July 30, 2002

Mr. A. C. Bakken III
Senior Vice President
Nuclear Generation Group
American Electric Power Company
500 Circle Drive
Buchanan, MI 49107-1395

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 50-315/02-03(DRP); 50-316/02-03(DRP)

Dear Mr. Bakken:

On June 30, 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 July 9, 2002, 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, eight issues 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, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the D. C. Cook facility.

The NRC has increased security requirements at D.C. Cook in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

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).

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

Sincerely,

/RA/

David Passehl, Acting Chief
Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 50-315/02-03(DRP);
50-316/02-03(DRP)

cc w/encl: J. Pollock, Site Vice President
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R. Whale, Michigan Public Service Commission
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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:	50-315/02-03(DRP); 50-316/02-03(DRP)
Licensee:	American Electric Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	1 Cook Place Bridgman, MI 49106
Dates:	April 1, 2002, through June 30, 2002
Inspectors:	B. Kemker, Senior Resident Inspector K. Coyne, Resident Inspector R. Krsek, Resident Inspector, Palisades J. Belanger, Senior Physical Security Inspector R. Gattone, Radiation Specialist D. Jones, Reactor Engineer D. Passehl, Senior Project Engineer W. Slawinski, Senior Radiation Specialist
Approved by:	D. Passehl, Acting Chief Branch 6 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000315-02-03(DRP), IR 05000316-02-03(DRP), on 04/01/2002-06/30/2002, Indiana Michigan Power Company, D. C. Cook Nuclear Power Plant, Units 1 and 2. Maintenance Risk Assessments and Emergent Work Evaluation, Personnel Performance During Non-routine Plant Evolutions, Event Follow-up.

This report covers a 12-week period of inspection by resident and region based inspectors. 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: Barrier Integrity

- Green. A Non-Cited Violation of Unit 2 Technical Specification (TS) 3.9.4.c was self-revealed for the licensee's failure to have the nitrogen to pressurizer relief tank containment penetration isolated prior to commencing core alterations. An operator incorrectly opened the instrument root shutoff containment isolation valve and removed the "Do Not Operate" tag from the valve without verifying the required position of the valve for local leak rate testing. This resulted in an inoperable containment penetration during refueling and resulted in the plant being in a higher risk configuration than that planned by the licensee.

  The inspectors determined that this issue had a credible impact on safety because the licensee failed to have the containment penetration isolated as required by the TSs and the valve was not in the correct position to fulfill its design safety function. The inspectors utilized the event information in conjunction with Appendix G, “Shutdown Operations Significance Determination Process,” of Manual Chapter 0609, Table T-1, “Pressurized Water Reactor (PWR) Refueling Operation Reactor Coolant System (RCS) Level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours AND Inventory in the Pressurizer.” This self-revealed issue was determined to be of very low significance (Green) by the significance determination process because (1) the issue did not increase the likelihood of a loss of primary coolant system inventory; (2) the issue did not degrade the licensee’s ability to terminate a leak path or add RCS inventory when needed; and (3) the issue did not degrade the licensee’s ability to recover decay heat removal once lost. Although this issue affected the integrity of the reactor containment during core alterations, the inspectors concluded that because the small diameter penetration would be a very small leakage path, this issue was of very low safety significance. (Section 1R14.1)

- Green. A Non-Cited Violation of Unit 1 TS 3.4.11.c was self-revealed. An operator incorrectly positioned the control switches for pressurizer power operated relief valves (PORVs) 1-NRV-152 and 1-NRV-153, rendering the valves unavailable for automatic pressure control. With Unit 1 in Mode 1 and two PORVs inoperable due to causes other than excessive seat leakage, the licensee failed to restore at least one of the inoperable PORVs to operable status within the following 72 hours or be in Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours.

The inspectors assessed this event using the Significance Determination Process (SDP). The inspectors determined that this issue had a credible impact on safety because the two PORVs were not capable of automatically controlling reactor coolant system (RCS) pressure below the setting of the pressurizer code safety valves, thereby reducing challenges to these valves. At the time of this event, the third pressurizer PORV (1-NRV-151) was already unavailable (automatic function only) with its manual isolation valve closed due to excessive seat leakage. Therefore the automatic function of all three PORVs was disabled. Although all three PORVs were not capable of automatic operation, the valves were still capable of manual operation to mitigate a steam generator tube rupture accident or as an alternate means of decay heat removal during plant shutdown. The inspectors concluded that this issue affected the operability of the pressurizer PORVs, which are barrier integrity components under the SDP designed to maintain the integrity of the RCS. The inspectors performed a Phase 2 SDP analysis for this finding using the following assumptions: (1) manual operation of the PORVs for primary heat removal using the feed and bleed safety function was not affected; therefore, the inspectors only evaluated the Anticipated Transients Without Scram (ATWS) initiator which considered the primary relief safety function; (2) the duration of the performance deficiency was 13 days; and (3) operator action to manually actuate the failed automatic function of the PORVs was credited. Results of the Phase 2 ATWS worksheet determined that only one accident sequence was affected and resulted in this issue being characterized as having very low safety significance. In accordance with NRC Inspection Manual Chapter 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 1R14.2)

- Green. A Non-Cited Violation of Unit 2 TS 3.6.1.3 was self-revealed for the licensee’s failure to have at least one containment airlock door closed while the airlock was inoperable with Unit 2 in Mode 3. The mechanical interlock on the lower containment personnel airlock malfunctioned and personnel opening the inner airlock door challenged the interlock by not verifying the outer door was closed prior to opening the inner door. This created a direct access path from the containment atmosphere to the outside atmosphere.

  The inspectors assessed this event using the Significance Determination Process (SDP). The inspectors determined that this issue had a credible impact on safety because the licensee failed to have at least one airlock door closed while the containment airlock was inoperable as required by the TSs and the resultant rapid containment pressure change also affected the operability of the ice condenser. The inspectors reviewed the guidance in NRC Inspection Manual Chapter 0609, Appendix H, "Containment Integrity SDP," and determined the finding was a Type "B" finding. Type "B" findings have no impact on the determination of Core Damage Frequency (CDF) and therefore they are not processed through the CDF based SDP. These findings, however, are potentially important to Large Early Release Frequency (LERF) determinations. The initial screening of the finding determined that the issue was potentially risk significant based on containment and ice condenser integrity which can be affected by the finding. The issue was therefore referred to the regional Senior Reactor Analyst (SRA) for further review. The analyst evaluated the circumstances of the issue to determine the actual duration of the finding. It was determined that the T/2 approach for fault exposure was not appropriate as the containment airlock doors were not discovered in the open position. In addition, the T/2 approach is generally used to estimate when a condition first occurred. The analyst therefore used the 5 second duration of time that the doors were actually opened, as each entry through the containment airlock is a deliberate, monitored activity (rather than a random event) and the licensee would be expected to identify the problem (both containment airlock doors opened simultaneously) as soon as it occurs. In determining the actual risk significance the SRA with the assistance of the headquarters containment risk analyst, utilized the LERF methodology identified in Appendix H for Type "B" findings. Utilizing this approach with actual plant specific probabilistic risk assessment values, the issue was determined to be of very low safety significance. (Section 1R14.3)

- Green. A Non-Cited Violation of Unit 2 TS 3.9.4.c was self-revealed for the licensee's failure to maintain refueling integrity configuration control of containment penetration CPN-74 during core alterations when containment isolation valve 2-XCR-101 was stroked open for testing. Opening this valve created a direct access path from the containment atmosphere to the outside atmosphere.

  The inspectors determined that this issue had a credible impact on safety because the licensee failed to have the containment penetration isolated as required by the TSs. The inspectors utilized the event information in conjunction with Appendix G, "Shutdown Operations Significance Determination Process," of Manual Chapter 0609, Table T-1, "Pressurized Water Reactor (PWR) Refueling Operation Reactor Coolant System (RCS) Level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours AND Inventory in the Pressurizer." This issue was determined to be of very low significance (Green) by the significance determination process because (1) the issue did not increase the likelihood of a loss of primary coolant system inventory; (2) the issue did not degrade the licensee's ability to terminate a leak path or add RCS inventory when needed; and (3) the issue did not degrade the licensee's ability to recover decay heat removal once lost. Although this issue affected the integrity of the reactor containment during core alterations, the inspectors concluded that because 2-XCR-101 was open for a short period of time and the small diameter penetration would be a very small leakage path, this issue was of very low safety significance. (Section 4OA3.3)

- Green. The inspectors identified a Non-Cited Violation of Unit 1 Technical Specification 4.6.5.3.1.b.3, 4.6.5.3.1.b.4, and 4.6.5.3.1.b.5 requirements associated with testing of the ice condenser lower inlet doors. Contrary to the TS requirements, previous TS 4.6.5.3.1.b surveillance testing performed in Unit 1 on November 21, 2000, failed to adequately measure the door opening torque and the door closing torque in accordance with the TS requirements. Specifically, the methodology used by the licensee to perform TS 4.6.5.3.1.b.3 and 4.6.5.3.1.b.4 testing resulted in door closing torques that were greater in magnitude than the door opening torques, contrary to the TS description of these torque values. The inspectors identified that the measured opening torque values for 36 of 48 Unit 1 lower inlet doors were less than the associated door closing torque values. Because calculation of the door frictional torque required accurate measurement of the door opening and closing torques, the licensee was unable to demonstrate compliance with the requirements of TS 4.6.5.3.1.b.5.

  The inspectors assessed this finding using the Significance Determination Process. The inspectors determined that the failure to adequately implement TS 4.5.6.3.b requirements for testing of the Unit 1 lower inlet doors had a credible impact on safety and was more than a minor concern. As stated in the TS 3.6.5 bases, operability of the ice condenser doors ensures that reactor coolant fluid released during a loss of coolant accident (LOCA) will be diverted through the ice condenser bays for heat removal. The ice condenser also augments the containment recirculation sump water inventory in the event of certain small break LOCAs and limits ice maldistributions within the ice condenser. Because the proper functioning of the ice condenser lower inlet doors was primarily associated with the heat removal function of the ice condenser, the inspectors determined that this issue was associated with the barrier integrity cornerstone. Based on a review of additional testing results for the Unit 1 lower inlet doors performed in May 2002, the inspectors concluded that there was no actual reduction in the atmospheric pressure control function of the reactor containment nor a loss of capability to provide additional recirculation sump inventory during certain small break LOCAs. Therefore, this issue was determined to be of very low safety significance. (Section 4OA3.5)

Cornerstone: Mitigating Systems

- Green. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed following the identification of foreign material in the Unit 1 West essential service water (ESW) pump. On June 24, 2002, the licensee identified a rapid degradation in the performance of the Unit 1 West ESW pump. Subsequent investigation identified that plastic barrier tape, a foreign material, had been ingested by the pump and had become wound tightly around the pump’s impeller. The inspectors concluded that the licensee failed to establish appropriate work controls to control foreign material in areas adjacent to the Unit 1 West ESW pump in accordance with the requirements of PMI-2220, "Foreign Material Exclusion."

The inspectors evaluated this failure to establish appropriate foreign material controls in the vicinity of the Unit 1 West ESW pump using the Significance Determination Process. The inspectors determined that this issue had a credible impact on safety and was more than a minor concern. Specifically, ingestion of foreign material by the Unit 1 West ESW pump degraded pump performance and rendered the pump inoperable, which affected the reliability and capability of the ESW system. The safety function of the ESW system is to provide sufficient cooling capacity for continued operation of safety-related equipment during normal and accident conditions. Consequently, the inspectors determined that this issue affected the objectives of the mitigating systems cornerstone. The inspectors concluded that this issue did not result in an actual loss of the safety function of a single train of ESW for greater than the TS allowed outage time. Additionally, because of the continued availability of ESW capability from both of the Unit 2 ESW trains and the Unit 1 East ESW train, the inspectors concluded that the foreign material ingestion did not result in an actual loss of the ESW system safety function. Consequently, the inspectors concluded that this issue was of very low safety significance. (Section 1R13)

- Green. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for maintenance procedures inappropriate to the circumstances, was self-revealed following gas binding of the Unit 2 West centrifugal charging pump. On February 16, 2002, the running charging pump became gas bound following attempts to switch the suction source from the volume control tank to the refueling water storage tank. Follow-up investigation revealed that valve 2-CS-369 (reactor coolant pump seal water heat exchanger to volume control tank shutoff valve) was partially open, allowing transfer of volume control tank cover gas directly to the suction of the Unit 2 charging pumps. The licensee later determined that the position of the 2-CS-369 stem stop nut prevented full closure of the valve. Approximately two weeks prior to this event, the licensee replaced the diaphragm in 2-CS-369 using instructions provided in maintenance procedure 12 MHP-5021-001-023. However, the instructions contained in 12 MHP-5021-001-023 were inconsistent with vendor recommendations for stem stop nut adjustment and contributed to the failure to maintain proper positioning of the stem stop nut. The inspectors determined that the failure to provide procedures appropriate to the circumstances for the adjustment of the 2-CS-369 stem stop nut was a violation of NRC requirements.

  The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue had a credible impact on safety and was therefore more than a minor concern. In particular, the gas intrusion into the suction of the running Unit 2 West centrifugal charging pump while aligned to the refueling water storage tank, a potential common cause failure mechanism for both of the Unit 2 charging pumps, impacted the capability of the high head injection system to provide the inventory and reactivity control safety functions. Therefore, the inspectors determined that this issue was associated with the mitigating systems cornerstone. During the Phase 1 SDP review, the inspectors concluded that this issue degraded the licensee's ability to add inventory to the reactor coolant system and therefore a Phase 2 SDP analysis was required. The Phase 2 shutdown risk SDP analysis, performed with the assistance of the Region III Senior Reactor Analyst and headquarters probabilistic risk assessment staff, determined that the total change in Core Damage Frequency associated with this condition was estimated to be approximately 3E-7 per year. The risk analysts reviewed several shutdown accident scenarios and determined that drain down to mid-loop operation after refueling to support vacuum refill of the reactor coolant system was the most limiting scenario. Based on the overall change in Core Damage Frequency, this issue was determined to be of very low safety significance. (Section 4OA3.7)

Cornerstone: Occupational Radiation Safety

- Green. A Non-Cited Violation of 10 CFR 20.1701 was identified for the licensee's failure to utilize all intended radiological engineering controls to limit the concentration of radioactive material in air during steam generator eddy current testing, resulting in intakes to four workers.

  This finding was determined to be of very low safety significance since radiation exposures to involved workers were low relative to regulatory limits, and because radiological conditions were not of a magnitude sufficient to create a substantial potential for an overexposure. (Section 2OS2.7)

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

Violations of very low safety significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in Section 40A7 of this report.

Report Details

Summary of Plant Status:

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

- On April 25, 2002, the licensee reduced power to approximately 8 percent of rated thermal power and took the main generator off-line to secure components of a 345 kilo-volt line disconnect that were damaged during switchyard maintenance activities the previous day. The licensee synchronized the unit to the grid on April 26, 2002.

- On May 3, 2002, the licensee conducted a reactor shutdown for the Cycle 18 refueling outage (U1C18). Following completion of the refueling outage, the licensee synchronized the unit to the grid on June 9, 2002.

- On June 12, 2002, a loss of the preferred reserve offsite power source to Unit 1 occurred when a 345 kilo-volt breaker exploded and a fire ensued in the 345 kilo-volt switchyard. Unit 1 was maintained at 70 percent power during the event. Power ascension to full power resumed on June 13, 2002.

- On June 14, 2002, operators manually tripped Unit 1 in response to a main feedwater pump trip. The feedwater pump condenser became clogged with zebra mussels and lost vacuum when a circulating water pump was started. The licensee performed a reactor startup and synchronized the unit to the grid on June 18, 2002.

Unit 2 operated at or near full power during this inspection period with the following exceptions:

- On April 4, 2002, the licensee initiated a reactor shutdown as required by Technical Specification (TS) 3.8.2.3.b due to an inoperable safety-related 250 volt battery. The licensee reduced power to approximately 41 percent of rated thermal power prior to receiving approval of a Notice of Enforcement Discretion (NOED). The licensee returned the unit to full power on April 5, 2002.

- On May 12, 2002, Unit 2 experienced an automatic reactor trip due to a power supply failure that caused the number 21 steam generator feedwater regulating valve to fail closed. The power supply failure also affected the operation of the steam dumps, causing the loss of the normal heat sink. Following the replacement of failed power supply drawers, the licensee synchronized the unit to the grid on May 15, 2002.

- On May 25, 2002, the licensee performed a reactor shutdown to isolate a steam leak and replace one of the main turbine reheat stop valves due to a failed weld. The main steam stop valves were shut after the trip to isolate the steam leak, causing the loss of the normal heat sink. Following the valve replacement, the licensee synchronized the unit to the grid on June 2, 2002.

- On June 12, 2002, a loss of the preferred reserve offsite power source to Unit 2 occurred when a 345 kilo-volt breaker exploded and a fire ensued in the 345 kilo-volt switchyard. Unit 2 was maintained at full power during the event. The licensee received an NOED to extend the 2-hour allowed action time of TS 3.0.5 to preclude shutting down the unit until an operable train of essential service water (ESW) was restored.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's procedures and preparations for high temperature, high wind, and flooding conditions. The inspectors reviewed severe weather procedures, emergency plan implementing procedures related to severe weather, annunciator response procedures, and performed general area walkdowns. During the walkdowns, the inspectors observed housekeeping conditions and verified that material capable of becoming an airborne missile hazard during high wind conditions or severe weather was appropriately restrained. Additionally, the inspectors reviewed condition reports (CRs) and the identification and resolution of equipment deficiencies associated with adverse weather mitigation.

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:

Mitigating Systems Cornerstone

- Unit 2 Train AB and CD Station Batteries
- Unit 2 Turbine Driven and East Motor Driven Auxiliary Feedwater (AFW) Pump Trains

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

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

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

Initiating Events Cornerstone

- Unit 2 Circulating Water System

The inspectors reviewed ongoing system maintenance, open job orders, and design issues for potential effects on the ability of the Unit 2 circulating water system to perform its design functions. The inspectors ensured that the configuration of the system was in accordance with applicable operating checklists. The inspectors verified acceptable material condition of system components, availability of electrical power to system components, and that ancillary equipment or debris did not interfere with system performance.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours

a. Inspection Scope

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

Mitigating Systems Cornerstone

- Unit 1 Lower Containment Building
- Unit 2 Auxiliary Cable Vault (Zone 59)
- Unit 2 Switchgear Room Cable Vault (Zone 60)
- Unit 2 East and West ESW Pump Rooms and Adjacent Screenhouse Areas (Zones 29C, 29D, and 142)

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.

.2 Annual Fire Drill Observation

a. Inspection Scope

The inspectors assessed fire brigade performance and the drill evaluators' critique during a fire brigade drill conducted in the Auxiliary Building entry/exit area on April 10, 2002. The drill simulated a trash receptacle fire in the center room of the radiological protection (RP) offices. The inspectors focused on command and control of fire brigade activities, fire fighting and communication practices, material condition and use of fire fighting equipment, and implementation of pre-fire plan strategies.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors evaluated whether the licensee took appropriate precautions to mitigate the risk from external flooding events. Specifically, the inspectors performed the following:

- reviewed the Updated Final Safety Analysis Report (UFSAR) and other selected design basis documents to identify those areas susceptible to external flooding;
- performed a walkdown of the 569 foot elevation of the Turbine Building, the Lake Screen House (including the ESW pump rooms), and general plant yard to evaluate whether appropriate flood protection controls were being maintained;
- reviewed selected station operating procedures used to identify and mitigate flooding events; and
- interviewed selected operating and engineering staff regarding external flooding protection controls.

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

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors observed the licensee perform an inspection of the following heat exchanger:

- 1-HE-15E Unit 1 East Component Cooling Water Heat Exchanger

The inspectors selected this heat exchanger to inspect because the component cooling water system was identified as risk significant in the licensee's risk assessment and the heat exchanger is required to support the operability of other risk significant safety-related equipment. During this inspection, the inspectors observed the as-found condition of the cooler and verified that no deficiencies existed that would mask degraded performance. In addition, the inspectors observed that no conditions were present that would indicate a potential for common cause problems. The inspectors discussed the as-found condition as well as the historical performance of the cooler with engineering department personnel and reviewed applicable documents and procedures.

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 heat sink performance related issues documented in selected CRs.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspector conducted a review of the licensee's inservice inspection program for monitoring degradation of the reactor coolant system (RCS) boundary and risk significant piping system boundaries. Specifically, the inspectors conducted a record review of the following examinations:

WELD NUMBER	CONFIGURATION	NON-DESTRUCTIVE EXAMINATION TYPE
1-RH-27-05S	Pipe to Elbow	UT & PT
1-CTS-2-13S	Pipe to Elbow	UT & PT
1-SI-23-17F	Pipe to Valve	UT & PT
1-FW-13-09S	Elbow to Pipe	UT & MT

These examinations were evaluated for compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspector also reviewed inservice inspection procedures, equipment certifications, personnel certifications, and NIS-2 forms for Code repairs performed during the last outage to confirm that ASME Code requirements were met. The inspector also conducted a review of the radiographs of 1-RC-131 (Job Order 1180005-09) 3/4-inch pipe replacement.

A sample of inservice inspection related problems documented in the licensee's corrective action program was also reviewed to assess conformance with 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action" requirements. In addition, the inspector determined that operating experience was correctly assessed for applicability by the inservice inspection group.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors evaluated the licensee's implementation of 10 CFR 50.65 (the Maintenance Rule). The inspectors assessed: (1) functional scoping in accordance with the Maintenance Rule, (2) characterization of system functional failures, (3) safety significance classification, (4) 10 CFR 50.65 (a)(1) or (a)(2) classification for system functions, and (5) performance criteria for systems classified as (a)(2) or goals and corrective actions for systems classified as (a)(1). The inspectors reviewed the following risk-significant system:

Mitigating Systems Cornerstone

- Diesel Generator Ventilation System

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 Rule related issues that were documented in selected CRs.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

Initiating Events Cornerstone

- Unit 1 Main Generator Output Breaker 1-52-K1 Replacement

Mitigating Systems Cornerstone

- Unit 2 West Motor Driven AFW Pump
- Unit 1 Turbine Driven AFW Pump
- Unit 2 East and Unit 1 West ESW Pump Replacements

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

b. Findings

The inspectors identified one finding of very low safety significance (Green) associated with a self-revealed failure of the Unit 1 West ESW pump. This finding was associated with foreign material exclusion controls. The finding was determined to be a violation of NRC requirements and was dispositioned as a Non-Cited Violation.

Description

On June 24, 2002, the Unit 1 control room operators noted that the performance of the Unit 1 West ESW pump had significantly decreased. Specifically, the control room operators noted that at 5:30 p.m. the Unit 1 West ESW pump header pressure dropped from approximately 64 pounds-per-square-inch gauge (psig) to 50 psig, resulting in actuation of the associated low ESW header pressure alarm. The operators also noted that total pump flow decreased from approximately 8200 gallons-per-minute (gpm) to 7200 gpm. Based on these indications, the operators declared the Unit 1 West ESW pump inoperable and initiated CR 02175037. During follow-up testing, the licensee determined that the pump developed head at the reference inservice testing flowrate was 55.6 pounds-per-square-inch differential (psid), which was significantly less than the low action limit of 63.8 psid. The Unit 1 West ESW pump had just been replaced and satisfactorily tested 3 days prior to this event. Following the event, the licensee replaced the ESW pump and returned the train to an operable status on June 26, 2002. During pump replacement activities, the licensee discovered that plastic barrier tape had been ingested by the pump and had become tightly wound around the impeller. The barrier tape was of a type used in the screenhouse to cordon off hazardous areas. The inspectors determined that the loss of foreign material controls that resulted in the Unit 1 West ESW pump ingesting a significant quantity of plastic barrier tape was a violation of NRC requirements.

Analysis

The inspectors evaluated this failure to establish appropriate foreign material controls in the vicinity of the Unit 1 West ESW pump using the Significance Determination Process (SDP). The inspectors determined that this issue had a credible impact on safety and was more than a minor concern. Specifically, ingestion of foreign material by the Unit 1 West ESW pump degraded pump performance and rendered the pump inoperable, which affected the reliability and capability of the ESW system. The safety function of the ESW system is to provide sufficient cooling capacity for continued operation of safety-related equipment during normal and accident conditions. Consequently, the inspectors determined that this issue affected the objectives of the mitigating systems cornerstone. The inspectors concluded that this issue did not result in an actual loss of the safety function of a single train of ESW for greater than the TS allowed outage time. Additionally, because of the continued availability of ESW capability from both of the Unit 2 ESW trains and the Unit 1 East ESW train, the inspectors concluded that the foreign material ingestion did not result in an actual loss of the ESW system safety function. Consequently, 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. Section 4.1.1. of Plant Manager's Instruction (PMI) 2220, "Foreign Material Exclusion," a procedure prescribing activities affecting quality, stated, in part, that appropriate work controls shall be established for areas adjacent to open systems or components to control foreign material which may be generated from any facet of plant work. Contrary to the above, the licensee failed to establish appropriate work controls to control foreign material for areas adjacent to the Unit 1 West ESW pump in accordance with the requirements of PMI-2220, Section 4.1.1. Specifically, appropriate work controls were not established to prevent the intrusion of the foreign material that was discovered in the impeller of the Unit 1 West ESW pump on June 25, 2002. The foreign material significantly degraded pump performance and resulted in the inoperability of the pump. This is a violation of 10 CFR 50, Appendix B, Criterion V. 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 50-315-02-03-01(DRP)). The licensee entered this violation into its corrective action program as CR 02176058 and CR 02175037.

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

.1 Containment Isolation Valve Alignment Error During Local Leak Rate Testing

a. Inspection Scope

On January 26, 2002, during reactor core offload for refueling outage U2C13, an operator performing a valve lineup for local leak rate testing on the nitrogen to pressurizer relief tank containment isolation valve (2-N-159) incorrectly opened the instrument root shutoff containment isolation valve (2-GPX-301-V1) and removed the "Do Not Operate" tag from the valve without verifying the required position of the valve for testing. This created a direct access path from the containment atmosphere to the outside atmosphere and violated the TS requirement for refueling integrity. This event was selected for review to evaluate the operator human performance errors that caused the event. The inspectors interviewed operations and licensing department personnel and reviewed the licensee's apparent cause evaluation, licensee event report, applicable procedures, and the CR to understand the details of the event. The licensee event report is discussed in Section 4OA3.1 of this report.

b. Findings

A finding of very low safety significance (Green) was self-revealed and is tied to human performance. An operator incorrectly opened an instrument root shutoff containment isolation valve and removed the "Do Not Operate" tag from the valve without verifying the required position of the valve for local leak rate testing. This resulted in an inoperable containment penetration during refueling and resulted in the plant being in a higher risk configuration than that planned by the licensee. This finding was dispositioned as a Non-Cited Violation.

Description

On January 26, 2002, while performing a valve lineup for local leak rate testing on the nitrogen to pressurizer relief tank containment penetration, an operator incorrectly opened 2-GPX-301-V1 and removed the "Do Not Operate" tag from the valve without verifying the required position of the valve for testing. The operator who had been involved with previous testing that required the lifting of similar tags from instrument root valves failed to self check and verify the required position for this valve in the test procedure. As a result, the licensee failed to have the containment penetration isolated for approximately 10 hours during core alterations. A different operator who performed the restoration for the valve lineup identified that 2-GPX-301-V1 was open and notified the control room. The licensee suspended core alterations and closed 2-GPX-301-V1.

The licensee's original outage risk evaluation reflected a "yellow" risk configuration (i.e., acceptable but reduced level of defense) by maintaining all of the containment penetrations closed for refueling integrity. By not maintaining 2-GPX-301-V1 closed, the licensee inadvertently entered a higher "red" risk configuration (i.e., less than minimum acceptable level of defense). The licensee's plant shutdown safety and risk management procedure, Plant Manager's Procedure (PMP) 4100-SDR-001, "Plant Shutdown Safety and Risk Management," did not permit voluntary entry into a "red" risk configuration and required the implementation of additional risk management actions to maintain an adequate level of defense for an emergent entry into a "red" risk configuration.

Analysis

The inspectors determined that this issue had a credible impact on safety because the licensee failed to have the containment penetration isolated as required by the TSs and the valve was not in the correct position to fulfill its design safety function. The inspectors utilized the event information in conjunction with Appendix G, “Shutdown Operations Significance Determination Process,” of Manual Chapter 0609, Table T-1, “Pressurized Water Reactor (PWR) Refueling Operation Reactor Coolant System (RCS) Level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours AND Inventory in the Pressurizer.” This self-revealed issue was determined to be of very low significance (Green) by the significance determination process because (1) the issue did not increase the likelihood of a loss of primary coolant system inventory; (2) the issue did not degrade the licensee’s ability to terminate a leak path or add RCS inventory when needed; and (3) the issue did not degrade the licensee’s ability to recover decay heat removal once lost. Although this issue affected the integrity of the reactor containment during core alterations, the inspectors concluded that because the small diameter penetration would be a very small leakage path, this issue was of very low safety significance.

Enforcement

Technical Specification 3.9.4.c, states, in part, that each containment penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed by either an isolation valve, blind flange, manual valve, or equivalent during core alterations or movement of irradiated fuel within the containment. With the above requirement not satisfied, the TS requires that the licensee immediately suspend all operations involving core alterations or movement of irradiated fuel in the Containment Building. Contrary to the above, on January 26, 2002, the licensee failed to have containment isolation valve 2-GPX-301-V1 closed to isolate the nitrogen to pressurizer relief tank containment penetration during core alterations. How long did this condition exist? This is a violation of TS 3.9.4.c. 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 50-316-02-03-02(DRP)). The licensee entered this violation into its corrective action program as CR 02027006.

.2 Pressurizer Power Operated Relief Valves (PORVs) Inoperable Due to Mis-positioned Control Switches

a. Inspection Scope

On February 19, 2002, during stroke time testing of the Unit 1 pressurizer PORV block valves, an operator noticed that the control switches for pressurizer PORVs 1-NRV-152 and 1-NRV-153 were not correctly positioned to the "auto" position. This rendered the two PORVs inoperable. Subsequent investigation identified that the control switches had been mis-positioned since February 6, 2002. This event was selected for review to evaluate the operator human performance errors that caused the event. The inspectors interviewed operations and licensing department personnel and reviewed the licensee's apparent cause evaluation, licensee event report, applicable procedures, and the CR to understand the details of the event. The licensee event report is discussed in Section 4OA3.2 of this report.

b. Findings

A finding of very low safety significance (Green) was self-revealed and is tied to human performance. An operator incorrectly positioned the control switches for pressurizer PORVs 1-NRV-152 and 1-NRV-153, rendering the valves unavailable for automatic pressure control. This finding was dispositioned as a Non-Cited Violation.

Description

On February 19, 2002, during stroke time testing of Unit 1 pressurizer PORV block valves 1-NMO-152 and 1-NMO-153, an operator noticed that the control switches for 1-NRV-152 and 1-NRV-153 were positioned slightly to the left of the "auto" position. Upon completion of the testing, the PORV control switches were restored to the "auto" position and the PORVs declared operable. The licensee's apparent cause evaluation concluded that operators failed to verify that the control switches were fully engaged in the "auto" position after previous testing on February 6, 2002. The inspectors also noted that this was not an isolated occurrence. During the licensee's extent of condition review, the licensee identified the same condition with Unit 2 pressurizer PORVs 2-NRV-152 and 2-NRV-153. However, that condition was identified while the valves were out-of-service during a refueling outage, and the valves were not being relied on for low temperature over-pressure protection of the RCS.

Analysis

The inspectors assessed this event using the SDP. The inspectors determined that this issue had a credible impact on safety because the two PORVs were not capable of automatically controlling RCS pressure below the setting of the pressurizer code safety valves, thereby reducing challenges to these valves. At the time of this event, the third pressurizer PORV (1-NRV-151) was already unavailable (automatic function only) with its manual isolation valve closed due to excessive seat leakage. Therefore the automatic function of all three PORVs was disabled. Although all three PORVs were not capable of automatic operation, the valves were still capable of manual operation to mitigate a steam generator tube rupture accident or as an alternate means of decay heat removal during plant shutdown. The inspectors concluded that this issue affected the operability of the pressurizer PORVs, which are barrier integrity components under the SDP designed to maintain the integrity of the RCS. The inspectors performed a Phase 2 SDP analysis for this finding using the following assumptions: (1) manual operation of the PORVs for primary heat removal using the feed and bleed safety function was not affected; therefore, the inspectors only evaluated the Anticipated Transients Without Scram (ATWS) initiator which considered the primary relief safety function; (2) the duration of the performance deficiency was 13 days; and (3) operator action to manually actuate the failed automatic function of the PORVs was credited. Results of the Phase 2 ATWS worksheet determined that only one accident sequence was affected and resulted in this issue being characterized as having very low safety significance. 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 (LERF) because the accident sequence result was less than 1E-7 per year.

Enforcement

Technical Specification 3.4.11 states that three PORVs and their associated block valves shall be operable in Modes 1, 2, and 3. Technical Specification 3.4.11.c states that with two PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORVs to operable status or close the associated block valves and remove power from the block valves; restore at least one of the inoperable PORVs to operable status within the following 72 hours or be in Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours. Contrary to the above, on February 19, 2002, with Unit 1 in Mode 1 and two PORVs inoperable due to causes other than excessive seat leakage, the licensee failed to restore at least one of the inoperable PORVs to operable status within the following 72 hours or be in Hot Standby within the next 6 hours and in Hot Shutdown within the following 6 hours. This is a violation of TS 3.4.11.c. 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 50-315-02-03-03(DRP)). The licensee entered this violation into its corrective action program as CR 02050022.

.3 Failure of Lower Containment Airlock Door Interlock and Failure to Follow Instructions Resulted in Inadvertent Opening of Both Airlock Doors

a. Inspection Scope

During the removal of plant equipment from the Unit 2 Containment Building on January 23, 2001, the mechanical interlock on the personnel airlock, which is designed to prevent opening both inner and outer lower containment airlock doors at the same time, malfunctioned and personnel opening the inner airlock door challenged the interlock by not verifying the outer door was closed prior to opening the inner door. This event was selected for review to evaluate the human performance errors that caused the event. The inspectors interviewed licensing department personnel and reviewed the licensee’s root cause evaluation, licensee event report, applicable procedures, and the CR to understand the details of the event. The licensee event report is discussed in Section 4OA3.4 of this report.

b. Findings

A finding of very low safety significance (Green) was self-revealed. The mechanical interlock on the Unit 2 lower containment personnel airlock malfunctioned and personnel opening the inner airlock door challenged the interlock by not verifying the outer door was closed prior to opening the inner door. This created a direct access path from the containment atmosphere to the outside atmosphere. This finding was dispositioned as a Non-Cited Violation.

Description

On January 23, 2001, the mechanical interlock on the Unit 2 lower containment personnel airlock malfunctioned as personnel were transferring equipment out of the Containment Building. Unit 2 had just entered Mode 3 to resolve a problem with the rod control system and two work crews were exiting the Containment Building. Personnel opening the inner airlock door failed to follow posted instructions to verify the outer door was closed prior to opening the inner door. The mechanical interlock was challenged and failed allowing both airlock doors to be open at the same time. The licensee had posted instructions on the outer airlock door security gate for proper operation of the airlock. Personnel using the airlock were expected to verify the readiness of the airlock doors for opening by using the door position indicator lights. In this event, personnel failed to verify that the outer door's indicating light was illuminated prior to opening the inner door. Because the interlock malfunctioned, both airlock doors were able to be opened at the same time. A rapid containment pressure change caused 10 lower ice condenser doors to go open, resulting in an inoperable ice condenser for a brief period of time. The lower ice condenser doors were subsequently reseated and the overall impact on the ice condenser was minimal.

A review of interlock maintenance history back to the late 1980s found interlock failures to be recurring. The licensee identified this adverse trend in 1999 and documented it in its corrective action program. The licensee had accepted the interlock failure rate as meeting standards and expectations for that time period. The root cause for the recurring interlock failures is that the interlock mechanism was vulnerable to slipping out of adjustment with door use. The specific failure involved gradual loosening of the setscrews that held the interlock gears in place on the interlock gear shafts. The frequency of airlock door use was greater than originally expected when the plant was designed. Preventive maintenance was not effective to maintain the interlock in good working order. Past corrective actions and oversight had not improved the interlock failure rate. The licensee noted in its root cause evaluation that other plants have upgraded their containment airlock interlocks, including installation of gears that use keyways rather than setscrews, and have not had significant interlock problems after the upgrades.

Analysis

The inspectors determined that this issue had a credible impact on safety because the licensee failed to have at least one airlock door closed while the containment airlock was inoperable as required by the TSs and the resultant rapid containment pressure change affected the operability of the ice condenser. The inspectors reviewed the guidance in IMC 0609, Appendix H, "Containment Integrity SDP," and determined the finding was a Type "B" finding. Type "B" findings have no impact on the determination of Core Damage Frequency (CDF) and therefore they are not processed through the CDF based SDP. These findings, however, are potentially important to LERF determinations. The initial screening of the finding determined that the issue was potentially risk significant based on the affect on containment and ice condenser integrity. The issue was referred to the regional Senior Reactor Analyst (SRA) for further review. The analyst determined that the T/2 approach for fault exposure was not appropriate as the containment airlock doors were not discovered in the open position. The analyst therefore used the 5 seconds duration of time that the doors were actually opened, as each entry through the containment airlock is a deliberate, monitored activity (rather than a random event), and the licensee would be expected to identify the problem (both containment airlock doors opened simultaneously) as soon as it occurs. In determining the actual risk significance the SRA, with the assistance of the headquarters containment risk analyst, utilized the LERF methodology identified in Appendix H for Type "B" findings. Utilizing this approach with actual plant specific probabilistic risk assessment values, the issue was determined to be of very low safety significance.

Enforcement

Technical Specification 3.6.1.3 requires, in part, that each containment air lock shall be operable with both containment airlock doors closed, except when the airlock is being used for normal transit entry and exit through containment, then at least one airlock door shall be closed. With an air lock inoperable, the TS requires that at least one door be maintained closed. This TS requirement is applicable with Unit 2 in Modes 1, 2, 3 and 4. Contrary to the above, on January 23, 2001, the licensee failed to have at least one airlock door closed while the containment airlock was inoperable with Unit 2 in Mode 3. This is a violation of TS 3.6.1.3. 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 50-316-02-03-04(DRP)). The licensee entered this violation into its corrective action program as CR 01023054.

.4 Unit 1 Power Reduction to Support Repairs to the Unit 1 Main Generator K1 Breaker Disconnect

a. Inspection Scope

On April 25, 2002, during maintenance activities on Unit 1 main generator output breaker K1, maintenance workers damaged the breaker disconnects. At the time, the breaker disconnects were opened to provide electrical safety isolation for maintenance on the K1 breaker. In order to facilitate repairs to the disconnect, the licensee reduced Unit 1 power from 100 percent to approximately 8 percent to allow removal of the main generator from service. The inspectors observed portions of the power reduction and assessed the operator response to this event.

b. Findings

No findings of significance were identified.

.5 Unit 1 Reactor Trip and Restart Following Loss of Main Feedwater Pump Vacuum

a. Inspection Scope

On June 14, 2002, control room operators manually tripped Unit 1 in response to a low vacuum automatic trip of the East main feedwater pump. Immediately prior to the main feedwater pump trip, the operators started circulating water pump13, which had been idled since October 2001. The licensee determined that an influx of zebra mussel shells and debris following the circulating water pump start caused blockage of the main feedwater pump condensers. The licensee restarted Unit 1 on June 17, 2002, after cleaning out the main feedwater pump condenser water boxes. The inspectors assessed control room operator performance immediately following the reactor trip, reviewed the post trip report, and observed portions of the reactor restart activities.

b. Findings

No findings of significance were identified.

.6 Unit 2 Station Battery AB Cell Cracking Operator Response

a. Inspection Scope

On April 23, 2002, operations personnel identified indications of cracking on battery cell 31 of the 2AB station battery. The licensee declared the 2AB battery inoperable and entered the action statement for TS 3.8.2.3, which required the battery to be returned to an operable status within 2 hours or the plant to be placed in at least Hot Standby within the next 6 hours. The licensee replaced the cracked cell and returned the battery to an operable status within the allowed outage time. On April 4, 2002, the licensee had requested and was granted an NOED for similar cell case cracking on the 2AB battery. (Refer to Section 4OA3.6 for the inspectors' review the NOED.) Because of the repetitive occurrence of this issue and the short allowed outage time for an inoperable station battery, the inspectors assessed the licensee's response to this issue.

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.

Barrier Integrity Cornerstone

- CR 02115002 Unit 1 Ice Basket 24-1-7 As-found Weight Below TS Requirements

Mitigating Systems Cornerstone

- CR 02136014 2-FW-160, West Motor Driven AFW Pump Emergency Leakoff Check Valve Leaked By During the Performance of Test 02-OHP-4030.STP.017E
- CR 02109003 Non-seismic Scaffolding Built in the Vicinity of 2AB DG [Diesel Generator] and 2CD DG Components

- 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
- CR 02137063 — Unit 2 Steam Stop Valve 2-MRV-220 Detent Bar/Guide Rod Bushing Has Fallen Out

The inspectors also reviewed the licensee's justification for not correcting existing degraded and nonconforming conditions during refueling outage U1C18 consistent with the timeliness guidance contained in Generic Letter 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," Revision 1.

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 structures, systems, and components 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 requirements associated with the following scheduled maintenance activities:

Mitigating Systems Cornerstone

- Job Order C0051164, "Replace 1-BATT-CD During Year 2002 Outage"
- 1-DCP [Design Change Procedure] 4504, "Replace Reserve Auxiliary Transformers 101AB and 102CD with Load Tap Changing Transformers"
- 1-DCP 4504, "Install New Undervoltage Protection Relays"
- Job Order 02093039, "Unit 2AB Station Battery Cell 46 Replacement"
- Unit 1 Turbine Driven AFW Pump Maintenance (Multiple Job Orders)

The inspectors verified that test methodology and acceptance criteria were appropriate for the scope of work performed. Documented test data was reviewed to verify that the testing was complete and that the equipment was able to perform the intended safety functions.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Unit 1 Refueling Outage (U1C18)

a. Inspection Scope

The inspectors evaluated the licensee's conduct of Unit 1 refueling outage activities to assess the licensee's control of plant configuration and management of shutdown risk. The inspectors reviewed configuration management to verify that the licensee maintained defense-in-depth commensurate with the shutdown risk plan; reviewed major outage work activities to ensure that correct system lineups were maintained for key mitigating systems; and observed refueling activities to verify that fuel handling operations were performed in accordance with the TSs and approved procedures. Other major outage activities evaluated included the licensee's control of the following:

- Containment penetrations in accordance with the TSs
- Systems, structures, and components (SSCs) which could cause unexpected reactivity changes
- Flow paths, configurations, and alternate means for RCS inventory addition and control of SSCs which could cause a loss of inventory
- RCS pressure, level, and temperature instrumentation
- Spent fuel pool cooling during and after core offload
- Switchyard activities and the configuration of electrical power systems in accordance with the TSs and shutdown risk plan
- SSCs required for decay heat removal

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

- Verification that RCS boundary leakage requirements were met prior to entry into Mode 4 (Cold Shutdown) and subsequent operational mode changes
- Verification that containment integrity was established prior to entry into Mode 4
- Inspection of the Containment Building to assess material condition and search for loose debris, which if present could be transported to the containment recirculation sumps and cause restriction of flow to the emergency core cooling system (ECCS) pump suctions during loss-of-coolant accident conditions
- Verification that the material condition of the Containment Building ECCS recirculation sumps met the requirements of the TSs and was consistent with the design basis
- Observation and review of reactor physics testing to verify that core operating limit parameters were consistent with the core design so that the fuel cladding barrier would not be challenged

The inspectors interviewed operations, engineering, work control, radiological protection, and maintenance department personnel and reviewed selected procedures and documents.

In addition, the inspectors reviewed the issues that the licensee entered into the 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 refueling outage issues documented in selected CRs.

b. Findings

No findings of significance were identified.

.2 Unit 2 Forced Outage

a. Inspection Scope

On May 25, 2002, the licensee performed a reactor shutdown to isolate a steam leak and replace one of the main turbine reheat stop valves. The steam leak on the "C" low pressure turbine was due to a cracked weld (approximately 90 degrees around the circumference) on a flange to the right reheat stop valve (2-OME-92). The unit was ramped down at 20 percent per hour and the reactor was tripped at 7:51 p.m. from 15 percent power. The main steam stop valves were shut after the trip to isolate the steam leak, causing a loss of the normal heat sink. Operators maintained temperature using feed and bleed with steam generator blowdown and cycling the steam generator PORVs. Following the reheat stop valve replacement, the licensee synchronized the unit to the grid on June 2, 2002.

The inspectors evaluated the licensee's conduct of forced outage activities to assess the licensee's control of plant configuration and risk management actions. The inspectors reviewed the cause for the weld failure as well as the extent of condition of other reheat stop valve welds. The inspectors observed portions of the restart activities to verify that requirements of the TSs and administrative procedure requirements were met prior to changing operational modes or plant configurations.

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

- 12 MHP 4030-10-03, "Ice Condenser Lower Inlet Door Surveillance"

Mitigating Systems Cornerstone

- 01 OHP 4030-108-008R, Attachment 8, "Accumulator Check Valve Test"
- 01 OHP 4030-STP-017R, "Auxiliary Feedwater Pump Time Response Test"
- 02 OHP 4030-214-029, Attachment 1, "PPC [Plant Process Computer] Derived Reactor Thermal Power Evaluation"
- 02 OHP 4030-214-029, Attachment 4," Power Range NI [Nuclear Instruments] Adjustments"
- PMI 5070, "Inservice Testing," [Valve Stroke Testing of 1-MCM-221]
- 12 IHP 4030-082-003, "AB, CD and N Train Battery Discharge Test and 18 Month Surveillance Requirements"
- 01 OHP 4030.001.002, "Containment Inspection Tours"

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. The inspectors also reviewed CRs concerning surveillance testing activities to verify that identified problems were appropriately characterized.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

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

- 12-TM-01-23-R0, "Install Splash Shield on the Unit 1 and Unit 2 AFW Pumps"

The temporary modification installed a bearing housing port shield and shaft flinger to prevent water intrusion from water spray and shaft packing leakage. The inspectors verified that configuration control of the modification was correct by reviewing design modification documents and confirmed that appropriate post-installation testing was accomplished. The inspectors reviewed the design modification documents and the 10 CFR 50.59 evaluation against the applicable portions of the UFSAR.

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 second quarter unannounced emergency planning drill that was conducted in the licensee's control room simulator and emergency response facilities on April 16, 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 and Radiological Boundary Verifications

a. Inspection Scope

The inspector conducted walkdowns of selected radiologically controlled areas to verify the adequacy of radiological boundaries and postings. The inspector reviewed both the administrative controls specified in radiation work permits (RWPs) and the physical controls (radiological postings and boundaries) for access to these areas, and assessed worker adherence to these controls through direct observation. Specifically, the inspector walked down several radiologically significant work area boundaries (high and locked high radiation areas) in the Unit 1 and Unit 2 Auxiliary Building and in the Unit 1 Containment Building, and performed confirmatory radiation measurements in the Auxiliary Building to verify that these areas and selected radiation areas were properly posted and controlled in accordance with 10 CFR Part 20 and the licensee's TSs. Additionally, the inspector reviewed two incidents that involved locked high radiation area access control problems that occurred in April and May 2002, assessed performance indicator applicability for the incidents and the adequacy of the licensee's problem identification, extent of condition evaluation and corrective actions for each event. (Refer to Section 4OA7).

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Radiation Dose Goals and Trending

a. Inspection Scope

The inspector reviewed the licensee's outage exposure data for the last several refueling outages to establish its prior performance relative to the industry. Job specific and cumulative exposure data and exposure trends for the first three weeks of the scheduled four week Unit 1 Spring 2002 refueling outage (U1C18) were reviewed to assess the licensee's current dose performance compared to pre-outage exposure goals and projections. The inspector also reviewed the licensee's dose forecasting practices for selected radiologically significant jobs (those with dose expenditure projected greater than approximately 3.5 rem) which were being performed during the outage. The review was performed to determine if adequate technical bases for outage dose estimates existed, and to determine if prior outage experiences and job scope and resource estimates were accurate and used properly to establish reasonable dose projections. Additionally, the inspector reviewed the effectiveness of the RP organization's exposure tracking for the outage to verify that the licensee could timely identify problems with its exposure performance and take actions to address identified deficiencies.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspector reviewed the licensee's ALARA program procedures which included a recently implemented ALARA control procedure for radiological risk significant work. Also, several U1C18 ALARA plans were evaluated to verify consistency with the procedures and to assess their overall adequacy relative to prior licensee practices and industry standards. Specifically, the inspector selected the following outage jobs that were projected to accrue in excess of 5 rem, and assessed the adequacy of the radiological controls and the work planning developed for each:

- Scaffold Erection/Removal in Containment (RWP 021136)
- Steam Generator Manway and Diaphragm Activities (RWP 021140)
- Valve Maintenance and Repair Activities in Containment (RWP 021139)
- Insulation Removal, Reinstallation and Modification in Containment (RWP 021134)
- Steam Generator Primary Work - Platform Activities (RWP 021141)

The inspector reviewed the RWP and the ALARA plan developed for each job, and assessed the radiological engineering controls and other dose mitigation techniques specified in these documents to verify that the plans were completed in compliance with procedures, included appropriate controls to reduce dose, and were sufficiently comprehensive. These documents were also reviewed to determine if job history files, lessons the licensee learned from its recent Unit 2 outage, and industry operating experiences were adequately integrated into each work package. Additionally, the inspector discussed ALARA planning with several RP staff to verify that adequate interface between contractors, station work groups, and RP ALARA staff occurred during job planning.

b. Findings

No findings of significance were identified.

.3 Implementation of ALARA Controls and Radiological Oversight of Work

a. Inspection Scope

The inspector selected the following high exposure or high radiation area jobs conducted during the outage and reviewed the execution of the ALARA program:

- Reactor Head Die Penetrant Testing (RWP 021152)
- Steam Generator Platform/Manway Activities (RWP 021140 and RWP 021141)
- Installation of Temporary Shielding in Containment (RWP 021119)
- Valve Activities in Containment (RWP 021139)
- Scaffold Erection in Containment (RWP 021136)

The inspector discussed the radiological performance for each activity with RP ALARA staff and reactor head die penetrant testing and various steam generator platform activities were observed. Also, total effective dose equivalent (TEDE) ALARA evaluations completed for these activities and for other outage work activities were assessed for technical adequacy. Work in progress reports and radiological survey data for these and other selected jobs, as applicable, were also reviewed to assess their adequacy and consistency with the licensee's procedures. The pre-job briefings for head die penetrant testing and for steam generator manway installation were attended to verify that the work activities were adequately planned and that radiological information was exchanged effectively. The inspector evaluated the licensee's radiological engineering controls utilized at selected work locations to determine if the controls were consistent with those specified in the ALARA plans. Additionally, the inspector reviewed a radiological intake incident that occurred during steam generator eddy current testing, and assessed the licensee's response to the incident and the adequacy of the RP staff's evaluation of the problem and its corrective actions. (Refer to Section 20S2.7)

b. Findings

No findings of significance were identified.

.4 Verification of Exposure Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspector reviewed the methodology and specific assumptions used by the ALARA group to develop U1C18 dose estimates, and compared collective outage and individual job dose performance for the first three weeks of the outage to assess dose performance and determine the accuracy of pre-outage projections. The inspector selectively reviewed job dose history files and dose reduction techniques applied to selected jobs to verify that previous problems had been adequately addressed. In particular, the inspector reviewed those jobs which accrued greater than 5 rem and which the dose expenditure significantly differed from original dose projections, to determine whether revised dose estimates were justified and could not reasonably have been accurately projected initially. The inspector also reviewed the process used to revise dose estimates and capture lessons learned to verify compliance with the licensee's ALARA procedure. As of May 24, 2002, the licensee had recorded an outage exposure of approximately 95 rem compared to its estimate of about 105 rem for that stage of the outage, and projected that its revised outage dose estimate of approximately 130 rem would be met. Selected work in progress reports were examined to evaluate the licensee's ability to assess the effectiveness of a job, to execute its ALARA plan, and to institute changes in work plans, if warranted. The licensee's exposure tracking system was also reviewed to determine if the level of exposure tracking detail, exposure report timeliness, and report distribution were sufficient to support the control of outage exposures.

b. Findings

No findings of significance were identified.

.5 Source Term Reduction and Control

b. Inspection Scope

The inspector reviewed some of the exposure reduction initiatives taken for the outage such as flushing and installation of temporary shielding. Also, the licensee's water chemistry control program implemented during the Unit 1 shutdown was selectively evaluated to determine its impact on outage source term reduction. The evaluation was conducted to determine whether the shutdown chemistry program was implemented consistent with station procedures and industry practices. In particular, the effectiveness of a new CRUD burst chemistry initiative that involved the use of a deborating demineralizer loaded with powdered resins to supplement the mixed bed demineralizer system was reviewed by the inspector.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed the results of an ALARA group root cause analysis which assessed work planning and work execution problems experienced during the licensee's previous outage in February 2002. The inspector also reviewed outage related Performance Assurance Department field observations, RP program related CRs generated during the outage and rapid event response and apparent cause evaluation reports related to the access control problems discussed in Section 4OA7 and the intake incident described in Section 2OS2.7. This review was performed to verify that the licensee adequately identified individual problems and trends, determined contributing causes and extent of condition, and developed appropriate corrective actions.

b. Findings

No findings of significance were identified.

.7 Review of a Radiological Intake Incident During Steam Generator Eddy Current Testing

a. Inspection Scope

The inspector reviewed the circumstances associated with a radiological intake incident that occurred during the Unit 1 refueling outage on May 18, 2002, associated with steam generator tube eddy current testing. Specifically, the inspector reviewed the licensee's preliminary rapid event response report, the ALARA plan and RWP that governed the work activity, and discussed the incident with RP staff. The inspector also independently calculated the committed effective dose equivalent (CEDE) assigned to the workers to verify the accuracy of the licensee's assessments. Additionally, the inspector independently evaluated the potential for an exposure in excess of regulatory limits based on the radiological conditions present.

b. Findings

A Green finding and an associated Non-Cited Violation were identified for the failure to use all intended radiological engineering controls during steam generator eddy current testing to control contamination and airborne radioactivity. In addition, the finding is tied to human performance.

On May 18, 2002, two contract workers involved in positioning steam generator eddy current test equipment in the number 11 steam generator were contaminated while performing the task. Shortly thereafter and prior to recognizing the problem, two other contract workers were contaminated as they cleaned-up the steam generator platform areas that were just vacated by the first two workers.

The two workers positioning eddy current equipment relocated a robotic device (termed ROGER) from the generator's hot leg to the cold leg. The ROGER was used to position and maneuver eddy current test probes within the steam generators. The equipment was highly contaminated (1Rad/hour of removable contamination/100 square centimeters or about 20 million disintegrations per minute) and necessitated proper radiological engineering controls to prevent airborne radioactivity. The ALARA staff's evaluation showed that respiratory protection equipment was not warranted for the work activity provided the necessary radiological controls were in-place, so the workers wore only face shields to reduce the potential for facial contamination. The radiological engineering controls to be used for the work included a high efficiency particulate air (HEPA) filtered ventilation system installed on the opposite generator manway leg and the spraying/wiping-down of all items removed from the generator. While the latter controls were not specified in either the RWP or the ALARA plan due to an oversight, these specific controls were communicated to the work crews during pre-job briefings and had also been a standard industry practice for any equipment removed from the generators. Despite these instructions, the ROGER was not wiped-down or wetted as it was removed from the generator hot leg. The licensee's event response found that the workers reasoned that since the ROGER was being relocated from one leg of the generator to another, it was not necessary to wipe or spray the equipment down. The problem compounded because the radiation protection technician assigned to cover the work activity was providing assistance on another generator platform at the time the ROGER was removed from the hot leg. Since the relocation and positioning of the equipment was physically intensive and the equipment was handled roughly, the licensee concluded that contamination dried-out and was jarred loose and became airborne. Once airborne, contaminated dust-like particles became an ingestion and inhalation hazard unbeknownst to the two workers.

Installation of the ROGER in the cold leg was completed near the end of the work shift on May 18 and the two workers left the platform without recognizing the radiological hazard that was created. The workers alarmed the personnel contamination monitors as they attempted to leave the Containment Building. The two other workers that subsequently cleaned the platforms also alarmed the monitors as they left the work area. Positive nasal smears and/or facial contamination prompted whole body count analyses of all four workers and each showed small intakes of radioactive material. Further evaluation by the licensee disclosed intakes through both inhalation and ingestion pathways with the maximum dose calculated at about 60 mrem CEDE.

This issue had an actual impact on radiological safety and if not corrected would become a more significant concern should other radiological engineering controls not be implemented as intended. Also, the issue involved unintended dose (from intakes) which resulted from the failure to implement the radiological controls required by regulatory requirements and those that were intended by the RWP/ALARA plan for the work activity. Therefore, the issue represents a finding which was evaluated using the SDP for the occupational radiation safety cornerstone. Since radiation exposures to involved workers were low relative to regulatory limits and because radiological conditions (removable contamination levels) were not of a magnitude sufficient to create a substantial potential for an overexposure (as evaluated by the inspector), the issue was determined to be of very low safety significance.

10 CFR 20.1701 requires that the licensee use, to the extent practical, process or engineering controls to control the concentration of radioactive material in air. The failure to implement all intended radiological engineering controls communicated to the work crew is a violation of that regulatory requirement. However, because the

licensee documented this issue in its corrective action program (CR 02139007) and because the violation is of very low safety significance, it is being treated as a Non-Cited Violation (NCV 50-315-02-03-05(DRS)).

Cornerstone: Public Radiation Safety

2PS2 Radioactive Waste (Radwaste) Processing and Transportation (71122.02)

.1 Walkdowns of Radwaste Systems

a. Inspection Scope

The inspector reviewed the liquid and solid radioactive waste system descriptions in the UFSAR and the annual radiological effluent release reports for calendar years 2000 and 2001, for information on the types and amounts of radwaste disposed. The inspector walked down the liquid and solid radwaste processing systems located in the Auxiliary Building, including the abandoned in-place radwaste evaporator system, to verify that the systems that remained in-use and operable were consistent with the descriptions in the UFSAR and the Process Control Program (PCP) and to assess their material condition. These walkdowns were also performed to determine if radiological postings were proper and if radiological access was controlled into these areas in accordance with the requirements in 10 CFR Part 20 and the licensee's TSs. The inspector reviewed the current processes for transferring radwaste resins and sludge into shipping containers to determine if appropriate waste stream mixing and sampling methods were utilized and to verify that representative samples were obtained of the waste product.

b. Findings

No findings of significance were identified.

.2 Waste Characterization and Classification

a. Inspection Scope

The inspector reviewed the licensee's methods and procedures for determining the classification of radwaste shipments, including the use of scaling factors to quantify "difficult to measure" radio-nuclides (e.g., pure alpha and low energy beta emitting materials). Specifically, the inspector reviewed the licensee's 2000 and 2001 radio-chemical analysis results for the plant's waste streams which consisted of primary and secondary (radwaste) system resins, sludge, filter media and cartridges, and dry active waste (DAW). The inspector reviewed these analyses to ensure that the scaling factors were accurately determined to allow waste shipments to be classified in accordance with the requirements of 10 CFR Part 61, consistent with the licensee's procedure. The inspector also reviewed the licensee's practices to ensure that changes in reactor operating parameters that could produce changes to the waste stream radio-nuclide composition were identified between annual scaling factor reevaluation. Additionally, the inspector performed independent calculations to verify proper scaling factor application and to determine if the activities of certain difficult to detect radio-nuclides were accurate and if waste streams were classified in accordance with 10 CFR Part 61. An inspector identified deficiency with the recent application of the scaling factor for carbon-14 activity determinations were assessed to verify that it did not result in mis-classified waste streams.

b. Findings

No findings of significance were identified.

.3 Shipment Preparation and Observation of Radwaste Processing Activities

a. Inspection Scope

The inspector observed two pre-job briefs and evaluated the preparations, including the operations department interface, associated with the sluice of primary system resins from the spent resin storage tank into a high integrity container. The evaluation was conducted to assess the overall adequacy of the work planning and to verify that work preparations were completed consistent with the resin transfer procedure. The inspector witnessed the sluice operation and discussed its performance with involved staff to verify that the work was executed in accordance with station procedure, to determine if supervisory oversight was adequate, and to assess the adequacy of the radiological controls for the work activity. Since there were no radioactive material shipments during the inspection, the inspector reviewed training and qualification records for those staff involved in radwaste processing and shipment activities. Specifically, the inspector reviewed training certificates for the licensee’s four authorized shippers and training lesson plans and qualification records for environmental technicians involved in the processing and shipment of radwaste and radioactive material. The documents were reviewed to verify that the licensee’s program provided hazardous material training to those personnel responsible for radioactive material shipments and shipment preparation, as required by Subpart H of 49 CFR Part 172.

b. Findings

No findings of significance were identified.

.4 Shipment Records

a. Inspection Scope

The inspector reviewed radioactive material and radwaste shipment manifests and associated records for eight non-excepted shipments (Low Specific Activity II, Surface Contaminated Object II and a Type B package shipment) completed between September 1999 and February 2002. The review was performed to verify compliance with NRC requirements contained in 10 CFR Parts 20, 61 and 71, and the Department of Transportation (DOT) requirements of 49 CFR Parts 172 and 173. Specifically, records were reviewed and those staff involved in shipment activities were interviewed to verify that packages were labeled and marked properly, that package and transport vehicle surveys satisfied DOT requirements, that cask certificate of compliance requirements were satisfied, and that shipment manifests were completed in accordance with the regulations and included appropriate emergency response information.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed a self-assessment, Performance Assurance Department audits and field observations, and CRs completed since January 2001, which addressed the areas of radwaste processing and radioactive material/radwaste shipping. The inspector reviewed these documents to assess compliance with the quality assurance program audit requirements of 10 CFR Part 71 and Appendix G of 10 CFR Part 20 and to evaluate the licensee's ability to identify problems, to determine contributing causes and extent of condition, and to implement corrective actions to prevent recurrence. The inspector also discussed the licensee's audit and field observation program with Performance Assurance staff including the scope of their recent activities and plans to enhance the program.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP1 Access Authorization (AA) Program (Behavior Observation Only) (71130.01)

a. Inspection Scope

The inspectors interviewed five supervisors and five non-supervisors (both licensee and contractor employees) to determine their knowledge level and practice of implementing the licensee's behavior observation program responsibilities. Selected procedures pertaining to the Behavior Observation Program and associated training activities were also reviewed. Also licensee fitness-for-duty semi-annual test results were reviewed. In addition, the inspectors reviewed a sample of licensee self-assessments, audits, and security logged events. The inspectors also interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

3PP2 Access Control (Identification, Authorization and Search of Personnel, Packages, and Vehicles) (71130.02)

a. Inspection Scope

The inspectors reviewed the licensee's protected area access control testing and maintenance procedures. The inspectors observed licensee testing of all access control equipment to determine if testing and maintenance practices were performance based. On two occasions, during peak ingress periods, the inspectors observed in-processing search of personnel, packages, and vehicles to determine if search practices were conducted in accordance with regulatory requirements. Interviews were conducted and records were reviewed to verify that security staffing levels were consistently and appropriately implemented. Also the inspectors reviewed the licensee's process for limiting access to only authorized personnel to the protected area and vital equipment by a sample review of quarterly access authorization reviews performed by managers. The inspectors reviewed the licensee's program to control hard-keys and computer input of security-related personnel data.

The inspectors reviewed a sample of licensee self-assessments, audits, maintenance request records, and security logged events for identification and resolution of problems. In addition, the inspectors interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors verified the data for the Physical Protection performance indicators pertaining to Fitness-For-Duty Personnel Reliability, Personnel Screening Program, and Protected Area Security Equipment. Specifically, a sample of plant reports related to security events, security shift activity logs, fitness-for-duty reports, and other applicable security records were reviewed for the period between October 1, 2001 and April 1, 2002.

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report (LER) 50-316-2002-001-00: “Containment Isolation Valve Alignment Error During Local Leak Rate Testing." This event is discussed in Section 1R14.1 of this report. In addition, this issue was identified as an input to a significant cross-cutting issue as discussed in Section 4OA4 of this report. This LER is closed.

.2 (Closed) LER 50-315-2002-002-00: “Pressurizer Power Operated Relief Valves Inoperable Due to Control Switch Position." This event is discussed in Section 1R14.2 of this report. In addition, this issue was identified as an input to a significant cross-cutting issue as discussed in Section 4OA4 of this report. This LER is closed.

.3 (Closed) LER 50-316-2002-002-00: "Technical Specification 3.9.4.c Was Violated During Core Alterations." The licensee failed to maintain refueling integrity configuration control of containment penetration CPN-74 during core alterations when containment isolation valve (2-XCR-101) was stroked open for testing. Opening this valve created a direct access path from the containment atmosphere to the outside atmosphere. The licensee reported this event as a condition prohibited by the plant's TSs in accordance with 10 CFR 50.73(a)(2)(i)(B). The inspectors determined that this issue had a credible impact on safety because the licensee failed to have the containment penetration isolated as required by the TSs.

The inspectors utilized the event information in conjunction with Appendix G, “Shutdown Operations Significance Determination Process,” of Manual Chapter 0609, Table T-1, “Pressurized Water Reactor (PWR) Refueling Operation Reactor Coolant System (RCS) Level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours AND Inventory in the Pressurizer.” This issue was determined to be of very low significance (Green) by the significance determination process because (1) the issue did not increase the likelihood of a loss of primary coolant system inventory; (2) the issue did not degrade the licensee’s ability to terminate a leak path or add RCS inventory when needed; and (3) the issue did not degrade the licensee’s ability to recover decay heat removal once lost. Although this issue affected the integrity of the reactor containment during core alterations, the inspectors concluded that because 2-XCR-101 was open for a short period of time and the small diameter penetration would be a very small leakage path, this issue was of very low safety significance (Green).

Technical Specification 3.9.4.c, states, in part, that each containment penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either closed by an isolation valve, blind flange, manual valve, or equivalent during core alterations or movement of irradiated fuel within the containment. With the above requirement not satisfied, immediately suspend all operations involving core alterations or movement of irradiated fuel in the Containment Building. Contrary to the above, on February 12, 2002, the licensee failed to maintain containment isolation valve 2-XCR-101 closed to isolate containment penetration 2-CPN-74 during core alterations. This is a violation of TS 3.9.4.c. 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 50-316-02-03-06(DRP)). The licensee entered this violation into its corrective action program as CR 02043026. This LER is closed.

.4 (Closed) LER 316-2001-002-00 and LER 316-2001-002-01: "Failure of Lower Containment Airlock Door Interlock Results in Inadvertent Opening of Both Doors". Supplement 1 of this LER was submitted to provide revised information from the completed root cause evaluation and replaced the original LER in its entirety. This event is discussed in Section 1R14.3 of this report. In addition, this issue was identified as an input to a significant cross-cutting issue as discussed in Section 4OA4 of this report. This LER and its supplement are closed.

.5 Ice Condenser Lower Inlet Door Testing

a. Inspection Scope

As discussed in Section 1R22.b.3 of NRC Inspection Report 50-315/316-01-20(DRP), the inspectors previously identified that the licensee's ice condenser lower inlet door testing methodology had not been capable of demonstrating compliance with the requirements of TS 4.6.5.3.1.b. The licensee subsequently revised its lower inlet door testing methodology in February 2002. At the time this condition was discovered, Unit 2 was in a shutdown mode and the licensee was able to retest the Unit 2 lower inlet doors using the revised test methods and demonstrate compliance with TS 4.6.5.3.1.b. Based on the results on the Unit 2 door retests, the inspectors were able to assess the safety significance of this issue and identified NCV 50-316-01-20-07(DRP).

Because lower inlet door testing requires the associated unit to be in at least Mode 5 (Cold Shutdown), the licensee was unable to retest the Unit 1 lower inlet doors in February 2002. Instead, the licensee requested, and was granted, Emergency TS Amendment 265 to defer Unit 1 lower inlet door testing until Unit 1 entered the Cycle 18 refueling outage or a Mode 5 entry of sufficient duration. Because as found door testing data had not been obtained for Unit 1, the inspectors were unable to assess the significance of this issue for Unit 1 and identified Unresolved Item (URI) 50-315-01-20-08(DRP) pending completion of a final significance determination. During the Unit 1 Cycle 18 refueling outage, the licensee tested the Unit 1 lower inlet doors using the revised test methodology. The inspectors reviewed the testing results and assessed the safety significance of the previously identified finding using the as-found results of this testing. The inspectors also reviewed two LERs that were issued as a result of this finding.

b. Findings

The inspectors identified a finding of very low safety significance (Green) associated with the licensee's failure to establish an ice condenser lower inlet door testing methodology capable of demonstrating compliance with Unit 1 TS surveillance requirements. This finding was dispositioned as a Non-Cited Violation.

b.1 (Closed) URI 50-315-01-20-08(DRP): "Failure to Adequately Measure the Ice Condenser Lower Inlet Door Opening Torque and Closing Torque in Accordance with TS Requirements."

Description

As discussed in NRC Inspection Report 50-315/316-01-20(DRP), the inspectors previously determined that the licensee's ice condenser lower inlet door testing methodology was inadequate. Specifically, the methodology used to perform previous ice condenser lower inlet door testing on November 21, 2000 under Job Order R0087658 did not accurately determine door opening and closing torques in accordance with TS 4.6.5.3.1.b.3 and TS 4.6.5.3.1.b.4. Consequently, the licensee was unable to adequately calculate inlet door friction in accordance with TS 4.6.5.1.b.5. During a review of the door testing data obtained under Job Order R0087658, the inspectors noted that the door opening torques for 36 of the 48 inlet doors were less than the associated door closing torques. Based on the inlet door design and configuration, the inspectors concluded that door opening torque must be greater than door closing torque. Based on these previous testing results, the inspectors concluded that the licensee's failure to adequately demonstrate compliance with the requirements of TS 4.6.5.3.1.b.3, TS 4.6.5.3.1.b.4, and TS 4.6.5.3.1.b.5 was a violation of NRC requirements.

In accordance with Unit 1 Licensee Amendment 265, the licensee measured the as-found lower inlet door torque and friction using the revised methodology under Job Order R0210872 in May 2002. During this testing, all friction and closing torque measurements were found to be within the TS allowable values. However, the licensee determined that the opening torque for one lower inlet door (bay 15 right) failed to meet the TS allowable opening torque. Specifically, the as-found opening torque for door 15 right was 212.6 inch-pounds, or approximately 18 inch-pounds greater than the TS 4.6.5.3.1.b.3 requirement of less than 195 inch-pounds.

Analysis

The inspectors evaluated this failure to adequately perform testing required by TS 4.6.5.3.1.b using the SDP. The inspectors determined that the failure to adequately implement TS 4.5.6.3.1.b testing requirements for the Unit 1 ice condenser lower inlet doors had a credible impact on safety and was more than a minor concern. Specifically, failing to adequately execute surveillance test requirements could credibly result in the failure to identify and correct degraded or inoperable equipment. As stated in the TS 3.6.5 bases, operability of the ice condenser doors ensures that reactor coolant fluid released during a loss of coolant accident (LOCA) will be diverted through the ice condenser bays for heat removal. The ice condenser also augments the containment recirculation sump water inventory in the event of certain small break LOCAs and limits ice maldistributions within the ice condenser. Because the proper functioning of the ice condenser lower inlet doors was primarily associated with the heat removal function of the ice condenser, the inspectors determined that this issue was associated with the barrier integrity cornerstone. Based on a review of the as-found Unit 1 lower inlet door testing performed in May 2002, the inspectors concluded that there was no reduction in the atmospheric pressure control function of the reactor containment nor a loss of capability to provide additional recirculation sump inventory during certain small break LOCAs. Specifically, all inlet doors, except for door 15 right, met TS 4.5.6.3.1.b requirements during as-found testing. Although the door 15 right opening torque exceeded the TS maximum allowable opening torque, the as-found opening torque of

212.6 inch-pounds was bounded by the 252 inch-pound upper design limit described in the NRC staff's safety evaluation associated with Unit 1 License Amendment 265. Consequently, the inspectors concluded that this issue was of very low safety significance.

Enforcement

Technical Specifications 4.6.5.3.1.b.3, 4.6.5.3.1.b.4, and 4.6.5.3.1.b.5 require testing of the ice condenser lower inlet doors at least once per 18 months in order to measure the torque required to open the door, the torque required to keep the door from closing, and the door frictional torque. Technical Specification 4.6.5.3.1.b.3 stated that the door opening torque was equal to the nominal door torque plus a frictional torque component. Technical Specification 4.6.5.3.1.b.4 stated that the door closing torque was equal to the nominal door torque minus a frictional torque component. Contrary to the above, previous TS 4.6.5.3.1.b surveillance testing performed in Unit 1 on November 21, 2000, failed to adequately measure the door opening torque and the door closing torque in accordance with TS requirements. Specifically, the methodology used by the licensee to perform TS 4.6.5.3.1.b.3 and TS 4.6.5.3.1.b.4 resulted in door closing torques that were greater in magnitude than the door opening torques, contrary to the TS description of these torque values. The inspectors identified that the measured opening torque values for 36 Unit 1 lower inlet doors were less than the associated door closing torque. Because calculation of the door frictional torque required accurate measurement of the door opening and closing torques, the licensee was unable to demonstrate compliance with the requirements of TS 4.6.5.3.1.b.5. Consequently, Unit 1 was operated in a Mode requiring operability of the lower ice condenser inlet doors for approximately 16 months between December 2000 and May 2002 without meeting the requirements of TS 4.6.5.3.1.b. During subsequent testing conducted on May 12, 2002, the licensee identified that lower inlet door 15 right had an opening torque in excess of the requirements of TS 4.6.5.3.1.b.3. 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 50-315-02-03-07(DRP)). The licensee entered this violation into its corrective action program as CR 02032016. This URI is closed.

b.2 (Closed) LER 50-315-2002-004-00: "Unit 1 Ice Condenser Lower Inlet Door Test Failure." The licensee issued LER 50-315-2002-004-00 to document the failure of ice condenser lower inlet door 15 right to meet TS 4.6.5.3.1.b.3 requirements. The licensee stated that the cause of the excessive door opening torque failure was incorrect door closing spring adjustment following refurbishment during the 1997-2000 extended dual unit shutdown. Consequently, lower inlet door 15 right failed to meet TS requirements for door opening torque for approximately 15 months. Although the licensee tested door 15 right following these previous refurbishment activities, the inspectors determined that the licensee's failure to identify the incorrect door closing spring adjustment was due to the use of an inadequate testing methodology. The inspectors have already concluded that the licensee's use of an inadequate door testing methodology was a violation of NRC requirements (refer to Section 4OA3.5.b.1 of this report) and, therefore, no additional enforcement action is warranted. The inspectors identified no other issues of significance during this review. This LER is closed.

b.3 (Closed) Licensee Event Report (LER) 50-315-2002-001-00: "Failure to Perform Ice Condenser Door Testing In Accordance With TSs." The inspectors reviewed this event and issued NCV 50-316-01-20-07(DRP) and NCV 50-315-02-03-07(DRP) for the licensee's failure to adequately test ice condenser lower inlet doors as required by TS 4.6.5.3.1.b.3. The inspectors determined that the information provided in LER 50-315-2002-001-00 did not raise any new issues or change the conclusions of the initial reviews, which were documented in NRC Inspection Report 50-315/316-01-20(DRP) and in Section 4OA3.5.b.1 of this report. This LER is closed.

.6 Cell Cracking Rendered the Unit 2 AB 250 Volt Station Battery Inoperable and Review of Associated NOED

a. Inspection Scope

On April 3, 2002, a licensee maintenance electrician identified cracks on the top covers of two battery cells in the 2AB battery. Although the cracking indicated abnormal deterioration of the battery as described in TS 4.8.2.3.2.c.1, the licensee did not declare the battery inoperable until April 4, 2002, and enter the appropriate TS action statement until approximately 30 hours after the initial identification of the cell deterioration. During follow-up investigations, the licensee identified that a third cell on the 2AB battery also exhibited abnormal cracking. The licensee requested, and was granted, an NOED to extend the TS 3.8.2.3 allowable outage time of 2 hours to 13 hours in order to support repair and testing activities necessary to return the 2AB station battery to an operable status. As required by TS 3.8.2.3, the licensee had initiated a shutdown of Unit 2, but terminated the shutdown at approximately 40 percent power when NOED-02-3-01 was issued. The licensee returned the 2AB battery to an operable status within the extended allowable outage time and returned the unit to full power on April 7, 2002. On May 29, 2002, the licensee issued LER 50-316-2002-003-00, which reported this issue as an operational NOED condition prohibited by the TSs because the battery was inoperable for longer than allowed by TS 3.8.2.3. The inspectors reviewed the cause of the licensee's delayed entry into the TS 3.8.2.3 Limiting Condition for Operation on April 4, 2002, the basis for the licensee's NOED request, and the licensee's compliance with the compensatory actions of the NOED.

b. Findings

b.1 (Closed) LER 50-316-2002-003-00: "2AB 250 D.C. [Direct Current] Volt Battery Inoperable For Longer Than Allowed By Plant's TSs."

The action statement for TS 3.8.2.3.2, "D.C. Distribution - Operating," required that an inoperable battery be restored to operable status within 2 hours or the plant be placed in Hot Standby within the next 6 hours. The inspectors previously documented a review of this issue in NRC Inspection Report 50-315/316-02-004(DRP), Section 4OA2.1.2. The inspectors determined that the failure to take prompt action to address abnormal deterioration of the safety-related battery constituted a violation of NRC requirements and issued NCV 50-316-02-04-01(DRP) for this condition. The licensee's corrective actions for this condition were reviewed and considered adequate. The inspectors reviewed the associated LER and did not identify any additional significant issues. This LER is closed.

b.2 (Closed) URI 50-316-02-03-08 (DRP): "Review of NOED-02-3-01 Regarding D.C. Cook, Unit 2, Compliance With TS 3.8.2.3."

The inspectors opened URI 50-316-02-03-08(DRP) to track documentation of the root cause for the NOED request, review of the NOED approval basis, and verification of licensee activities associated with NOED implementation. As discussed in NRC Inspection Report 50-315/316-02-04(DRP), the inspectors reviewed the performance history of the station batteries to determine if the licensee had prior opportunities to identify and correct the battery cell cracking prior to requesting an NOED. The inspectors determined that the licensee's actions to address the condition prior to April 3, 2002, did not constitute a violation of NRC requirements. The inspectors concluded that the licensee provided a reasonable basis for the NOED and appropriately implemented compensatory measures. This URI is closed.

.7 Significance Determination Process Review for Gas Binding of Unit 2 Centrifugal Charging Pump (CCP) Due to Inadequate Valve Maintenance Activity

a. Inspection Scope

On February 16, 2002, the Unit 2 West CCP exhibited indications of gas binding following swap over of the suction source from the volume control tank to the refueling water storage tank. The inspectors concluded that the cause of the CCP gas binding was the licensee's failure to ensure that valve 2-CS-369 (reactor coolant pump seal water heat exchanger to volume control tank shutoff valve) was fully closed, resulting in transfer of volume control tank cover gas directly to the suction of the Unit 2 CCPs. This issue was identified as URI 50-316-02-02-01(DRP) pending completion of the safety significance determination for the gas binding event. The inspectors, with the assistance of the Region III Senior Reactor Analysts, performed additional safety significance reviews of this issue and reviewed the licensee's completed apparent cause evaluation for this event.

b. Findings

(Closed) URI 50-316-02-02-01(DRP): "Failure to Perform Adequate Maintenance and Testing on Valve 2-CS-369 Resulted in Gas Binding the Unit 2 West Centrifugal Charging Pump."

A self-revealed finding of very low safety significance (Green) was identified for the licensee's failure to provide instructions of a type appropriate to the circumstances for maintenance on valve 2-CS-369. 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 issue as a Non-Cited Violation.

Description

Following the February 16, 2002, CCP gas binding event, the licensee determined that the position of the valve stem stop nut prevented full closure of 2-CS-369 and allowed gas to vent from the volume control tank directly to the CCP suction line. Approximately two weeks before this event, on February 1, 2002, the licensee performed preventative maintenance activities on 2-CS-369 in accordance with Job Order 01094018 and procedure 12 MHP 5021.001.023, "Manual Diaphragm Valve Maintenance," Revision 6. Steps 6.6.3, 6.6.4, and 6.6.5 of 12 MHP 5021.001.023, which positioned the stop nut after valve reassembly, required that the stem stop nut be positioned in contact with the valve handwheel after the valve was turned clockwise 1/8 of a turn beyond the closed seat contact point by tightening the stem lock nut. The stem lock nut was then tightened into the stem stop nut to lock the stop nut into position. The purpose of these steps was to lock the stop nut in a position that would allow full closure of 2-CS-369 without applying excessive compressive force on the valve diaphragm.

The licensee's apparent cause evaluation for this condition, which was documented in CR 02047050, concluded that the 12 MHP 5021.001.023 stem stop nut adjustment instructions were inconsistent with vendor recommendations and left the valve susceptible to stop nut loosening if the valve was tightly closed. Specifically, the licensee determined that tightly closing the valve with the stop nut positioned per this guidance resulted in high contact forces between the handwheel and stem stop nut. Binding between the stem stop nut and handwheel during subsequent opening operations could then cause the stop nut to loosen, preventing full valve closure. The licensee determined that use of the vendor recommended procedure for stem stop nut adjustment left a gap between the handwheel and stem stop nut and therefore prevented binding that could loosen the stem stop nut. Consequently, the inspectors determined that the failure to provide stem stop nut adjustment instructions in 12 MHP 5021.001.023 of a type appropriate to the circumstances constituted a violation of NRC requirements.

Analysis

The inspectors assessed this issue using the SDP. The inspectors concluded that this issue had a credible impact on safety and was therefore more than a minor concern. Specifically, gas intrusion into the common suction lines of both Unit 2 CCPs with the suction source aligned to the refueling water storage tank impacted the capability of the high head injection system to provide the inventory and reactivity control safety functions. Furthermore, the inspectors determined that this issue was associated with the mitigating systems cornerstone. The inspectors concluded that 2-CS-369 was in a degraded condition from February 1, 2002, when the valve diaphragm was replaced, to February 16, 2002, when the condition was identified and corrected. Because Unit 2 was in a shutdown mode during this period, the inspectors performed a Phase 1 SDP review of this issue using the guidance provided in IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." During this Phase 1 review, the inspectors concluded that this issue degraded the licensee's ability to add inventory to the RCS and therefore a Phase 2 SDP analysis was required. A modified Phase 2 shutdown risk SDP analysis was performed with the assistance of the Region III Senior Reactor Analyst and headquarters probabilistic risk assessment staff. The following factors were considered during this shut down risk assessment:

- Shutdown initiating event frequencies were obtained from NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1."

- A loss of the operating train of residual heat removal caused by gas intrusion was not considered to be credible due to the torturous path the gas would have to follow in order to bind the residual heat removal system. Although the safety injection (SI) pumps share a common suction with the CCPs from the refueling water storage tank, check valve 2-SI-185 would prevent migration of gas to the suction of the SI pumps. The licensee seat leak tested 2-SI-185 on February 8, 2002, and measured a seat leakage rate of 0 gpm. Consequently, the estimation of risk significance assumed that only the charging system was affected by the performance deficiency.

- This deficiency does not increase the likelihood or severity of a loss of offsite power event, so this initiating event is not considered.

- The risk assessment considered the following plant configuration states: (1) 5 days in Mode 6 (Refueling) with the refueling cavity full and SI pumps available, (2) 2.5 hours in Mode 6 with the refueling cavity full and SI pumps unavailable, (3) draining to mid-loop operations after refueling to support RCS vacuum refill, (4) 1 day of mid-loop operations after refueling, and (5) a reactivity accident during mid-loop operations.

- Because recovery of the charging system would require identifying the performance deficiency and implementation of appropriate corrective actions, no recovery credit was applied.

- The use of the opposite unit's high head injection via a unit cross tie was credited in the analysis. Although procedures did not specifically address use of the charging system cross tie for loss of shutdown cooling events, the cross-connect valves were regularly tested and the operators were trained on the cross-connection procedures.

- The 2-CS-369 diaphragm was replaced on February 1, 2002, with Unit 2 defueled. Unit 2 entered Mode 6 on February 10, 2002, and completed core reload on February 12, 2002. Because the degraded condition of 2-CS-369 was identified and corrected on February 16, 2002, the safety function provided by the CCPs was degraded for approximately 6 days with fuel in the reactor vessel.

- Based on the observed Unit 2 West CCP performance during the gas intrusion event on February 16, 2002, (decreased pump amperage and near 0 gpm flowrate), the inspectors concluded that the degraded condition of 2-CS-369 would render the CCPs unavailable when aligned to the refueling water storage tank.

Based on a consideration of the above factors, the total change in Core Damage Frequency associated with this issue was estimated to be 3E-7 per year. Therefore, this issue was considered to be 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 procedures of a type appropriate to the circumstances for the adjustment of the stem stop nut on 2-CS-369, which is an activity affecting quality. Specifically, the instructions for stem stop nut adjustment contained in 12 MHP 5021.001.023, "Manual Diaphragm Valve Maintenance," Section 6.6, Revision 6 were inconsistent with vendor recommendations and rendered the valve susceptible to loosening of the stem stop nut. The stem stop nut for 2-CS-369 was adjusted in accordance with these instructions on February 1, 2002. This issue was self-revealed on February 16, 2002, when the Unit 2 West CCP became gas bound due to leakage of volume control tank cover gas through the partially opened valve 2-CS-369 into the common suction header for the Unit 2 CCPs. Subsequent investigation identified that the 2-CS-369 stem stop nut was loose and in a position that prevented full closure of the valve. 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 50-316-02-03-09(DRP)). The licensee entered this violation into its corrective action program as CR 02047050. This URI is closed.

.8 (Closed) URI 50-316-00-19-02; 50-316-00-19-02: "Potentially Non-Conservative Engineered Ventilation TS 3.7.6.1." During a review of an operability determination for the component cooling pump ventilation fans documented in CR 00-6947, the inspectors noted an inconsistency between the Auxiliary Building ventilation calculation assumptions and TS 3.7.6.1, "ESF [Engineered Safety Features] Ventilation System," operability requirements. Specifically, the Auxiliary Building ventilation calculation of record, MD-12-HV-002-N, credited airflow from the ESF ventilation unit in the non-accident unit during a design basis accident. Because TS 3.7.6.1 was applicable only in Modes 1 through 4, TS 3.7.6.1 was non-conservative relative to the lowest ESF ventilation functional capability assumed in MD-12-HV-002-N with one unit in Mode 5 (Cold Shutdown) or Mode 6 (Refueling). On August 12, 2000, the licensee initiated CR 00-11265 to investigate this issue and determine the lowest functional capability required from the ESF ventilation system to mitigate a design basis event.

The licensee included resolution of this issue within the scope of the revised Auxiliary Building ventilation system calculation, TH-01-05, which was completed on January 18, 2002. The licensee concluded that, although several ECCS pump room temperatures were increased if the ESF ventilation in the non-accident unit was removed from service in accordance with TS 3.7.6.1, the mitigating equipment remained operable. The inspectors reviewed the results of this calculation and the resolution of CR 00-11265 and CR 00-6947 and determined that the licensee's evaluation and conclusions were reasonable. Consequently, the inspectors concluded that the licensee has developed an adequate basis to justify the lowest functional capability of the ESF ventilation system as currently defined in TS 3.7.6.1. Although the licensee had been unable to adequately justify the lowest ESF ventilation system functional capability as defined in TS 3.7.6.1 prior to the issuance of TH-01-05, this issue is considered to be of minor safety significance and is not subject to formal enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This URI is closed.

4OA5 Other

.1 Institute of Nuclear Power Operations (INPO) Mid-Cycle Report

The inspectors reviewed the INPO Mid-Cycle Report for the D. C. Cook Plant conducted in April 2002. During this review, the inspectors did not identify any safety significant issues.

.2 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (Temporary Instruction 2515/145)

a. Inspection Scope

The inspector performed a review of the licensee's activities in response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," to verify compliance with applicable regulatory requirements. In accordance with the guidance of NRC Bulletin 2001-01, D. C. Cook Unit 1 was characterized as belonging to the sub-population of plants (Bin 4) that were considered to have a low susceptibility to primary water stress corrosion cracking (PWSCC). Although the anticipated low likelihood of PWSCC degradation at the Bin 4 facilities indicates that enhanced examination beyond the present requirements is not currently necessary, the licensee responded to NRC Bulletin 2001-01 by performing a remote visual examination of the reactor vessel head and a qualified volumetric (ultrasonic and eddy current) examination of the 79 vessel head penetrations and head vent. Full volumetric coverage was achieved on 77 vessel head penetrations. Liquid penetrant examination was performed on penetrations 70 and 73 J-welds due to the inability to perform 100 percent volumetric coverage of these welds.

The inspector interviewed inspection personnel, reviewed procedures and inspection reports, including photographic documentation, to assess the licensee's efforts in conducting an "effective" visual and volumetric examination of the reactor vessel head. The inspector reviewed the qualifications and certification of personnel performing the volumetric examinations to ensure that they were in accordance with approved procedures and techniques (ultrasonic and eddy current) demonstrated for the NRC at the Electric Power Research Institute (EPRI). The inspector reviewed the inspection procedures, equipment certifications, and personnel certifications.

Evaluation of Visual Head Inspection Requirements

1. *Were the licensee's examinations performed by qualified and knowledgeable personnel?*

The inspector determined that the examinations were performed by individuals certified as Level II and Level III in the VT-2 Method. The specific guidelines described in the EPRI, "Visual Examination for Leakage of PWR [Pressurized Water Reactor] Reactor Head Penetrations," were also used for the inspections.

2. *Were the licensee's examinations performed in accordance with approved and adequate procedures?*

The inspector verified that the examinations were conducted in accordance with an approved plant procedure, "Reactor Vessel Head Penetration Remote Visual Inspections for Cook Unit 1", MRS-SSP-1319, and the guidelines established in EPRI Document 1006296, "Visual Examination for Leakage of PWR Reactor Head Penetrations." The inspector determined that the procedure and supplemental guidance was appropriate for the examinations.

3. *Were the licensee's examinations adequately able to identify, disposition, and resolve deficiencies?*

The inspector determined through a review of post-examination records, discussions with the personnel that conducted the examinations, and a review of the procedure, that the examinations were sufficient to identify any deficiencies. The licensee's examinations identified two deficiencies that were documented in CR 021360423 and CR 02135066. The inspector assessed the licensee's efforts to disposition and resolve the deficiencies.

4. *Were the licensee's examinations capable of identifying the primary stress corrosion cracking phenomenon described in the Bulletin?*

The inspector determined through interviews with inspection personnel, and reviews of procedures and inspection reports, including photographic documentation of the examinations, that the licensee's efforts were capable of identifying the phenomenon described in the Bulletin. The inspector determined that the inspection personnel had 360 degree access to all 80 vessel head penetrations, with no obstructions or interferences.

5. *What was the condition of the reactor vessel head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?*

The vessel head had block contoured vessel head insulation, consisting of mirror panels fabricated of 3-inch thick Type 304 stainless steel insulation. The inspector determined that the licensee had complete viewable coverage. The inspector also determined through discussions with the inspection personnel and review of the inspection photographs that the as-found pressure vessel head condition showed evidence of boric acid residues from known canopy seal and thermocouple column conoseal leakage. Foreign material (CR 02135066) in the form of 2 screws, 1 bolt, several pieces of wire, dust/dirt, paint chips and metal filings were found during the video inspection. These items were removed by vacuuming and cleaning of the reactor vessel head.

6. *Could small boron deposits, as described in the bulletin, be identified and characterized?*

The inspector verified, through interviews with inspection personnel and review of the photographic record of the examination, that small boron deposits, as described in the Bulletin, could be identified; given the accessibility of the pressure vessel head penetrations. However, no evidence of boric acid deposits characteristic of active leakage were found during the inspection.

7. *What materiel deficiencies (associated with the concerns identified in the bulletin) were identified that required repair?*

Through a review of the examination records, the inspector determined the inspection personnel did not identify any materiel deficiencies. No wastage or corrosion was noted other than very minor inactive surface rusting.

8. *What, if any, significant items that could impede effective examinations and/or ALARA issues were encountered?*

The inspector verified that there were no impediments to the examinations. Collective radiation doses received as a part of the examinations were 3.958 rem.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Interim Exits

The results of the Public Radiation Safety - Radwaste Processing and Transportation Inspection were presented to Mr. J. Pollock and other members of licensee management at the conclusion of the inspection on April 12, 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 inspector subsequently discussed changes to the original characterization of the findings by telephone with Mr. J. Long on April 17, 2002.

The results of the Safeguards Access Authorization Program/Access Control Inspection were presented to Mr. J. McMahon and other members of the licensee management at the conclusion of the inspection on April 26, 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 Unit 1 Biennial Inservice Inspection and TI-145 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (NRC Bulletin 2001-01) Inspection were presented to Mr. J. Pollock and other members of licensee management at the conclusion of the inspection on May 23, 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 Occupational Radiation Safety - Access Controls for Radiologically Significant Areas and ALARA Planning/Controls Inspection were presented to Mr. C. Bakken and other members of licensee management at the conclusion of the inspection on May 24, 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 inspector subsequently discussed changes to the original characterization of the findings by telephone with Mr. D. Noble on June 6, 2002.

.2 Resident Inspectors' Exit

The inspectors presented the inspection results to Mr. J. Pollock and other members of licensee management at the conclusion of the inspection on July 9, 2002. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was examined during this inspection but is not specifically discussed in this report.

.3 Annual Assessment Meeting

On April 12, 2002, the NRC presented the results of its annual assessment of D. C. Cook Plant's performance to Mr. C. Bakken and other members of licensee management during a public meeting held at the Hampton Inn in Stevensville, Michigan. The results of the annual assessment were previously documented in a letter to the licensee dated March 4, 2002. The slides presented by the NRC are available in ADAMS (accession number ML021120165).

4OA7 Licensee Identified Violations. The following findings of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a Non-Cited Violations (NCVs).

If the licensee contests these NCVs, the licensee should provide a response within 30 days of the date of this inspection report, with the basis for the denial, to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region III; 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.

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-315/316-02-03-10	Technical Specification 6.12 requires that high radiation areas accessible to personnel with radiation levels greater than 1000 mrem/hour be provided with locked doors to prevent unauthorized entry and be conspicuously posted. Doors shall remain locked except during periods of access by personnel under an approved RWP.

Contrary to the above, for an approximate 20 hour period beginning the afternoon of April 10, 2002, the licensee failed to maintain the door leading into the 587 foot elevation radioactive waste drumming room in the Auxiliary Building (an area that had radiation levels up to 2500 mrem/hour) locked or under direct surveillance to prevent unauthorized entry. This is a violation of TS 6.12. During the 20 hour period, one unauthorized entry into the drumming room occurred but without dose consequence. The problem was identified during a routine RP surveillance. The licensee entered this violation into its corrective action program as CR 02101048.

NCV 50-315/316-02-03-11

Technical Specification 6.12 requires that high radiation areas accessible to personnel with radiation levels greater than 1000 mrem/hour be provided with locked doors to prevent unauthorized entry and be conspicuously posted. Doors shall remain locked except during periods of access by personnel under an approved RWP.

Contrary to the above, for an approximate 6-8 hour period beginning the evening of May 5, 2002, during the Unit 1 CRUD burst cleanup of the RCS, the licensee failed to properly post and maintain the door leading into the 617 foot elevation demineralizer valve gallery in the Auxiliary Building (an area that had radiation levels up to 3000 mrem/hour) locked or under direct surveillance to prevent unauthorized entry. This is a violation of TS 6.12. No unauthorized entry was made into the area while it was not properly posted or controlled. The problem was identified during follow-up CRUD burst surveys by the RP staff. The licensee entered this violation into its corrective action program as CR 02126020.

The inspector concluded that the maximum radiation levels in both the drumming room and valve gallery coupled with the limited accessibility of the high radiation area in the drumming room precluded a substantial potential for an overexposure. Both incidents were therefore determined to be of very low safety significance. The licensee correctly concluded that the events described in NCV 50-315/316-02-03-10 and NCV 50-315/316-02-03-11 each represented a high radiation area occurrence under the Occupational Exposure Control Effectiveness performance indicator, both of which the licensee planned to report in its second quarter 2002 performance indicator submittal to the NRC.

KEY POINTS OF CONTACT

Licensee

M. Allen, Assistant Maintenance Director
G. Arent, Regulatory Affairs Manger
C. Bakken, Senior Vice President, Nuclear Generation
G. Borlodan, Plant Programs Manager
J. Bradshaw, Security/Support Services Performance Supervisor
K. Burkett, Security/Support Services Access Control Supervisor
P. Cowan, Regulatory Affairs Licensing Supervisor
R. Gaston, Regulatory Affairs Compliance Supervisor
J. Gebbie, System Engineering Manager
G. Gibson, Site Protective Services Manager
S. Greenlee, Nuclear Technical Services Director
R. Hall, Inservice Inspection Program Specialist
G. Harland, Work Control/Maintenance Director
N. Jackiw, Regulatory Affairs Specialist
E. Lamoureut, Westinghouse Project Manager
C. Lane, Inservice Inspection Supervisor
E. Larson, Operations Director
J. Long, Environmental Compliance General Supervisor
R. Meister, Regulatory Affairs Specialist
D. Moul, Operations Shift Technical Advisor Supervisor
D. Noble, Radiation Protection Technical Support
T. Noonan, Performance Assurance Director
J. Pollock, Site Vice President
A. Rodriguez, Security/Support Services Manager
M. Schaefer, Nuclear Specialist
L. Smead, Security Operations Analyst
R. Smith, Plant Engineering Assistant Director
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L. Weber, Performance Oversite Manager
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D. Passehl, Acting Chief, Reactor Projects Branch 6
S. Burgess, Senior Reactor Analyst, Division of Reactor Safety
M. Parker, Senior Reactor Analyst, Division of Reactor Safety

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315-02-03-01	NCV	Failure to implement adequate foreign material exclusion controls resulted in degradation of Unit 1 West ESW pump
50-316-02-03-02	NCV	Containment isolation valve alignment error during local leak rate testing resulted in inoperable containment penetration during refueling and violation of TS 3.9.4.c
50-315-02-03-03	NCV	Pressurizer power operated relief valves inoperable due to mis-positioned control switches
50-316-02-03-04	NCV	Failure of lower containment airlock door interlock and failure to follow instructions resulted in inadvertent opening of both airlock doors
50-315-02-03-05	NCV	Failure to implement all intended radiological engineering controls during steam generator eddy current testing, as required by 10 CFR 20.1701
50-316-02-03-06	NCV	TS 3.9.4.c was violated during core alterations when containment isolation valve (2-XCR-101) was stroked open for testing
50-315-02-03-07	NCV	Failure to measure Unit 1 lower ice condenser inlet door opening torque and closing torque in accordance TS requirements
50-316-02-03-08	URI	Review of NOED-02-3-01 regarding D.C. Cook, Unit 2, compliance with TS 3.8.2.3
50-316-02-03-09	NCV	Failure to provide work instructions appropriate to the circumstances for adjustment of stem lock nut on 2-CS-369
50-315/316-02-03-10	NCV	Failure to adequately control access to a locked high radiation area for the radioactive waste drumming room
50-315/316-02-03-11	NCV	Failure to adequately control access to a locked high radiation area for the demineralizer valve gallery

Closed

50-315-02-03-01	NCV	Failure to implement adequate foreign material exclusion controls resulted in degradation of Unit 1 West ESW pump
50-316-02-03-02	NCV	Containment isolation valve alignment error during local leak rate testing resulted in inoperable containment penetration during refueling and violation of TS 3.9.4.c
50-315-02-03-03	NCV	Pressurizer power operated relief valves inoperable due to mis-positioned control switches
50-316-02-03-04	NCV	Failure of lower containment airlock door interlock and failure to follow instructions resulted in inadvertent opening of both airlock doors
50-315-02-03-05	NCV	Failure to implement all intended radiological engineering controls during steam generator eddy current testing, as required by 10 CFR 20.1701
50-316-2002-001-00	LER	Containment Isolation valve alignment error during local leak rate testing
50-315-2002-002-00	LER	Pressurizer power operated relief valves inoperable due to control switch position
50-316-2002-002-00	LER	TS 3.9.4.c was violated during core alterations
50-316-02-03-06	NCV	TS 3.9.4.c was violated during core alterations when containment isolation valve (2-XCR-101) was stroked open for testing
50-316-2001-002-00	LER	Failure of lower containment airlock door interlock results in inadvertent opening of both doors
50-316-2001-002-01	LER	Failure of lower containment airlock door interlock results in inadvertent opening of both doors
50-315-01-20-08-00	URI	Failure to adequately measure the ice condenser lower inlet door opening torque and closing torque in accordance with TS requirements
50-315-02-03-07	NCV	Failure to measure Unit 1 lower ice condenser inlet door opening torque and closing torque in accordance TS requirements
50-315-2002-004-00	LER	Unit 1 ice condenser lower inlet door test failure
50-315-2002-001-00	LER	Failure to perform ice condenser door testing in accordance with TS

Closed

50-316-2002-003-00	LER	2AB 250 volt D.C. battery inoperable for longer than allowed by plant's TS
50-316-02-03-08	URI	Review of NOED-02-3-01 regarding D.C. Cook, Unit 2, compliance with TS 3.8.2.3
50-316-02-02-01	URI	Failure to perform adequate maintenance and testing on valve 2-CS-369 resulted in gas binding the Unit 2 West centrifugal charging pump
50-316-02-03-09	NCV	Failure to provide work instructions appropriate to the circumstances for adjustment of stem lock nut on 2-CS-369
50-315/316-00-19-02	URI	Potentially non-conservative engineered ventilation TS 3.7.6.1
50-315/316-02-03-10	NCV	Failure to adequately control access to a locked high radiation area for the radioactive waste drumming room
50-315/316-02-03-11	NCV	Failure to adequately control access to a locked high radiation area for the demineralizer valve gallery

Discussed

50-316-01-20-07	NCV	Failure to adequately measure the ice condenser lower inlet door opening torque and closing torque in accordance with TS requirements

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
AEP	American Electric Power
AFW	Auxiliary Feedwater
ALARA	As Low As Is Reasonably Achievable
ANSI	American National Standards Institute
ATR	Administrative Technical Requirement
ATWS	Anticipated Transients Without Scram
ASME	American Society of Mechanical Engineers
CCP	Centrifugal Charging Pump
CDF	Core Damage Frequency
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Regulations
CR	Condition Report
DAC	Derived Air Concentration
DAW	Dry Active Waste
D. C.	Direct Current
DCP	Design Change Procedure
DG	Diesel Generator
DIT	Design Information Transmittal
DOT	Department of Transportation
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EHP	Electrical Maintenance Head Procedure
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ESW	Essential Service Water
FFD	Fitness-for-Duty
FME	Foreign Material Exclusion
gpm	gallons-per-minute
IEEE	Institute of Electrical and Electronics Engineers
IHP	Instrument Maintenance Head Procedure
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
ISI	Inservice Inspection
LER	Licensee Event Report
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LOP	Loss-of-Off-site Power
MHP	Maintenance Head Procedure
MT	Magnetic Particle Examination
MVAR	Megavolt-Amperes Reactive
NCV	Non-Cited Violation
NI	Nuclear Instrument
NOED	Notice of Enforcement Discretion
NRC	Nuclear Regulatory Commission
OA	Other Activities

OHP	Operations Head Procedure
PADS	Personnel Access Data System
PARS	Publically Available Records
PCP	Process Control Program
PI	Performance Indicator
psid	pounds-per-square-inch differential
psig	pounds-per-square-inch gauge
PMI	Plant Manager’s Instruction
PMP	Plant Manager’s Procedure
PORV	Power Operated Relief Valve
PPC	Plant Process Computer
PT	Die Penetrant Examination
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
Radwaste	Radioactive Waste
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RP	Radiation Protection
RWP	Radiation Work Permit
SDP	Significance Determination Process
SI	Safety Injection
SPP	Special Plant Procedure
SRA	Senior Reactor Analyst
SSC	Structures, Systems, and Components
STP	Surveillance Test Procedure
TEDE	Total Effective Dose Equivalent
TS	Technical Specification
U1C18	D. C. Cook Unit 1, 18th Refueling Outage
U2C13	D. C. Cook Unit 2, 13th Refueling Outage
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Examination

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.

<u>1R01</u> <u>Adverse Weather</u>

Plant Manager's Procedure (PMP) 2080 SWM.001	Severe Weather Guidelines	Revision 0
12-OHP 4022.001.010	Severe Weather	Revision 0
12-OHP 4022.001.009	Seiche	Revision 0
Calculation MD-12-SCRN-001-N	Screen House Internal Flood Levels	Revision 0
Condition Report (CR) 00-11073	NRC Identified That Entry Conditions for Severe Weather Procedure Could Be Overly Restrictive	August 9, 2001
CR 01194005	1-HV-CIR-3 Is Not Maintaining Instrument Room Temperature	July 13, 2001
CR 01213055	East ESW [Essential Service Water] Pump Room Temperature Alarm Is Standing on a Normal Summer Day	August 1, 2001
CR 01297117	Control Rooms Were Not Notified That a Tornado Watch and Warning Had Been Issued	October 24, 2001
CR 02152042	Evaluate West ESW Pump Room Ventilation and High Temperature Alarm	June 1, 2002
CR 02174006	NRC Identified That Unit 1 Switchgear Drain Cover Is Broken	June 21, 2002
CR 02174008	NRC Identified Fouling of Unit 2 Switchgear Ventilation Inlet Screens and Questioned Timeliness of Installation of Switchgear Vent Hood Proposed Modification	June 21, 2002

1R04 Equipment Alignment

Unit 2 Turbine Driven and East Motor Driven Auxiliary Feedwater (AFW) Pumps

	D. C. Cook Nuclear Plant Updated Final Safety Analysis Report (UFSAR)	
PMP 5020.RTM.001	Restraint of Transient Material	Revision 1
12-MHP-5021.SCF.001	Scaffolding Guidelines	Revision 0b
01-OHP-5030.001.001	Operations Plant Tours	Revision 19b
02-OHP-4030.STP.017E	East Motor Driven Auxiliary Feedwater System Test	Revision 10
02-OHP-5030-001-001	Operations Plant Tours	Revision 19a
DB-12-AFWS	Auxiliary Feedwater System Design Basis Document	Revision 0
Flow Diagram OP-2-5106-45	Auxiliary Feedwater	Revision 45

Unit 2 Train AB and CD Station Batteries

Technical Specification (TS) 3.8.2.3	D.C. Distribution - Operating	Amendment 249
UFSAR Section 8.0	Electrical Systems	Revision 17.2
CR 02116008	NRC Identified Minor Discrepancies in the Unit 2 Station Battery Rooms	April 25, 2002
CR 02116010	NRC Identified That the Gaitronics Speaker in the Unit 2 CD Battery Room Is Not Functioning Properly	April 25, 2002

Unit 2 Circulating Water System

Flow Diagram 02-12-5119-51	Circulating Water, Priming System and Screenwash Units 1 and 2	Revision 51
02-OHP-4021-057-001	Circulating Water System Operation	Revision 20
02-OHP-4021-057-002	Placing In/Removing from Service the Circulating Water Deice System	Revision 8a
02-OHP-4024-223	Annunciator 223 Response: Circulating Water	Revision 7

CR 01271065	Circulating Water Pump PP-21 Discharge Shutoff Valve 2-WMO-21 Is Not Consistently Staying in Manual Operation	September 28, 2001
CR 02113041	Unit 2 Operations Does Not Appear to Have the Seal Injection Filters Valved in Correctly	April 23, 2002
CR 02114035	Configuration Control Issue Related to Misalignment of Unit 2 Seal Water Injection Filters Following Unit 2 Refueling	April 24, 2002
CR 02120087	The Incorrect Main Condenser Waterbox Was Removed for Service Due to Improper Sample Collection	April 29, 2001

<u>1R05</u> <u>Fire Protection</u>

UFSAR, Section 9.8.1	Fire Protection System	
	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
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
Plant Manager's Instruction (PMI) 2270	Fire Protection	Revision 26
Fire Training Exercise 21	Radiological Control Area [RCA] Access 609 Foot Elevation Auxiliary Entry/Exit Area	April 10, 2002
CR 02131015	During Troubleshooting of ERS-2300, VRA-2310 Was Rendered Inoperable Without the Control Rooms Knowledge and Without Entering the Appropriate TS	May 11, 2002

<u>1R06</u> <u>Flood Protection</u>

	D. C. Cook Nuclear Plant UFSAR	

Calculation MD-12-SCRN-001-N	Screen House Internal Flood Levels	Revision 0 April 30, 2000
12-OHP 4022.001.009	Seiche	Revision 0
NRC Information Notice 2002-12	Submerged Safety-related Electrical Cables	March 21, 2002
CR P-99-07656	ESW Strainer Backwash Outlet Shutoff Valves Are Below the Flood Protected Level of 595 Feet	April 6, 1999
CR 01323022	Program Controls for Protection Against Plant Flooding Need to Be Reviewed for Adequacy and Understanding by Plant Personnel	November 19, 2001
CR 02088011	Tracking CR for Development of a Design Basis Document for Flood Protection	March 29, 2002

<u>1R07</u> <u>Heat Sink Performance</u>

12-MHP-5021-005-009	Heat Exchanger Tube Plugging	Revision 2
12-MHP-5030-016-001	Component Cooling Water Heat Exchanger Inspection, Cleaning and Tube Plugging	Revision 4
AEP-BOP-208-ET	D. C. Cook Component Cooling Water Heat Exchanger Eddy Current Testing	Revision 0
Generic Letter 89-13	Service Water System Problems Affecting Safety-Related Equipment	July 18, 1989
Job Order R0221325	1-HE-15E, Inspect and Clean Heat Exchanger as Required	May 26, 2002
Job Order C0178129	Perform Eddy Current Testing for Component Cooling Water Heat Exchanger 1-HE-15E	May 23, 2002
CR 02126069	ESW Lines to 1-HV-AFP-T1AC	May 6, 2002
CR 02138028	Found the Divider Plate in 1-HE-15E Bowed Approximately 1/2 Inch From Inlet to Outlet and 3/8 Inch When Measured From Tubesheet to Cover Plate	May 18, 2002

CR 02138031	While Performing a Generic Letter 89-13 Inspection, Found Sea Grass and Sand Fouling the Return End of the 1-HE-15E Tubesheet	May 18, 2002
CR 02143053	Two Additional Tubes Require Plugging in the East Component Cooling Water Heat Exchanger	May 23, 2002
CR 02143088	Eddy Current Testing Was Performed on 100 Percent of the Tubes in 1-HE-15E. As a Result of this Testing, 27 Tubes Were Plugged	May 23, 2002

1R08 Inservice Inspection

1278909A	BWI Replacement Steam Generator Secondary Side Inspection Procedure	January 28, 2002
51-5004764-03	D. C. Cook Units 1 and 2 Appendix H Review	April 17, 2002
MDS-609	Steam Generator Tube Plugging	May 16, 2002
01-EHP-5037-SGP-003	Steam Generator Primary Side Inspections	May 1, 2002
	Site Specific Eddy Current Data Analysis Guidelines D. C. Cook Nuclear Plant Unit 1	May 13, 2002
SGP-DA-U1-C18	Steam Generator Degradation Assessment - Unit 1 Cycle 18	May 10, 2002
83A6218	Ultrasonic Examination Procedure for Ferritic Piping Welds and Vessels (Less Than or Equal) 2 Inches Thickness for Cook Nuclear Plant	August 30, 2001
83A6118	Magnetic Particle Examination for D. C. Cook Nuclear Plant	March 18, 2002
83A6228	Ultrasonic Examination Procedure for Austenitic Piping and Vessels (Less than or Equal) 2 Inches Thickness	March 18, 2002
83A6108	Liquid Penetrant Examination for D. C. Cook Nuclear Plant	August 30, 2001
80A9055	Thermometer Check Record	April 18, 2002

AR 02115006	Identified Discrepant Conditions During ISI [Inservice Inspection] Examinations on 1-GRH-V-14	April 25, 2002
AR 02115008	Identified Discrepant Conditions During ISI Examinations on 1-GSI-R-50	April 25, 2002
AR 02115016	Identified Discrepant Conditions During ISI Examinations on 2-GCCW-S-843	April 25, 2002

<u>1R12</u> <u>Maintenance Rule Implementation</u>

PMP 4030-001-001	Impact of Safety Related Ventilation on the Operability of Technical Specification Equipment	Revision 4
	Maintenance Rule (a)(1) Action Plan Diesel Generator Ventilation System	Revision 1 January 24, 2002
	Maintenance Rule Scoping Document Diesel Generator Room Ventilation System (VDG)	Revision 2 February 28, 2002
CR 01152061	Apparent Failure of 2-HV-SGR-MD-2 Damper to Open Created a High Temperature Condition in the CRID and Control Rod Drive Equipment Rooms	June 1, 2001
CR 01191011	Diesel Generator Tempering Damper 1-HV-DDP-CD-1 Does Not Function Properly	July 10, 2001
CR 01194022	Maintenance Rule Review for Diesel Generator Ventilation System Was Not Adequate	July 13, 2001
CR 01194029	Maintenance Rule Review for Diesel Generator Ventilation System Was Not Adequate	July 13, 2001
CR 01199073	Unit 1 CD Diesel Generator Supply Fan Tempering Damper 30 Percent Open with Outside Air Temperature Approximately 90 Degrees	July 18, 2001
CR 01207001	2-HV-SGRS-9 Smells Hot/no Air Flow Due to Damper Not Opening	July 26, 2001
CR 01289032	Diesel Generator Tempering Dampers Have No Preventative Maintenance	October 16, 2001

CR 01329018	2AB Diesel Generator Exhaust Tampering Damper, 2-HV-DDP-AB1, Was Discovered with One Louver Detached	November 25, 2001
CR 01331035	Diesel Generator Ventilation System Unavailability Exceeds Maintenance Rule Performance Criteria	November 27, 2001
CR 01341132	Inlet Damper to Unit 2 CRID Fans 2-HV-SGRS-1A and 2-HV-SGRS-4A Appears to Have Failed	December 7, 2001
CR 99-12474	Lack of Preventative Maintenance Program for the Diesel Generator Ventilation Motor Operated Dampers	May 19, 1999

<u>1R13 Maintenance and Emergent Work Control</u>

PMP-2291-OLR-001	On-Line Risk Management	Revision 2
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

<u>Unit 1 Main Generator Output Breaker 1-52-K1 Replacement</u>

PMI-4090	Criteria for Conducting Infrequently Performed Tests or Evolutions, Attachment 1, "Briefing Guide for Removal of Unit 1 345 Kilovolt Output Breaker K1 from Service for Replacement with the Unit On Line"	April 5, 2002
CR 02120049	Unit 1 in an Orange Risk Status on Large Early Release Frequency, With Unit 2 in a Very High Yellow Status Due to Predicted Severe Weather Expected in the Area	April 30, 2002

<u>Unit 2 West Motor Driven AFW Pump</u>

CR 020134025	Failed to Have Quality Control Verify Freedom of Movement of Check Valve	May 14, 2002

PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 41, Week 6	May 12-18, 2002
	Unit 2 Control Room Logs	May 15-17, 2002
	Unit 2 Supervisors Turnover Logs	May 15-17, 2002
	Unit 2 Abnormal Position Log	May 15-17, 2002

<u>Unit 1 Turbine Driven AFW Pump</u>

PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 41, Week 1	April 7-13, 2002

<u>Replacement of the Unit 2 East and Unit 1 West Essential Service Water Pumps</u>

PMI-2220	Foreign Material Exclusion	Revision 11
PMP 2220-001-001	Foreign Material Exclusion (FME)	Revision 2a
PMP 2291-OLR.001 Data Sheet 1	Work Schedule Review and Approval Form Cycle 41, Week 10 and Cycle 41, Week 11	
12 MHP 5021.DIV.002	Divers Safety Net Installation and Restoration	Revision 2
CR 01048011	An 8 Foot Piece of Herculite Was Dropped in the Unit 1 Forebay While Work Was Being Performed in the 1-WMO-13 Pit.	February 16, 2001
CR 01093002	Herculite Found Between the Inlet and Grating Inside the Unit 2 East Main Feed Pump Condenser Water Box	April 3, 2001
CR 02175037	Step Change in Unit 1 West ESW Pump Performance with No Associated Flow Change	June 24, 2002
CR 02176058	FME - Red Danger Barrier Tape Was Found in the Suction Bell of the Unit 1 West ESW Pump	June 25, 2002

1R14 Personnel Performance During Nonroutine Evolutions

1R14.1 Containment Isolation Valve Alignment Error During Local Leak Rate Testing

Licensee Event Report (LER) 50-316-2002-001-00	Containment Isolation Valve Alignment Error During Local Leak Rate Testing	March 28, 2002
CR 02027006	Valve 2-GPX-301-V1 Was Misaligned During the Performance of Step 53 of 02-EHP-4030-234-203, Unit 2 B&C Leak Rate Testing that Violated Containment and Refueling Integrity While Fuel Movement Was in Progress	January 27, 2001

1R14.2 Pressurizer Power Operated Relief Valves (PORVs) Inoperable Due to Mis-Positioned Control Switches

LER 50-315-2002-002-00	Pressurizer Power Operated Relief Valves Inoperable Due to Control Switch Position	April 19, 2002
01-OHP-4023.FR-S.1	Response to Nuclear Power Generation ATWS [Anticipated Transient Without Scram]	Revision 7
CR 02050022	The Automatic Function of All Three Unit 1 PORVs Was Defeated	February 19, 2002

1R14.3 Failure of Lower Containment Airlock Door Interlock and Failure to Follow Instructions Resulted in Inadvertent Opening of Both Airlock Doors

LER 50-316-2001-002-00	Failure of Lower Containment Airlock Door Interlock Results in Inadvertent Opening of Both Doors	March 16, 2001
LER 50-316-2001-002-01	Failure of Lower Containment Airlock Door Interlock Results in Inadvertent Opening of Both Doors	October 26, 2001
PMP 4010.CAC.001	Containment Access and Cleanliness	Revision 0
Job Order R0202758	Perform Preventive Maintenance Task 24, Steps 1.0 thru 1.11.8	October 31, 2000
Job Order R0210015	Perform 6-Month Airlock Preventive Maintenance Task	April 17, 2001

CR 01023054	While Leaving Unit 2 Lower Containment Through the Lower Containment Airlock, the Inner Airlock Door Was Able to Be Opened While the Outer Airlock Door Was Also Opened	January 23, 2001
CR 01023055	Lower Containment Airlock Interlock Did Not Prevent Opening the Inner Door While the Outer Door Was Open	January 23, 2001

<u>1R14.4</u> <u>Unit 1 Power Reduction to Support Repairs to Unit 1 Main Generator Breaker K1 Disconnect</u>

CR 02115039	Insulator for Disconnect for K1 Breaker Damaged During Maintenance	April 25, 2002
CR 02115030	Rod Control Non-urgent Failure Alarm Received in Unit 1 During Down Power Operation	April 25, 2002

<u>1R14.5</u> <u>Unit 1 Reactor Trip and Restart Following Loss of Main Feed Pump Vacuum</u>

NRC Event Notification 38993	Manual Reactor Trip from 88 Percent Power	June 14, 2002
PMP 4010.TRP.001 Data Sheet 1	Unit One Reactor Trip Review Report (June 14, 2002)	June 15, 2002
CR 02165064	Manual Reactor Trip Due to Loss of East Main Feedwater Pump	June 14, 2002
CR 02166009	Unit 1 Reactor Trip Resulted in Excessive Cooldown	June 14, 2002
CR 02166016	Thermal Overload Tripped on 1-MRV-230 Hydraulic Actuator	June 15, 2002
CR 02165063	Turbine Driven AFW Pump Speed Oscillates Approximately 200 Revolutions-per-Minute While Running	June 14, 2002

<u>1R14.6</u> <u>Unit 2 Station Battery 2AB Cell Cracking Operator Response</u>

	Unit 2 Control Room Logs	April 23, 2002
CR 02113067	Crack Found on Unit 2 AB Battery Cell 31	April 23, 2002

1R14.7 Unit 2 Reactor Trip, May 13, 2002

CR 02133031	2-RU-27 Failed Low After Unit 2 Reactor Trip	May 13, 2002
CR 02133034	2-MRV-240 Started to Drift Closed After Reactor Trip	May 13, 2002
CR 02133048	2-DRV-250 the Bleed Steam Drain Valve for the 5A Heater Failed to Open Automatically During a Turbine Trip	May 13, 2002

1R15 Operability Evaluations

Unit 1 Ice Basket 24-1-7 As-Found Weight Below TS Requirements

12-EHP-4030-010-262 Data Sheet 6	Ice Condenser Surveillance and Operability Evaluation - Expanded Ice Weighing Results for Basket 24-1-7	May 14, 2002
CR 02134066	As-found Ice Basket Weighing Surveillance - the As-found Weighing Results for Ice Basket 24-1-7 Is below the TS Minimum Required Amount	May 14, 2002
CR 02115002	Unit 1 Ice Basket 24-1-7 As-found Weight Was below the TS Limit and Structural Analysis Limit	April 25, 2002
Calculation SD-990826-003	Ice Condenser Ice Basket Design	Revision 0

2-FW-160, West Motor Driven AFW Pump Emergency Leakoff Check Valve Leaked by During the Performance of Test 02-OHP-4030.STP.017E

CR 02136014	2-FW-160, West Motor Driven AFW Pump Emergency Leakoff Check Valve Leaked by During the Performance of Test 02-OHP-4030.STP.017E	
02-OHP-4030.STP.017E	East Motor Driven Auxiliary Feedwater System Test	Revision 10

Non-Seismic Scaffolding Built in the Vicinity of 2AB DG [Diesel Generator] and 2CD DG Components

CR 02109003	Non-Seismic Scaffolding Built in the Vicinity of 2AB DG and 2CD DG Components	April 19, 2002

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

Velan Valve Corporation Letter to the NRC	10 CFR Part 21 Notification for Potential Safety Related Problem With 2½-Inch, 3-Inch, and 4-Inch Forged Swing Check Valves	January 18, 1991
CR 96-0094	Assess the Applicability, Significance, and Probability for an Event Similar to Operating Experience 7640 Occurring at Cook Nuclear Plant	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 02138029	The Extent of Condition Which Was Originally Identified Under CR 02134021 Against Four Unit 2 Charging Line Check Valve Failures Is Considered to Extend to All 3-Inch Velan Valves of Model B10-3114B-13M	May 18, 2002

Unit 2 Steam Stop Valve 2-MRV-220 Detent Bar/guide Rod Bushing Has Fallen Out

CR 02137063	Steam Stop Valve 2-MRV-220 Detent Bar Guide Has Fallen out	May 17, 2002
CR 02172042	NRC Identified That a Work Request Is Needed to Replace 2-MRV-220 Bushing During the Next Refueling Outage	June 21, 2002

Degraded and Nonconforming Conditions Remaining After Refueling Outage U1C18

	D. C. Cook Nuclear Plant Unit 2 Technical Specifications	
	D. C. Cook Nuclear Plant Updated Final Safety Analysis Report	
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
Calculation EVAL-MD-12-RHR-905-N	Residual Heat Removal Shutdown Cooling Line Vibration Fatigue Evaluation	Revision 0
Safety Evaluation 2000-1534-00	Unit 1 Residual Heat Removal System Restart Assessment	August 23, 2000
CR P-99-02455	Residual Heat Removal Pumps May Be Experiencing Cavitation	February 11, 1999
CR 02108069	Design Change 1-DCP-720 Has Been Removed From the Scope of U1C18. This Will Delay Resolution of the Operability Issue Documented in CR 99-2455	April 18, 2002
CR 02123015	This CR Is Written to Document an Aggregate Operability Determination to Support Unit 1 Restart Following the U1C18 Refueling Outage	May 3, 2002

Miscellaneous Condition Reports Reviewed

CR 02124008	While Performing 01-OHP 4030-102-060 (Pressurizer Relief Valve Testing) for 1-NRV-152 on Step 4.26.2 the Acceptance Criteria Was Not Met	May 4, 2002
CR 02126093	Starting Air Check Valve Sealing Surface Defect on the 6R Cylinder Head	May 6, 2002

1R19 Post Maintenance Testing

Job Order C0051164, Replace 1-BATT-CD During Year 2002 Outage

UFSAR Chapter 8	Electrical Systems	Revision 17.2
VTD-CDBA-001	C&D Technologies Standby Battery Vented Cell Installation and Operating Instructions	Revision 2
Purchase Order NU04-0000020621	C&D Technologies Certificate of Compliance	May 9, 2002
Institute of Electrical and Electronics Engineers (IEEE) Standard 450-1995	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications	May 31, 1995

American National Standards Institute (ANSI)/IEEE Standard 484-1987	IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations	May 18, 1987
12-IHP 5021-EMP-006	Battery Cell Replacement	Revision 2
CR 02143005	Electro Alarm for Annunciator Panel 120, Drop 101 Did Not Work Correctly During Battery Draw Down Testing	May 23, 2002
12-IHP-4030-082-003	AB, CD, and N-Train Battery Discharge Test and 18 Month Surveillance Requirements	Revision 2
Job Order C0051164	Replace 1-BATT-CD During Year 2002 Outage	
Job Order R0209107	Perform 1-BATT-CD 18 Month Surveillance	

<u>Design Change Procedure (DCP) 4504, Replace Reserve Auxiliary Transformers 101AB and 102CD with Load Tap Changing Transformers</u>

Job Order 01159017	1-DCP-4504/Replace Unit 1 TR101AB	
Job Order 01159019	1-DCP-4504/Replace Unit 1 TR101CD	
01 OHP 4030-182-026 Attachment 1	Auxiliary Power Transfer Test Surveillance Procedure, Automatic Transfer of Reactor Coolant Buses to Reserve Feed By Simulated or Intended Unit Trip	Revision 0a, Performed May 30, 2002
01 OHP 4030-132-217A	DG1CD Load Sequence & ESF [Engineered Safety Features] Testing	Revision 2
01 OHP 4030-132-217B	DG1AB Load Sequence & ESF Testing	Revision 2
1-DCP-4504-TP-1	Reserve Auxiliary Transformer 101AB Functional Test	Revision 0
1-DCP-4504-TP-2	Reserve Auxiliary Transformer 101CD Functional Test	Revision 0
1-DCP-4504-TP-3	Bus T11A and DCP-4505 Relay Change Out Functional Test	Revision 0
1-DCP-4504-TP-4	Bus T11D and DCP-4505 Relay Change Out Functional Test	Revision 0

1-DCP-4504	Replace Auxiliary Transformers 101AB and 101CD with Load Tap Changing Transformers	Revision 0

<u>DCP 4504, Install New Undervoltage Protection Relays</u>

Job Order R0228922	Perform 1-IHP-6030-IMP-309 4kV Bus Undervoltage Relay Calibration	May 23, 2002
Job Order R0229719	Perform 1-IHP-6030-IMP-309 4kV Bus Undervoltage Relay Calibration	June 4, 2002
01 IHP 6030-IMP-309	4KV Bus Loss of Voltage and 4KV Bus Degraded Voltage Relay Calibration	Revision 5
CR 02154052	Degraded Voltage Relay 1-27-T11A1 Failed to Actuate During Train B LOP [Loss of Off-site Power]/LOCA Testing	June 3, 2002
01 OHP 4022.082.004	Degraded Offsite AC Voltage Response	Revision 1
01 OHP 4024-119	Annunciator #119 Response: Station Auxiliary AB	Revision10
01 OHP 4024-120	Annunciator #120 Response: Station Auxiliary CD	Revision xx
01 OHP 4024-121	Annunciator #121 Response: Generator	Revision 17
01 OHP 4023-SUP-010	Starting Reactor Coolant Pumps	Revision 1
01 OHP 4021-002-003	Reactor Coolant Pump Operation	Revision 14
TS Table 3.3-4, Functional Unit 8	Engineered Safety Feature Actuation System Instrumentation Trip Setpoints	Amendment 268

<u>Job Order 02093039, Unit 2AB Station Battery Cell 46 Replacement</u>

UFSAR Chapter 8	Electrical Systems	Revision 17.2
IEEE Standard 450-1995	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications	May 31, 1995
ANSI/IEEE Standard 484-1987	IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations	May 18, 1987

12-IHP-5021-EMP-006	Battery Cell Replacement	Revision 2 Change 3
12-IHP-4030-082-001	AB, CD and N Train Battery Weekly Surveillance and Maintenance	Revision 0 Change 1
Job Order 02093039	Unit 2AB Station Battery Cell 46 Replacement	

Unit 1 TDAFP Maintenance

Job Order 01333060	Replace Valve 1-MS-326 By Welding	April 11, 2002
Job Order 01303055	Repair Trap 1-T-76-2 Lack of Flow	April 12, 2002
Job Order 0130357	Repair Trap 1-T-76-1 Lack of Flow	April 12, 2002
Job Order R0221579	1-PP-4- Lube Pump Bearings and Coupling, Sample Oil	April 12, 2002
Job Order R0212013	1-T-132 Perform Steam Trap Internal Inspection	April 12, 2002
Job Order R0221666	1-QT-506 Generic Letter 89-10 Perform External Preventive Maintenance	April 11, 2002
Specification DCCPV102QCS	Shop and Field Fabrication and Erection of Conventional Piping	Revision 11
12-IHP-5030-EMP-001	Limitorque Valve Operator Preventive Maintenance	Revision 4 Change 2

1R20 Refueling and Outage Activities

	D. C. Cook Nuclear Plant Unit 2 Technical Specifications	
	D. C. Cook Nuclear Plant Updated Final Safety Analysis Report	
01-OHP-4021-001-004	Plant Cooldown From Hot Standby to Cold Shutdown	Revision 36
01-OHP-4030-114-030	Daily and Shiftly Surveillance Checks	Revision 0
12-OHP-4050-FHP-001	Refueling Procedure Guidelines	Revision 3
12-OHP-4050-FHP-005	Core Unload/Reload and Incore Shuffle	Revision 3
12-OHP-4050-FHP-023	Reactor Vessel Head Removal With Fuel in the Vessel	Revision 0

12-OHP-4050-FHP-026	Upper Internals Removal With Fuel in the Vessel	Revision 1
01-OHP-4030-STP-041	Refueling Integrity	Revision 8
PMP 4100-SDR-001	Plant Shutdown Safety and Risk Management	Revision 5, C1
	Daily Shift Manager's Logs	May 3, 2002 through June 9, 2002
	U12C18 Outage Schedule Shutdown Risk Review	
01 OHP 4021-017-002	Placing In Service The Residual Heat Removal System	Revision 16
01 OHP 4021-001-004	Plant Cooldown From Hot Standby To Cold Shutdown	Revision 36
01-EHP-4030-102-386	Multiple Rod Drop Measurements	Revision 0a
01-OHP-4021-001-002	Reactor Startup	Revision27a
12-EHP-4030-002-356	Low Power Physics Tests with Dynamic Rod Worth Measurement	Revision 0b
1-DCP-5075	Unit 1 Cycle 18 Reload Cord Design	Revision 0
CR 02111020	Generator Voltage Dropped From 116 Volts to 110 Volts	April 21, 2002
CR 02118009	MVAR's [Megavolt-Amperes Reactive] Dropped From 60 in to 700 in While Attempting to Raise the Main Generator Voltage	April 28, 2002
CR 02124001	1-MRV-240 (Number 4 Main Steam Stop Valve) Drifted Open Just After Reactor Trip	May 4, 2002
CR 02124003	Unit 1 Main Turbine High Vibration After Manual Reactor Trip Required Partial Condenser Vacuum Breaking	May 4, 2002
CR 02124004	Two Steam Plums Coming From the Fitting on the Top of the Transmitter Approximately 1 to 2 Feet Long	May 4, 2002

CR 02124023	Four Control Rod Bottom Lights Failed to Illuminate Following the Reactor Trip to Enter the Unit 1, Cycle 18 Refueling Outage	May 4, 2002
CR 02125005	"B" Reactor Trip Breaker Control Switch was Inadvertently Turned to the "Close" Instead of the "Trip" Position	May 5, 2002
CR 02037026	2-PW-275 Is Not Expected to Performed Adequately as a Containment Isolation Valve Throughout the Next Cycle	February 6, 2002
CR 02114043	Design Change Number 12-RFC-2718 Was Initiated in 1989 to Replace Carbon Steel Valve Studs	April 24, 2002
CR 02123056	Pipe Cap Leak in Containment Annulus Quad 4	May 3, 2002
CR 02123059	Brown Oily-like Substance Leaking from Overhead in Accumulator Number 2	May 5, 2002
CR 02124047	An Oxygen Alarm Received While Venting Nitrogen From the Accumulators in the Unit 1 Annulus	May 4, 2002
CR 02127075	NRC Identified the Shutdown Risk Reporting Database in Lotus Notes Is Missing a Row From the Safety Function Table	May 7, 2002
CR 02130012	1-DCR-304 Process was Breached Without Sufficient Clearance Protection	May 10, 2002
CR 02131015	During Troubleshooting of ERS-2300, VRA-2310 Was Rendered Inoperable Without the Control Rooms Knowledge and Without Entering the Appropriate TS	May 11, 2002
CR 02133015	Improper Lineup on SI [Safety Injection]/Charging Suction Lead to the Volume Control Tank and Refueling Water Storage Tank Being Operated in a Cross-Tied Configuration	May 13, 2002
CR 02134003	Fuel Assembly GG02 Identified as Leaking by In-mast Fuel Sipping	May 14, 2002

CR 02134039	U1 Refueling Water Sequence Was Initiated when a Drain (1-CS-348) Was Opened Prior to the CCP [Centrifugal Charging Pump] Suction Valves (1-IMO-910/911) De-energized	May 14, 2002
CR 02135066	Debris Was Found and Then Removed on Top of the Reactor Head During Remote Visual Inspection	May 15, 2002
CR 02136042	Inactive/Passive Boric Acid Was Found on the Top of the Reactor Vessel Head	May 16, 2002
CR 02136095	In-core Fuel Source Secondary Source (SS17) Has Been Found with on Finger Missing 4-5 Feet in Length	May 16, 2002
CR 02141047	During U1C18 Steam Generator Eddy Current Inspections 4 Tubes Were Identified with Abnormal Eddy Current Signals	May 22, 2002
CR 02143095	NRC Identified That Reactor Operator Returned Control Power to Residual Heat Removal Pump Refueling Water Storage Tank Suction Contrary to Procedural Guidance	May 5, 2002
CR 02159004	NRC Identified Concerns Associated with Access Controls to Reactor Vessel Head During Sub-critical Control Rod Withdrawals	June 7, 2002
CR 02177047	NRC Identified Adverse Housekeeping and Work Practice in the Control Room Back Panel Areas on June 7, 2002. A CR Was Not Promptly Written to Trend This Condition	June 26, 2002

1R22 Surveillance Testing

12 MHP 4030-10-03, "Ice Condenser Lower Inlet Door Surveillance"

12 MHP 4030-010-003	Ice Condenser Lower Inlet Door Surveillance	Revision 2a
CR 02149034	NRC identified that the test acceptance criteria for the ice condenser lower inlet door testing was incorrectly determined	May 29, 2002

CR 02132047	Unit 1 lower ice condenser inlet door 15 right failed the 40 degree opening force test	May 12, 2002
CR 02133017	Unit 1 lower ice condenser inlet door 4 right failed the 40 degree opening force test	May 13, 2002
CR 02136018	Unit 1 lower ice condenser inlet door 21 right door spring dragging	May 16, 2002
DIT S-00105-02	Ice Condenser Lower Inlet Door Surveillance Requirements and Basis	Revision 2
	Donald C. Cook Nuclear Plant, Unit 1 - Issuance of Amendment RE: Ice Condenser Lower Inlet Doors (TAC Number MB3989)	February 14, 2002

<u>01-OHP-4030-108-008R, Attachment 8, "Accumulator Check Valve Test"</u>

01-OHP-4030-108-008, Attachment 8	Accumulator Check Valve Test	Revision 0
PMI 5070	Inservice Testing	Revision 1
Westinghouse Nuclear Safety Advisory Letter NSAL-02-6	Nitrogen Release to Residual Heat Removal During SI Accumulator Low Pressure Blowdown Tests	April 8, 2002
Westinghouse Letter LTR-SEE-02-110	AEP [American Electric Power] Units 1 and 2 Accumulator Check Valve Blowdown Test Report	April 24, 2002
Design Information Transmittal (DIT) S-00885-03	Information Requested by Westinghouse to Support Analysis of Proposed Accumulator Check Valve Blowdown Testing	April 18, 2002
DIT B-02320-02	Engineering Limitations on Conduct of Accumulator Check Valve Testing	May 1, 2002

<u>01-OHP-4030-STP-017R, "Auxiliary Feedwater Pump Time Response Test"</u>

01-OHP-4030-STP-017R	Auxiliary Feedwater Pump Response Time	Revision 9
PMP 4030.TRT.001	Time Response and Verification of Engineered Safety Features	Revision 2, Change 7

02-OHP-4030-214-029, Attachment 1, "PPC [Plant Process Computer] Derived Reactor Thermal Power Evaluation" and 02-OHP-4030-214-029, Attachment 4," Power Range NI [Nuclear Instruments] Adjustments"

02-OHP-4030-214-029, Attachment 1	PPC [Plant Process Computer] Derived Reactor Thermal Power Evaluation	Revision 1
02-OHP-4030-214-029, Attachment 4	Power Range NI [Nuclear Instrument] Adjustments	Revision 1
	Daily Shift Manager's Log	April 5, 2002
CR P-00-10476	During Investigation of the June 30, 2000 Event Identified CR 00-9437, the Question Was Raised Regarding Whether the Power Range NIs, Which Were Found to Be Reading Above the TS 2.2.1 Limit, Should Have Been Declared Inoperable	July 26, 2000
CR 01065006	TS 3.0.3 Was Entered Twice Due to NI Trip Set Point Being Greater Than 110 Percent Power	March 6, 2001
CR 02095001	When Performing Power Reduction in Unit 2 Control Room Received Rod Sequence Violation Annunciator	April 5, 2002
CR 02095045	Discovered Three Power Range NIs With a Calculated Trip Greater Than 110 Percent While Performing a Thermal Power Calculation	April 5, 2002
CR 02107060	NRC Identified Sequencing of Sign-offs for Attachments and Listed Acceptance Criteria in Attachment 1 Appear to Indicate That It Is Acceptable to Sign-off the Acceptance Criteria As Satisfactory Prior to Performing Attachment 4 If Required	April 17, 2002

PMI 5070, "Inservice Testing," [Valve Stroke Testing of 1-MCM-221]

PMI 5070	Inservice Testing	Revision 2

12 IHP 4030-082-003, "AB, CD and N Train Battery Discharge Test and 18 Month Surveillance Requirements"

12-IHP-4030-082-003	AB, CD and N Train Battery Discharge Test and 18 Month Surveillance Requirements	Revision 2
Job Order R0209107	Perform 1-BATT-CD 18 Month Surveillance	
IEEE Std 450-1995	IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications	
CR 02143010	While Performing Drawdown on 1-BATT-CD, Redundant Test Data Was Not Acquired Within the Required 2 Minute Time Frame at Test Initiation	May 23, 2002
CR 02151069	NRC Identified That Battery Performance Testing Steps Associated with Comparison to Previous Test Data May Have Been Inappropriately Marked as Not Applicable	May 31, 2002
CR 02151074	NRC Identified That No Process or Procedures Exist to Document or Control the Use of Vendor Testing to Satisfy TS Requirements	May 31, 2002
CR 02143005	Electro Alarm for Annunciator Panel 120, Drop 101 Did Not Work Correctly During Battery Draw Down Testing	May 23, 2002
12-IHP-4030-082-003	AB, CD, and N-Train Battery Discharge Test and 18 Month Surveillance Requirements	Revision 2
Job Order C0051164	Replace 1-BATT-CD During Year 2002 Outage	
Job Order R0209107	Perform 1-BATT-CD 18 Month Surveillance	

01-OHP 4030.001.002, "Containment Inspection Tours"

01 OHP 4030.001.002	Containment Inspection Tours	Revision 17
12 MHP 5040-010-003	Ice Condenser Support Activities	Revision 2

CR 02156016	NRC Identified Small Boric Acid Buildup on 1-RC-102-L2	June 4, 2002
CR 02156014	NRC Identified Small Amount of Boric Acid Buildup on 1-SI-141-L3	June 5, 2002
CR 02156013	NRC Identified Small Amount of Boric Acid Buildup on 1-IMO-130	June 4, 2002
CR 02156012	NRC Identified Small Amount of Boric Acid Buildup on 1-SI-141-L2	June 4, 2002
CR 02154016	NRC Identified Minor Housekeeping and Transient Material Storage Issues in the Auxiliary Building	June 2, 2002
CR 02156023	NRC Identified Standing Water by Ice Condenser Inlet Doors	June 4, 2002
CR 02155082	NRC Identified a Puddle of Liquid Below Steam Generator Snubber 1-OME-3-3-HSD-3L	June 4, 2002
CR 02156008	NRC Identified 1-QPX-200-V1 Has Valve Stem Leakage	June 5, 2002
CR 02156010	NRC Identified 1-IMO-315 Had Boric Acid Buildup on the Stem	June 5, 2002
CR 02156011	NRC Identified 1-QRV-114 Has Boric Acid Build-up on the Stem	June 5, 2002
CR 02156017	NRC Identified Missing Screws From an Electrical Box Cover on the North Wall of the Regenerative Heat Exchanger Room Near 1-QRV-51 and -1-QRV-62	June 5, 2002
CR 02156019	NRC Identified Foreign Material Embedded in Concrete Against Outer Wall in the Overhead	June 5, 2002
CR 02156021	NRC Identified Water Beneath the Number 13 Reactor Coolant Pump	June 4, 2002
CR 02156083	NRC Identified Deficiencies When Inspecting the Unit 1 Containment	June 5, 2002

1R23 Temporary Plant Modifications

D. C. Cook Nuclear Plant Updated Final Safety Analysis Report

Temporary Modification 12-TM-01-23-R0	Install Water Splash Shields on the Unit 1 and Unit 2 AFW Pumps	July 14, 2001
12-EHP-5040-MOD-001	Temporary Modifications	Revision 7a
10 CFR 50.59 Safety Screening 2000-0604-00	AFW Pump Bearing Housing Shield and Shaft Flinger Ring	July 13, 2001
10 CFR 50.59 Applicability Determination 2001-0604-00	Temporary Modification 12-TM-01-23-RO	July 13, 2001
CR 01184086	Unit 1 West Motor Driven AFW Pump Inboard and Outboard Pump Bearing Have Water in the Bearing Reservoirs	July 3, 2001

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

	Cook Nuclear Plant Unannounced Drill Scenario April 16, 2002	Revision 1
CR 02107026	Perform Emergency Preparedness Self-assessment SA-2002-SPS-027, "Second Quarter 2002 Off Hours Emergency Preparedness Drill"	April 17, 2002

<u>2OS1</u> <u>Access Controls For Radiologically Significant Areas</u>

PMP-6010-RPP-003	High, Locked High, and Very High Radiation Area Access	Revision 10
Apparent Cause Evaluation	CR 021101048 Condition Evaluation - Unlocked High Radiation Area in Drumming Room	May 22, 2002 (Draft)
01-OHP-4021-004-001, Attachment 4	South Deborating Demineralizer Operation as Parallel Flow Mixed Bed Demineralizer	Revision 10
Rapid Event Response Investigation Report	Inadequate Radiological Control and Postings	May 22, 2002 (Draft)
	Post Crud Burst Surveys of 617 Foot Demineralizer Gallery	May 5 - 18, 2002
	587 Foot Drumming Room Survey	April 10, 2002

CR 02101048	Door to 587 Foot Drumming Room Posted as Locked High Radiation Area But Card Reader Allowed Door to Open	April 11, 2002
CR 02126004	Locked High Radiation Area Posting	May 5, 2002
CR 02126018	Radiological Posting Around Pressurizer Hatch Didn't Reflect Conditions	May 6, 2002
CR 02126020	Un-posted Locked High Radiation Area After Peroxide Flush	May 6, 2002
CR 02041007	Radiological Area Status Sheets With Inaccurate Information	February 9, 2002

<u>2OS2</u> As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls

	U1C18 Outage ALARA Guide	May 2002
	U1C18 RWP Dose Totals Reports and Daily ALARA Dose Reports/Graphs	May 16 - 24, 2002
	Listing of Outage Generated CRs Coded to RP Issues	May 1 - 23, 2002
PMP-6010.ALA.001	ALARA Program - Review of Plant Work Activities	Revision 11
12-THP-6010-RPP-018	Controls For Radiological Risk Significant Work Activities	Revision 0
12-THP-6010.RPP.006	Radiation Work Permit Processing	Revision 17
RWP 021141 and Associated ALARA Plan	U1C18 Steam Generator Primary Work - Platform Activities	RWP Revisions 0 - 4
TEDE ALARA Evaluations For Steam Generator Platform Work	Install Steam Generator Tube Plugs, Install Diaphragms and Manways, Steam Generator Nozzle Installation and Removal, Decontamination Activities on Platform, ROGER Removal	Various Dates Between April 30 - May 20, 2002
12-THP-6010.RPP.014	Total Effective Dose Equivalent Evaluation	Revision 3(a)
RP Calculation 96-07	Contamination to DAC [Derived Air Concentration] Fraction Conversions	December 10, 1996
ALARA In-Progress Review	U1C18 Steam Generator Primary Work Activities	May 20, 2002

RWP 021136 and Associated ALARA Plan	U1C18 Containment Install, Modify and Remove Scaffold	RWP Revision 1
RWP 021140 and Associated ALARA Plan	U1C18 Steam Generator Manway and Diaphragm Activities	RWP Revision 6
ALARA In-Progress Reviews	U1C18 Steam Generator Manway and Diaphragm Activities	May 15, 19 and 21, 2002
TEDE ALARA Evaluations For Steam Generator Manway and Diaphragm Activities	Steam Generator Manway and Diaphragm Removal, Installation and Support Work	April 30 and May 20, 2002
RWP 021139 and Associated ALARA Plan	U1C18 Valve Maintenance/Repair	RWP Revision 3
ALARA In-Progress Review	Valve Maintenance/Repair	May 15 and 18, 2002
RWP 021134 and Associated ALARA Plan	U1C18 Containment Remove, Reinstall and Modify Insulation	RWP Revision 1
RWP 021149 and Associated ALARA Plan	U1C18 Regenerative Heat Exchanger Activities	RWP Revision 2
ALARA In-Progress Review	Regenerative Heat Exchanger Maintenance	May 15, 2002
CR 02139007	Contamination Event During Eddy Current Testing	May 18, 2002
Rapid Event Response Report	Personnel Contamination Event Resulting In Internal Contamination of Steam Generator Eddy Current Workers	May 18, 2002 (Draft)
12-THP-6010-RPP-006, Data Sheet 1	Pre-job ALARA Briefing Checklist and Attendance Roster	May 6, 2002
	Whole Body Count Analyses Results and Corresponding Dose Calculations	May 22 - 24, 2002
	Personnel Contamination Log	May 7 - 23, 2002
12-THP-6020-CHM-110	RCS Chemistry - Shutdown/Refueling	Revision 8(b)
Performance Assurance Audit PA-02-06	Radiation Protection	February 22, 2002 through March 15, 2002
Root Cause Analysis	U2C13 ALARA Dose Estimates Exceeded	May 2002 (Draft)

Performance Assurance Field Observation FO-02-E-030	CRUD Burst Activities During U1C18	May 7, 2002
Performance Assurance Field Observation FO-02-E-088	Personnel Use of Contamination Monitors	May 17, 2002
Performance Assurance Field Observation FO-02–E-099	Follow-up of Actions as a Result of Personnel and Internal Contamination	May 20, 2002
Performance Assurance Field Observation FO-02-E-054	Upper Internals Removal With Fuel in the Vessel	May 11, 2002
Performance Assurance Field Observation FO-02-E-047	RP Practices When Exiting Contaminated Areas	May 10, 2002
Performance Assurance Field Observation FO-02-E-057	Transfer Canal Pre-job ALARA Brief	May 7, 2002
Performance Assurance Field Observation FO-02-E-020	ALARA Plan For Scaffold Activities	May 5, 2002

<u>2PS2</u> <u>Radwaste Processing and Transportation</u>

12-THP-6010-RPP-901	Resin Transfer to Qualified Shipping Container	Revision 4A
12-THP-6010-RPP-904	High Integrity Containers	Revision 1C
PMP-6010-PCP-900	Radioactive Waste Process Control Program	Revision 4B
12-THP-6010-RPP-909	Filter Packaging	Revision 1B
12-THP-6010-RPP-906	Processing Wet Radioactive Wastes	Revision 1A
12-THP-6010-RPP-902	De-watering of High Integrity Containers	Revision 3
PMP-6010-PCP-901	Shipment of Radioactive Materials and Waste	Revision 1A
12-THP-6010-RPP-900	Preparation of Radioactive Shipments	Revision 7A
12-THP-6010-RPP-903	Activity Determination and Waste Classification	Revision 3

12-THP-6010-RPP-913	Scaling Factor Determination	Revision 0A
Shipment RMC-01-048	Waste Manifest and Associated Shipment Preparation Documents	October 25, 2001
Shipment RMC-01-070	Waste Manifest and Associated Shipment Preparation Documents	December 15, 2001
Shipment RMC-01-003	Waste Manifest and Associated Shipment Preparation Documents	January 17, 2001
Shipment RMC-00-106	Waste Manifest and Associated Shipment Preparation Documents	May 5, 2000
Shipment RMC-99-100	Waste Manifest and Associated Shipment Preparation Documents	September 17, 1999
Shipment RMC-00-293	Waste Manifest and Associated Shipment Preparation Documents	December 18, 2000
Shipment RMC-01-009	Waste Manifest and Associated Shipment Preparation Documents	February 1, 2001
Shipment RMC-02-025	Waste Manifest and Associated Shipment Preparation Documents	February 11, 2002
EA-C-R-RW01	Radioactive Waste Lesson Plan	November 2001
EA-O-509005	Qualification Card - Survey a Shipment of Radioactive Material	Various Dates and Individuals
EA-O-509007	Qualification Card - Prepare Radioactive Waste Containers	Various Dates and Individuals
EA-O-509024	Qualification Card - Perform Checks on Radioactive Materials Shipping Containers	Various Dates and Individuals
EA-O-509029	Qualification Card - Load Radioactive Waste onto Vehicles	Various Dates and Individuals
EA-O-509035	Qualification Card - Sort Radioactive Waste in Preparation for Shipment	Various Dates and Individuals
PA-02-06	Performance Assurance Audit - Radiation Protection	February 22 - March 15, 2002
PA-01-14	Performance Assurance Audit - Radiation Protection	February 9 - March 16, 2001
FO-99-K-173	Field Observation - Environmental Radioactive Shipment	October 20, 1999

FO-00-C-154	Field Observation - Review of Findings from Audit 99-10/NSDRC 268. Receipt, Packaging and Shipment of Radioactive and Fissile Material	March 7 - 10, 2000
FO-00-I-046	Field Observation - Radwaste Laundry Shipment	September 8 - 11, 2000
FO-00-L-072	Field Observation - High Level Radioactive Waste Shipment	December 17, 2000
SA-2002-REA-001	Draft Report of Self Assessment - Packaging and Shipping of Radioactive Waste	March 11 - 15, 2002
CR 02074040	Potential Declining Trend in Procedure Adherence by Radiation Protection	March 15, 2002
CR 01043011	High Integrity Container Rigging	February 12, 2001
CR 01068026	Radioactive Source Potentially Improperly Controlled	March 9, 2001
CR 01081034	Certificate of Compliance Minor Discrepancy	March 22, 2001
CR 01129036	Drum Being Prepared for Shipment With Unexpected Dose Rate	May 9, 2001
CR 01205043	Contamination on Resin High Integrity Container	July 24, 2001
CR 01345038	Unexpected Dose Rate on Side of Container Being Loaded for Shipment	December 11, 2001
CR 02066016	Issues from Self Assessment SA-2002-REA-001, Packaging and Shipping of Radioactive Waste	March 7, 2002

3PP1 Physical Protection (Access Authorization)

12 PMP 2060.ACS.002	Access Authorization Program	January 13, 1997 Revision 1
AEP:NRC:2691-01	FFD [Fitness-for-Duty] Six Month Data (July 1, 2001 to December 31, 2001)	February 22, 2002
CO801-03	FFD Six Month Data (January 1, 2001 - June 30, 2001)	August 3, 2001
Performance Assurance Audit PA-01-11	Access Authorization/ Personnel Access Data System (PADS)	May 25, 2001

Performance Assurance Audit PA-00-15	Fitness-for-Duty Program	December 11, 2000
	Quarterly Security Event Log	1st Quarter 2002
	Quarterly Security Event Log	4th Quarter 2001
Performance Assurance Surveillance SR-02-0005	Access Authorization	April 16, 2002

<u>3PP2</u> <u>Physical Protection (Access Control)</u>

12 PP2060 SEC 008	Tests of Security Related Equipment	Revision 6
12 PMP 2060 SEC.006	Security Requirements for Plant Personnel	Revision 0
CR 02113070	Unauthorized Vehicle Past Post One	April 23, 2002
	Quarterly Security Event Log	1st Quarter 2002
	Quarterly Security Event Log	4th Quarter 2002
	Security Initiated Condition Reports	January 1, 2002 to April 26, 2002

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

Special Plant Procedure (SPP) 2060 SFI 101	Performance Indicator Data Gathering	Revision 0
PMP 7110.PIP.001	Regulatory Oversight Program Performance Indicators	Revision 1
	PI Camera Submittal	October 1, 2001 to March 31, 2002
	Perimeter PI Submittal	October 1, 2001 to March 31, 2002

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

LER 50-316-2001-002-00	Failure of Lower Containment Airlock Door Interlock Results in Inadvertent Opening of Both Doors	March 16, 2001
LER 50-316-2001-002-01	Failure of Lower Containment Airlock Door Interlock Results in Inadvertent Opening of Both Doors	October 26, 2001

LER 50-316-2002-001-00	Containment Isolation Valve Alignment Error During Local Leak Rate Testing	March 28, 2002
LER 50-315-2002-002-00	Pressurizer Power Operated Relief Valves Inoperable Due to Control Switch Position	April 19, 2002
02-OHP-4030.STP.041	Refueling Integrity	Revision 8
Drawing OP-2-5120D-25	Flow Diagram Containment Control Air 85 Pound and 50 Pound Ring Headers Unit 2	Revision 25
CR 02043026	Refueling Integrity Lost When 2-XCR-101 Was Stroked During Core Alterations	February 12, 2002
LER 50-316-2002-002-00	Technical Specification 3.9.4.c Was Violated During Core Alterations	April 12, 2002

4OA3.5 Ice Condenser Lower Inlet Door Testing

CR 02032016	NRC Identified That Ice Condenser Lower Inlet Door Testing Performed Prior to the Unit 1 and Unit 2 Restart in 2000 Was Inadequate	January 31, 2002
NRC Letter to Mr. A.C. Bakken	Donald C. Cook Nuclear Plant, Unit 1 - Issuance of Amendment Re: Ice Condenser Lower Inlet Doors (TAC Number MB3989)	February 14, 2002
Job Order R0210872	Unit 1 - Perform Lower Ice Condenser Inlet Door Surveillance	May 30, 2002
Job Order R0087658	Perform Lower Inlet Door Surveillance 12 MHP 4030.010.003	November 22, 2000
CR 02091007	NRC Identified Incorrect Title in Cover Letter for LER 50315-2002-001-00	April 1, 2002
CR 02132047	Unit 1 Lower Ice Condenser Inlet Door 15 Right Failed As-found Opening Torque Test Required by TS 4.6.5.3.1.b.3	May 12, 2002
CR 02133017	Unit 1 Lower Ice Condenser Inlet Door 4 Right Failed As-found Opening Torque Test Acceptance Criteria of 12 MHP 4030.010.003	May 13, 2002

CR 02150052	NRC Identified Error in Reportability Evaluation for Ice Door 15 Right Failure. Condition Is Reportable.	May 30, 2002

<u>4OA3.6</u> <u>Cell Cracking Rendered 2AB 250 VDC Station Battery Inoperable and Review of Associated Notice of Enforcement Discretion (NOED)</u>

AEP Letter AEP:NRC:2016-01	Donald C. Cook Nuclear Plant Unit 2 Request for Notice of Enforcement Discretion for the Unit 2 AB Station Battery	April 8, 2002
NRC Letter to Mr. A.C. Bakken	Notice of Enforcement Discretion for Indiana Michigan Power Company Regarding D.C. Cook, Unit 2 (NOED 02-3-001), EA 02-065	April 10, 2002
CR 01347067	Internal Degradation Found on 23 Cells of 2-BATT-AB During Surveillance	December 13, 2001
CR 02093039	2AB Station Battery Cells 102 and 27 Have Top Cover Cracks	April 3, 2002
CR 02095021	Inoperability of Three 2AB Battery Cells Not Reported in a Timely Manner	April 4, 2002
CR 02107063	NRC Identified Minor Inconsistency Between Verbal and Written NOED Request for 2AB 250 VDC Station Battery	April 17, 2002

<u>4OA3.7</u> <u>Significance Determination Process Review for Gas Binding of Unit 2 Centrifugal Charging Pumps Due to Inadequate Valve Maintenance Activity</u>

CR 02047050	The Unit 2 West CCP Showed Signs of Air Entrainment During Attempts to Swap Its Suction From the Volume Control Tank to the Refueling Water Storage Tank	February 16, 2002
12 MHP 5021-001-023	Manual Diaphragm Valve Maintenance	Revision 6
Job Order 01094018	2-CS-369 Replace Diaphragm	
02 OHP 4021.002.013	Reactor Coolant System Vacuum Fill	Revision 1
Memo from R.W. Hennan to Shift Technical Advisor	Unit 2 Time to 200°F and Time to Boil Graphs for the Refueling Outage	January 4, 2002
Vendor Manual VTD-ITEV-0027	DIA-FLO Handwheel Operated Diaphragm Valves	

Vendor Manual VTD-ITEV-0017	DIA-FLO Diaphragm Valves Installation, Operation, and Maintenance Manual	
Vendor Manual VTD-ITEV-0016	ITT Engineered Valves Maintenance and Instruction Manual for Handwheel Operated Diaphragm Valves	
	Unit 2 Control Room Logs	February 2002

4OA3.8 URI 50-316-00-19-02; 50-316-00-19-02: "Potentially Non-Conservative Engineered Ventilation TS 3.7.6.1."

CR 01138078	Calculation 12-HV-042-N Was Issued to Address CR 00-6947. The Calculation Acceptance Criteria Is Not Traceable to an Approved Design Input.	May 18, 2001
CR 00-11265	NRC Questioned Inconsistency Between Design Basis Calculation and TS 3.7.6.1 Limiting Conditions for Operation	August 12, 2000
CR 98-6364	New Calculation of Heat Gain of the Auxiliary Building Ventilation System Was Performed and Results Show Several ESF Equipment Rooms Exceed 125°F [Degrees Fahrenheit]	November 2, 1998
Calculation TH-01-05	Auxiliary Building Temperature Analysis	Revision 0
DIT B-00501-03	Time-Temperature Profiles for Plant Areas During Normal Conditions	Revision 3
DIT B-00197-26	Time-Temperature Profiles for Plant Areas During Accident Conditions	Revision 26
CR 02046034	Calculation TH-01-05 Supercedes Old Auxiliary Building Calculations and Results in Higher Temperatures	February 15, 2002

40A5 Other

MRS-SSP-1319	Reactor Vessel Head Penetration Remote Visual Inspections for D. C. Cook Unit 1	May 8, 2002
MRS-SSP-1320	Reactor Vessel Head Penetration Inspection Tool Operation D. C. Cook Unit 1	May 8, 2002

MRS-SSP-1321	Penetration Thermal Sleeve Removal and Installation at D. C. Cook Unit 1	May 8, 2002
	D. C. Cook Unit 1 Reactor Vessel Head Penetration Inspection Acquisition and Analysis Training Outline	May 6, 2002
ISI-ET-001	Eddy Current Inspection of J-Groove Welds in Vessel Head Penetrations	January 7, 2002
ISI-ET-002	Eddy Current Procedure for Detection of Cracks in Vessel Head Penetrations With or Without Thermal Sleeves-Differential Gap Probe	January 7, 2002
ISI-UT-003	Ultrasonic Inspection of Reactor Vessel Head Penetrations Using Pulse Echo Techniques	October 22, 2001
WDI-UT-007	Ultrasonic Procedure for Detection of Circumferential Indications in Reactor Head Penetration Welds - 0 Degree to 20 Degree Sword Probes	January 14, 2002
ISI-UT-002	Time of Flight Ultrasonic Inspection of Reactor Head Penetrations	January 13, 2002
CR 02135066	Debris Was Found and Then Removed on Top of the Reactor Head During Remote Visual Inspection	May 15, 2002
CR 02136042	Inactive/passive Boric Acid Was Found on the Top of the Reactor Vessel Head	May 16, 2002