January 27, 2003

Mr. A. C. Bakken III
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107

SUBJECT: D.C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 50-315/02-09(DRP); 50-316/02-09(DRP)

Dear Mr. Bakken:

On December 28, 2002, the NRC completed an inspection at your D. C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on January 3, 2003, with Mr. J. Pollock 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, six 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 Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III; 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 facility.

Since the terrorist attacks on September 11, 2001, the NRC has issued two Orders (dated February 25, 2002, and January 7, 2003) and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance access authorization. The NRC also issued Temporary Instruction 2515/148 on August 28, 2002, that provided guidance to inspectors to audit and inspect licensee implementation of the interim compensatory measures (ICMs) required by the February 25th Order. Phase 1 of TI 2515/148 was completed at all commercial nuclear power plants during calendar year (CY) '02, and the remaining inspections are scheduled for completion in CY '03. Additionally, table-top security drills were conducted at several licensees to evaluate the impact of expanded adversary characteristics and the ICMs on licensee protection and mitigative strategies. Information gained and discrepancies identified during the audits and drills were reviewed and dispositioned by the Office of Nuclear Security and Incident Response. For CY ‘03, the NRC will continue to monitor overall safeguards and security controls, conduct inspections, and resume force-on-force exercises at selected power plants. Should threat conditions change, the NRC may issue additional Orders, advisories, and temporary instructions to ensure adequate safety is being maintained at all commercial power reactors.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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/

Laura Collins, Acting Chief
Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 50-315/02-09(DRP);
50-316/02-09(DRP)

cc w/encl: J. Pollock, 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:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/02-09(DRP); 50-316/02-09(DRP)
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	1 Cook Place Bridgman, MI 49106
Dates:	October 1, 2002 through December 28, 2002
Inspectors:	B. Kemker, Senior Resident Inspector I. Netzel, Resident Inspector R. Azua, Project Engineer, Region IV M. Bielby, Operations Engineer J. Ellegood, Resident Inspector, Perry R. Gibbs, Senior Reactor Analyst, NRR R. Jickling, Emergency Preparedness Inspector R. Krsek, Resident Inspector, Palisades J. Maynen, Physical Security Inspector W. Poertner, Operations Engineer, NRR R. Powell, Senior Resident Inspector, Perry P. Prescott, Operations Engineer, NRR R. Schmidt, Radiation Specialist H. Walker, Senior Reactor Engineer R. Winter, Reactor Engineer S. Wong, Senior Reactor Analyst, NRR D. Wrona, Operations Engineer, NRR
Approved by:	L. Collins, Acting Chief Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000315-02-09(DRP), IR 05000316-02-09(DRP), on 10/01/2002-12/28/2002, Indiana Michigan Power Company, D. C. Cook Nuclear Power Plant, Units 1 and 2. Maintenance Effectiveness, Identification and Resolution of Problems, Event Follow-up.

This report covers a 13-week period of inspection by resident, regional, and headquarters based inspectors. The inspectors identified six Green findings. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process," (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. **Inspector Identified Findings**

Cornerstone: Initiating Events

- Green. The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to assure that prompt corrective actions were taken to address age-related failures of reactor control instrumentation power supplies to prevent repetition of power supply failures, a significant condition adverse to quality. This issue was self-revealed on May 12, 2002, when an automatic reactor trip of Unit 2 occurred due to the failure of redundant 24-volt direct current power supplies in reactor control instrumentation cabinet 2-PS-CGC-16. The failure of both power supplies caused the number 21 steam generator feedwater regulating valve to close. Unit 2 subsequently tripped on low steam generator water level coincident with low feedwater flow.

  The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue, if left uncorrected, would become a more significant safety concern with the likelihood of continued failures of reactor control instrumentation power supplies and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective. Specifically, the failure of redundant power supplies in reactor control instrumentation cabinets would upset plant stability (cause a reactor trip) and challenge the function of critical safety equipment. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding contributes to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, the inspectors determined that this finding required a Phase 2 SDP analysis. After a review of additional information, the inspectors determined that a Phase 3 analysis was required. The Phase 3 SDP analysis, performed with the assistance of the NRC probabilistic risk analysis staff, determined that the resultant Core Damage Frequency and Large Early Release Frequency associated with this finding were less than 1E-6 per year and 1E-7 per year, respectively. Based on these results, this issue was determined to be of very low safety significance. (Section 1R12)

- Green. The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to take corrective action to preclude the repetition of reactor control instrumentation 24-volt direct current power supply failures. Specifically, the licensee failed to perform weekly verification of control group power supplies to ensure that the "power available" status lights were lit. This corrective action was identified by the licensee in response to the Unit 2 reactor trip on May 12, 2002, which was caused by the failure of redundant power supplies in reactor control instrumentation cabinet 2-PS-CGC-16. The licensee subsequently performed this check on November 22, 2002, and discovered a failed 24-volt direct current power supply in Unit 1 cabinet 1-PS-CGC-16.

  The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue could be reasonably viewed as a precursor to a significant event (i.e., potentially result in a reactor trip similar to the Unit 2 trip on May 12, 2002), and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective. Specifically, the failure of redundant power supplies in reactor control instrumentation cabinets would upset plant stability (cause a reactor trip) and challenge the function of critical safety equipment. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding contributes to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, the inspectors determined that this finding required a Phase 2 SDP analysis. After a review of additional information, the inspectors determined that a Phase 3 analysis was required. The Phase 3 SDP analysis, performed with the assistance of the NRC probabilistic risk analysis staff, determined that the resultant Core Damage Frequency and Large Early Release Frequency associated with this finding were less than 1E-6 per year and 1E-7 per year, respectively. Based on these results, this issue was determined to be of very low safety significance. (Section 1R12)

- Green. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed. The licensee failed to provide an appropriate procedure for testing the Unit 1 pressurizer power operated relief valves (PORVs), causing an uncontrolled release of reactor coolant system inventory to the pressurizer relief tank. This issue was self-revealed on June 5, 2002, when pressurizer PORV 1-NRV-153 inadvertently opened while testing actuation logic circuitry for pressurizer PORV 1-NRV-151. The surveillance test procedure failed to provide adequate control of 1-NRV-151 and 1-NRV-153, which have a common automatic opening signal. The release rate exceeded the 25 gallons-per-minute limit established for declaring an Unusual Event in accordance with the licensee's Emergency Plan.

The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue could be reasonably viewed as a precursor to a significant event and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective. Specifically, the uncontrolled release of reactor coolant system inventory upset plant stability and challenged the inventory control safety function. Because Unit 1 was in a shutdown mode during this period, the inspectors performed a Phase 1 SDP review of this issue using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process." Based on the plant conditions at the time, the inspectors concluded that the most appropriate Appendix G checklist to use for this issue was the checklist for "Pressurized Water Reactor Hot Shutdown Operation - Time to core boiling less than 2 hours." Because, operator intervention was required to manually close the affected PORV block valve, the inspectors concluded that the unit was in a configuration where a single active failure or personnel error could have resulted in a rapid loss of reactor coolant system inventory as described in Section II.B.(2) of the checklist. Consequently, the inspectors concluded that this issue increased the likelihood of a loss of reactor coolant system inventory and therefore required a Phase 2 SDP analysis. The inspectors discussed the safety significance of this issue with the Regional Senior Reactor Analyst (SRA). The SRA reviewed the finding and determined that the drain path could be easily isolated, accurate reactor coolant system level indication was available, all steam generators were available for cooling, and all trains of standby injection were available and not impacted by the finding. Based on these factors the finding was determined to be of very low safety significance. (Section 4OA3.1)

- Green. The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." The licensee failed to provide appropriate instructions for conducting a planned shutdown of Unit 2 on January 19, 2002, which resulted in unnecessarily challenging the automatic start function of Unit 2 turbine driven auxiliary feedwater pump (TDAFWP). This issue was self-revealed when the TDAFWP unexpectedly started due to low steam generator levels following the manual reactor trip.

  The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective and was therefore more than a minor concern. Specifically, the function of critical safety equipment was challenged and plant stability was upset during the performance of a normal plant shutdown by the automatic start of Unit 2 TDAFWP. The inspectors performed a Phase 1 SDP review of this issue using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding did not cause or contribute to the likelihood of an initiating event, the inspectors concluded that this issue was of very low safety significance. (Section 4OA3.3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to assure that corrective actions were taken to preclude repetition of emergency diesel generator (EDG) starting air system relay failures, a significant condition adverse to quality. This issue was self-revealed when the failure of a starting air system relay for the Unit 2 AB EDG occurred on October 16, 2002, causing the engine to roll without a valid start signal. The inspectors subsequently identified that appropriate corrective actions to prevent repetition had not been taken following two previous age-related EDG air start relay failures in January 1999 and September 2000.

  The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue, if left uncorrected, would become a more significant safety concern and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the mitigating systems cornerstone and adversely affected the cornerstone objective. Specifically, the repetitive EDG air start relay failures affected the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was a licensee performance deficiency of very low safety significance because the finding: (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time; (4) did not represent an actual loss of safety function of one or more Non-Technical Specification trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. (Section 4OA2.1)

Cornerstone: Barrier Integrity

- Green. The inspectors identified a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to identify and take appropriate corrective actions to preclude the failure of four Unit 1 reactor coolant system pressure boundary charging line check valves (Velan Model B10-3114B-13M), which were at risk of common cause failure due to industry identified design and manufacturing defects, a significant condition adverse to quality. This issue was self-revealed when the check valves were all found to be stuck in either the full or partially open position during radiographic nonintrusive testing in May 2002.

  The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this finding was associated with the barrier integrity cornerstone and adversely affected the cornerstone objective, and as such it was more than a minor concern. Specifically, the charging line check valves perform a safety-related function of limiting the release of reactor coolant inventory should a charging line failure occur. The failure of the valves in the open position would prohibit the performance of this function and therefore affects the objective of the barrier integrity cornerstone. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding involved the integrity of the reactor coolant system barrier, the inspectors determined that this finding required a Phase 2 SDP analysis. After consulting with the Regional Senior Reactor Analyst, the inspectors determined that this issue was of very low safety significance because no actual loss of safety function occurred. The inspectors concluded that no actual loss of safety function occurred based on the reported minimal force required to shut the valves (indicating they would have shut given the differential pressure applied during accident conditions) and the redundancy provided by a third check valve (1-CS-321) in the charging line. In accordance with IMC 0609, Appendix A, Attachment 1, Step 2.6, the SDP results were not evaluated for potential risk contribution due to Large Early Release Frequency because the accident sequence result was less than 1E-7 per year. (Section 4OA2.2)

B. **Licensee Identified Violations**

None

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near full power during this inspection period with the following exceptions:

- On October 5, 2002, the licensee reduced power to approximately 90 percent of rated thermal power to repair a steam leak on a feedwater heater. Following the repair, the licensee returned the unit to full power on October 7, 2002.
- On November 10, 2002, the licensee initiated a power reduction to approximately 30 percent of rated thermal power to enter the Containment Building and add oil to a reactor coolant pump motor. Following the maintenance activity, the licensee returned the unit to full power on November 12, 2002.
- On December 21, 2002, the licensee initiated a power reduction to approximately 53 percent of rated thermal power to enter the Containment Building and add oil to a reactor coolant pump motor. Following the maintenance activity, the licensee returned the unit to full power on December 22, 2002.
- On December 24, 2002, the licensee reduced power to approximately 55 percent of rated thermal power to remove a main feedwater pump from service and repair a failed weld on a small diameter instrument line at the discharge of the pump. Following the repair, the licensee returned the unit to full power on December 25, 2002.

Unit 2 operated at or near full power during this inspection period.

- On November 4, 2002, the licensee received approval of a Notice of Enforcement Discretion to extend the 72-hour allowed action time of Technical Specification 3.8.1.1.b to preclude shutting down the unit until the CD emergency diesel generator could be restored to an operable status.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's procedures and preparations for cold weather conditions. The inspectors reviewed winterization procedures, severe weather procedures, emergency plan implementing procedures related to severe weather, and performed general area walkdowns. Specifically:

- During general pre-winterization walkdowns conducted the week of October 14, 2002, the inspectors toured selected buildings and areas to verify the licensee had identified all discrepant conditions such as damaged doors, windows, or vent louvers. Additionally, the inspectors observed housekeeping conditions and verified that materials capable of becoming airborne missile hazards during high wind conditions, or impacting snow removal, were appropriately located and restrained.

- During post-winterization walkdowns conducted the week of November 18, 2002, the inspectors verified that all items on the licensee's pre-winterization checklist were completed with appropriate corrective actions taken for identified discrepant conditions. Additionally, the inspectors verified that outside water storage tanks (e.g., refueling water storage tanks, primary water storage tanks, and condensate storage tanks) and associated valve houses and piping had no missing or damaged insulation and were serviced by operable heat trace circuits.

- During post-winterization walkdowns conducted the week of December 9, 2002, the inspectors toured plant areas to monitor the physical condition of cold weather protection features following a period of extended freezing temperatures. The inspectors observed insulation, heat trace circuits, space heater operation, and weatherized enclosures to ensure operability of affected systems.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

Initiating Events Cornerstone

- Unit 2 AB Emergency Diesel Generator (EDG)

Mitigating Systems Cornerstone

- Unit 2 Turbine Driven and West Auxiliary Feedwater (AFW) System Trains
- Unit 1 West Essential Service Water (ESW) System Train

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones. The inspectors reviewed operating procedures, Technical Specification (TS) requirements, Administrative Technical Requirements, system diagrams, 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 reviewed the issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for equipment alignment related issues documented in selected condition reports (CRs).

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours

a. Inspection Scope

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

Initiating Events Cornerstone

- Lake Screen House (Zone 142)

Mitigating Systems Cornerstone

- Fire Pump Building
- Auxiliary Building North 609 Foot Elevation (Zone 44N)
- Auxiliary Building South 609 Foot Elevation (Zone 44S)
- Unit 1 East ESW Pump Room (Zone 29A)
- Unit 1 West ESW Pump Room (Zone 29B)
- Unit 1 Turbine Building Southeast (Zone 91)
- Unit 2 Turbine Building Northeast (Zone 96)
- Unit 1 Turbine Building Southwest (Zone 92)
- Unit 2 Turbine Building Northwest (Zone 99)
- Unit 1 CD EDG Room (Zone 15)
- Unit 2 CD EDG Room (Zone 18)

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

b. Findings

No findings of significance were identified.

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 annual requalification evaluations in the D. C. Cook Plant operations training simulator on October 29, 2002. The inspectors focused on alarm response, command and control of crew activities, communication practices, procedural adherence, and implementation of emergency plan requirements.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

Initiating Events Cornerstone

- Control Group Power Supply Failures

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 management of handling of SSC performance or condition problems in terms of:

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

b. Findings

b.1 Failure to Address Age-related Failures of Reactor Control Instrumentation Power Supplies to Prevent Repetition of Power Supply Failures

The inspectors identified a finding of very low safety significance (Green) associated with a self-revealed event. The licensee failed to assure that prompt corrective actions were taken to address age-related failures of reactor control instrumentation power supplies to prevent repetition of power supply failures, a significant condition adverse to quality. The inspectors determined that this issue constituted a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," and therefore dispositioned this finding as a Non-Cited Violation.

Discussion

On May 12, 2002, an automatic reactor trip of Unit 2 occurred due to the failure of redundant 24-volt direct current (DC) power supplies in reactor control instrumentation cabinet 2-PS-CGC-16. The failure of both power supplies caused the number 21 steam generator feedwater regulating valve to close. Unit 2 subsequently tripped on low steam generator water level coincident with low feedwater flow.

Each reactor control instrumentation cabinet contains two separate power supplies (originally Lambda Model LRS-57-24 or LMS-9120). The two power supplies are interconnected through auctioneering diodes, such that the cabinet remains energized in the event of the failure of one of the power supplies. The cabinets provide indication and control functions for various plant systems including: steam generator feedwater control, automatic steam generator power operated relief valve (PORV) control, automatic steam dump control, reactor coolant system volume control tank automatic make-up, automatic switch-over of the charging pump suction to the refueling water storage tank on low-low volume control tank level, automatic pressurizer pressure control using spray valves and heaters, automatic pressurizer level control, and automatic pressurizer PORV controls. Detection of a single power supply failure was inhibited because there was no annunciation on the loss of a single power supply. There were "power available" status lights connected with each of the power supplies located inside the normally closed cabinet doors; however, the licensee did not routinely check the status lights prior to the Unit 2 reactor trip.

The reactor control instrumentation cabinet power supplies in question were originally installed in both units in 1994 as part of a modification to replace obsolete equipment. In 1999 and 2000, several of these power supplies failed and were sent to a vendor for repair. Repair reports were generated by the vendor which identified the existence of internal components that were much older than expected. However, these repair reports were apparently not forwarded to the system engineering department when they were received at D. C. Cook. Following the Unit 2 reactor trip, the two failed power supplies from 2-PS-CGC-16 and several other failed 24-volt DC power supplies were sent to the vendor for detailed analysis. All of the failures were determined to be age-related. In all cases, capacitors with date codes as early as 1989 were found. Hence, these power supplies were already several years old when they were first installed and energized.

The licensee recognized in August 2001 that there had been a significant number of DC power supply failures during the 24-month period prior to August 2001. The licensee collectively documented a total of 20 power supply failures in CR 01236037, including six reactor control instrumentation power supply failures, stating that the failures should be investigated for a common cause. Other power supply failures were in nuclear instrumentation, radiation monitoring instrumentation, reactor protection instrumentation, rod control/rod position indication, steam generator PORV indication, reactor coolant pump vibration monitoring instrumentation, and main generator hydrogen and carbon dioxide (CO_2) purity monitoring. The inspectors noted that the licensee did not complete its evaluation of CR 01236037 until after the Unit 2 reactor trip 9 months later.

It is also noteworthy that Unit 2 was started up following the Cycle 13 refueling outage in February 2002, with one of the two power supplies known to be failed in reactor control instrumentation cabinet 2-PS-CGC-19. This failed power supply was discovered by instrument technicians during a routine cleaning and inspection of the cabinet on February 16, 2002, (12 days prior to completion of the outage). According to the licensee's root cause evaluation, replacement of the power supply was not performed due to perceived time pressure associated with the refueling outage schedule. Considering that a second power supply failure in that cabinet would result in a reactor trip and that the licensee should have been aware of the power supply history based on CR 01236037, the inspectors concluded that this decision was not conservative in that it increased the likelihood an initiating event (i.e., a reactor trip).

The inspectors determined that the licensee's failure to assure that corrective actions were taken to preclude repetitive age-related failures of reactor control instrumentation power supplies is a licensee performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting issue of problem identification and resolution.

<u>Analysis</u>

The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue, if left uncorrected, would become a more significant safety concern with the likelihood of continued failures of reactor control instrumentation power supplies and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective. Specifically, the failure of redundant power supplies in reactor control instrumentation cabinets would upset plant stability (cause a reactor trip) and challenge the function of critical safety equipment. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter (IMC) 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding contributes to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available, the inspectors determined that this finding required a Phase 2 SDP analysis.

Using the current risk-informed inspection notebook for D. C. Cook (Revision 0) for the Phase 2 SDP analysis, the inspectors determined that this finding was potentially greater than very low safety significance. Specifically, the inspectors determined that this issue caused the likelihood for transients involving a loss of the primary conversion system (PCS) to be increased by an order of magnitude using Usage Rule 1.2 of IMC 0609, Appendix A, Attachment 2. The initiating event likelihood was evaluated for a greater than 30-day period because the condition had existed since the last refueling outage which occurred in February 2002. However, after a review of additional information the inspectors determined that a Phase 3 analysis was required. Specifically, for transient sequences, including those involving a loss of the PCS, the inspectors noted that additional credit that was not assumed in the risk-informed notebook could be given to the AFW function. This additional credit involves the operators' ability to cross-tie the opposite unit's motor-driven AFW pumps to the affected unit's AFW system. This cross-tie evolution is probabilistically limited to the operators' ability to perform the evolution and is given a failure probability of 0.1. Applying this additional credit (i.e., one point), the inspectors determined that this finding was of very low safety significance from a Core Damage Frequency (CDF) perspective.

The inspectors also evaluated the effect of this finding on the Large Early Release Frequency (LERF) while factoring in the additional AFW credit discussed above. Using IMC 0609, Appendix H, "Containment Integrity SDP," the inspectors determined that this finding was also potentially greater than very low safety significance. Specifically, for ice condenser plants involving transient accident sequences, the LERF result is a direct correlation to the CDF result. However, the inspectors determined through discussions with NRC risk analysts that due to refinements of LERF estimations, the impact from transient sequences were no longer being considered as a direct correlation to the LERF result. In fact, this refinement indicates that the LERF contribution from most transient sequences for ice condenser plants is not risk significant. The refinement indicates that only station blackout accident sequences would result in this one-to-one correlation between the CDF result and the LERF estimation. When considering this refinement for LERF estimations, the inspectors determined that this finding was also of very low safety significance from a LERF perspective.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the above, the licensee failed to promptly take corrective action to address age-related failures of reactor control instrumentation power supplies to prevent repetition of power supply failures, a significant condition adverse to quality. Specifically, for a 24-month period prior to August 2001, the licensee documented six reactor control instrumentation power supply failures. All of these failures were subsequently determined to be age-related. Consequently, four additional reactor control instrumentation power supply failures have occurred for the same cause since August 2001: (1) two redundant power supplies failed in reactor control instrumentation cabinet 2-PS-CGC-16, which resulted in a reactor trip and challenged the function of critical safety equipment; (2) one power supply failed in reactor control instrumentation cabinet 2-PS-CGC-19; and (3) one power supply failed in reactor control instrumentation cabinet 1-PS-CGC-16. Because of the

very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-316-02-09-01(DRP)). The licensee entered this violation into its corrective action program as CR 02133001 and CR 02133002.

b.2 Failure to Implement a Corrective Action to Prevent Recurrence Associated with Reactor Control Instrumentation Power Supply Failures

The inspectors identified a finding of very low safety significance (Green). The licensee failed to take corrective action in response to a Unit 2 reactor trip on May 12, 2002, to preclude the repetition of reactor control instrumentation 24-volt DC power supply failures, a significant condition adverse to quality. Specifically, the licensee did not perform prescribed weekly verifications of reactor control instrumentation power supplies to identify failed power supplies in lieu of no annunciation on the loss of a single power supply. The inspectors determined that this issue constituted a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," and therefore dispositioned this finding as a Non-Cited Violation.

Discussion

Following the Unit 2 reactor trip on May 12, 2002, the licensee initiated the following corrective actions to prevent recurrence:

- The failed 24-volt DC power supplies in reactor control instrumentation cabinet 2-PS-CGC-16 and 2-PS-CGC-19 were replaced.

- All 24-volt DC control group power supplies in Unit 2 were inspected and components were verified to be no older than 2 years old. One power supply was replaced as a result of the inspection.

- All 24-volt DC control group power supplies in Unit 1 were replaced prior to the Unit 1 reactor startup on June 8, 2002.

- The licensee identified the need to establish a recurring task to perform a weekly verification of control group power supplies to ensure that the "power available" status lights were lit. This could afford the licensee an opportunity to take compensatory measures or replace a failed power supply prior to the failure of the redundant power supply.

During the inspectors' review of the licensee's corrective actions for the Unit 2 reactor trip and in response to the inspectors' questions, the licensee discovered that weekly verifications of the control group power supply "power available" status lights were not being performed. Verification of the power supply status lights was stipulated as a restart action by the licensee's Plant Operations Review Committee following the reactor trip and was specified as a corrective action in the Licensee Event Report (LER) that reported the event. The licensee subsequently performed this check on November 22, 2002, and discovered a failed 24-volt DC power supply in Unit 1 cabinet 1-PS-CGC-16.

The inspectors concluded that the licensee's failure to implement this corrective action to prevent recurrence for a significant condition adverse to quality was a performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting issue of problem identification and resolution.

Analysis

The inspectors assessed this finding using the SDP. The inspectors concluded that this issue could be reasonably viewed as a precursor to a significant event (i.e., potentially result in a reactor trip similar to the Unit 2 trip on May 12, 2002), and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective. Specifically, the failure of redundant power supplies in reactor control instrumentation cabinets could upset plant stability (i.e., cause a reactor trip) and challenge the function of critical safety equipment. Consistent with the SDP evaluation performed for the finding described in Section 1R12.b.1, this finding was determined to be of very low safety significance.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the above, the licensee failed to take corrective action to preclude the repetition of reactor control instrumentation 24-volt DC power supply failures. Specifically, the licensee failed to perform a weekly verification of control group power supplies to ensure that the "power available" status lights were lit. This corrective action was identified by the licensee in response to the Unit 2 reactor trip on May 12, 2002, which was caused by the failure of redundant power supplies in reactor control instrumentation cabinet 2-PS-CGC-16. The licensee subsequently performed this check on November 22, 2002, and discovered a failed 24-volt DC power supply in Unit 1 cabinet 1-PS-CGC-16. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-316-02-09-02(DRP)). The licensee entered this violation into its corrective action program as CR 02325058.

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 maintenance activities affecting the following equipment:

Initiating Events Cornerstone

- Unit 1 AB EDG
- Unit 1 AB EDG ESW Supply Valves
- Unit 2 CD EDG

Mitigating Systems Cornerstone

- Unit 1 East ESW Pump
- Unit 1 West ESW Pump
- Unit 2 East ESW Pump
- Unit 2 East AFW System Train

These activities were selected based on their potential risk significance relative to the reactor safety cornerstones. The maintenance associated with the Unit 2 CD EDG was emergent work to replace the engine's governor that was identified as failed during a scheduled surveillance test. 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.

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

b. Findings

No findings of significance were identified.

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

.1 Unit 1 Power Reduction to Support Oil Addition to a Reactor Coolant Pump Motor

a. Inspection Scope

On November 10, 2002, the licensee initiated a power reduction on Unit 1 to approximately 30 percent of rated thermal power to enter the Containment Building and add oil to the number 14 reactor coolant pump motor. Following the maintenance activity, the licensee returned the unit to full power on November 12, 2002. The inspectors observed portions of the power reduction and assessed operator performance.

b. Findings

No findings of significance were identified.

.2 Unit 1 Control Group Power Supply Replacement

a. Inspection Scope

On December 3, 2002, the licensee replaced one of two redundant 24-volt DC power supplies in reactor control instrumentation cabinet 1-PS-CGC-16. The licensee identified the failed power supply on November 22, 2002 and installed a temporary back-up power supply until replacement of the failed power supply could be performed. This was the first time that replacement of a control group power supply was performed with the unit on line. The inspectors reviewed the licensee's preparations for this evolution and assessed operator performance when the temporary back-up power supply unexpectedly failed, causing a complete loss of power to the instrumentation cabinet.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

Mitigating Systems Cornerstone

- CR 02136008 — Wire Found Disconnected in 1-RPS-A
- CR 02131018 — Review Operability and Reportability Issues for Two Items Dealing with Feedwater Pressure Indication and the Plant Process Computer Calorimetric Program
- CR 02290012 — Steam Generator PORV Actuator Capability Calculation Revealed Negative Calculated Margin for Full Stroke Capability
- CR 02339016 — Ultrasonic Examination on Unit 1 ESW to West Motor Driven AFW Pump Piping Found Some Silt/Sand in the Piping

Barrier Integrity Cornerstone

- CR 02135049 — 1-CCR-462 Leaking Excessively During Local Leak Rate Testing

- CR 02300002 Unit 2 Control Room Access Door 2-DR-AUX411B Latch Has Broken and Door Will Not Shut

In addition, the inspectors reviewed the issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for issues potentially affecting the operability of safety-related SSCs that were documented in selected CRs.

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 operator work-arounds (OWAs) listed below to identify any potential affect on the functionality of mitigating systems or on the operators' response to initiating events:

- OWA 01-02 Feedwater Preheat Valves Cause Cooldown During a Reactor Trip
- OWA 02-03 Need to Open Condenser Vacuum Breakers in Unit 1 to Control Main Turbine Vibration Post Trip

The inspectors selected OWA 01-02 to review the potential affect that leakby past the feedwater preheat control valves has on contributing to excessive plant cooldowns following reactor trips from low power. The inspectors selected OWA 02-03 to review the potential for loss of the secondary heat sink with operation of the main turbine vacuum breakers. 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 its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for issues potentially affecting the functionality of mitigating systems or on the operators' response to initiating events that were documented in selected CRs.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

Barrier Integrity Cornerstone

- Unit 1 Leak Test of Post Accident Containment Hydrogen Monitoring System

Mitigating Systems Cornerstone

- Unit 2 CD EDG Governor Replacement
- Unit 2 East AFW Pump Maintenance
- Unit 2 West ESW Pump Maintenance

The inspectors selected these post maintenance testing activities because the systems were identified as risk significant in the licensee's risk analysis. 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. During this inspection, the inspectors interviewed operations, maintenance, and engineering department personnel and reviewed the completed post maintenance testing documentation.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

For the surveillance test procedures listed below, the inspectors observed selected portions of the surveillance test and/or reviewed the test results to determine whether risk significant systems and equipment were capable of performing their intended safety functions and to verify that testing was conducted in accordance with applicable procedural and TS requirements:

Barrier Integrity Cornerstone

- 01-OHP-4030-STP-011, "Containment Isolation and Inservice Inspection Valve Operability Test"
- 02-IHP-4030-234-001, "Unit 2 Distributed Ignition System Surveillance and Baseline Testing"
- 12-IHP-4030-046-227, "Unit 1 and 2 Personnel Airlock Door Seal Leak Rate Surveillance"

Mitigating Systems Cornerstone

- 01-EHP-4030-ATR-225-020, "Unit 1 Auxiliary Cable Vault CO_2 Fire Suppression Test"
- 02-IHP-4030-SMP-219, "Steam Generator 1 & 2 Steam/Feed Flow Mismatch and Steam Pressure Protection Set I Functional Test and Calibration"
- 02-IHP-4030-SMP-222, "Steam Generator 2 & 4 Steam/Feed Flow Mismatch and Steam Pressure Protection Set II Functional Test and Calibration"
- 02-IHP-4030-SMP-227, "Steam Pressure Protection Set III Functional Test and Calibration"
- 02-IHP-4030-SMP-228, "Steam Pressure Protection Set IV Functional Test and Calibration"
- 12-EHP-5030-CAR-001, "Characterization Testing Program"

The inspectors reviewed the test methodology and test results in order to verify that equipment performance was consistent with safety analysis and design basis assumptions.

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

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the temporary modifications listed below to verify that the installations were consistent with design modification documents and that the modifications did not adversely impact system operability or availability:

Barrier Integrity Cornerstone

- 12-EHP-5040-EMP-006, "Disable Bridge East Travel Limit Switch on East Auxiliary Building Crane 12-QM-3E"

Mitigating Systems Cornerstone

- 1-TM-02-85-R0, "Install Backup Power Supply for Control Group 1"
- 12-TM-00-61-R2, "Winterization/De-Winterization Temporary Modification to Support 12-IHP-5040-EMP-004"

The first temporary modification disabled the end of travel limit switch to allow the Auxiliary Building crane to move far enough to allow resin shipping casks to be lowered into the drumming room. The second temporary modification installed a backup power supply in reactor control instrumentation cabinet 1-PS-CGC-16 until a permanent power supply replacement could be performed for one of the two redundant 24-volt DC power supplies. The third temporary modification installed ventilation system covers and other cold weather system protection measures for the upcoming winter season.

The inspectors verified that configuration control of the modifications were correct by reviewing design modification documents and confirmed that appropriate post-installation testing was accomplished. The inspectors interviewed engineering and operations department personnel and reviewed the design modification documents against the applicable portions of the Updated Final Safety Analysis Report (UFSAR).

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors discussed with Emergency Preparedness (EP) staff the design, equipment, and periodic testing of the public ANS for the D. C. Cook reactor facility emergency planning zone to verify that the system was properly tested and maintained. The inspectors also reviewed procedures and records for a 24-month period ending September 2002 related to ANS testing, annual preventive maintenance, and non-scheduled maintenance. The inspectors reviewed the licensee’s documentation for determining whether each model of siren installed in the emergency planning zone would perform as expected if fully activated. Records used to document and trend component failures for each model of installed siren were also reviewed to ensure that corrective actions were taken for test failures or system anomalies.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the licensee's ERO augmentation testing to verify that the licensee maintained and tested its ability to staff the ERO during an emergency in a timely manner. Specifically, the inspectors reviewed quarterly, off-hours staff augmentation test procedures, dated December 14, 2001, August 23, 2001, March 14, 2002, April 16, 2002, and July 17, 2002 drill records, primary and backup provisions for off-hours notification of the D. C. Cook reactor facility emergency responders, and the current ERO rosters for D. C. Cook. The inspectors reviewed and discussed the facility EP staff's provisions for maintaining ERO call out lists.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the Performance Assurance staff's 2001 - 2002 audits to ensure that these audits complied with the requirements of 10 CFR 50.54(t) and that the licensee adequately identified and corrected deficiencies. The inspectors also reviewed the EP staff's 2001 and 2002 self-assessments, 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, CRs, and action requests 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 the conduct of the licensee's annual announced emergency training exercise that was conducted in the licensee's control room simulator and emergency response facilities on October 23, 2002. The inspection effort was focused on evaluation of the licensee's classifications, notifications, and protective action recommendations for the simulated event. The inspectors also evaluated the licensee's conduct of the training evolution, including the licensee's critique of performance to identify weaknesses and deficiencies.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns, Radiological Boundary Verifications, and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors conducted walkdowns of the radiologically protected area to verify the adequacy of radiological area boundaries and postings. Specifically, the inspectors walked down radiologically significant work area boundaries (radiation, high and locked high radiation areas) in the Auxiliary Building, radwaste area, spent fuel pool/refuel floor, as well as the Unit 2 Containment Building. The inspectors performed confirmatory radiation surveys in selected portions of these areas to verify that these areas were properly posted and controlled in accordance with 10 CFR 20, licensee procedures, and TSs. The inspectors also examined the radiological conditions of work areas within those radiation, high and locked high radiation areas to assess the adequacy of licensee implemented contamination controls. Additionally, the inspectors reviewed radiation work permits (RWPs) for general tours, access to locked high radiation areas for work on spent fuel pool demineralizers, drumming room clean-up activities, an at power entry into Unit 1 for work on a reactor coolant pump; and for another at power entry into Unit 2 for work on a safety injection system accumulator. The RWPs were evaluated for protective clothing requirements, respiratory protection concerns, electronic dosimetry alarm set points, use of remote telemetry dosimetry, radiation protection hold points, and As-Low-As-Reasonably-Achievable considerations, to verify that work instructions and controls had been adequately specified and that electronic dosimeter set points were in conformity with survey results.

b. Findings

No findings of significance were identified.

.2 Job-In-Progress Reviews, Observations of Radiation Worker Performance, and Radiation Protection Technician Proficiency

a. Inspection Scope

The inspectors observed selected portions of the following radiologically significant work activities performed during the inspection and evaluated the licensee's use of radiological controls:

- number 21 safety injection system accumulator level indicator repair, and
- preparations for spent fuel pool demineralizer work.

The inspectors reviewed the pre-job briefing package for the work evolutions, reviewed the radiological requirements for the activities and assessed the licensee's performance with respect to those requirements. The inspectors reviewed survey records, including radiation, contamination, and airborne surveys, to verify that appropriate radiological controls were effectively utilized. The inspectors also reviewed in-process surveys and applicable postings and barricades to verify their accuracy. The inspectors observed radiation protection technician (RPT) and worker performance during the work evolution at the job sites to verify that the technicians and workers were aware of the significance of the radiological conditions in their workplace and RWP controls/limits, and that they were performing adequately given the radiological hazards present and the level of their training.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed licensee CRs written since the last inspection (July 2002) to the date of the current inspection, which focused on access control to radiologically significant areas (i.e., problems concerning activities in high radiation areas, radiation protection technician performance, and radiation worker practices). The inspectors also reviewed the recently revised "High, Locked High, and Very High Radiation Area Access" procedure, which addressed new requirements for specific locking devices for these areas. The inspectors reviewed these documents to assess the licensee's ability to identify repetitive problems, contributing causes, and the extent of conditions, and then implement corrective actions in order to achieve lasting results.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (71121.03)

.1 Walkdowns of Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors reviewed the UFSAR and performed walkdowns of continuous air monitors in the Auxiliary Building, radwaste area, spent fuel pool/refuel floor, Radioactive Material Building, and one area radiation monitor (ARM) in the Unit 2 Containment Building. Additionally, the inspectors examined a representative number of portable radiation survey instruments staged throughout the licensee's facility to verify that those instruments had current calibrations, were operable, and in good physical condition. The inspectors also reviewed the status of repair or troubleshooting activities associated with selected radiation monitoring instruments (i.e., small article monitors and portal monitors that had work request tags) to verify that instrumentation problems were being addressed in an appropriate and timely manner.

b. Findings

No findings of significance were identified.

.2 Calibration, Operability, and Alarm Set Points of Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors examined radiological instrumentation associated with monitoring transient high and/or very high radiation areas to verify that the instrumentation was operating consistent with industry standards and in accordance with station procedures. Specifically, the inspectors assessed the operability of the following instrumentation:

- Unit 2 In-Core Instrumentation Room ARM.

The inspectors reviewed the licensee's alarm set point for this specific ARM to verify that the set point was established consistent with the UFSAR, TSs, and the licensee's Emergency Plan.

The inspectors discussed surveillance practices with licensee personnel and reviewed calender year 2001 - 2002 calibration records and procedures for selected radiation monitors used for assessment of internal exposure. The inspectors also reviewed calibration records and procedures for those instruments utilized for surveys of personnel and equipment prior to egress from the radiologically controlled area. These instruments included:

- AMS-4 Air Monitoring System,
- APTEC PMW-3 Personnel Monitor, and
- Gamma 40/60 Portal Monitor.

Additionally, the alarm set points for these instruments were reviewed to verify that they were established at levels consistent with industry standards and regulatory guidance provided in Health Physics Positions 72 and 250 of NUREG/CR-5569.

The inspectors evaluated the calibration procedures and calibration records for selected portable radiation survey instruments to verify that they had been properly calibrated consistent with the licensee’s procedures. Specifically, the inspectors reviewed the calibrations of the following instruments:

- Emergency Plan designated RO-7 ion chamber, and
- Smart Radiation Monitor general area dose rate meter.

The inspectors also assessed periodic performance tests completed for selected portable radiation survey instruments to verify that they had been tested consistent with the licensee’s procedures. Specifically, the inspectors observed the performance testing of the following instruments:

- extender instruments, and
- bicron RSO survey instruments.

b. Findings

No findings of significance were identified.

.3 Radiation Protection Technician Instrument Use

a. Inspection Scope

The inspectors observed RPTs performing in-field source checks of portable radiation survey instruments to verify that those source checks were adequately completed using appropriate radiation sources and station procedures. The inspectors assessed the RPTs use of radiation/contamination detection instruments as they provided radiological job coverage for risk significant work (e.g., the safety injection system accumulator repair work in the Unit 2 Containment Building), as well as routine work, to ensure that the RPTs were utilizing the appropriate instruments. The inspectors monitored RPTs performing functional tests of selected contamination monitors, portal monitors, and small article monitors (i.e., for surveys of personnel and equipment prior to unconditional release from the radiologically controlled area) to verify that they were source tested and calibrated as required by station procedures and industry standards.

b. Findings

No findings of significance were identified.

.4 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed calendar year 2001-2002 CRs that addressed radiation monitoring instrument deficiencies to determine if any significant radiological incidents involving instrument deficiencies had occurred. The inspectors examined the results of a self-assessment (i.e., the Summary Report for performance Assurance Audit PA-0206, "Radiation Protection") that focused on the licensee's CR database and several individual CRs related to radiation monitoring instrumentation generated during the current assessment period. The inspectors also interviewed plant staff and examined closed CRs to verify that radiological instrumentation related issues were adequately addressed by the licensee. The inspectors evaluated these documents to verify the licensee's ability to identify repetitive problems, contributing causes, extent of conditions, and the implementation of corrective actions to achieve lasting results.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP3 Response to Contingency Events (71130.03)

a. Inspection Scope

The inspectors reviewed the status of security operations and assessed licensee implementation of the protective measures in place as a result of the current, elevated threat environment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures

a. Inspection Scope

Mitigating Systems Cornerstone

The inspectors verified the Safety System Functional Failures performance indicator for both units. The inspectors reviewed each LER from October 2001 to September 2002, determined the number of safety system functional failures that occurred, evaluated each LER against the performance indicator definitions, and verified the number of safety system functional failures reported.

b. Findings

No findings of significance were identified.

.2 Reactor Coolant System Leakage

a. Inspection Scope

Barrier Integrity Cornerstone

The inspectors verified the Reactor Coolant System Leakage performance indicator for both units. The inspectors reviewed operating logs and the results of reactor coolant system water inventory balance calculations performed from October 2001 through September 2002 and verified the licensee's calculation of reactor coolant system leakage for both units.

b. Findings

No findings of significance were identified.

.3 Reactor Coolant System Specific Activity

a. Inspection Scope

Barrier Integrity Cornerstone

The inspectors verified the Reactor Coolant System Specific Activity performance indicator for both units. The inspectors reviewed specific activity results reported from October 2001 through September 2002 and verified the licensee's calculation of reactor coolant system activity for both units. In addition, the inspectors observed staff chemistry technicians collecting reactor coolant system samples to verify that the technicians had complied with applicable procedures during the collection and processing of the samples.

b. Findings

No findings of significance were identified.

.4 ANS, Drill and Exercise Performance (DEP), and ERO Drill Participation

a. Inspection Scope

Emergency Preparedness Cornerstone

The inspectors verified that the licensee had accurately reported the ANS, DEP, and ERO Drill Participation performance indicators for both units. Specifically, the inspectors reviewed the licensee's performance indicator records, data reported to the NRC, and CRs for the period July 2001 through September 2002 to identify any occurrences that were not identified by the licensee. Records of relevant control room simulator training sessions, periodic ANS tests, and excerpts of drill and exercise scenarios and evaluations were also reviewed.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 EDG Starting Air Relay Failures

a. Inspection Scope

A failure of a starting air system relay for the Unit 2 AB EDG occurred on October 16, 2002, causing the engine to roll without a valid start signal. The inspectors reviewed previous failures associated with starting air system relays for the EDGs. In addition, the inspectors reviewed the root cause evaluation for the following CR:

- CR P-99-01279, "Unit 2 AB EDG Rolled with Air by Itself. An Auxiliary Equipment Operator Was Dispatched Who Reported the Engine Rolling with Air. No Indication of a Start Signal Was Detected Locally or in the Control Room. Starting Air Continued to Blow Down Engine Until Air Depleted."

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

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

The inspectors discussed the corrective actions and associated CR evaluations with site personnel including the CR evaluators and system engineers.

b. Findings

The inspectors identified a finding of very low safety significance (Green) associated with this self-revealed event. The licensee failed to assure that corrective actions were taken to preclude repetition of EDG starting air system relay failures, a significant condition adverse to quality. The inspectors determined that this issue constituted a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," and therefore dispositioned this finding as a Non-Cited Violation.

Description

On October 16, 2002, one of the two Unit 2 AB EDG air receivers (AB2) depressurized when starting air valve 2-XRV-222 inadvertently opened and consequently air rolled the engine. The second safety-related air receiver for the EDG remained pressurized and auxiliary equipment operators manually isolated 2-XRV-222, which re-pressurized the AB2 air receiver. A subsequent licensee investigation determined that starting air system relay 2-19-DGAB had failed causing the starting air valve to open.

The starting air system provides the motive force for starting the EDGs. The inspectors noted that each of the four plant EDGs have two of the same model relays installed in the starting air system, which are designated as relays 19 and 19-1. The inspectors determined that while the failure of relay 19 resulted in depressurization of only one of the two air start receivers for an EDG, the failure of relay 19-1 resulted in depressurization of both air start receivers for an EDG.

The inspectors reviewed work order history and the corrective action program database to determine whether or not previous failures had occurred on starting air system relays. The inspectors found two recent occurrences:

- in January 1999, relay 19-1 failed on the Unit 2 AB EDG starting air system, which resulted in depressurization of the starting air system; and
- in September 2000, relay 19 failed on the Unit 1 CD EDG starting air system, which resulted in depressurization of one air start receiver.

The inspectors reviewed the licensee’s assessment of CR P-99-01279 associated with the January 1999 failure and noted the following:

- the root cause evaluation concluded that the most likely failure scenario of the relay was long term overheating of the continuously energized coil in relay 19-1;
- the root cause evaluation concluded that while the relay was designed and rated for 250-volt DC, equalization of the station batteries at the plant had raised DC bus voltages to 280-volt DC. The evaluator determined that from a thermal deterioration perspective, this would increase the heat generated in the relay by approximately 25 percent;
- the root cause evaluation concluded that given the root cause of the failure, the population of similarly aged relays that were continuously energized should be considered suspect and candidates for failure;
- the root cause evaluation concluded that similar relays should be expected to function for an equivalent duration given similar operating conditions; and
- the root cause evaluation recognized that the recommendations in the evaluation were based on only one data point and that prior to a wholesale changeout of relays, the analysis of additional relays of the same type in the same configuration should be considered.

The inspectors determined that the licensee's root cause evaluation addressed the potential extent of condition and the generic implications of the relay failure. However, the only corrective action taken for this significant condition adverse to quality was the replacement of the failed relay 19-1 for the Unit 2 AB EDG. The licensee did not evaluate the condition of other relays of the same type in the same configuration, nor investigate the need for the implementation of a preventive maintenance program or replacement program for this type of relay used in the EDG starting air system.

The inspectors also reviewed CR 00266004, which documented the failure of relay 19 for the Unit 1 CD EDG in September 2000. The inspectors noted that neither an apparent cause evaluation nor a root cause evaluation was performed and that the licensee did not evaluate the potential extent of condition considering the previous failure in January 1999. The licensee concluded that the failure was age related and the only corrective action completed was replacement of the failed relay.

The inspectors determined that the licensee's failure to assure that corrective actions were taken to preclude repetition of starting air system relay failures for the EDGs is a licensee performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting issue of problem identification and resolution.

Analysis

The inspectors assessed this finding using the SDP. The inspectors concluded that this issue, if left uncorrected, would become a more significant safety concern with the continued failures of the starting air system relays and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the mitigating systems cornerstone and adversely affected the cornerstone objective. Specifically, the repetitive EDG air start relay failures affected the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was a licensee performance deficiency of very low safety significance because the finding: (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; (4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. Contrary to the above,

the licensee failed to take corrective action to prevent repetitive failures of EDG starting air system relays 19 and 19-1, a significant condition adverse to quality. Specifically, in January 1999, relay 19-1 failed on the Unit 2 AB EDG starting air system, which resulted in depressurization of the starting air system. The licensee failed to promptly perform corrective actions to preclude a repetition of starting air system relay failures for the EDGs. Consequently, two additional starting air system relay failures have occurred: (1) in September 2000, relay 19 failed on the Unit 1 CD EDG starting air system, which resulted in depressurization of one air start receiver; and (2) in October 2002, relay 19 failed on the Unit 2 AB EDG starting air system, which resulted in depressurization of one air start receiver. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-315/316-02-09-03(DRP)). The licensee entered this violation into its corrective action program as CR 02289033.

.2 Common Cause Failure of Four Unit 1 Charging System Check Valves

a. Inspection Scope

During the Unit 1 Cycle 18 refueling outage in May 2002, the licensee discovered the failure of four reactor coolant system pressure boundary charging line check valves. The inspectors reviewed the circumstances relating to the common cause failure of these valves documented in the root cause evaluation for the following CR:

- CR 02134021, "Check Valves 1-CS-328-L1, 1-CS-328-L4, 1-CS-329-L1, and 1-CS-329-L4 Were Found Open During Radiographic Nonintrusive Testing."

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

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

The inspectors also reviewed the corrective actions and associated CR evaluations with applicable site personnel including the CR evaluators and system engineers.

b. Findings

The inspectors identified a finding of very low safety significance (Green) associated with this self-revealed event. The licensee failed to identify and take appropriate corrective actions to preclude the failure of reactor coolant system pressure boundary charging line check valves (Velan model B10-3114B-13M), which were at risk of common cause failure due to industry identified design and manufacturing defects, a significant condition adverse to quality. The inspectors determined that this issue constituted a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," and therefore dispositioned this finding as a Non-Cited Violation.

Description

On May 12, 2002, the licensee conducted radiographic nonintrusive testing on check valve 1-CS-329-L1 in accordance with procedure 12-EP-4030-001-001, "Check Valve Examination Surveillance." Results of the examination indicated that the valve's disk was stuck in the open position. Subsequent radiography of the remaining three identical charging line check valves, 1-CS-329-L4, 1-CS-328-L1, and 1-CS-328-L4, identified that all three were stuck in either the full or partially open position. The licensee's examination of the valves identified several as-found condition discrepancies including:

- discs binding against the valve body due to oversize discs;
- improper bushing positioning; and
- problems with disc to arm clearances.

The licensee documented the condition and entered it into its corrective action program as CR 02134021. In response to the CR, the licensee conducted a thorough root cause analysis, which the inspectors concluded properly identified substantial industry operating experience documenting similar failures. Further, the licensee's evaluation identified instances of D. C. Cook operating experience involving similar failures which were not considered when evaluating operating experience. Specifically:

- The Velan Valve Corporation published Service Bulletin 104 on October 10, 1990. The licensee received the bulletin on October 12, 1990 and entered it into the its corrective action program as problem report 90-1503. The licensee took corrective action including one-time vendor training on the identified valve deficiencies and the return of on-site spare parts to the vendor for examination. No positive actions, however, such as specific dimensional checks, were taken to determine if the condition existed in the in-service components.

- On February 10, 1992, the licensee received correspondence from Westinghouse Electric Corporation informing them of a 10 CFR Part 21 Report filed by Velan. The correspondence was entered into the licensee's corrective action program as problem report 92-157. The issue was closed with no action taken based on the previous evaluation of problem report 90-1503.

- On January 24, 1996, Operating Experience 7640 was received by the licensee which documented Sequoia's discovery of four Velan Model B10-3114B-13MS 3-inch charging injection check valves in the stuck open position. The OE was entered into the licensee's corrective action program as CR 96-0094. The licensee's evaluation concluded that although similar valves were installed at D. C. Cook, similar problems had not been observed and therefore no corrective action was required. Again, no specific dimensional checks of the installed valves were performed.

The inspectors determined that the licensee's failure to assure that corrective actions were taken to preclude the failure of reactor coolant system pressure boundary charging line check valves due to industry identified design and manufacturing defects was a licensee performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting issue of problem identification and resolution. The inspectors noted that the majority of the instances of missed opportunities to identify and correct potential valve deficiencies occurred in the early to mid 1990s time frame.

Analysis

The inspectors assessed this finding using the SDP. The inspectors concluded that this finding was associated with the barrier integrity cornerstone and adversely affected the cornerstone objective, and as such it was more than a minor concern. Specifically, the charging line check valves perform a safety-related function of limiting the release of reactor coolant inventory should a charging line failure occur. The failure of the valves in the open position would prohibit the performance of this function and therefore affects the objective of the barrier integrity cornerstone. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding involved the integrity of the reactor coolant system pressure boundary, the inspectors determined that this finding required a Phase 2 SDP analysis. After consulting with the Regional SRA, the inspectors determined that this issue was of very low safety significance because no actual loss of safety function occurred. The inspectors concluded that no actual loss of safety function occurred based on the reported minimal force required to shut the valves (indicating that they would have shut given the differential pressure applied during accident conditions) and the redundancy provided by a third check valve (1-CS-321) in the charging line. In accordance with IMC 0609, Appendix A, Attachment 1, Step 2.6, the SDP results were not evaluated for potential risk contribution due to LERF because the accident sequence result was less than 1E-7 per year.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, the licensee failed to take corrective action to preclude the failure of reactor coolant system pressure boundary charging line check valves (Velan model B10-3114B-13M), which were at risk of common cause failure due to industry identified design and manufacturing defects, a condition adverse to quality. Specifically, industry operating experience was published in October 1990, February 1992, and January 1996 and subsequently entered into the licensee's corrective action program. However, the licensee took no positive actions, such as specific dimensional checks, to determine if the condition existed in the in-service components. Consequently, during the Unit 1 Cycle 18 refueling outage in May 2002, the licensee discovered the failure of four reactor coolant system pressure boundary charging line check valves (1-CS-329-L1, 1-CS-329-L4, 1-CS-328-L1, and 1-CS-328-L4), which were the result of conditions identified in the industry operating experience. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with

Section VI.A of the NRC Enforcement Policy (NCV 50-315-02-09-04(DRP)). The licensee entered this violation into its corrective action program as CR 02134021.

4OA3 Event Follow-up (71153)

.1 (Closed) Unresolved Item (URI) 50-315-02-06-01(DRP): "Pressurizer Power Operated Relief Valve (PORV) Inadvertently Opened During Testing Resulting in a Loss of Reactor Coolant System Inventory and an Unusual Event."

a. Inspection Scope

On June 5, 2002, with Unit 1 in Mode 4 (Hot Shutdown), pressurizer PORV 1-NRV-153 inadvertently opened while testing actuation logic circuitry for pressurizer PORV 1-NRV-151. Approximately 100 gallons of reactor coolant was released to the pressurizer relief tank. The release rate exceeded the 25 gallons-per-minute (gpm) limit established for declaring an Unusual Event in accordance with the licensee's Emergency Plan. The inspectors reviewed the circumstances associated with this event, including the root cause determination, operator response during the event, and corrective actions.

b. Findings

A finding of very low safety significance (Green) was self-revealed. The licensee failed to provide an appropriate procedure for testing the Unit 1 pressurizer PORVs, causing an uncontrolled release of reactor coolant system inventory to the pressurizer relief tank. This finding was dispositioned as a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings."

Discussion

As discussed in NRC Inspection Report 50-315/316-02-06(DRP), the inspectors originally documented this finding as an Unresolved Item pending a final safety significance determination. The inspectors referred this finding to the Regional SRA to perform the additional analysis.

Analysis

The inspectors assessed this finding using the SDP. The inspectors concluded that this issue could be reasonably viewed as a precursor to a significant event and was therefore more than a minor concern. The inspectors also concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective. Specifically, the uncontrolled release of reactor coolant system inventory upset plant stability and challenged the inventory control safety function. Because Unit 1 was in a shutdown mode during this period, the inspectors performed a Phase 1 SDP review of this issue using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process." Based on the above information, the inspectors concluded that the most appropriate Appendix G checklist to use for this issue was the checklist for "Pressurized Water Reactor Hot Shutdown Operation - Time to core boiling less than 2 hours."

Because operator intervention was required to manually close the affected PORV block valve, the inspectors concluded that the unit was in a configuration where a single active failure or personnel error could have resulted in a rapid loss of reactor coolant system inventory as described in Section II.B.(2) of the checklist. Consequently, the inspectors concluded that this issue increased the likelihood of a loss of reactor coolant system inventory and therefore required a Phase 2 SDP analysis. The inspectors discussed the safety significance of this issue with the Regional SRA. The SRA reviewed the finding and determined that the drain path could be easily isolated, accurate reactor coolant system level indication was available, all steam generators were available for cooling, and all trains of standby injection were available and not impacted by the finding. Based on these factors the finding is characterized as having very low safety significance.

Enforcement

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, the licensee failed to provide a procedure of a type appropriate to the circumstances for testing the Unit 1 pressurizer PORVs, which is an activity affecting quality. Specifically, the instructions contained in 1-IHP-4030-102-017, "Pressurizer Power Operated Relief Valve (PORV) Actuation Channel Calibration with Valve Operation (for Modes 1, 2, and 3)," Revision 1, failed to provide adequate control of pressurizer PORVs 1-NRV-151 and 1-NRV-153, which have a common automatic opening signal. This issue was self-revealed on June 5, 2002, when pressurizer PORV 1-NRV-153 inadvertently opened while testing actuation logic circuitry for 1-NRV-151, causing an uncontrolled release of reactor coolant system inventory to the pressurizer relief tank. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-316-02-09-05(DRP)). The licensee entered this violation into its corrective action program as CR 02157039.

.2 (Closed) LER 50-315-1999-010-01: "Reactor Coolant System Leak Detection System Sensitivity Not in Accordance with TS Basis," Supplement 1. On May 3, 1999, the licensee documented that the Unit 1 and Unit 2 lower containment sump level and flow monitoring capabilities were not consistent with the recommendations of Regulatory Guide 1.45 as stated in TS Basis 3/4.4.6.1. Specifically, the containment sump level and flow monitoring systems were not able to detect a 1 gpm reactor coolant system leak within 1 hour. The licensee's review of this issue determined that, other than the TS Basis statement that the containment sump level and flow leak detection systems are consistent with the recommendations of Regulatory Guide 1.45, the most restrictive requirement was that the leak detection system be able to detect a 1 gpm leak within 4 hours to meet Generic Letter 84-04 requirements. The licensee has changed the TS Basis statement using the 10 CFR 50.59 process and documented this corrective action in Supplement 1 to LER 50-315-1999-010-00. The inspectors concluded that this change was appropriate. This event did not constitute a violation of NRC requirements. This LER is closed.

.3 Unanticipated Start of the Unit 2 Turbine Driven Auxiliary Feedwater Pump (TDAFWP) During a Normal Plant Shutdown for Refueling Outage

a. Inspection Scope

On January 19, 2002, in preparation for a Unit 2 refueling outage, operators initiated a planned manual reactor trip of Unit 2 from 22 percent power. Shortly thereafter, an automatic start of the TDAFWP occurred due to a valid low level condition in two of the four steam generators. The inspectors reviewed the circumstances associated with this event, including the root cause determination, operator response leading to and during the event, and corrective actions.

b. Findings

(Closed) LER 50-316-2002-004-00: "Unanticipated Start of the Turbine Drive Auxiliary Feedwater Pump."

(Closed) LER 50-316-2002-004-01: "Unanticipated Start of the Turbine Drive Auxiliary Feedwater Pump," Supplement 1.

(Closed) LER 50-316-2002-004-02: "Unanticipated Start of the Turbine Drive Auxiliary Feedwater Pump," Supplement 2.

The inspectors identified a finding of very low safety significance (Green) associated with this self-revealed event. The licensee failed to provide instructions of a type appropriate to the circumstances for a planned shutdown of Unit 2, which resulted in unnecessarily challenging the automatic start function of Unit 2 TDAFWP. The inspectors determined that this issue constituted a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and therefore dispositioned this finding as a Non-Cited Violation.

Discussion

On January 19, 2002, an automatic start of the Unit 2 TDAFWP occurred due to a valid low level condition in two of the four steam generators following a planned manual reactor trip from 22 percent power in preparation for a refueling outage. The licensee had recently revised its plant shutdown procedure (02-OHP-4021-001-003, "Power Reduction," Revision 15) to trip the reactor from less than 22 percent power in order to enter the refueling outage more expeditiously. The licensee had previously performed the manual reactor trip between 1 percent and 4 percent power. Water levels in each of the four steam generators were within the program band prior to the reactor trip and rapidly shrank to below the TDAFWP auto-start actuation setpoint. Following the reactor trip and TDAFWP start, steam generator water levels rapidly recovered with three AFW pumps supplying water from the condensate storage tank and cooled the reactor coolant system below the system "no-load" temperature of 547 degrees Fahrenheit. Pressurizer level also dropped to below 17 percent as a result of the cooldown, which resulted in reactor coolant system letdown isolation. The automatic start of the TDAFWP and reactor coolant system letdown isolation were both unexpected occurrences for a normal plant shutdown and unnecessarily challenged the operators.

The inspectors previously reviewed the licensee’s apparent cause evaluation for this event. The NRC concluded in NRC Inspection Report 50-315/316-02-04(DRP) that the licensee’s ability to consistently identify reasonable causes for conditions adverse to quality in apparent cause evaluations performed for Category 3 CRs was inadequate and documented a finding (FIN 50-315/316-02-04-03). The licensee’s apparent cause evaluation for this event was one of the four examples included in that finding. The inspectors concluded that the licensee’s apparent cause evaluation failed to adequately address the cause for the unexpected TDAFWP start. The inspectors noted that the evaluation was limited to the 10 CFR 50.73 reportability aspects of the unexpected actuation of an engineered safety features component. The licensee subsequently wrote CR 02107016 to evaluate the operational aspects of the unexpected automatic pump start and to identify appropriate corrective actions.

The inspectors reviewed the licensee’s evaluation documented in CR 02107016 and concurred with the licensee’s conclusion that a planned shutdown should not challenge critical safety equipment to automatically start. The licensee submitted Supplement 1 to LER 50-316-2002-004-00 to provide this conclusion and the corrective actions. The licensee subsequently revised the plant shutdown procedure to initiate the reactor trip from less than 17 percent power and has successfully performed the procedure on both units without challenging the automatic start function of an TDAFWP. The licensee submitted Supplement 2 to LER 50-316-2002-004-00 to identify the cause for the engineered safety features component actuation and to clarify statements made in earlier revisions of the LER. The inspectors concluded that 02-OHP-4021-001-003, "Power Reduction," Revision 15, was not appropriate to the circumstances because initiating the reactor trip from 22 percent power unnecessarily challenged the automatic start function of Unit 2 TDAFWP.

Analysis

The inspectors assessed this finding using the SDP. The inspectors concluded that this finding was associated with the initiating events cornerstone and adversely affected the cornerstone objective and was therefore more than a minor concern. Specifically, the function of critical safety equipment was challenged and plant stability was upset during the performance of a normal plant shutdown by the automatic start of Unit 2 TDAFWP. The inspectors performed a Phase 1 SDP review of this issue using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Because this finding did not cause or contribute to the likelihood of an initiating event, the inspectors concluded that this issue was of very low safety significance.

Enforcement

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary

to the above, the licensee failed to provide instructions of a type appropriate to the circumstances for conducting the Unit 2 plant shutdown on January 19, 2002, which is an activity affecting quality. Specifically, the instructions contained in 02-OHP-4021-001-003, "Power Reduction," Revision 15, failed to ensure that the automatic start function of Unit 2 TDAFWP would not be unnecessarily challenged during a normal plant shutdown. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-316-02-09-06(DRP)). The licensee entered this violation into its corrective action program as CR 02019036 and CR 02107016.

.4 (Closed) LER 50-316-2002-005-00: "Unit 2 Trip Due to Instrument Rack 24-Volt DC Power Supply Failure." The event described in this LER was discussed in Section 1R12 of this report. The inspectors concluded that the licensee's failure to assure that prompt corrective actions were taken to address age-related failures of reactor control instrumentation power supplies to prevent repetition of power supply failures was a finding of very low safety significance and a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action". The licensee reported this event as a condition that resulted in an automatic actuation of the reactor protection system in accordance with 10 CFR 50.73(a)(2)(iv)(A). This LER is closed.

.5 (Closed) LER 50-316-1997-004-02: "Analysis Demonstrates Design Basis Impact of Inadequate Refueling Outage Safety Evaluation Was Negligible," Supplement 2. The licensee submitted Supplement 2 to LER 50-316-1997-004-00 to provide additional information concerning the analysis of this event. The inspectors determined that the information provided in Supplement 2 to LER 50-316-1997-004-00 did not raise any new issues or change the conclusion of previous NRC reviews documented in Inspection Reports 50-315/316-97-02(DRP), 50-315/316-98-09(DRS), 50-315/316-99-029(DRS), and 50-315/316-00-01(DRP). This LER is closed.

.6 (Closed) LER 50-315-1997-005-00: "Reactor Coolant Pump Fire Protection Inoperable for Extended Period Without Compensatory Actions Due to Improperly Fabricated Gasket in Spray Header Line." The closure of Supplement 1 to this LER is discussed below in Section 4OA3.7. This LER is closed.

.7 (Closed) LER 50-315-1997-005-01: "Reactor Coolant Pump Fire Protection Inoperable for Extended Period Without Compensatory Actions Due to Improperly Fabricated Gasket in Spray Header Line," Supplement 1. On March 5, 1997, it was discovered that gaskets in the fire protection water spray system for the number 13 reactor coolant pump had not been properly fabricated prior to installation during the 1995 refueling outage. The gaskets for a spectacle flange were fabricated out of a sheet of red rubber, without the center removed to provide a flow path for the water spray. The center area, which should have been cut away during gasket fabrication, was found to be torn. The torn gasket was attributed to the pressure of the system supervisory air. The licensee reported this event in accordance with 10 CFR 50.73(a)(2)(i)(B) as an operation prohibited by the plant's TSs. Fire Protection requirements were subsequently removed from the TSs in March of 1996. However, at the time the gaskets were installed, the Fire Protection TSs were still in effect. During review of this event, the licensee identified that personnel error was the root cause of the event. The personnel involved did not properly incorporate the information contained in the job order activity regarding fabrication of the gaskets into their actions. At the time of discovery, the licensee corrected the problem and entered the issue into its corrective action program as CR 97-0586. The failure to enter previous TS 3.7.9.2, Table 3.7-5B for an inoperable reactor coolant pump sprinkler system whenever the reactor coolant pump was operable was a violation TS 3.7.9.2. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section VI of the NRC Enforcement Policy. This LER is closed.

.8 (Closed) LER 50-315-1998-056-01: "Inadequate Control and Processing of Design Information Results in Unanalyzed Hot Leg Recirculation Switchover," Supplement 1. On December 11, 1998, the licensee identified an unanalyzed condition related to the post-loss-of-coolant accident emergency core cooling system hot leg switchover sub-criticality analysis. The inspectors reviewed the original LER in NRC Inspection Report 50-315/316-99-29(DRS) and concluded that this was a minor issue. The licensee submitted Supplement 1 to LER 50-315-1998-056-00 to provide new information concerning the analysis of the event and corrective actions. The cause of this event was the licensee's failure to adequately control design basis calculations and supporting documentation, which is a violation of 10 CFR 50, Appendix B, Criteria III, "Design Control." The licensee entered this event into its corrective action program as CR 98-7848. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section VI of the NRC Enforcement Policy. This LER is closed.

.9 (Closed) LER 50-315-1998-029-01: "Fuel Handling Area Ventilation System Inoperable Due to Original Design Deficiency," Supplement 1. The inspectors reviewed the original LER in NRC Inspection Report 50-315/316-99-29(DRS) and concluded that this was a minor issue. The licensee submitted Supplement 1 to LER 50-315-1998-029-00 to provide additional information concerning the analysis of the event, the cause, and the corrective actions. The inspectors determined that the information provided in Supplement 1 to LER 50-315-1998-029-00 did not raise any new issues or change the conclusion of the initial review. This LER is closed.

.10 (Closed) LER 50-315-1999-003-00: "Control Room Pressurization System Surveillance Test Does Not Test System in Normal Operating Condition." On January 7, 1999, the licensee identified that the TS surveillance test procedure for testing the control room pressurization function (12 EHP 4030 STP.229, "Control Room Emergency Ventilation Test," Revision 3,) did not test the control room pressurization system in the normal operating configuration. Specifically, one of the prerequisites prior to performing the surveillance test was to verify that the pressure boundary door that separated the Unit 1 and Unit 2 control rooms (12DR-AUX415) was closed. However, it was identified that the door was normally maintained open to facilitate access and egress between the two control rooms with no procedural guidance to close the door during an event. It was identified that a failure to recognize the door as part of both units' control room pressure boundary design resulted in the door being maintained open since initial plant start-up. The inspectors concluded that this constitutes a violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings". The bases for TS Surveillance Requirement 4.7.5.1.e.3 was to demonstrate control room operability such that radiation exposure to personnel occupying the control room would be limited to 5 rem or less whole body, or its equivalent. The licensee as part of its corrective actions closed and labeled pressure boundary door 12DR-AUZ415. In addition, the licensee performed a tracer gas test under several conditions, including one in which door 12DR-AUX415 was left open between the Unit 1 and Unit 2 control rooms with one of the unit's control room ventilation systems treated as inoperable. The test showed that the amount of unfiltered in-leakage was not highly dependent on pressurization and the dose consequences of having the door open during a postulated accident would remain within 10 CFR 50, Appendix A, General Design Criteria 19 allowable limits. Therefore, this issue is considered to be of minor safety significance. The licensee entered this event into its corrective action program as CR 99-0275. This finding constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section VI of the NRC Enforcement Policy. This LER is closed.

.11 (Closed) LER 50-315-1999-003-01: "Control Room Pressurization System Surveillance Test Does Not Test System in Normal Operating Condition," Supplement 1. The licensee submitted Supplement 1 to LER 50-315-1999-003-00 to provide additional information concerning the analysis of the event, the cause, and the corrective actions. The inspectors determined that the information provided in Supplement 1 to LER 50-315-1999-003-00 did not raise any new issues or change the conclusion of the initial review which was documented above in Section 4OA3.10. This LER is closed.

.12 (Closed) LER 50-315-2000-004-00: "Circuit Design Could Result in Failure of Emergency Diesel Generators to Load Properly After Loss of Offsite Power." On July 19, 1999, an unanalyzed condition was identified by the Expanded System Readiness Review Teams wherein a sneak electrical circuit existed that could cause improper EDG load sequencing of equipment onto the vital buses following a loss-of-coolant accident concurrent with a loss of offsite power. The licensee reported this event as a condition that was outside the design basis in accordance with 10 CFR 50.73(a)(2)(ii). The licensee implemented a design change in both Unit 1 and Unit 2 to eliminate the possibility of the sneak circuit. This event did not constitute a violation of NRC requirements. This LER is closed.

4OA5 Other

.1 (Open) URI 50-316-02-09-07(DRP): "Review of NOED-02-3-058 Regarding D. C. Cook, Unit 2, Compliance With Technical Specification 3.8.1.1." By letter dated November 6, 2002, the licensee requested that the NRC exercise discretion not to enforce compliance with the actions of TS 3.8.1.1 regarding operability of the Unit 2 CD EDG. The inspectors opened URI 50-316-02-09-06 to track documentation of the root cause for the Notice of Enforcement Discretion (NOED) request, review the NOED approval basis, and verify licensee activities associated with NOED implementation.

.2 Completion of Appendix A to Temporary Instruction 2515/148, Revision 1

The inspectors completed the pre-inspection audit for interim compensatory measures at nuclear power plants, dated September 13, 2002.

.3 (Closed) Inspector Follow-up Item (IFI) 50-315/316-99-29-01: "Review and Approval of Dose Calculation for General Design Criteria 19 Control Room Habitability Issue." The inspectors reviewed calculation RD-01-05, "Adjusted Dose Consequences for Changes to Control Room," Revision 1. The calculation stated that the control room dose consequences for all events would be below the acceptance criteria required by 10 CFR 50.67. No findings of significance were identified. This item is closed.

.4 <u>(Closed) IFI 50-316-00-07-03</u>: "Failure to Perform Post Modification Checks to Verify Adequate Clearance Between the Pressurizer Surge Line Whip Restraints and the Surge Line Under Hot Plant Conditions." The inspectors performed a limited review of calculation SD-990825-001, "HELB [High Energy Line Break]: Structural Evaluation of Surge Line Pipe Whip Restraints," Revision 3. This calculation included determination of necessary clearances for the pressurizer surge line whip restraints. Additional design engineering documents were reviewed to verify that the calculation results were properly incorporated into the plant design. No findings of significance were identified. This item is closed.

.5 <u>(Closed) URI 50-315/316-00-16-04</u>: "Determine Whether the Latent Failure of a Test Relay Should Be Treated Under the Category of a Single Failure." The NRC staff reviewed this issue and determined that the failure of a K-800 relay would not prohibit the proper operation of an engineered safety features actuation circuit in response to a valid actuation signal. No findings of significance were identified. This item is closed.

.6 <u>(Closed) URI 50-315/316-01-15-01</u>: "A Change Was Made to the UFSAR Without a 10 CFR 50.59 Evaluation." The licensee changed the UFSAR and inappropriately used 10 CFR 50.71 (e) rather than 10 CFR 50.59 to evaluate the UFSAR change. The inspectors reviewed CR 01291058 which described this issue. The licensee corrected this problem by performing a 10 CFR 50.59 screening and concluded that the change did not require NRC approval prior to implementation. No findings of significance were identified. This item is closed.

4OA6 <u>Meetings</u>

.1 <u>Interim Exits</u>

The results of the Emergency Preparedness Program Inspection were presented to Mr. J. Molden and other members of licensee management at the conclusion of the inspection on December 6, 2002. 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.

The results of the Radiological Protection Instrumentation and Access Control Inspection were presented to Mr. J. Molden and other members of licensee management at the conclusion of the inspection on December 6, 2002. 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.

.2 <u>Resident Inspectors' Exit</u>

The inspectors presented the inspection results to Mr. J. Pollock and other members of licensee management at the conclusion of the inspection on January 3, 2003. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee

J. Gebbie, Plant Engineering Assistant Director
G. Gibson, Site Protective Services
C. Graffenius, Emergency Planner
S. Greenlee, Nuclear Technical Services Director
G. Harland, Work Control/Maintenance Director
R. Hershberger, Chemistry Supervisor
R. LaBurn, Radiation Protection General Supervisor
E. Larson, Operations Director
B. McIntyre, Regulatory Assurance Manager
R. Meister, Regulatory Affairs Specialist
J. Molden, Acting Plant Manager
D. Moul, Operation Work Control Manager
T. Noonan, Performance Assurance Director
S. Partin, Emergency Planning Manager
J. Pollock, Site Vice President
B. Robinson, Radiation Protection Superintendent
M. Scarpello, Regulatory Compliance Supervisor
S. Simpson, Operations Staff Manager
D. Wood, Radiation / Environmental Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-316-02-09-01	NCV	Failure to assure that prompt corrective actions were taken to address age-related failures of reactor control instrumentation power supplies to prevent repetition of power supply failures (Section 1R12)
50-316-02-09-02	NCV	Failure to implement a corrective action to prevent recurrence associated with reactor control instrumentation power supply failures (Section 1R12)
50-315/316-02-09-03	NCV	Failure to assure that corrective actions were taken to preclude repetition of EDG starting air system relay failures (Section 4OA2.1)
50-315-02-09-04	NCV	Failure to identify and take appropriate corrective actions to preclude the failure of reactor coolant system pressure boundary charging line check valves which were at risk of common cause failure due to industry identified design and manufacturing defects (Section 4OA2.2)
50-315-02-09-05	NCV	Failure to provide an appropriate procedure for testing the Unit 1 pressurizer power operated relief valves causing an uncontrolled release of reactor coolant system inventory to the pressurizer relief tank (Section 4OA3.1)
50-316-02-09-06	NCV	Failure to provide appropriate instructions for a planned shutdown of Unit 2 which resulted in unnecessarily challenging the automatic start function of Unit 2 turbine auxiliary feedwater pump (Section 4OA3.3)
50-316-02-09-07	URI	Review of NOED-02-3-058 regarding D. C. Cook, Unit 2, compliance with Technical Specification 3.8.1.1 (Section 4OA5.1)

Closed

50-316-02-09-01	NCV	Failure to assure that prompt corrective actions were taken to address age-related failures of reactor control instrumentation power supplies to prevent repetition of power supply failures (Section 1R12)
50-316-02-09-02	NCV	Failure to implement a corrective action to prevent recurrence associated with reactor control instrumentation power supply failures (Section 1R12)

50-315/316-02-09-03	NCV	Failure to assure that corrective actions were taken to preclude repetition of EDG starting air system relay failures (Section 4OA2.1)
50-315-02-09-04	NCV	Failure to identify and take appropriate corrective actions to preclude the failure of reactor coolant system pressure boundary charging line check valves which were at risk of common cause failure due to industry identified design and manufacturing defects (Section 4OA2.2)
50-315-02-06-01	URI	Pressurizer power operated relief valve inadvertently opened during testing resulting in a loss of reactor coolant system inventory and an Unusual Event (Section 4OA3.1)
50-315-02-09-05	NCV	Failure to provide an appropriate procedure for testing the Unit 1 pressurizer power operated relief valves causing an uncontrolled release of reactor coolant system inventory to the pressurizer relief tank (Section 4OA3.1)
50-315-1999-010-01	LER	Reactor coolant system leak detection system sensitivity not in accordance with TS [Technical Specification] Basis (Section 4OA3.2)
50-316-02-09-06	NCV	Failure to provide appropriate instructions for a planned shutdown of Unit 2 which resulted in unnecessarily challenging the automatic start function of Unit 2 turbine auxiliary feedwater pump (Section 4OA3.3)
50-316-2002-04-00	LER	Unanticipated start of the turbine drive auxiliary feedwater pump (Section 4OA3.3)
50-316-2002-04-01	LER	Unanticipated start of the turbine drive auxiliary feedwater pump (Section 4OA3.3)
50-316-2002-04-02	LER	Unanticipated start of the turbine drive auxiliary feedwater pump (Section 4OA3.3)
50-316-2002-05-00	LER	Unit 2 trip due to instrument rack 24-volt DC [direct current] power supply failure (Section 4OA3.4)
50-316-1997-004-02	LER	Analysis demonstrates design basis impact of inadequate refueling outage safety evaluation was negligible (Section 4OA3.5)
50-315-1997-005-00	LER	Reactor coolant pump fire protection inoperable for extended period without compensatory actions due to improperly fabricated gasket in spray header line (Section 4OA3.6)
50-315-1997-005-01	LER	Reactor coolant pump fire protection inoperable for extended period without compensatory actions due to improperly fabricated gasket in spray header line (Section 4OA3.7)

50-315-1998-056-01	LER	Inadequate control and processing of design information results in unanalyzed hot leg recirculation switchover (Section 4OA3.8)
50-315-1998-029-01	LER	Fuel handling area ventilation system inoperable due to original design deficiency (Section 4OA3.9)
50-315-1999-003-00	LER	Control room pressurization system surveillance test does not test system in normal operating condition (Section 4OA3.10)
50-315-1999-003-01	LER	Control room pressurization system surveillance test does not test system in normal operating condition (Section 4OA3.11)
50-315-2000-004-00	LER	Circuit design could result in failure of emergency diesel generators to load properly after loss of offsite power (Section 4OA3.12)
50-315/316-99-29-01	IFI	Review and approval of dose calculation for General Design Criteria 19 control room habitability issue (Section 4OA5.3)
50-316-00-07-03	IFI	Failure to Perform post modification checks to verify adequate clearance between the pressurizer surge line whip restraints and the surge line under hot plant conditions (Section 4OA5.4)
50-315/316-00-16-04	URI	Determine whether the latent failure of a test relay should be treated under the category of a single failure (Section 4OA5.5)
50-315/316-01-15-01	URI	A change was made to the UFSAR without a 10 CFR 50.59 evaluation (Section 4OA5.6)

<u>Discussed</u>

50-315/316-02-04-03	FIN	Green finding regarding the failure to consistently identify a reasonable apparent cause for conditions adverse to quality (Section 4OA3.1)

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
AFW	Auxiliary Feedwater
ALARA	As-Low-As-Reasonably-Achievable
ANS	Alert and Notification System
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CO_2	Carbon Dioxide
CR	Condition Report
CY	Calender Year
DC	Direct Current
DEP	Drill and Exercise Performance
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EHP	Electrical Maintenance Head Procedure
EP	Emergency Preparedness
ERO	Emergency Response Organization
ESW	Essential Service Water
EWS	Early Warning System
FIN	Finding
HELB	High Energy Line Break
IFI	Inspector Follow-up Item
IHP	Instrument Maintenance Head Procedure
IMC	Inspection Manual Chapter
LER	Licensee Event Report
LERF	Larger Early Release Frequency
LHRA	Locked High Radiation Area
MHP	Maintenance Head Procedure
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NOED	Notice of Enforcement Discretion
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OA	Other Activities
OHP	Operations Head Procedure
OWA	Operator Workaround
PARS	Publically Available Records
PCS	Power Conversion System
PI	Performance Indicator
PMI	Plant Manager’s Instruction
PMP	Plant Manager’s Procedure
PORV	Power Operated Relief Valve
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RPT	Radiation Protection Technician
RWP	Radiation Work Permit
SDP	Significance Determination Process

SPP	Special Plant Procedure
SRA	Senior Reactor Analyst
SSC	Structures, Systems, and Components
STP	Surveillance Test Procedure
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. 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, unless specifically stated in the inspection report.

1R01 Adverse Weather Protection

PMP 5055-001-001	Winterization/Summerization	Revision 0
12-OHP 4022.001.010	Severe Weather	Revision 1
12-IHP 4022.001.009	Plant Winterization and De-Winterization	Revision 0
CR P-96-00229	Coils Were Found Frozen and Ruptured on Various Air Handling Units	February 14, 1996
CR P-98-06318	There is No Freeze Protection for the Condensate Storage Tanks	October 29, 1998
CR P-99-01516	Plant Does Not Have adequate Winterization Policies	January 26, 1999
CR P-99-16338	Winter Storm Damage to Intake Structures	June 10, 1999
CR P-00-01638	Unit 1 and 2 Screenhouse Water Level Sensing Lines Freezing	January 28, 2000
CR 02087029	Reviewed Winterization Program Per Condition Report 01318056, Action 1, and Determined That Two Procedure Enhancements Are Necessary to Make the Program Successful in the Future	March 28, 2002

1R04 Equipment Alignment

1R04.1 Partial System Walkdowns

Unit 2 Turbine Driven and West Auxiliary Feedwater (AFW) System Trains

02-OHP-4021-056-001	Filling and Venting Auxiliary Feedwater System	Revision 15
12-PMP-4030.001.001	Impact of Safety Related Ventilation in the Operability of Technical Specification Equipment	Revision 5

02-OHP-4030-STP-017W	West Motor Driven Auxiliary Feedwater System Test	Revision 11
02-OHP-4030-STP-017T	Turbine Driven Auxiliary Feedwater System Test	Revision 15
OP-2-5106A	Flow Diagrams - Auxiliary Feedwater	Revision 45
DB-12-AFWS	Design Basis Document - Auxiliary Feedwater System	Revision 0
	D. C. Cook Nuclear Plant Updated Final Safety Analysis Report (UFSAR), Section 10.5.2, "Auxiliary Feedwater System"	Revision 17
CR 02296004 (1)	2-FW-244-2 West Motor Driven AFW Pump Suction Strainer OME-32W South Basket Vent Valve Leaks at a Rate of Less than One Drop per Minute	October 23, 2002
CR 02296006 (1)	2-FW-244-1 West Motor Driven AFW Pump Suction Strainer OME-32W North Basket Vent Valve Leaks at a Rate of Less than One Drop per Minute	October 23, 2002
CR 02298053 (1)	2-HV-AFP-FD-4B, Fire Damper for the U2 West Motor Driven Aux Feed Pump Room, Was Found to Have a Small Plastic Security Sign Laying in the Fire Damper Track	October 25, 2002

Unit 2 AB Emergency Diesel Generator (EDG)

02-OHP-4021-032-008AB	Operating DG2AB Subsystems	Revision 2
OP-2-5251B-59	Flow Diagram Emergency Diesel Generator AB Unit 2	Revision 59
OP-2-5151A-51	Flow Diagram Emergency Diesel Generator AB Unit 2	Revision 51
CR 02308032 (1)	2-DG-136A (2AB EDG Starting Air Receiver Number 2 to Flywheel Air Jack Connection Shutoff Valve) Was Found Open During a Procedure Walkdown Being Performed By an NRC Inspector	November 4, 2002

Unit 1 West Essential Service Water (ESW) System Train

12-OHP-4021-019-001	Operation of the Essential Service Water System	Revision 25
01-OHP-4030-066-4025	Unit 1 Appendix R and Ventilation Requirements for Unit 2	Revision 3
OP-1-5113-74	Flow Diagram Essential Service Water	Revision 74

Miscellaneous Condition Reports

CR 022700009 (1)	Tell Tale Drains From Unit 1 Steam Generator 1 and 4 Safety Valves Have Signs of Leakage	September 26, 2002

1R05 Fire Protection

1R05.1 Routine Resident Inspector Tours

	D. C. Cook Nuclear Plant UFSAR, Section 9.8.1, "Fire Protection System"	Revision 17
	D. C. Cook Nuclear Plant Fire Hazards Analysis, Units 1 and 2	Revision 8
	D. C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook	February 1995
	D. C. Cook Nuclear Plant Administrative Technical Requirements Manual, Sections 1-FP-7 and 2-FP-7, "Fire Rated Assemblies"	
PMP 2270.CCM.001	Control of Combustible Materials	Revision 1
PMP 2270.FIRE.002	Responsibilities for Cook Plant Fire Protection Program Document Updates	Revision 0
PMP 2270.WBG.001	Welding, Burning and Grinding Activities	Revision 0
PMP 5020.RTM.001	Restraint of Transient Material	Revision 1
PMI 2270	Fire Protection	Revision 26
12-PPP-2270-066-001	Portable Fire Extinguisher Inspections	Revision 0a
Job Order R0216014	18 Month Surveillance of Fire Dampers in Accordance with 12-PPP-4030-066-021	June 9, 2002

12-PPP-4030-066-021	Inspection of Fire Dampers Protecting Safety-Related Areas	Revision 1
Job Order R0234423	Perform 6 Month Surveillance of Fire Detection Circuits in Accordance with 12-IHP-4030-STP-206	November 1, 2002
12-IHP-4030-STP-206	Fire Detection Instrumentation Channel Functional Test	Revision 3
02-IHP-4030-266-052	Unit 2 Control Rod Drive, Transformer, Switchgear Room Carbon Dioxide Fire Suppression Test	Revision 1

Miscellaneous Condition Reports

CR 02305079 (1)	Misinterpretation of the Administrative Technical Requirement Surveillance Requirements for Fire Damper Closure Testing	November 1, 2002
CR 02317181 (1)	Tracking CR to Add Fire Pump House to Fire Hazard Analysis Fire Zone Designations	November 13, 2002

1R11 Licensed Operator Requalification

1R11.1 Resident Inspector Quarterly Review

Licensed Operator Requalification Training Simulator Evaluation Scenario for October 29, 2002

1R12 Maintenance Effectiveness

PMP-5035-MRP-001	Maintenance Rule Program Administration	Revision 4
PMI-5035	Maintenance Rule Program	Revision 9

Control Group Power Supply Failures

	Maintenance Rule (a)(1) Action Plan Reactor Control 7 Instrumentation System - Control Group Power Supplies	Revision 0
	Maintenance Rule Scoping Document Reactor Control System	Revision 1
LER 316-2002-005-00	Unit 2 Trip Due to Instrument Rack 24-Volt DC (Direct Current) Power Supply Failure	July 10, 2002
NRC Information Notice 94-24	Inadequate Maintenance of Uninterruptible Power Supplies and Inverters	March 24, 1994
NRC Information Notice 95-10, Supplement 2	Potential for Loss of Automatic Engineered Safety Features Actuation	August 11, 1995
NRC Event Notification 38915	D. C. Cook Unit 2 Tripped From Full Power Due to an Instrument Rack Power Supply Failure	May 12, 2002
NRC Event Notification 38915, Revised	D. C. Cook Unit 2 Tripped From Full Power Due to an Instrument Rack Power Supply Failure	May 14, 2002
PMP 4010.TRP.001	Unit Two Reactor Trip Review Report	May 12, 2002
	Unit 2 Control Room Logs	May 12, 2002 through May 13, 2002
CR 01236037	There Have Been a Significant Number of Electronic DC Power Supply Failures During the Past 24 Months	August 24, 2001
CR 02047020	2-CG-2-19 Power Supply PS2 Power Available Lamp Is Off	February 16, 2002
CR 02133001	Both 24-Volt DC Power Supplies in Control Group 1 for Rack 16 Failed	May 12, 2002
CR 02133002	Unit 2 Trip From 100 Percent Power Level Due to Low Feedwater Flow Coincident With Low Steam Generator Level on Loop 1	May 12, 2002

CR 02133035	After Unit 2 Trip, No Auto-makeup to Volume Control Tank, 2-QRV-303 Went to Full Divert, and No Refueling Water Sequence Due to 2-QLC-451	May 12, 2002
CR 02133058	The Procedure for Volume Control Tank Instrument Malfunction Directs the Bistable to Be Tripped Which Does Not Result in a Conservative Condition	May 12, 2002
CR 02134014 (1)	TS 3.0.3 Was Entered Erroneously Following the Unit 2 Reactor Trip on May 12, 2002	May 13, 2002
CR 02137004	This CR Written at Plant Operations Review Committee Chairman's Request to Drive a Design Engineering Evaluation of the Instrument Control Power Electrical Distribution Based on the Equipment Operability During the Unit 2 Reactor Trip on May 12, 2002	May 17, 2002
CR 02138002	24-Volt DC Power Supplies in Control Group 1 Are Only Production 4 Volts DC	May 18, 2002
CR 02139034	80-Volt DC Power Supply in Control Group 3 1-PS-CGC-20 PS-1 Is Dead	May 19, 2002
CR 02142030	Unit 2 Tripped Due to Loss of Control Room Control Group 1 Power Supplies	May22, 2002
CR 02325058 (1)	Weekly Recurring Tasks to Walkdown Taylor Mod 30 Power Supplies - No Documented Performance of Walkdown Since September 30, 2002	November 21, 2002
CR 02326025	PS2 Available Lamp Not Lit	November 22, 2002

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

PMP-2291-OLR-001	On-Line Risk Management	Revision 2
PMP-2291-OLR-001	On-Line Risk Management	Revision 3
NUMARC 93-01	Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Section 11, "Assessment of Risk Resulting From Performance of Maintenance Activities"	Revision 2

Unit 1 AB EDG

PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 43, Week 3, with Revisions	October 6, 2002 through October 12, 2002
	Unit 2 Control Room Logs	October 8, 2002 through October 10, 2002
	Unit 2 Supervisors Turnover Logs	October 8, 2002 through October 10, 2002
	Unit 2 Abnormal Position Log	October 8, 2002 through October 10, 2002
	Online Integrated Work Schedule	October 8, 2002 through October 10, 2002

Unit 1 AB EDG ESW Supply Valves

	Daily Shift Manager's Logs	December 6, 2002
PMP-2291-OLR-001, Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 43, Week 11	December 1, 2002 through December 7, 2002

Unit 2 CD EDG

	Daily Shift Manager's Logs	November 4, 2002 through November 5, 2002
PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 43, Week 7	November 3, 2002 through November 9, 2002

Unit 1 East ESW Pump

	Daily Shift Manager's Logs	December 15, 2002 through December 19, 2002
PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 44, Week 1	December 15, 2002 through December 21, 2002

	NRC Letter to A. Christopher Bakken III, Subject: "Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments (TAC NOS. MB5729 and MB5730)"	September 9, 2002

Unit 1 West ESW Pump

	Daily Shift Manager's Logs	October 27, 2002 through October 31, 2002
PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 43, Week 6	October 27, 2002 through November 2, 2002

Unit 2 East ESW Pump

	Daily Shift Manager's Logs	November 17, 2002 through November 23, 2002
PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 43, Week 9	November 17, 2002 through November 23, 2002
	NRC Letter to A. Christopher Bakken III, Subject: "Donald C. Cook Nuclear Plant, Units 1 and 2 - Issuance of Amendments (TAC NOS. MB5729 and MB5730)"	September 9, 2002
	D. C. Cook Nuclear Plant UFSAR, Section 9.8.3, "Service Water Systems"	Revision 17
CR 02324116 (1)	Training for the Sky Jack Fork Truck Was Observed Being Performed in Close Proximity of the Unit 1 Station Transformer	November 20, 2002

Unit 2 East AFW System Train

PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 43, Week 5, with Revisions	October 20, 2002 through October 26, 2002
	Unit 2 Control Room Logs	October 20, 2002 through October 26, 2002

	Unit 2 Supervisors Turnover Logs	October 20, 2002 through October 26, 2002
	Unit 2 Abnormal Position Log	October 20, 2002 through October 26, 2002
	Online Integrated Work Schedule	October 20, 2002 through October 26, 2002

Miscellaneous Condition Reports

CR 02344008	1-HE-8 Turbine Auxiliary Cooling Water Heat Exchanger Became Air Bound on the Circulating Water Side of the Heat Exchanger Resulting in a Secondary Transient in Unit 1	December 10, 2002
CR 02346017	There Have Been Repeat Instances of Inconsistencies in Regards to Application of Cascading TS	December 12, 2002
CR 02350015	Unit 2 Containment Hydrogen Recombiner Number 2 Maintenance Was Scheduled During the 1E ESW Pump Replacement	December 16, 2002

1R14 Personnel Performance During Non-routine Plant Evolutions

1R14.1 Unit 2 Power Reduction to Support Oil Addition to a Reactor Coolant Pump Motor

02-OHP-4021-001-003	Power Reduction	Revision 15
	Daily Shift Manager's Logs	November 10, 2002 through November 11, 2002
CR 02315078 (1)	Rising Pressure Indications on 1-IPI-260/-265, Safety Injection Pump Discharge Pressures	November 11, 2002

1R14.2 Unit 1 Control Group Power Supply Replacement

PMI-4090	Infrequently Performed Test or Evolution Briefing Guide for Replacement of PS2 Power Supply at 1-PS-CGC-16	December 3, 2002

	D. C. Cook Nuclear Plant Unit 1 Technical Specifications	
	Daily Shift Manager's Logs	December 3, 2002

1R15 Operability Evaluations

	D. C. Cook Nuclear Plant Unit 1 and 2 Technical Specifications	
	D. C. Cook Nuclear Plant UFSAR	Revision 17
Generic Letter 91-18	Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions	Revision 1
PMP-7030-ORP-001	Operability Determinations	Revision 9
R219215-02	Ultrasonic Test Report - Unit 2 Essential Service Water Line to West Motor Driven Auxiliary Feedwater Pump	December 4, 2002
CR 01101066	Flush Unit 2 ESW Line to West Motor Driven AFW Pump due to Silt/Sand in Piping	April 11, 2001
CR 02136008	While Performing Inspection of Termi-point Connection, a Wire Found Disconnected in 1-RPS-A	May 15, 2002
CR 02131018	Review Operability and Reportability Issues for Two Items Dealing with Feedwater Pressure Indication and the Plant Process Computer Calorimetric Program	May 11, 2002
CR 02135049	1-CCR-462 Leaking Excessively During Local Leak Rate Testing	May 15, 2002
CR 02290012	Steam Generator PORV Actuator Capability Calculation Revealed Negative Calculated Margin for Full Stroke Capability	October 17, 2002
CR 02300002	Unit 2 Control Room Access Door 2-DR-AUX411B Latch Has Broken and Door Will Not Shut	October 27, 2002

CR 02339016	Ultrasonic Examination on Unit 1 ESW to West Motor Driven AFW Pump Piping Found Some Silt/Sand in the Piping	December 5, 2002

1R16 Operator Workarounds

PMP-4010-OWA-001	Oversight and Control of Operator Workarounds	Revision 1
NRC Inspection Manual Temporary Instruction 2515/138	Evaluation of the Cumulative Effect of Operator Workarounds	
	Work Around Review Board Meeting Agenda	October 23, 2002
	Work Around Review Board Meeting Minutes	October 23, 2002
	Unit 1 Operator Workarounds	October 23, 2002
	Unit 2 Operator Workarounds	October 23, 2002
	Unit 1 Operator Workarounds Contingency Actions for Reactor Operators	October 23, 2002
	Unit 1 Operator Workarounds Compensatory Actions for Auxiliary Equipment Operators	October 23, 2002
	Unit 2 Operator Workarounds Contingency Actions for Reactor Operators	October 23, 2002
	Unit 2 Operator Workarounds Compensatory Actions for Auxiliary Equipment Operators	October 23, 2002
CR 01048019	Unit 1 Main Turbine Was Deliberately Slowed Due to High Vibration Using the Vacuum Breakers	February 17, 2002
CR 01280031	Bypass Steam to Feedwater Heater Valves Leak and Must be Manually Isolated	October 7, 2001
CR 02023083	Feedwater Preheating Valves Will Not Isolate Sufficiently to Prevent Cooldown	January 23, 2002

CR 02312059 (1)	Update Operations Lesson Plan RQ-C-KNOW. Lesson Plan Contains Incorrect Information	November 8, 2002

1R19 Post Maintenance Testing

Unit 2 CD EDG Governor Replacement

02-OHP-4030-STP-027CD	CD Diesel Generator Operability Test (Train A)	Revision 20
PMP-2291-PNT-001	Post Maintenance Testing	Revision 3
12-MHP-5021-032-017	Attachment 1, Guideline To Run Diesel At Slow Speed	Revision 4
CR 02306005	CD EDG Exhibited 150 Kilowatt Oscillations at Full Load During Surveillance Testing	November 2, 2002

Unit 2 East AFW Pump Maintenance

Job Order 02228012-01	2-PP-3E-MTR - Unit 2 East Motor Driven AFW Pump Motor - Check For Soft Foot and Perform Alignment	August 16, 2002
Job Order 02136108-01	2-FRV-255 Adjust Packing and Perform Diagnostic Testing	May 16, 2002
02-OHP-4030-STP-017E	East Motor Driven Auxiliary Feedwater System Test	Revision 10
02-OHP-4021-056-002, Attachment 2	Auxiliary Feed Pump Operation - East Motor Driven Auxiliary Feedwater Pump Long-Term Minimum Flow	Revision 13
02-OHP-4021-056-002, Attachment 9	Auxiliary Feed Pump Operation - East Motor Driven Auxiliary Feedwater Pump Operation	Revision 13
	Unit 2 East Motor Driven AFW Pump Historical Vibration Data	January 2001 through October 2002
OHI-4030	Removal and Restoration of Technical Specification Related Equipment - Unit 2 East Motor Driven AFW Pump	October 23, 2002

CR 02296051 [(1)]	The Motor Mounting Bolts on the Unit 2 East Motor Driven AFW Pump Were Not Tightened per the Torque Selection Procedure 12-MHP-5021-001-009	October 23, 2002

Unit 2 West ESW Pump Maintenance

Job Order 02269031-01	2-PP-7W - Uncouple-Inspect Coupling Gap on Pump	September 26, 2002
Job Order 02269031-02	2-PP-7W Run Pump/Perform Operability Test	September 26, 2002
02-OHP-4030-219-022W	West Essential Service Water System Test	Revision @

Unit 1 Post Accident Containment Hydrogen Monitor

12-EHP-4030-STP-236-010	Leak Test of Unit 1 and Unit 2 Post Accident Containment Hydrogen Monitoring System	Revision 1
OP-1-5141D-19	Flow Diagram Post-Accident Sampling Containment Hydrogen Unit No. 1	Revision 19
OP-2-5141D-14	Flow Diagram Post-Accident Sampling Containment Hydrogen Unit No. 2	Revision 14

1R22 Surveillance Testing

	D. C. Cook Nuclear Plant UFSAR	Revision 17
	D. C. Cook Nuclear Plant Unit 1 and 2 Technical Specifications	

Unit 1 Auxiliary Cable Vault CO_2 [Carbon Dioxide] Fire Suppression Test

01-EHP-4030-ATR-225-020	Unit 1 Auxiliary Cable Vault CO_2 Fire Suppression Test	Revision 0
Administrative Technical Requirements	Section 1-FP-5, Low Pressure CO_2 Systems	Revision 25
CR 02345019	Procedure Needs Enhancement to Avoid Unwanted CO_2 Discharge	December 11, 2002
CR 02345020	Procedure Needs Fixes	December 11, 2002

Unit 2 Distributed Ignition System Surveillance and Baseline Testing

02-IHP-4030-234-001	Unit 2 Distributed Ignition System Surveillance and Baseline Testing	Revision 0
	D. C. Cook Nuclear Plant UFSAR, Section 14.3.6.6, "Distributed Ignition System"	Revision 17
Job Order R232934-01	Perform Quarterly Distributed Ignition System Surveillance Unit 2	

Steam Generator Steam/Feed Flow Mismatch and Steam Pressure Protection Functional Testing

02-IHP-4030-SMP-219	Steam Generator 1&2 Steam/Feed Flow Mismatch and Steam Pressure Protection Set I Functional Test and Calibration	Revision 6
02-IHP-4030-SMP-222	Steam Generator 2&4 Steam/Feed Flow Mismatch and Steam Pressure Protection Set II Functional Test and Calibration	Revision 4
Job Order R0235115-01	Perform 2IHP-4030-SMP-219	
PMI-4030 Performance Review and Acceptance Sheet	Performance Review and Acceptance Sheet for Job Order R0235115-01	
Job Order R0235114-01	Perform 02-IHP-4030-SMP-222	
OP-2-99012	Steam Generator 1 & 2 Mismatch Channel 1 Functional Diagram	Revision 1
	D. C. Cook Nuclear Plant UFSAR, Chapter 7, "Instrumentation and Control"	Revision 17

Containment Isolation and In-service Inspection Valve Operability Testing

01-OHP-4030-STP-011	Containment Isolation and In-service Inspection Valve Operability Test	Revision 23

Steam Pressure Protection Functional Testing

02-IHP-4030-SMP-227	02-IHP-4030-SMP-227 Steam Pressure Protection Set III Functional Test and Calibration	Revision 2

02-IHP-4030-SMP-228	02-IHP-4030-SMP-228 Steam Pressure Protection Set IV Functional Test and Calibration	Revision 2

Unit 2 East AFW Pump Characterization Testing

12-EHP-5030-CAR-001	Characterization Testing Program	Revision 0
Motor Analysis Report, Sequence 24	Motor-Driven AFW Pump, 1-PP-3W-MTR	December 4, 2002

Unit 1 and 2 Personnel Airlock Door Seal Leak Rate Surveillance Testing

12-IHP-4030-046-227	Unit 1 and Unit 2 Personnel Airlock Door Seal Leak Rate Surveillance	Revision 0
Nuclear Energy Institute (NEI) 94-01	Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J	Revision 0
Regulatory Guide 1.163	Performance-Based Leak-Test Program	September 1995
NUREG-1493	Performance-Based Containment Leak-Test Program	September 1995

Miscellaneous Condition Reports

CR 02269002	Unit 2 Main Turbine "B" Control Valve Opened Unexpectedly From 50 Percent to 75 Percent, Causing an Unintended Power Rise and Reactor Coolant System Temperature Reduction and Automatic Control Rod Withdrawal	September 25, 2002

1R23 Temporary Plant Modifications

	D. C. Cook Nuclear Plant UFSAR	Revision 17
12-EHP-5040-MOD-001	Temporary Modifications	Revision 9

Disable East Travel Limit Switch on East Auxiliary Building Crane

12-EHP-5040-EMP-006	Disable Bridge East Travel Limit Switch on East Auxiliary Building Crane 12-QM-3E	Revision 0

2002-1065-00	10 CFR 50.59 Applicability Determination for 12-IHP-5040-EMP-006, Revision 0, "Disable Bridge East Travel Limit Switch on East Auxiliary Building Crane"	
12-TC-02-64-R0	Design Packet for Temporary Condition to Disable East Travel Limit Switch on East Auxiliary Building Crane	
PMP-4050-CHL-001	Control of Heavy Loads	Revision 1
	D. C. Cook Nuclear Plant UFSAR, Section 9.7, "Reactor Components and Fuel Handling System"	Revision 17
	D. C. Cook Nuclear Plant UFSAR, Section 12.2.1, "Control of Heavy Loads"	Revision 17

Plant Winterization

12-TM-00-61-R2	Winterization/De-Winterization Temporary Modification to Support 12-IHP-5040-EMP-004	December 30, 2000
Job Order R0235054	Plant Winterization, Perform PM Task 30	September 27, 2002
RPA005058	Winterization of Tank Vents	
12-EHP-5040-MOD-001	Temporary Modifications	Revision 9
12-IHP-5040-EMP-004	Plant Winterization and De-winterization	Revision 3
	D. C. Cook Nuclear Plant UFSAR, Section 10.5, "Condensate and Feedwater Systems"	Revision 17

Install Backup Power Supply for Control Group 1

1-TM-02-85-R0	Install Backup Power Supply for Control Group 1	November 23, 2002
2002-1637-00	10 CFR 50.59 Applicability Determination for 1-TM-02-85-R0, "Install Backup Power Supply for Control Group 1"	November 23, 2002
Job Order 02326025-01	Install Temporary Modification 1-TM-02-85-R0	November 23, 2002

1EP2 Alert and Notification System (ANS) Testing

	Berrien County Early Warning System (EWS) Operations Manual	December 12, 2001
	D. C. Cook Sirens and Contour Maps	January 2002 through September 2002
	Berrien County Monthly EWS Test Reports	April 2002 through October 2002

1EP3 Emergency Response Organization (ERO) Augmentation Testing

	D. C. Cook Emergency Plan, Section E	Revision 17
	D. C. Cook Emergency Plan, Section N	Revision 17
PMP-2080-EPP-100	Emergency Response	Revision 0
PMP-2080-EPP-107	Notification	Revision 16
SA-2001-SPS-014	Unannounced Drill	December 14, 2001
SA-2001-SPS-032	Semi-Annual Unannounced Drill	August 23, 2001
SA-2002-SPS-026	Unannounced Drill	March 14, 2002
SA-2002-SPS-027	Unannounced Drill	April 16, 2002
SA-2002-SPS-028	On-Shift Unannounced Drill	July 17, 2002
CR 02032031	December 14, 2001 Drill, Emergency Operations Facility and Technical Support Center Did Not Activate Within 60 Minutes	February 1, 2001

1EP5 Correction of Emergency Preparedness Weakness and Deficiencies

PA-01-18	PA Audit - Emergency Planning	February 25, 2002
PA-02-15	PA Audit - Emergency Planning	November 22, 2002
PMP-7030-CAP-001	Corrective Action Program Process Flow	Revision 13
SA-2001-SPS-026	Self-Assessment - Emergency Plan Graded Exercise	July 25, 2001

SA-2001-SPS-036	Self-Assessment - 4th Quarter 2001 ERO Drill	December 27, 2001
SA-2001-SPS-037	Self-Assessment - On-Shift Emergency Planning Staffing Survey	February 28, 2002
SA-2002-SPS-013	Self-Assessment - 1th Quarter 2002 Accountability Drill	March 27, 2002
SA-2002-SPS-021	Self-Assessment - Emergency Plan Drill	August 7, 2002
SA-2002-SPS-022	Self-Assessment - Emergency Plan Drill	September 24, 2002
SA-2002-SPS-031	Self-Assessment - Review of NRC IN 2002-14	October 21, 2002
CR 01247001	During The ESW Flow Restriction Event on 8/29/01, a More Conservative Decision Regarding Emergency Plan Entry Would Have Been Appropriate	September 3, 2001
CR 02157101	Evaluate Timeliness of NRC Notification For Unusual Event Based On PORV Opening On June 5, 2002	June 6, 2002
CR 02163045	Catastrophic Failure Resulting in a Loss Of Offsite Power Sources Supplied to Reserve Feed	June 12, 2002
CR 02010029	Personnel Manning the New Maintenance Contractor Building Did Not Report a Failure with the Public Address System to the Emergency Plan System Engineer During the Accountability Drill on January 8, 2002	January 10, 2002
CR 02204003	ERO Dialogic System Failed to Perform as Required During a Forced Outage Initiation	July 22, 2002
CR 02214013	Power Was Lost at the Buchanan Office Building Due to a Storm Resulting in a Loss of Power to the Emergency Operations Facility	August 2, 2002

CR 02268022	The Process for Ensuring Qualified Individuals Report for ERO Duties Failed and as a Result the Operations Support Center Was Activated During the 9/18/02 Drill with Unqualified Individuals and Individuals That Had Not Been Confirmed to be ERO Qualified	September 25, 2002
CR 02276062	Error in Minimum Staffing Requirement Found in Table 1, Revision 17 of the Emergency Plan	October 3, 2002
CR 02284015	Acceptable Interim Actions Have Not Been Initiated and a Request for Project Authorization Has Not Been Implemented to Correct 20 Plant Locations Where it Has Been Reported That the Public Address System Is Not in Regulatory Compliance	October 11, 2002
CR 02340005	Document Errors Reported in 2nd and 3rd Quarter 2002 NRC DEP Performance Indicator Data	December 5, 2002

<u>1EP6</u> <u>Drill Evaluation</u>

	Exercise Scope and Objectives for October 24, 2002 Annual Exercise	
	Emergency Notification Forms Completed During Annual Exercise	October 24, 2002
	Desktop Guide for Emergency Planning Performance Indicators	Revision 2
PMP-2080-EPP-107	Notifications	Revision 16

<u>2OS1</u> <u>Access Control to Radiologically Significant Areas</u>

PMP-6010-RPP-003	High, Locked High, and Very High Radiation Area Access	Revision 11
PMP-6010-RPP-006	Radiation Work Permit Program	Revision 7a
RP 014-01	Total Effective Dose Equivalent Evaluation Worksheet for Work at 587 Foot Drumming Room Clean-up	September 23, 2002

RP 014-01	Total Effective Dose Equivalent Evaluation Worksheet for Work on Spent Fuel Pool Demineralizer High Pressure Spray of the Inlet Retention Element	October 23, 2002
RP 014-01	Total Effective Dose Equivalent Evaluation Worksheet for Unit 2 at Power Entry to Work on Number 1, Safety Injection Accumulator	November 8, 2002
RWP 020504	Restricted Area NRC Tours and Inspections	Revision 10
RWP 021016	Resin Sluice activities - Locked High Radiation Areas	Revision 6
RWP 021037; 617’	617 Foot Demineralizer Locked High Radiation Area Work Activities	Revision 3
RWP 021046; 587’	Drumming Room Activities	Revision 1
RWP 021052	Unit 2 At Power Entry	Revision 2
RWP 02-1037	Radiation Protection ALARA [As-Low-As-Reasonably-Achievable] Plan for Work on Spent Fuel Pool Demineralizer High Pressure Spray of the Inlet Retention Element	Revision 1
RWP 02-1046	Radiation Protection ALARA Plan for Work at 587 Foot Drumming Room Clean-up	Revision 0
RWP 02-1052	Radiation Protection ALARA Plan for Work on Reactor Coolant Pumps 11 and 14	Revision 1
12-THP-6010-RPP-006	Radiation Work Permit Processing	Revision 17
12-THP-6010-RPP-401	Performance of Radiation and Contamination Surveys	Revision 10
12-THP-6010-RPP-418	Radiological Postings	Revision 9
CR 02217009	Modifications to Radiological Posting Program	August 5, 2002
CR 02226075	Unnecessary Locked High Radiation Areas	August 14, 2002
CR 02308023	Valve Released to Unrestricted Area	November 4, 2002

CR 02337041	Improper Receipt of Package Containing Radioactive Source	November 27, 2002

2OS3 Radiation Monitoring Instrumentation

PA-02-06	Performance Assurance Audit, "Radiation Protection"	April 16, 2002
12-THP06010-RPI-500	Instrument Issue and Operation Testing	Revision 13
12-THP06010-RPI-500	Instrument Issue and Operation Testing; Data from Portal Monitor Operational Checks Performed on December 5, 2002	Revision 13
12-THP06010-RPC.512	Calibration of the Eberline Smart Portable Survey Meter(s)	Revision 5
12-THP06010-RPC.512	Calibration of the Eberline Smart Portable Survey Meter(s); Data Sheet from December 3, 2002	Revision 5
12-THP06010-RPC-513	Calibration of the Eberline Model R0-7 Survey Meter	Revision 2
12-THP06010-RPC-513	Calibration of the Eberline Model R0-7 Survey Meter; Data Sheet from December 3, 2002	Revision 2
	ALARA Radiation Protection Daily Dose Report and Schedule	December 2, 2002
CR 02246017	Foot and Hand Monitor Found Out of Service	September 3, 2002
CR 02249037	Instrument Missing from Work Area	September 6, 2002
CR 02287056	Discrepancies Between Laboratory Cross-check Program	September 30, 2002
CR 02304023	Failure to Follow Procedural Requirements for Instrument Accountability	October 31, 2002

Stations Radiation Protection Instrumentation

	Blitz Team Bulletin; Weekly Station Performance Bulletin	December 3, 2002

	Calibration Packages from a Selection of Station's Radiation Protection Instruments	December 2001 through December 2002
	Online Quality Control Schedule	November 27, 2002
	Radiation Protection Instrument Use History Analysis Forms; Selections from Number 858-953	January 2002 through December 2002
	Report of Instruments Due for Calibration	December 31, 2002

<u>4OA1</u> <u>Performance Indicator (PI) Verification</u>

NEI 99-02	Regulatory Assessment Performance Indicator Guideline	Revision 2
SPP-2060-SFI-101	PI Data Gathering	Revision 0
PMP-7110.PIP.001	Regulatory Oversight Program PI	Revision 1
	Letter from J. Pollock, American Electric Power, to the US NRC, Subject: "Cook Unit 1 and 2 -- 4Q2001 -- PI Data Elements (QR and CR)"	January 17, 2002
	Letter from J. Pollock, American Electric Power, to the US NRC, Subject: "Cook Unit 1 and 2 -- 1Q2002 -- PI Data Elements (QR and CR)"	April 19, 2002
	Letter from J. Pollock, American Electric Power, to the US NRC, Subject: "Cook Unit 1 and 2 -- 2Q2002 -- PI Data Elements (QR and CR)"	July 18, 2002
	Letter from J. Pollock, American Electric Power, to the US NRC, Subject: "Cook Unit 1 and 2 -- 3Q2002 -- PI Data Elements (QR and CR)"	October 21, 2002
	Administrative Technical Requirements Units 1 and 2, Reactor Coolant System, Supplemental Operational and Surveillance Requirements	Revision 18
	Results of Gamma Spectrometry Count of Units 1 and 2 Reactor Coolant System Specific Activity Samples	December 4, 2002

OHI-4032	Leakage Monitoring Program	Revision 2
12-THP-6020-CHM-101	Reactor Coolant System	Revision 14
12-EHP-5030-001-008	Recirculation Loop Total Leak Rate	Revision 3
12-THP-6020-INS-026	Gamma Spectrometry System	Revision 1
	Licensee Event Reports	October 1, 2001 through September 30, 2002
NRC Information Notice 94-46	Non conservative Reactor Coolant System Leakage Calculation	June 20, 1994
CR 01201019	Enhancements May Be Needed in the Documentation of Test Results of 4 Alert and Notification System Sirens in the Two State Parks	July 20, 2001
CR 01325066	Resident Inspector Observations of Cook Operations Training with Regard to EP Performance Indicator Data Gathering	November 21, 2001
CR 02193022	Declining Trend in DEP Emergency Planning NRC Performance Indicator	July 12, 2002
CR 2009038	Declining Trend in DEP Emergency Planning NRC Performance Indicator	January 9, 2002
CR 02019069	Exceeding Limits for Hard Gammas in Reactor Coolant System Filtrate Isotopic Mixture	January 19, 2002
CR 02219004	E-BAR Determinations Found to Be Slightly Erroneous	August 6, 2002
CR P-00-29181	Control Room Operability Evaluation, with Subsequent Lowering of TS for Reactor Coolant System Specific Activity	December 15, 1999

4OA2 Identification and Resolution of Problems

4OA2.1 EDG Starting Air Relay Failures

JO 00266004	Job Order - Unit 1 CD Diesel Failed to Stop (Suspect 1-19-DGCD)	October 6, 2000

CR P-99-01279	Unit 2 AB EDG Rolled With Air by Itself. No indication of a Start Signal was Detected Locally or the Control Room. Starting Air Continued to Blow Down Engine Until Air Depleted.	January 21, 1999
CR P-99-01336	Found a Relay (EDG - Start Failure Relay 2-19-DGAB) in a De-energized State. Voltage was Present at the Coil Terminals Which Should Have Kept the Relay in an Energized State	January 22, 1999
CR 00266004	Unit 1 CD Diesel Generator Failed to Stop from the Control Room	September 22, 2000
CR 02289033	Diesel Generator 2AB 2-OME-150-AB Starting Rolling Unexpectedly on Starting Air	October 16, 2002
CR 02296037	Replace 1-19-1-DGCD. There Have Been Three 19/19-1 Relay Failures Since January 1999. They Are Original Installation and Normally Energized.	October 23, 2002

<u>4OA2.2</u> <u>Common Cause Failure of Four Unit 1 Charging System Check Valves</u>

PMP 7030.OE.001	Industry Operating Experience	Rev. 5
	Part 21 Notification, Swing Check Valves - Forged Steel	January 18, 1991
Problem Report 90-1503	Velan Valve Designs	October 12, 1990
Problem Report 92-157	Part 21 Issue Concerning Velan Valves	February 20, 1992
CR 96-0094	Operating Experience 7640, "Charging Injection Valves Found Stuck Open"	January 24, 1996
CR 02132050	Disc on Valve 1-CS-329-L1 Was Found in the Open Position	May 12, 2002
CR 02134021	Check Valves 1-CS-328-L1, 1-CS-328-L4, 1-CS-329-L1, and 1-CS-329-L4 Were Found Open During Radiographic Nonintrusive Testing	May 14, 2002
CR 02205061	10 CFR 21 Evaluation Required for Manufacturing Defects Discovered on Velan 3-inch Check Valves	July 24, 2002
CR P-00-07039	NRC Information Notice 2000-08	May 16, 2000

CR 00278072	Operating Experience Number 11420	October 4, 2000
CR 01067004	Operating Experience 11950	March 1, 2001
CR 01136027	NRC Information Notice 2001-06	May 16, 2001
CR 01198006	Operating Experience 12454	July 17, 2001
CR 01198020	Operating Experience 12451	July 17, 2001
CR 01206021	Operating Experience 12454	July 25, 2001
CR 01362008	NRC Information Notice 2001-14	December 28, 2001
CR 02002020	NRC Information Notice 2001-19	January 2, 2002

4OA3 Event Follow-up

4OA3.1 Pressurizer Power Operated Relief Valve (PORV) Inadvertently Opened During Testing Resulting in a Loss of Reactor Coolant System Inventory and an Unusual Event

NRC Event Notification 38967	Unit 1 Declared an Unusual Event Due to Reactor Coolant System Leakage Greater Than 25 Gallons-Per-Minute During Surveillance Testing	June 6, 2002
01-OHP-4030-102-017	Pressurizer PORV Actuation Channel Calibration with Valve Operation (for Modes 1, 2, and 3)	Revision 0
01-OHP-4030-102-017	Pressurizer PORV Actuation Channel Calibration with Valve Operation (for Modes 1, 2, and 3)	Revision 1
	Daily Shift Manager's Logs	June 5, 2002 through June 6, 2002
CR 02157039	Pressurizer PORV 1-NRV-153 Opened During Testing with its Block Valve Open, Causing an Unexpected Release of Reactor Coolant System Inventory into the Pressurizer Relief Tank	June 6, 2002
CR 02157101	Evaluate Timeliness of NRC Notification for Unusual Event Declared Based on PORV Opening on June 5, 2002 at 23:00	June 6, 2002

4OA3.2 LER 315-1999-010-01, "Reactor Coolant System Leak Detection System Sensitivity Not in Accordance with TS Basis"

LER 315-1999-010-00	Reactor Coolant System Leak Detection System Sensitivity Not in Accordance with TS Basis	May 3, 1999
LER 315-1999-010-01	Reactor Coolant System Leak Detection System Sensitivity Not in Accordance with TS Basis	March 6, 2000
Generic Letter 84-04	Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR (Pressurized Water Reactors) Primary Main Loops	February 13, 1984

4OA3.3 Unanticipated Start of the Unit 2 Turbine Driven Auxiliary Feedwater Pump (TDAFWP) During a Normal Plant Shutdown for Refueling Outage

LER 50-316-2002-04-00	Unanticipated Start of the Turbine Drive Auxiliary Feedwater Pump	March 15, 2002
LER 50-316-2002-04-01	Unanticipated Start of the Turbine Drive Auxiliary Feedwater Pump, Supplement 1	June 28, 2002
LER 50-316-2002-04-02	Unanticipated Start of the Turbine Drive Auxiliary Feedwater Pump, Supplement 2	December 13, 2002
NRC Event Notification 38640	The Turbine Driven Auxiliary Feedwater Pump Auto Started After a Scheduled Reactor Trip From 20 Percent Power	January 19, 2002
02-OHP-4021-001-003	Power Reduction	Revision 15
2001-0985-00	10 CFR 50.59 Screening for Revision 15 to 02-OHP-4021-001-003, "Power Reduction"	January 11, 2002
	Daily Shift Manager's Logs	January 19, 2002
CR 02019036	During Planned Reactor Trip the TDAFWP Started	January 19, 2002
CR 02107016 (1)	CR 02019036 Evaluation Did Not Address the Operational Aspects of the TDAFWP Auto Start on the Planed Reactor Trip	April 17, 2002

4OA3.4 LER 50-316-2002-05-00, "Unit 2 Trip Due to Instrument Rack 24-Volt DC Power Supply Failure"

LER 50-316-2002-05-00	Unit 2 Trip Due to Instrument Rack 24-Volt DC Power Supply Failure	July 10, 2002

4OA3.5 LER 50-316-1997-004-02, "Analysis Demonstrates Design Basis Impact of Inadequate Refueling Outage Safety Evaluation Was Negligible"

LER 316-1997-04-02	Analysis Demonstrates Design Basis Impact of Inadequate Refueling Outage Safety Evaluation was Negligible	January 22, 1998
LER 316-1997-04-01	Change to Component Cooling Water Temperature Without Revision to UFSAR	November 17, 1997

LER 316-1997-04-00	Change to Component Cooling Water Temperature Without Revision to UFSAR Results in Condition Outside Design Basis	September 22, 1997
CR 97-2342	Inadequate Safety Review Performed for Establishment of a 90°F Upper Limit for Component Cooling Water During Unit 2 1996 Refueling Outage	August 26, 1997
Amendment Request 1202	Refueling Operations Decay Time Technical Specification	November 16, 1994
Amendment Request 1202B	Response to Request for Additional Information (RAI) Technical Specification Amendment Refueling Operations Decay Time	August 1, 1996
Amendment Request 1202A	Refueling Operations Decay Time Updated Analysis and Response to RAI	February 1, 1996
Amendment Request 1202D	Response to RAI Regarding Refueling Operations Decay Time	June 19, 1997
Amendment Request 1202F	Request to Withdraw the Refueling Operations Decay Time Technical Specification Amendment Request	January 27, 1998
Amendment Request 1146	Refueling Operations Decay Time Technical Specification	November 30, 2001

4OA3.6 LER 50-315-1997-005-00, "Reactor Coolant Pump Fire Protection Inoperable for Extended Period Without Compensatory Actions Due to Improperly Fabricated Gasket in Spray Header Line"

LER 315-1997-05-00	Reactor Coolant Pump Fire Protection Inoperable for Extended Period Without Compensatory Actions due to Improperly Fabricated Gasket in Spray Header Line	April 14, 1997
CR 97-0586	When Installing Blank Side of Spectacle Flange Rubber Gasket Not Properly Installed	March 5, 1997

4OA3.7 LER 50-315-1997-005-01, "Reactor Coolant Pump Fire Protection Inoperable for Extended Period Without Compensatory Actions Due to Improperly Fabricated Gasket in Spray Header Line"

LER 315-1997-05-00	Reactor Coolant Pump Fire Protection Inoperable for Extended Period Without Compensatory Actions due to Improperly Fabricated Gasket in Spray Header Line	April 14, 1997
LER 315-1997-05-01	Reactor Coolant Pump Fire Protection Inoperable for Extended Period Without Compensatory Actions due to Improperly Fabricated Gasket in Spray Header Line	October 23, 1997
CR 97-0586	When Installing Blank Side of Spectacle Flange Rubber Gasket Not Properly Installed	March 5, 1997

4OA3.8 LER 50-315-1998-056-01, "Inadequate Control and Processing of Design Information Results in Unanalyzed Hot Leg Recirculation Switchover"

LER 50-315-1998-056-00	Inadequate Control and Processing of Design Information Results in Unanalyzed Hot Leg Recirculation Switchover	January 6, 1999
LER 50-315-1998-056-01	Inadequate Control and Processing of Design Information Results in Unanalyzed Hot Leg Recirculation Switchover	November 24, 1999
CR P-98-7848	Unanalyzed Condition Pertaining to Post-Loss-of-Coolant Accident Emergency Core Cooling System Hot Leg Switchover	December 11, 1998

4OA3.9 LER 50-315-1998-029-01, "Fuel Handling Area Ventilation System Inoperable Due to Original Design Deficiency"

LER 50-315-1998-029-01	Fuel Handling Area Ventilation System Inoperable Due to Original Design Deficiency	August 4, 1999
Calculation No. RD-99-01	Control Room Dose Resulting from a Fuel Handling Accident for Off-Load Specific Conditions	Revision 1

CR P-98-01712	Fuel Handling Area Ventilation System Inoperable Due to Original Design Deficiency	April 22, 1998

4OA3.10 LER 50-315-1999-003-00, "Control Room Pressurization System Surveillance Test Does Not Test System in Normal Operating Condition"

LER 50-315-1999-003-00	Control Room Pressurization System Surveillance Test Does Not Test System in Normal Operating Condition	February 24, 1999
CN-CRA-99-78	D.C. Cook TID-14844 Source Term Loss of Coolant Accident Radiation Dose Analysis	February 29, 2000
12 EHP 4030 STP 229	Control Room Emergency Ventilation Test	Revision 3
American Electric Power Purchase Order A 10342	NCS Corporation - Control Room Envelope In-leakage Testing at D.C. Cook Nuclear Plant 1999 - Final Report	August 11, 1999
CR P-99-00275	Control Room Pressurization System Surveillance Test Does not Test System in Normal Operating Condition	January 7, 1999

4OA3.11 LER 50-315-1999-003-01, "Control Room Pressurization System Surveillance Test Does Not Test System in Normal Operating Condition"

LER 50-315-1999-003-01	Control Room Pressurization System Surveillance Test Does Not Test System in Normal Operating Condition	November 10, 2000

4OA3.12 LER 50-315-2000-004-00, "Circuit Design Could Result in Failure of Emergency Diesel Generators to Load Properly After Loss of Offsite Power"

LER 50-315-2000-004-00	Circuit Design Could Result in Failure of Emergency Diesel Generators to Load Properly After Loss of Offsite Power	July 3, 2000
NRC Information Notice 93-17	Safety Systems Response to Loss of Coolant and Loss of Offsite Power	March 8, 1993
CR P-99-18884	Certain Automatic Safety Systems Could Respond Inappropriately to Certain Sequences of Loss of Coolant and Loss of Offsite Power Events	July 19, 1999

4OA5.3 Inspector Follow-up Item (IFI) 50-315/316-99-29-01, "Review and Approval of Dose Calculation for General Design Criteria 19 Control Room Habitability Issue"

DIT-B-00069-00	Design Input for D. C. Cook Offsite and Control Room Dose Analysis	July 21, 1999
DIT-B-00069-09	Design Input for D. C. Cook Offsite and Control Room Dose Analysis	April 5, 2002
RD-01-05	Adjusted Dose Consequences for Changes to Control Room	Revision 1

4OA5.4 IFI 50-316-00-07-03, "Failure to Perform Post Modification Checks to Verify Adequate Clearance Between the Pressurizer Surge Line Whip Restraints and the Surge Line under Hot Plant Conditions"

2-DCP-4260	Modification to Surge Line Pipe Whip Restraints with Field Change Requests	Revision 0
SD-990825-001	HELB Structural Evaluation of Surge Line Pipe Whip Restraints	Revision 3
CR 01089055	Calculation Impact Assessment Did Not Adequately 03/30/2001 Address Impacts Associated with Calculation SD-990825-001	March 30, 2001

4OA5.5 URI 50-315/316-00-16-04, "Determine Whether the Latent Failure of a Test Relay Should Be Treated under the Category of a Single Failure"

NRC Task Interface Agreement No. 2000-12	Evaluation of the Engineering Safety Features Safeguards Test Cabinet	November 11, 2000

4OA5.6 URI 50-315/316-01-15-01, "A Change Was Made to the UFSAR Without a 10 CFR 50.59 Evaluation"

CR 01291058	UFSAR Change Request Number 969, Changed the Seismic Class of Components Within the ESW System, but Did Not Use CFR50.59 as a Basis for the Change	October 18, 2001

(1) Condition report written as a result of inspection activities.