
- EMD-32A, "Nuclear Plant Event Notification," Drill Messages for Declared Unusual Event, Alert and Site Area Emergency, June 15, 2004

<u>2PS2</u> <u>Radioactive Material Processing and Transportation</u>

- Updated Final Safety Analysis Report; Chapter 11; Revision 17
- Annual Radioactive Effluent Release Reports for 2002 and 2003; Tables Summarizing Solid Waste and Irradiated Fuel Shipments
- Radwaste/Radioactive Material Shipment Summary Log; April 2002 - May 6, 2004
- Documentation and Manifest for Shipment RMC-02-097; Reactor Coolant Pump Motor; July 31, 2002
- Documentation and Manifest for Shipment RMC-02-119; High Activity DAW; November 21, 2002
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- Documentation and Manifest for Shipment RMC-02-125; Spent Resin Storage Tank Resins; December 12, 2002
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- CR 04091045; Detachable Probe Not Used for Entire Transport Vehicle Survey; March 30, 2004
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<u>4OA1</u> <u>Performance Indicator Verification</u>

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- PMP-7110.PIP.001, "Regulatory Oversight Program Performance Indicators," Revision 1 and Revision 2
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- Key ERO Personnel and Drill Participation Dates; Third and Fourth Quarters 2003
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- EMD-32a; Nuclear Plant Event Notification Forms; July through December 2003
- Emergency Plan Performance Indicator Data Sheets; July through December 2003

<u>4OA3</u> <u>Event Follow-up</u>

- LER 50-316/2002-005-01: "Unit 2 Trip Due to Instrument Rack 24-Volt DC [Direct Current] Power Supply Failure," January 12, 2004
- CR 03038002, "Incorrect Information Transmitted in LER 50-316/2002-005-00," February 6, 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
- LER 50-315/2003-003-01, "Supplemental LER for Dual Unit Manual Trip Due to the Failure of the Intake Traveling Screens and Failure to Comply with Technical Specification 3.8.1.1," November 5, 2003
- NUREG 1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," Revision 2
- CR 03114018, "Both Unit Two EDGs Were Declared Inoperable at 0348 Due to Inadequate ESW Flow," April 24, 2003
- CR 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
- CR 03269028, "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

<u>4OA5</u> <u>Other Activities</u>

- CR P-00-10960; "Lifting of Unit 2 West Containment Spray Heat Exchanger Safety Valve;" August 6, 2000
- CR 01282046; "NRC Questioned Conduction of Generic Letter 89-13 As-found Inspections in Reference to Functionality During Previous Operating Period;" October 10, 2001
- CR 01046029; "Ineffective Evaluation of Past Condition Involving Integrity of the Component Cooling Water Heat Exchanger Baffle Plates;" February 16, 2001
- CR 03126016; "Channel Cover (Dollar Plate) Pass Partition Groove on Unit 1 West Component Cooling Water Heat Exchanger Deteriorated;" May 7, 2003
- CR 03083036; "NRC Questioned Approach for Assessing the As-found Condition of Generic Letter 89-13 Heat Exchangers;" March 25, 2003
- CR 03124006; "Divider Plate on U1 West Component Cooling Water Heat Exchanger Found Tore Away;" May 3, 2003
- CR 04076006; "Unresolved Items Opened in NRC Inspection Report;" March 16, 2004
- CR 04106040; "Organizational Ineffectiveness Allowed Incomplete Corrective Action to

Go Undetected;" April 15, 2004*

- CR 0410739; "Essential Service Water Pipe Support Has One of Four Bolts Not Perpendicular to Wall;" April 16, 2004*
- 1-ESW-43; Essential Service Water Isometric Auxiliary Building Elevation 628'-3"; Revision 12
- 1-ESW-44; Essential Service Water Isometric Auxiliary Building Elevation 620'-3"; Revision 10
- 1-ESW-64; Essential Service Water Isometric Auxiliary Building Elevation 643'-0"; Revision 9
- 2-ESW-53; Essential Service Water Isometric Auxiliary Building Elevation 628'-3"; Revision 13
- 2-ESW-58; Essential Service Water Isometric Auxiliary Building Elevation 616'-7"; Revision 8
- 1-GESW-R23; Hanger Detail Drawing; Revision 6
- 1-GESW-R24; Hanger Detail Drawing; Revision 7
- 1-GESW-R63; Hanger Detail Drawing; Revision 6
- 1-GESW-V16; Hanger Detail Drawing; Revision 6
- 2-GESW-R31; Hanger Detail Drawing; Revision 8
- 2-GESW-R32; Hanger Detail Drawing; Revision 5
- 2-GESW-V6; Hanger Detail Drawing; Revision 5
- 31760; ML Worthington Heat Exchanger Specification Sheet; September 5, 1972
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- MDS-607; Heat Exchanger Tube Plugging; Revision 5
- TS-O-3020; Perform Generic Letter 89-13 Program Field Inspection; Revision 0
- 12-MHP-5030-016-001; Component Cooling Water Heat Exchanger Inspection, Cleaning and Tube Plugging; Revision 5
- 12-MHP-5030-016-002; Emergency Diesel Generator Engine Jacket Water and Lube Oil Heat Exchanger Disassembly, Inspection, Cleaning, Tube Plugging and Assembly; Revision 4
- WO R0227595; Unit 2 East Component Cooling Water Heat Exchanger: Open, Inspect, Clean and Close Heat Exchanger; May 31, 2003
- WO R0227597; Unit 2 West Component Cooling Water Heat Exchanger: Open, Inspect, Clean and Close Heat Exchanger; May 27, 2003
- WO R0244993; Unit 1 West Component Cooling Water Heat Exchanger: Open, Inspect, Clean and Close Heat Exchanger; November 1, 2003
- WO R0245571; Unit 1 East Component Cooling Water Heat Exchanger: Open, Inspect, Clean and Close Heat Exchanger; November 8, 2003

C CR 03251002, "Significant Event Notification, SEN 242 - Loss of Grid Event, August 14, 2003," September 8, 2003

C PMP-3100-IOA-001, "Inter-Organizational Agreement Between the American Electric Power Energy Delivery and the Nuclear Generation Group for Assistance to Cook Nuclear Plant," Revision 0

C Maintenance Rule Scoping Document for Offsite Power System, Revision 2

C 01-OHP-4030-114-021 Data Sheet 20, "Event Initiated Surveillances - Inoperable Power
Supply," Revision 2

C 01-OHP-4030-114-031 Attachment 2, "Operation Weekly Surveillance Checks - Bus

Voltage Supplement," Revision 2

- 02-OHL-4030-SOM-041, "Unit 2 Tours," Revision 3
- 02-OHL-4030-SOM-012, "Unit 2 Tours," Revision 6
- D. C. Cook Nuclear Plant UFSAR, Revision 18

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
AC	Alternating Current
AEP	American Electric Power
AFW	Auxiliary Feedwater
ANS	Alert and Notification System
ASME	American Society of Mechanical Engineers
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRDM	Control Rod Drive Mechanism
CTS	Containment Spray
DAW	Dry Active Waste
DC	Direct Current
DOT	Department of Transportation
DRP	Division of Reactor Projects
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EP	Emergency Preparedness
ESW	Essential Service Water
ESF	Engineered Safety Feature
ERO	Emergency Response Organization
GDC	General Design Criteria
gpm	Gallons-Per-Minute
IMC	Inspection Manual Chapter
LER	Licensee Event Report
LCO	Limiting Condition for Operation
LSA	Low Specific Activity
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OA	Other Activities
OHP	Operations Head Procedure
OWA	Operator Work-Around
PARS	Publically Available Records
PI	Performance Indicator
PMI	Plant Manager's Instruction
PMP	Plant Manager's Procedure
PORV	Power Operated Relief Valve
psig	Pounds Per Square Inch Gauge

Radwaste	Radioactive Waste
RCS	Reactor Coolant System
ROP	Reactor Oversight Process
RP	Radiation Protection
RTO/TSO	Regional Transmission Organization/Transmission System Operator
SCO	Surface Contaminated Object
SDP	Significance Determination Process
SG	Steam Generator
SSCs	Structures, Systems, and Components
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WO	Work Order