December 18, 2003

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

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNIT 2
NRC INSPECTION REPORT 05000316/2003014(DRP)

Dear Mr. Nazar:

On November 21, 2003, the U. S. Nuclear Regulatory Commission (NRC) completed a supplemental inspection using Inspection Procedure 95001, “Inspection For One Or Two White Inputs In A Strategic Performance Area,” at your D. C. Cook Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on November 21, 2003, with you and other members of your staff.

In July 2003, your 2nd Quarter 2003 Performance Indicator submittal reported that recent Unit 2 plant trips with the loss of the normal heat removal path to the main condenser had exceeded the Green/White threshold for the Scrams With Loss of Normal Heat Removal performance indicator. This represented a reduction in safety margin characterized by a White performance indicator and adversely affected the Initiating Events cornerstone. The reduced safety margin associated with this performance indicator warranted a supplemental NRC inspection and assessment of your actions to improve performance in the Initiating Events cornerstone of the Operational Reactor Safety strategic performance area.

Based on our review of your root cause evaluations for the individual plant trips and your cumulative evaluation of all three events, we have concluded that your staff adequately identified the underlying root causes and contributing causes for these trips. The evaluations were determined to be generally thorough and followed a structured approach for performing such reviews. We also concluded that your staff’s planned corrective actions, if properly implemented, are sufficient to adequately address each of the identified root and contributing causes.

While the root cause evaluation was generally thorough, we identified that you failed to identify that post-trip reports did not consistently identify the root causes and contributing causes of plant trips. We also identified that your planned corrective actions to address the White performance indicator did not initially include an effectiveness review that contained the necessary elements to ensure that specific corrective actions were effective. Both of these areas warrant your attention.

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

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

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

Sincerely,

/RA/

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

Docket No. 50-316
License No. DPR-74

Enclosure: Inspection Report 05000316/2003014(DRP)
w/Attachment: Supplemental Information

cc w/encl: Site Vice President
M. Finissi, Plant Manager
R. Whale, Michigan Public Service Commission
Michigan Department of Environmental Quality
Emergency Management Division
MI Department of State Police
D. Lochbaum, Union of Concerned Scientists

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-316
License Nos:	DPR-74
Report No:	05000316/2003014(DRP)
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Unit 2
Location:	1 Cook Place Bridgman, MI 49106
Dates:	November 17 through November 21, 2003
Inspector:	G. Wilson, Senior Resident Inspector
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000316-03-14(DRP); 11/17/2003-11/21/2003, D. C. Cook Nuclear Power Plant, Unit 2; Supplemental Inspection - Scrams With Loss of Normal Heat Removal White Performance Indicator.

This report covers a supplemental inspection performed by the Duane Arnold Senior Resident Inspector. This inspection identified two Green findings which involved three associated Non-Cited Violations (NCVs). 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.

The NRC performed this supplemental inspection in accordance with Inspection Procedure 95001, “Inspection For One Or Two White Inputs In A Strategic Performance Area,” to assess the licensee’s evaluation associated with a White performance indicator in the Scrams With Loss of Normal Heat Removal area of the Initiating Events cornerstone. During this supplemental inspection, the inspector determined that the licensee’s overall evaluation of the Scrams With Loss of Normal Heat Removal performance indicator was acceptable. The licensee utilized a structured approach to evaluate the circumstances of the individual plant trips and the collective significance of the three trips which led to the White performance indicator to identify potential common causes.

The licensee’s corrective actions for each of the plant trips contributing to the White performance indicator were determined to correspond with the root and contributing causes identified by the root cause evaluations. At the conclusion of the inspection, the corrective actions were either completed or were being tracked for completion. The licensee had also established a process for performing reviews to assess the effectiveness of these corrective actions.

Given the licensee's acceptable performance in addressing the root causes and contributing causes of the individual plant trips which contributed to the Scrams With Loss of Normal Heat Removal White performance indicator, the White performance indicator will only be considered in the assessment of plant performance for a total of 4 quarters in accordance with the guidance in Inspection Manual Chapter 0305, "Operating Reactor Assessment Program."

The two findings of very low safety significance which were identified during the inspection are summarized below.

A. <u>Inspector-Identified and Self-Revealed Findings</u>

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspector when licensee personnel failed to have an adequate reactor trip response procedure and failed to take prompt and adequate corrective actions to address excessive reactor coolant system cooldown following a reactor trip. The primary cause of this finding was related to the cross-cutting area of Problem Identification and Resolution.

The finding was more than minor because the finding was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was of very low safety significance, since it did not impact equipment operability, did not represent an actual loss of safety function of a system or train of safety-related or risk-significant equipment, and was not potentially risk significant due to external events. Corrective actions to address this issue included revising emergency operating procedures to reduce reactor coolant system cooldown by means that did not result in the loss of the normal heat removal path to the main condenser. One Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified. One Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was also identified. (Section 2.0.1.b)

- Green. A finding of very low safety significance was identified by the inspector when licensee personnel failed to adhere to a procedure and closed main steam isolation valves prematurely following a reactor trip. The primary cause of this finding was related to the cross-cutting area of Human Performance.

  The finding was more than minor, because the finding was associated with the Human Performance attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was of very low safety significance, since it did not impact equipment operability, did not represent an actual loss of safety function of a system or train of safety-related or risk-significant equipment, and was not potentially risk significant due to external events. Corrective actions to address this issue included revising emergency operating procedures to reduce reactor coolant system cooldown by means that did not result in the loss of the normal heat removal path to the main condenser. One Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified. (Section 2.0.2.b)

B. **Licensee-Identified Violations**

None.

Report Details

01 Inspection Scope

This supplemental inspection was performed to assess the licensee's root cause evaluation associated with the Unit 2 performance indicator in the Scrams With Loss of Normal Heat Removal area of the Initiating Events cornerstone which exceeded the Green/White threshold in the 2nd quarter of 2003. The three trips which involved the loss of the normal heat removal path to the main condenser and which caused this performance indicator Green/White threshold to be exceeded are described below:

- An automatic reactor trip occurred on May 12, 2002, due to the failure of two redundant power supplies in a reactor control instrumentation cabinet. The failure of the power supplies resulted in the closure of a feedwater regulating valve and caused a reactor trip on low feedwater flow coincident with low steam generator level. In addition, condenser steam dump controls were disabled, resulting in a loss of the normal heat removal path to the main condenser.

- An automatic reactor trip occurred on July 22, 2002, due to a loss of condenser vacuum that occurred during condenser waterbox cleaning. Operators closed the main steam isolation valves (MSIVs) to address excessive reactor coolant system (RCS) cooldown. The closure of the MSIVs isolated the normal heat removal path to the main condenser.

- A manual reactor trip occurred on April 24, 2003, due to a large influx of fish which significantly degraded circulating water flow. The degraded circulating water flow resulted in a loss of condenser vacuum and a loss of the normal heat removal path to the main condenser.

This supplemental inspection was performed in accordance with Inspection Procedure 95001, "Inspection For One Or Two White Inputs In A Strategic Performance Area." The following inspection results are organized by the specific inspection requirements of Inspection Procedure 95001 which are noted in italics in each section.

02 Evaluation of Inspection Requirements

02.01 Problem Identification

a. *Determination of who (i.e., licensee, self-revealing, or NRC) identified the issue and under what conditions.*

The May 12, 2002, trip was credited as a Scram With Loss of Normal Heat Removal performance indicator occurrence due to a loss of the steam dump controls.

The July 22, 2002, trip was credited as a Scram With Loss of Normal Heat Removal performance indicator occurrence due to the closure of the MSIVs which isolated the normal heat removal path to the main condenser.

The April 24, 2003, trip was credited as a Scram With Loss of Normal Heat Removal performance indicator occurrence due to a loss of circulating water flow that resulted in a loss of main condenser vacuum.

All of the trips were self-revealing events.

b. *Determination of how long the issue existed, and prior opportunities for identification.*

The D.C. Cook Unit 2 Scrams With Loss of Normal Heat Removal performance indicator exceeded the Green/White threshold as reported in the 2nd Quarter 2003, D.C. Cook Performance Indicator submittal.

The practice of closing MSIVs to address post-trip RCS cooldown concerns rather than by other means, such as reducing auxiliary feedwater flow to the steam generators, was an identified root cause in the licensee's root cause evaluation report.

This practice was questioned by the D.C. Cook Senior Resident Inspector following reactor trips on May 12, 2002, and July 22, 2002, and the licensee generated condition report (CR) 02305075 to enter this issue into their corrective action program, but failed to take prompt and adequate corrective action to address this issue. This was considered a prior opportunity for identification of a problem and resulted in the following finding.

Introduction

A finding of very low safety significance (Green) and associated Non-Cited Violations (NCVs) of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," and 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," were identified for the failure to have an adequate reactor trip procedure and the failure to take prompt and adequate corrective actions to address excessive RCS cooldown following a reactor trip.

Discussion

While performing a review of the licensee's root cause evaluation for the Scrams With Loss of Normal Heat Removal performance indicator, the inspector identified that the licensee failed to have an adequate reactor trip procedure and failed to take prompt and adequate corrective actions to address an excessive post-trip RCS cooldown issue.

During an extended dual-unit shutdown in 2000, licensee personnel revised Emergency Operating Procedures (EOPs) to incorporate guidance contained in generic Emergency Response Guidelines (ERGs). During this effort, on June 4, 2000, Emergency Operating Procedure 02-OHP-4023-E-0, "Reactor Trip or Safety Injection," was revised to remove a step which throttled auxiliary feedwater flow to the steam generators. During the review of this revision, licensee personnel failed to consider a D.C. Cook design feature which immediately tripped the main feed pumps following a reactor trip which, when combined with the procedure revision, resulted in an excessive post-trip

RCS cooldown. The excessive RCS cooldown necessitated the closure of the MSIVs prior to reducing the cooldown rate by other means, which isolated the normal heat removal path to the main condenser.

The practice of closing MSIVs to address post-trip RCS cooldown was questioned by the D.C. Cook Senior Resident Inspector following reactor trips on May 12, 2002, and July 22, 2002. The licensee entered this issue into their corrective action program as CR 02305075. However, no action to address this issue was taken until after a third reactor trip occurred on February 5, 2003, when the MSIVs were again closed to address excessive RCS cooldown and more than 6 months after the inspector brought this issue to the attention of licensee personnel.

After the licensee reviewed the issue following the February 5, 2003, trip, Emergency Operating Procedure 02-OHP-4023-E-0, "Reactor Trip or Safety Injection," was revised on February 12, 2003. The revision added the capability to reduce auxiliary feedwater flow to the steam generators and effectively addressed post-trip RCS cooldown without requiring the closure of the MSIVs. The revised procedure was proved effective following the April 24, 2003, trip when auxiliary feedwater flow was reduced to address RCS cooldown and the MSIVs were not closed.

<u>Analysis</u>

The inspector determined that the failure to promptly and adequately address excessive post-trip RCS cooldown as a result of an inadequate reactor trip procedure was a performance deficiency warranting a significance evaluation. The Mitigating Systems cornerstone was impacted by this performance deficiency. The inspector also determined that this finding affected the cross-cutting area of Problem Identification and Resolution.

The inspector reviewed the examples 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 that appropriately described this issue. The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Disposition Screening," since the reactor trip procedure which failed to adequately address the excessive RCS cooldown and resulted in the closure of the MSIVs was associated with the Procedure Quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the normal heat removal path to the main condenser was isolated as a result of the deficiency.

In accordance with IMC 0609, "Significance Determination Process (SDP)," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," a Phase 1 SDP was initiated. In accordance with the SDP Phase 1, "Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barrier Integrity]," the inspector determined that since the finding was not a design or qualification deficiency that resulted in a loss of function per Generic Letter 91-18; did not represent an actual loss of safety function of a system; did not represent an actual loss of a safety function of a single train for greater than its Technical Specification Allowed Outage

Time; did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk-significant; and was not potentially risk significant due to seismic, fire, flooding or severe weather, that the finding screened out as Green.

Enforcement (1)

10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires, in part, that measures be established to assure that conditions adverse to quality, such as deficiencies, deviations, and nonconformances are promptly identified and corrected. Contrary to the above, licensee personnel failed to take prompt and adequate corrective actions to address excessive post-trip RCS cooldown, a condition adverse to quality, following reactor trips on May 12, 2002, and July 22, 2002, which resulted in closing main steam isolation valves and the loss of the normal heat removal path to the main condenser following a reactor trip on February 5, 2003.

However, because of the very low safety significance of this issue and because this issue was entered into the licensee’s corrective action program, this violation is being treated as a Non-Cited Violation (5000316/2003014-01), consistent with Section VI.A of the NRC Enforcement Policy. This issue was entered into the licensee’s corrective action program as CR 03325025. Corrective actions to address this issue included revising emergency operating procedures to reduce reactor coolant system cooldown by means that did not result in the loss of the normal heat removal path to the main condenser.

Enforcement (2)

10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented procedures of a type appropriate to the circumstances. Contrary to the above, Emergency Operating Procedure 02-OHP-4023-E-0, “Reactor Trip or Safety Injection,” which was revised on June 4, 2000, was not appropriate to the post-trip circumstances since it failed to address the immediate loss of main feedwater with a reactor trip which resulted in the premature closure of the main steam isolation valves and the isolation of the normal heat removal path to the main condenser.

However, because this violation was associated with a finding of very low safety significance and because this issue was entered into the licensee’s corrective action program, this violation is being treated as a Non-Cited Violation (5000316/2003014-02), consistent with Section VI.A of the NRC Enforcement Policy. This issue was entered into the licensee’s corrective action program as CR 03325026. Corrective actions to address this issue included revising emergency operating procedures to reduce reactor coolant system cooldown by means that did not result in the loss of the normal heat removal path to the main condenser.

c. *Determination of the plant-specific risk consequences (as applicable) and compliance concerns associated with the issue.*

In response to each of the three reactor trips, the resident inspectors evaluated plant parameters, operator actions, and overall plant status including the availability of mitigating systems. For the trips on May 12, 2002, and June 22, 2002, the inspectors determined that all systems responded as designed, the trips were not complicated by material condition deficiencies, and no human performance errors complicated the event response. Following the trip on April 24, 2003, a number of equipment performance anomalies occurred. As a result, the events were reviewed in more detail and were the subject of a special inspection conducted in accordance with Inspection Procedure 93812, "Special Inspection." The NRC concluded that each individual event was of low risk significance.

At the beginning of this inspection, the licensee had not evaluated the risk significance of the three individual events to determine whether a change in initiating event frequency for trips which resulted in a loss of the normal heat removal path caused a substantive increase in core damage frequency. Subsequently, on November 19, 2003, the licensee completed an assessment of the impact of these events on core damage frequency. Using Bayesian updating methods, the licensee calculated a revised initiating event frequency. The increase in the initiating event frequency only resulted in a very small increase in the core damage frequency of approximately 6.3E-08. The licensee's risk analysis was considered to be acceptable. No concerns were identified.

02.02 Root Cause and Extent of Condition Evaluation

a. *Evaluation of method(s) used to identify root cause(s) and contributing cause(s).*

The licensee performed a root cause evaluation for each of the three plant trips which caused the Scrams With Loss of Normal Heat Removal performance indicator to cross the Green/White threshold. A root cause evaluation was also performed to determine if any potential common causes for the three events existed. These root cause evaluations are listed below.

- CR 0213302, "Unit 2 Automatic Reactor Trip on May 12, 2002, Due to Dual Power Supply Failure"
- CR 02203001, "Unit 2 Automatic Reactor Trip on July 22, 2002, Due to Low Condenser Vacuum"
- CR 03114004, "Unit 2 Manual Reactor Trip on April 24, 2003, Due to Large Fish Intrusion"
- CR 03199051, "Loss of Normal Heat Removal Performance Indicator"

The four root cause evaluations were conducted using a structured methodology to evaluate the root causes and contributing causes of the events. These included event and casual factors analyses, failure mode identification, human failure mode analysis, change analysis, and WHY staircase methodologies. The licensee used a combination of these root cause analysis techniques to evaluate the trips in accordance with Plant Management Instruction (PMI) 7030, "Corrective Action Plan;" Plant Management Procedure (PMP) 7030-CAP-001, "Corrective Action Program Process Flow;" and the

"D.C. Cook Nuclear Plant Equipment Root Cause Analysis Desk Top Guide." The documented root cause evaluations adequately described the methods used to identify the root causes for the events.

The inspector reviewed the methods employed and concluded that the licensee had used a formal, structured approach to perform the root cause evaluations to identify root causes and contributing causes. No concerns were identified.

b. *Level of detail of the root cause evaluation.*

The four root cause evaluations were performed in accordance with PMI-7030, "Corrective Action Plan;" PMP-7030-CAP-001, "Corrective Action Program Process Flow;" and the "D.C. Cook Nuclear Plant Equipment Root Cause Analysis Desk Top Guide." These procedures provided sufficient guidance for personnel to follow a structured and methodical approach to evaluate the events. The inspector determined that the four root cause evaluations were performed with sufficient detail and analysis to support the conclusions reached. The root cause evaluations adequately considered previous operating experience, organizational response, human error, programmatic weaknesses, procedure and training adequacy, external events, and communications. In addition, each of the four root cause evaluations adequately incorporated internal and external operating experience into the scope of review. The analysis techniques chosen were considered to be appropriate to each particular event and each of the identified failure modes. These failure modes were then used to identify the root causes and contributing causes.

The license's root cause evaluations identified two primary root causes. The first was the failure to implement an effective Equipment Reliability Program and resolve long-standing and repetitive equipment problems. The second was that Emergency Operating Procedure 02-OHP-4023-E-0, "Reactor Trip or Safety Injection," was inadequate since it did not contain the steps necessary to mitigate excessive RCS cooldown without necessitating the early closure of the MSIVs, resulting in the loss of the normal heat removal path to the main condenser.

During the review of the root cause evaluation associated with the July 22, 2002, reactor trip the inspector identified that the operating crew failed to adhere to Operating Head Instruction (OHI) 4023, "Abnormal Emergency Procedures User Guide," during the implementation of 02-OHP-4023-E-0, "Reactor Trip or Safety Injection." This resulted in the following finding.

<u>Introduction</u>

A finding of very low safety significance (Green) and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified when licensee personnel failed to adhere to an operating procedure and closed main steam isolation valves prematurely following a reactor trip.

Discussion

Following a reactor trip, operators are directed to perform the actions prescribed by Emergency Operating Procedure 02-OHP-4023-E-0, "Reactor Trip or Safety Injection," and are then directed, if a safety injection is not required, to Emergency Operating Procedure 02-OHP-4023-ES-0.1, "Reactor Trip Response." Emergency Operating Procedure 02-OHP-4023-E-0, Revision 15a, did not include actions to address RCS cooldown. Actions to address RCS cooldown concerns, at the time this revision was in effect, were prescribed in Emergency Operating Procedure 02-OHP-4023-ES-0.1, Revision 16, and included actions to stop dumping steam, reduce flow to the steam generators from the auxiliary feedwater system, and finally to close the main steam isolation valves.

The inspector reviewed the root cause evaluation associated with the July 22, 2002, reactor trip, and determined that during operator response to the event, MSIVs were closed to address excessive RCS cooldown. The inspector determined that this action was accomplished prior to completing the actions prescribed by Emergency Operating Procedure 02-OHP-4023-E-0, and before entering Emergency Operating Procedure 02-OHP-4023-ES-0.1. As a result, actions prescribed by this procedure, such as reducing auxiliary feedwater flow to the steam generators, were not accomplished.

The inspectors reviewed the operators' actions in this matter and determined that the decision to take actions outside those specified in Emergency Operating Procedure 02-OHP-4023-E-0 isolated the normal heat removal path to the main condenser unnecessarily, since operators had the ability to address RCS cooldown by reducing auxiliary feedwater flow following completion of the actions specified in Emergency Operating Procedure 02-OHP-4023-E-0 and after entering Emergency Operating Procedure 02-OHP-4023-ES-0.1.

The inspectors also determined that Section 4.5.9 of OHI-4023, "Abnormal Emergency Procedures User Guide," prohibited the concurrent performance of multiple emergency operating procedures unless specifically directed by the procedure in effect. Therefore, since Emergency Operating Procedure 02-OHP-4023-E-0 did not provide direction to close MSIVs to address excessive RCS cooldown, by closing the MSIVs prior to entering 02-OHP-4023-ES-0.1, "Reactor Trip Response," licensee personnel effectively performed two emergency operating procedures concurrently.

Analysis

The inspector determined that the failure to adhere to the procedure for a reactor trip and close main steam isolation valves prematurely was a performance deficiency warranting a significance evaluation. The Mitigating Systems cornerstone was impacted by this performance deficiency.

The inspector reviewed the examples 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 that appropriately described this issue. The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Disposition Screening,” since the failure to adhere to procedures which prohibited the concurrent performance of emergency operating procedures was associated with the Human Performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the normal heat removal path to the main condenser was isolated as a result of the deficiency.

In accordance with IMC 0609, “Significance Determination Process (SDP)," Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," a Phase 1 SDP was initiated. In accordance with the SDP Phase 1, “Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barrier Integrity],” the inspector determined that since the finding was not a design or qualification deficiency that resulted in a loss of function per Generic Letter 91-18; did not represent an actual loss of safety function of a system; did not represent an actual loss of a safety function of a single train for greater than its Technical Specification Allowed Outage Time; did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk-significant; and was not potentially risk significant due to seismic, fire, flooding or severe weather, that the finding screened out as Green.

Enforcement

10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented procedures of a type appropriate to the circumstances and shall be accomplished in accordance with these procedure. Contrary to the above, on July 22, 2002, the operating crew failed to adhere to OHI-4023, “Abnormal Emergency Procedures User Guide,” a quality procedure, which prohibited concurrent performance of multiple emergency operating procedures and closed main steam isolation valves, an action prescribed by Emergency Operating Procedure 02-OHP-4023-ES-0.1, “Reactor Trip Response,” Revision 16 while performing Emergency Operating Procedure 02-OHP-4023-E-0, “Reactor Trip or Safety Injection,” Revision 15a, following a reactor trip.

However, because this violation was associated with a finding of very low safety significance and because this issue was entered into the licensee’s corrective action program, this violation is being treated as a Non-Cited Violation (5000316/2003014-03), consistent with Section VI.A of the NRC Enforcement Policy. This issue was entered into the licensee’s corrective action program as CR 03325028. Corrective actions to address this issue included revising emergency operating procedures to reduce reactor coolant system cooldown by means that did not result in the loss of the normal heat removal path to the main condenser.

c. *Consideration of prior occurrences of the problem and knowledge of prior operating experience.*

The licensee’s root cause evaluation identified that the lack of a timely response to industry and facility operating experience contributed to all three performance indicator occurrences.

There were several industry events related to fish intrusion and loss of circulating water prior to the April 24, 2003, reactor trip. In addition, the licensee experienced a previous fish intrusion event in 1996 and a silting/mud intrusion event in 2001. However, none of these previous events had been adequately considered to prevent the trips which contributed to the Scrams With Loss of Normal Heat Removal White Performance Indicator.

The licensee also identified that facility operating experience was available in reactor trip reports which involved post-trip excessive RCS cooldown events. The inspector reviewed this information and identified that these reports failed to identify that post-trip reports did not consistently identify the root causes and contributing causes of plant trips which could have aided in the assessment of the excessive RCS cooldown rate. The licensee acknowledged this weakness in their root cause evaluation and generated CR 03325029 to enter this issue in their corrective action program.

Overall, the inspector concluded that the licensee's root cause evaluations properly considered and evaluated prior operating experience.

d. Consideration of potential common cause(s) and extent of condition of the problem.

The licensee's common cause analysis was a collective evaluation of the events which caused the Scrams With Loss of Normal Heat Removal performance indicator to cross the Green/White threshold. The evaluation identified the following common causes associated with the events:

- An inadequate emergency operating procedure led to the premature closure of the MSIVs, resulting in the unnecessary isolation of the normal heat removal path to the main condenser.
- A lack of an effective equipment reliability program led to failures with power supplies and a loss of voltage to the control rod drive mechanism that resulted in reactor trips.
- Licensee staff and management inappropriately accepted MSIV closure as an adequate means to mitigate excessive RCS cooldown.
- Plant trip response training was inadequate due to simulator modeling weaknesses.

Overall, the inspector concluded that the licensee's root cause evaluation adequately evaluated the potential for common cause among the events.

The inspector also reviewed the licensee's extent of condition evaluation and concluded that licensee personnel adequately evaluated the extent of condition among the events.

02.03 Corrective Actions

a. *Appropriateness of corrective action(s).*

The inspector reviewed each of the four root cause evaluations and the associated corrective actions. The corrective actions were clearly described and were entered into the licensee's tracking system. The established corrective actions were determined to appropriately address the root causes and contributing causes of the events and if properly implemented would address the problem identified within each of the root cause evaluations. No concerns were identified.

b. *Prioritization of corrective actions.*

Prioritization of the corrective actions from the root cause evaluations were not directly based on risk perspectives or analysis, but on a deterministic approach considering the significance of the identified problem.

The inspector reviewed the prioritization of the corrective actions and verified that actions of a generally higher priority were scheduled for completion ahead of those of a lower priority. No concerns were identified.

c. *Establishment of schedule for implementing and completing the corrective actions*.

The licensee's corrective action program, as described in PMP-7030, "Corrective Action Program," identified the process for assigning significance levels for condition reports. Subsequently, condition reports were evaluated and corrective actions were identified. These corrective actions were assigned a priority level commensurate with their safety significance. These priority levels had corresponding time limits for implementing the corrective actions and the licensee had a process in place to track all corrective actions and priority levels. In addition, the inspector selected a number of corrective actions in each of the root cause evaluations and verified that they had been completed or were being tracked for resolution and closure consistent with PMP-7030. No concerns were identified.

d. *Establishment of quantitative or qualitative measures of success for determining the effectiveness of the corrective actions to prevent recurrence.*

The licensee established an effectiveness review to validate the effectiveness of the overall corrective action plan. This initial effectiveness review consisted of a simple verification that the Scrams With Loss of Normal Heat Removal performance indicator had returned to the Green category.

The inspector questioned the adequacy of the initial effectiveness review plan, since the plan did not include qualitative or quantitative measures of success specifically focused on the effectiveness of the corrective actions implemented to address the root causes. To address this concern, licensee personnel revised the effectiveness review plan by adding measures, such as simulator verifications and plant assessments, to better assess the effectiveness of the corrective actions. The inspector reviewed the licensee's revised effectiveness review plan. No additional concerns were identified.

03 **Management Meetings**

Exit Meeting Summary

On November 21, 2003, the inspector presented the inspection results to Mr. M. Nazar and other members of licensee management. The licensee acknowledged the findings presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Nazar, Senior Vice President
M. Finissi, Plant Manager
J. Zwolinski, Engineering & Regulatory Affairs Director
S. Simpson, Operations Director
J. Giessner, Plant Engineering Director
P. Cowan, System Engineering Manager
M. Scarpello, Regulatory Affairs
J. Kobyra, Learning Organization Director
L. Weber, Performance Assurance Director

Nuclear Regulatory Commission

B. Kemker, Senior Resident Inspector, D.C. Cook
I. Netzel, Resident Inspector, D.C. Cook

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000316/2003014-01	NCV	Failure to Address Excessive RCS Cooldown (Section 2.0.1.b)
05000316/2003014-02	NCV	Inadequate Reactor Trip EOP (Section 2.0.1.b)
05000316/2003014-03	NCV	Concurrent Performance of EOPs (Section 2.0.2.b)

Closed

05000316/2003014-01	NCV	Failure to Address Excessive RCS Cooldown (Section 2.0.1.b)
05000316/2003014-02	NCV	Inadequate Reactor Trip EOP (Section 2.0.1.b)
05000316/2003014-03	NCV	Concurrent Performance of EOPs (Section 2.0.2.b)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

CR 03199051; Loss of Normal Heat Removal Root Cause Report; July 18, 2003
CR 03037028; Unit 2 Reactor Trip from 100 Percent Power; February 05, 2003
CR 03036056; Unit 2 Reactor Trip from Dual Power Supply Failure; February 5, 2003
CR 02203001; Unit 2 Reactor Trip While Flushing Condenser Waterboxes; July 22, 2003
CR 03114044; Large Intrusion of Lake Fish; April 24, 2003
CR 02305075; Senior Resident Questioned D.C. Cook Plant Design; November 1, 2002
CR 02203007; Unit 2 Reactor Tripped Due to Low Condenser Vacuum; July 22, 2003
CR 01236037; Power Supply Failures; August 24, 2001
CR 01280017; Rapid Event Response; October 7, 2001
CR 02019036; During Planned Trip Turbine Driven Feed Pump Started; January 19, 2002
CR 02133001; Unit 2 Automatic Reactor Trip Due to Both Power Supplies Failed; May 12, 2002
CR 02133002; Unit 2 Trip; May 12, 2002
CR 01265064; Manual Reactor Trip Due to Loss of East Main Feed Pump; June 14, 2002
CR 02166009; Excessive Cooldown; June 15, 2002
CR 01280015; Valve Failed to Throttle Following a Flow Retention Signal; October 7, 2001
CR 03325025; Failure to Take Prompt Action; November 21, 2003
CR 03325026; Inadequate Procedure for E-0; November 21, 2003
CR 03325028; Inappropriate Action; November 21, 2003
CR 03325011; Power Supply Root Cause; November 21, 2003
CR 03325029; Reactor Trip Reports; November 21, 2003
Unit 2 Control Room Log; October 6 and 7, 2001
Unit 2 Control Room Log; May 11 and 12, 2002
Unit 2 Control Room Log; July 21 and 22, 2002
Unit 2 Control Room Log; February 4 and 5, 2003
Calculation Number EVAL-MD-02-RCS-019-S; Evaluation of RCS Cooldown; April 8, 2003
Reactor Trip Review; September 30, 2003
Deficiency Request 2002016; Steam Generator Level Shrink; January 20, 2002
OHI-4023; Abnormal Emergency Procedures User Guide; Revision 13
02-OHP-4021-057-006; Operation of Main and Feed Pump Condensers; Revision 13
02-OHP-4021-057-006; Operation of Main and Feed Pump Condensers; Revision 12
02-OHP-4021-057-006; Operation of Main and Feed Pump Condensers; Revision 11
02-OHP-4021-057-006; Operation of Main and Feed Pump Condensers; Revision 10
01-OHP-4021-057-006; Operation of Main and Feed Pump Condensers; Revision 9
02-OHP-4023-E-0; Reactor Trip or Safety Injection; Revision 12
02-OHP-4023-E-0; Reactor Trip or Safety Injection; Revision 15
02-OHP-4023-E-0; Reactor Trip or Safety Injection; Revision 16
02-OHP-4023-E-0; Reactor Trip or Safety Injection; Revision 17
02-OHP-4023-E-0; Reactor Trip or Safety Injection; Revision 18
02-OHP-4023-E-0; Reactor Trip or Safety Injection; Revision 19
02-OHP-4023-E-0; Reactor Trip or Safety Injection; Revision 20
01-OHP-4023-E-0; Reactor Trip or Safety Injection; Revision 19
01-OHP-4023-ES-0.1; Reactor Trip Response; Revision 17
02-OHP-4023-ES-0.1; Reactor Trip Response; Revision 14
02-OHP-4023-ES-0.1; Reactor Trip Response; Revision 15

LIST OF DOCUMENTS REVIEWED

02-OHP-4023-ES-0.1; Reactor Trip Response; Revision 16
02-OHP-4023-ES-0.1; Reactor Trip Response; Revision 17
ANS 3.5-1985; Nuclear Power Plant Simulators for Use in Operator Training, October 1985
Electric Power Research Institute, Power Supply Maintenance and Application Guide; December 2001
CR 030104036; Inadequate Action; January 14, 2003
Management Lessons Learned - Loss of Normal Heat Removal; November 7, 2003
Plant Management Instruction 1060; Equipment Reliability Steering Committee; Revision 0
Plant Management Instruction 7030; Corrective Action Program; Revision 30
Plant Management Procedure 7030 - Corrective Action Program 001; Corrective Action Program Process Flow; Revision 15
D.C. Cook Nuclear Plant Equipment Root Cause Analysis Desk Top Guide; Revision 0
Emergency Response Guidelines; Reactor Trip or Safety Injection; September 30, 1997
Emergency Response Guidelines; Reactor Trip Response; September 30, 1997
Plant Specific Background Document; Reactor Trip Response; May 3, 2002
Plant Specific Background Document; Reactor Trip or Safety Injection; February 12, 2003

LIST OF ACRONYMS

CFR	Code of Federal Regulations
CR	Condition Report
DRP	Division of Reactor Projects
EOP	Emergency Operating Procedure
ERG	Emergency Response Guidelines
GL	Generic Letter
IMC	Inspection Manual Chapter
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OHI	Operating Head Instruction
OHP	Operating Head Procedure
OI	Operations Instruction
PMI	Plant Management Instruction
PMP	Plant Management Procedure
RCS	Reactor Coolant System
SDP	Significance Determination Process