August 4, 2005

Mr. M. Nazar
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000315/2005004;
05000316/2005004

Dear Mr. Nazar:

On June 30, 2005, the U. S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your D. C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the inspection findings that were discussed on July 7, 2005, with Mr. J. Jensen 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, four findings of very low safety significance (Green) were identified, all of which involved violations of NRC requirements. However, because of their very low safety significance and because the issues were entered into your corrective action program, the NRC is treating the violations as Non-Cited Violations in accordance with Section VI.A.1 of the NRC Enforcement Policy.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000315/2005004; 05000316/2005004
w/Attachment: Supplemental Information

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

See Previous Concurrences

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	05000315/2005004; 05000316/2005004
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	One Cook Place Bridgman, MI 49106
Dates:	April 1, 2005, through June 30, 2005
Inspectors:	B. Kemker, Senior Resident Inspector J. Lennartz, Resident Inspector F. Ramirez, Resident Inspector T. Bilik, Reactor Inspector A. Garmoe, Reactor Engineer J. Jandovitz, Reactor Inspector M. Parker, Operations Engineer C. Phillips, Senior Operations Engineer R. Ruiz, Reactor Engineer W. Slawinski, Senior Radiation Specialist S. Sheldon, Reactor Engineer
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000315/2005004, IR 05000316/2005004; 04/01/2005-06/30/2005; D. C. Cook Nuclear Power Plant, Units 1 and 2; Inservice Inspection (ISI) Activities, Maintenance Effectiveness, Access Control to Radiologically Significant Areas.

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

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Barrier Integrity

C Green. The inspectors identified a finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion IX, "Control of Special Processes," when licensee personnel failed to use a Code qualified weld procedure for a weld overlay repair completed on a pressurizer nozzle-to-safe end weld. Specifically, the licensee staff failed to perform Charpy V-notch impact tests to support weld procedure qualification and failed to incorporate a supplemental essential welding variable into the weld procedure as required by the American Society of Mechanical Engineers (ASME) Code.

This finding was more than minor because if left uncorrected, the issue could have become a more significant safety concern since an unqualified weld process could have reduced the impact toughness of the pressurizer weldment such that it would be susceptible to brittle fracture. The finding was of very low safety significance because subsequent Charpy V-notch impact tests that were conducted as part of the licensee's immediate corrective actions demonstrated adequate impact toughness. (Section 1R08.1.b.1)

Cornerstone: Mitigating Systems

C Green. The inspectors identified a finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," when licensee personnel failed to perform maintenance on the Unit 1 west centrifugal charging pump with a procedure that was appropriate to the circumstances. The cumulative effect of several delays, including a need to disassemble and reassemble the outboard bearing mechanical seal due to improper installation and the lack of an appropriate fit check, resulted in the unavailability of the pump beyond the originally planned 58-hour maintenance window. The licensee was granted an emergency license amendment to extend the Technical

Specification 72-hour allowed outage time to preclude a plant shutdown. The licensee implemented appropriate changes to the maintenance procedure to prevent recurrence. This finding affected the cross-cutting area of human performance.

The inspectors determined that this finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the Unit 1 west charging pump was rendered unavailable for an extended period of time. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the additional outage time for the Unit 1 west charging pump was a degradation of the Mitigating System Cornerstone; however, this finding 1) was not a design deficiency or qualification deficiency confirmed to result in a loss of function per Generic Letter 91-18; 2) did not represent an actual loss of safety function of a system; 3) did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time; 4) did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk significant; and 5) did not screen as potentially risk significant due to seismic, flooding, or a severe weather initiating event. Therefore, the finding screened as Green and was considered to be of only very low safety significance. (Section 1R12.b.1)

C Green. The inspectors identified two examples of a finding of very low safety significance and a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," associated with the review of operating experience information. Licensee personnel failed to take prompt and effective corrective actions to address asbestos-filled spiral wound gaskets subject to limited shelf life, which resulted in a steam leak from the Unit 2 pressurizer manway cover. The licensee also failed to take prompt and effective corrective actions to address tempered 414 stainless steel centrifugal charging pump shafts susceptible to high cycle fatigue cracking, which resulted in the failure of the Unit 1 west charging pump. The licensee subsequently replaced the failed components. The inspectors considered each of the two examples separately when completing the SDP review since each example occurred apart in time and neither one influenced the other.

The failure of the Unit 2 pressurizer manway gasket was associated with the equipment performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during power operation. Specifically, the manway gasket failure resulted in reactor coolant system (RCS) leakage that necessitated the reactor be shut down for repair. The inspectors determined that this example was of very low safety significance during a Phase 1 SDP evaluation because it would not likely result in exceeding the Technical Specification limit for identified RCS leakage and would not likely affect other mitigation systems, resulting in a total loss of their safety function. As part of the licensees immediate corrective actions, the gasket was replaced.

The Unit 1 charging pump failure was associated with the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors performed a Phase 1 SDP review of this finding The inspectors determined that the additional outage time for the Unit 1 west charging pump was a degradation of the Mitigating System Cornerstone; however, this finding 1) was not a design deficiency or qualification deficiency confirmed to result in a loss of function per Generic Letter 91-18; 2) did not represent an actual loss of safety function of a system; 3) did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time; 4) did not represent an actual loss of safety function of one or more non-Technical Specification trains of equipment designated as risk significant; and 5) did not screen as potentially risk significant due to seismic, flooding, or a severe weather initiating event. Therefore, the finding was considered to be of very low safety significance. As part of the licensee's immediate corrective actions, the charging pump was replaced. (Section 1R12.b.2)

Cornerstone: Occupational Radiation Safety

C Green. The inspectors identified a finding of very low safety significance and an associated Non-Cited Violation of Technical Specification 6.12.2 when licensee personnel failed to provide an adequate physical barrier to prevent unauthorized entry into a locked high radiation area. The barrier for a locked high radiation area did not extend fully across an accessible area and allowed passage by an individual around the barrier.

The issue was more than minor because it was associated with the plant facilities/equipment attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. The issue represented a finding of very low safety significance because it did not involve As Low As Is Reasonably Achievable (ALARA) Planning or work controls, and there was no overexposure or substantial potential for an overexposure, nor was the licensee's ability to assess worker dose compromised. Corrective actions included the installation of a flashing light and temporary physical barrier pending plans to construct a permanent extension to the barrier. Since the issue was initially licensee-identified, but was not characterized correctly, the licensee's initial corrective actions were not adequate. Consequently, the finding was also related to the cross-cutting area of problem identification and resolution. (Section 2OS1)

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

None

REPORT DETAILS

Summary of Plant Status

Unit 1 was shut down in Mode 6 (Refueling) at the beginning of the inspection period for the Cycle 20 refueling outage (U1C20). Unit 1 automatically tripped on April 26, 2005, during plant startup preparations to synchronize the main generator to the grid due to a failed bistable relay driver card for nuclear instrumentation channel 35. Following replacement of the relay driver card, the licensee performed a reactor startup and synchronized the unit to the grid later the same day.

Following startup from the refueling outage, Unit 1 operated at or near full power until June 18, 2005, when the licensee reduced power to about 8 percent to remove the main generator from service to repair the main generator automatic voltage regulator and load limiter. Following the repairs, the licensee synchronized the unit to the grid on June 19, 2005. Unit 1 operated at or near full power for the remainder of the inspection period.

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

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Severe Weather Preparations

a. Inspection Scope

The inspectors reviewed the licensee's procedures and preparations for high temperature and high wind conditions. The inspectors reviewed severe weather and plant de-winterization procedures and performed general area walkdowns. During walkdowns of the plant and switchyard conducted the last 2 weeks of May 2005, 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) associated with adverse weather mitigation.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

.2 Severe Thunderstorm Warning

a. Inspection Scope

The inspectors reviewed the licensee's procedures and preparations for a severe thunderstorm warning with high winds on May 13, 2005. The inspectors reviewed severe weather procedures and performed general area walkdowns. During walkdowns of the plant and switchyard during the severe thunderstorm warning, the inspectors observed housekeeping conditions and verified that material capable of becoming an airborne missile hazard during the high winds and severe weather was appropriately restrained.

This activity represented one inspection sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

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

- Unit 2 AB Emergency Diesel Generator (EDG) during planned maintenance on the Unit 2 CD EDG
- Unit 1 East Residual Heat Removal (RHR) system train following system alignment for shutdown cooling
- Unit 2 West Component Cooling Water (CCW) system train during planned maintenance on the Unit 2 East Component Cooling Water System Train

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

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours

a. Inspection Scope

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

- Unit 1 Emergency Power System Motor Control Room (Zone 42C)
- Unit 2 Emergency Power System Motor Control Room (Zone 46C)
- Unit 1 Control Room (Zone 53)
- Unit 2 Control Room (Zone 54)
- Unit 1 Containment Lower Volume (Zone 67)
- Unit 1 Containment Upper Volume (Zone 68)
- Unit 1 Heating, Ventilation, and Air Conditioning Vestibule 633' Elevation (Zone 49)
- Unit 2 Heating, Ventilation, and Air Conditioning Vestibule 633' Elevation (Zone 50)

The inspectors verified that fire zone conditions were consistent with assumptions in the licensee's Fire Hazards Analysis. The inspectors walked down fire detection and suppression equipment, assessed the material condition of fire fighting equipment, and evaluated the control of transient combustible materials. In addition, the inspectors verified that fire protection related problems were entered into the licensee's corrective action program with the appropriate characterization and significance.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 External Flood Protection

a. Inspection Scope

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

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

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

- The inspectors performed a walkdown of the lower elevations of the Turbine Building and Auxiliary Building to assess the adequacy of watertight doors and verify that drains and sumps were clear of debris and were operable.

- The inspectors reviewed selected operating procedures used to identify and mitigate external flooding events and reviewed preparations for possible flooding of susceptible plant areas due to a seiche and high waves from Lake Michigan.

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

b. Findings

Potential External and Internal Flooding Impact on Safe Shutdown Equipment

Introduction

The inspectors identified that the plant design for flooding events may not mitigate the consequences of either external flooding or internal piping system failures. As a result of a seiche, high waves, or non-seismically qualified piping failures, water could collect in the lower Turbine Building and result in substantial damage to safe shutdown plant equipment through flooding.

Discussion

The licensee's flooding analysis assumed that the station drainage system would not negatively impact the analysis. In the flooding analysis, the licensee described the roadway on the west side of the plant, along with the shoreline buildup, as a flood protection feature for protection from external flooding. Two distinct flooding scenarios were postulated to occur, a seiche on Lake Michigan and a worst-case wave run-up. The plant external flood protection elevation was 594.6' New York Mean Datum; however, the licensee established the Turbine Building maximum credible flooding elevation as 583.5'. Previous licensee surveys of the roadway identified that portions of the roadway were about 0.45' below the highest probable water elevation of 594.5'. The licensee concluded that the amount of water that could breach the roadway and shoreline buildup barrier was inconsequential compared to the volume available for dispersion. However, the inspectors were not able to locate any formal licensee calculation to determine the amount of water for dispersion.

The inspectors questioned how water would be dispersed following an external flooding event. The licensee indicated that large roll-up doors on the west end of the Turbine Building would be secured prior to a seiche and that water would be dispersed in the roadway drains. The inspectors reviewed CR 03234074 that documented the licensee's response to previously identified plant flooding concerns. The licensee stated in the condition report evaluation that operators would receive advanced notification and implement Abnormal Operating Procedure 4022-001-009, "Seiche," and that this procedure would direct plant operators to close doors on the west side of the Turbine Building in order to slow the influx of water during an event. However, the inspectors noted during their review of the procedure that there was no guidance directing operators to close the west side Turbine Building doors in the event of a seiche warning.

As discussed above, the roadway and shoreline buildup protected the plant from most external flooding sources; however, the inspectors identified a potential breach in that barrier. The Turbine Building sump had an overflow box with a 30" overflow pipe that led to the lake by way of the Lake Screen House. This line had a 30" flapper-type check valve, 12-DR-129, located in the sump overflow box to prevent backflow from the lake. Failure of this nonsafety-related component during a design basis seiche could cause the Turbine Building sump to overflow and back up into the safe shutdown plant equipment rooms. All four of the Unit 1 and Unit 2 EDGs were located on the 587' elevation, with the lowest of the EDG room floor drains at the 584' elevation. The auxiliary feedwater (AFW) pumps for both Unit 1 and Unit 2 were located on the 591' elevation of the Turbine Building. All of these rooms were connected to the Turbine Building sump via floor drains and there were no check valves in the individual equipment room drain lines to prevent back-flow into the floor drain system.

The bottom of the overflow pipe was at the 583.5' elevation at its highest point. The highest recorded lake level was 583.5', however, the licensee's analysis assumed a worst case seiche of 11 feet or 594.5' elevation. The inspectors inquired into preventative maintenance history for this check valve because this was the only barrier from the lake to safe shutdown equipment. This valve was not included in the licensee's check valve preventative maintenance program. The inspectors noted that this valve was coded as a "run to fail" component. Review of the valve history identified that this valve has been subject to a harsh environment and had failed on at least two occasions in the last 3 years. In November of 2002, the valve was found broken with a piece of the flapper in the overflow box pit. In February 2004, the valve was found further degraded with a broken hinge pin, preventing the valve from operating. The hinge pin was replaced in 2004; however, the licensee was unable to complete repairs to the valve flapper due to excessive corrosion. Thus, the valve has been in a degraded state for over 2 years.

As a result of the degradation of this flood protection feature for protection from both external and internal flooding, high water level in the Turbine Building could flow into the AFW pump room and the EDG equipment rooms. The inspectors postulated that water could flow into the equipment rooms by way of leakage past non-watertight doors and the plant's floor-drain system. Water level in these rooms could potentially rise and render multiple trains of safe shutdown equipment unavailable. The licensee's corrective actions addressing this issue were in progress at the end of this inspection period. This issue is considered an Unresolved Item (URI 05000315/316/2005004-01) pending further review.

.2 Internal Flood Protection

a. Inspection Scope

The inspectors performed one inspection sample related to the licensee's precautions to mitigate the risk from internal flooding events. Specifically, the inspectors verified the adequacy of internal flood protection features for the AFW pump room and the EDG rooms. The following inspection activities were performed:

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

- The inspectors performed a walkdown of the lower elevations of the Turbine Building to assess the adequacy of watertight doors and verify that drains and sumps were clear of debris and were operable.

- The inspectors reviewed selected operating procedures used to identify and mitigate internal flooding events and verified that these procedures were adequate.

In addition, the inspectors verified that flood protection related problems were entered into the licensee's corrective action program with the appropriate characterization and significance.

b. Findings

Section 1R06.1 discusses an Unresolved Item based on the results of this inspection.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the completed test reports and observed the licensee perform selected portions of inspections for the following two heat exchangers. This activity represented one inspection sample.

- 1-HE-47-CDN CD EDG North Combustion Air Aftercooler
- 1-HE-47-CDS CD EDG South Combustion Air Aftercooler

The inspectors selected these heat exchangers because the EDGs were identified as risk significant in the licensee's risk assessment. During this inspection, the inspectors observed the as-found condition of the heat exchangers and verified that no deficiencies existed that would mask degraded performance. The inspectors discussed the as-found condition as well as the historical performance of the heat exchangers with engineering department personnel and reviewed applicable documents and procedures.

In addition, the inspectors verified that heat sink related problems were entered into the licensee’s corrective action 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 condition reports.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

From March 28, 2005, through April 15, 2005, the inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries for Unit 1. The inspectors selected the ASME Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of the inspection procedure, based upon the ISI activities available for review during the onsite inspection period.

The inspectors observed the following nondestructive examination activities:

- Dye penetrant examination (PT) of the No. 14 steam generator inlet and outlet nozzle-to-safe-end welds (STM-14-02R and STM-14-03R) to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements.

- Bare metal visual examination (VT) of the pressurizer penetrations to evaluate compliance with licensee commitments to NRC Bulletin 2004-01. (Section 4OA5.2).

The inspectors reviewed examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service to verify that the licensee's acceptance was in accordance with Section XI of the ASME Code. Specifically, the inspectors reviewed:

- a PT examination with relevant indications identified on a weld joint upstream of the west containment spray pump suction flushing connection shutoff valve (1-CTS-140W); and

- a PT examination with relevant indications identified on the Unit 1 west centrifugal charging pump (1-PP-50-W).

The inspectors reviewed pressure boundary welds for Class 1 or 2 systems that were completed since the beginning of the previous refueling outage to determine if the welding acceptance and pre-service examinations (e.g., pressure testing, visual, dye penetrant, and weld procedure qualification tensile tests and bend tests) were performed in accordance with ASME Code Sections III, V, IX, and XI requirements. Specifically, the inspectors reviewed welds associated with the following work activities:

- an ISI Class 1 weld repair of valve 1-IMO-315 (welds OW1 through OW3), a Unit 1 safety injection valve to reactor coolant loops No. 1 and No. 4; and

- replacement/welding of ISI Class 2 steam generator blow down under flow instrumentation piping 1-DFI-421 (welds OW1 through OW5).

The inspectors performed a review of ISI-related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff, and reviewed licensee corrective action records to determine if:

- the licensee had described the scope of the ISI related problems;

- the licensee had established an appropriate threshold for identifying issues;

- the licensee had evaluated industry generic issues related to ISI and pressure boundary integrity; and

- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the attachment to this report.

This activity represented one inspection sample.

b. Findings

(1) Unqualified Weld Procedure Used for Weld Overlay Repair

Introduction

The inspectors identified a finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion IX, “Control of Special Processes,” when licensee personnel failed to perform Charpy V-notch impact tests to support weld procedure qualification and failed to incorporate a supplemental essential welding variable into the weld procedure as required by the ASME Code.

Description

While performing an ultrasonic examination (UT) during the Unit 1, Cycle 20 refueling outage, the licensee identified an axial crack indication in the pressurizer nozzle-to-safe end weld 1-PRZ-23. To repair this weld, the licensee used a semi-automatic gas tungsten arc welding process to deposit a layer of Inconel 52 weld metal over the outside pipe diameter that covered the original dissimilar metal weld 1-PRZ-23, in accordance with welding procedure specification (WPS) 18-WBU/52 MC-GTAW, Revision 0.

On April 20, 2005, the inspectors identified that the licensee had not performed Charpy V-notch impact tests in the procedure qualification record (PQR) 757A, Revision 0, that was used to qualify WPS 18-WBU/52 MC-GTAW, Revision 0. Impact tests were required to be performed during weld procedure qualification by the construction Code (ASME Code Section III, 1965, 1966 Winter Addenda). Additionally, the inspectors identified that the licensee had not incorporated appropriate limits on weld parameters affecting heat input (supplemental essential welding variable) into the welding procedure to limit heat input as required by the ASME Code Section IX. The supplemental essential weld variables that limit the field welding process heat input to below that used for qualification welds ensure that the welding did not reduce the fracture toughness of the base material and weld heat-affected zone since a high heat input can reduce base metal toughness through increased grain size or promote high cooling rates that can lead to formation of brittle martensite.

Analysis

The inspectors determined that the failure to use a qualified weld procedure for fabrication of the weld overlay repair of 1-PRZ-23 was a licensee performance deficiency that warranted a significance determination. The inspectors reviewed this finding against the guidance contained in Appendix B, "Issue Dispositioning Screening," of Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports." In particular, the inspector compared this finding to the findings identified in Appendix E, "Examples of Minor Issues," of IMC 0612 to determine whether the finding was minor and concluded that none of the examples listed in Appendix E accurately represented this example. As a result, the inspector compared this performance deficiency to the minor questions contained in Section 3, "Minor Questions," to Appendix B of IMC 0612. The inspector concluded that the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," because the finding affected the Barrier Integrity cornerstone and if left uncorrected, it could have become a more significant safety concern because the unqualified welding process could have reduced the impact toughness of the pressurizer weldment such that it would be susceptible to brittle fracture.

The inspectors determined that the finding could not be evaluated using the Significance Determination Process (SDP) in accordance with NRC IMC 0609, "Significance Determination Process," because the SDP for the Barrier Integrity cornerstone only applied to degraded systems/components, not to deficiencies in the procedures that were designed to detect component degradation. Therefore, this finding was reviewed by a Regional Branch Chief in accordance with IMC 0612, Section 05.04c, who determined that this finding was of very low safety significance (Green). The inspectors concluded that the finding was of very low safety significance because the licensee subsequently performed Charpy V-notch impact tests that demonstrated adequate impact toughness.

Enforcement

10 CFR 50, Appendix B, Criterion IX, "Control of Special Processes,"” requires, in part, that measures be established to assure that special processes, including welding, are controlled and accomplished using qualified procedures in accordance with applicable

codes. The ASME Code Section IX, 2001 Edition, Table QW 256 implemented QW 409.1 requirements as a supplemental essential variable for qualification of a gas tungsten arc welding procedure. The ASME Code Section IX, 2001 Edition, paragraph QW 409.1, in part, limited the increase in weld heat input to not exceed that which was qualified. The ASME Code Section III, 1965 Edition, 1966 Winter Addenda, paragraph N331.2, states, in part, that Charpy V-notch tests shall be used to test materials 1/8-inch thick and greater.

Contrary to the above, on April 20, 2005, the licensee fabricated a weld overlay at weld 1-PRZ-23 (greater than 1/8-inch thickness) using WPS 18-WBU/52 MC-GTAW, Revision 0, which did not limit weld pass heat input such that it would not exceed that used in the qualification weld. Consequently, the licensee completed the overlay weld with an increase in heat input over that documented in PQR 757A, Revision 0, for the qualification weld. Additionally, the licensee did not perform Charpy V-notch impact testing of a qualification weld to qualify WPS 18-WBU/52 MC-GTAW prior to completing the overlay weld. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program, it is being treated as a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000315/2005004-02).

As a corrective action for this issue, the licensee completed impact tests on specimens retained from the original procedure qualification record. The results of the tests were satisfactory and represented valid qualification tests for the weld heat input used in the first four weld passes (total of seven passes). Because the original weld procedure did not contain restrictions for heat input, the heat input applied during the final three weld passes were higher than the heat input on the original qualification weld specimen. Therefore, the licensee had the vendor fabricate and test additional weld procedure qualification impact specimens with higher weld heat inputs that bounded the heat inputs used in the field weld for all seven weld passes. The results of this testing were documented in PQRs 757A, Revision 2, and 760, Revision 0, to complete qualification of WPS 18-WBU/52 MC-GTAW, Revision 3. The licensee entered this issue into their corrective action program as CR 05111034.

(2) Weld Flaw Returned to Service Without Code Evaluation or Repair

Introduction

The inspectors identified an Unresolved Item associated with the licensee's failure to complete a Code repair or flaw evaluation for a crack indication identified in the pressurizer safe end-to-elbow weld prior to returning this component to service. This issue is considered an Unresolved Item pending further review of the licensee's root cause investigation.

Description

On April 21, 2005, while performing a UT examination of the weld overlay of pressurizer weld 1-PRZ-23, the licensee identified a weld flaw indication in the downstream fusion zone of pressurizer weld 1-RC-9-01F. Weld 1-RC-9-01F was a stainless steel safe end-to-elbow weld on the pressurizer safety relief valve line which had been overlaid during the repair of the adjacent 1-PRZ-23 Inconel weld. The indication identified was axially oriented, located about 0.09-inch from the pipe inner diameter, and was contained in an area 0.29-inches in length by an area of 0.30-inches in width.

On April 28, 2005, the inspectors compared the flaw dimensions for the indication in weld 1-RC-9-01 with the acceptance criteria identified in the ASME Code 1989 Edition, Section XI, Table IWB-3514-2. Based upon this review, the inspectors identified that this flaw was rejectable and required a Code repair or flaw growth analysis, neither of which had been performed prior to the Unit 1 restart on April 22, 2005. In CR 05117045, written after the Unit 1 start-up, the licensee identified that weld 1-RC-9-01F had received a full structural weld overlay as part of the overlay for 1-PZR-23 and that this overlay became the structural weld (e.g., did not take credit for any remaining original weld or base material under the weld overlay). Therefore, the licensee staff believed that the lack of a Code repair or analysis of this flaw had not affected system operability.

The inspectors noted that if this weld flaw indication had been service-induced and identified during a scheduled Section XI Code weld examination, then the licensee would have been required by Section XI to expand the scope of UT examinations to include similar welds. The licensee staff did not expand the scope of the UT examinations to other similar stainless steel welds to determine the potential extent of condition. The licensee's decision to not expand the scope of UT examinations was, in part, based upon this flaw being subsurface and therefore not service-induced (e.g., original construction flaw). However, the inspectors noted that the very small ligament measured by UT (0.09-inch dimension) that remained at the inside surface may not be accurate. Specifically, the inspectors noted that the UT technique was not qualified to detect and size indications at the depth that this indication was identified.

On May 5, 2005, a telephone conference was held between the licensee staff, Region III NRC staff and the Office of Nuclear Reactor Regulation (NRR) staff to discuss the disposition of the flaw in weld 1-RC-9-01F. Following this telephone conference, the licensee completed a flaw evaluation in accordance with the ASME Code Section XI, Paragraph IWB-3640. Based upon this analysis, the licensee concluded that the flaw was acceptable for the duration of the plant life and that no repair was necessary.

The licensee initiated a root cause investigation to determine why this indication was not evaluated in accordance with the ASME Code requirements prior to returning the pressurizer weld to service. Pending NRC review of the licensee's root cause evaluation, this issue is considered an Unresolved Item (URI 05000315/2005004-03).

(3) Failure to Obtain NRC Approval For a Non-Code Weld Metal Overlay

Introduction

The inspectors identified an Unresolved Item related to the licensee's failure to obtain NRC approval for a non-Code compliant weld metal overlay applied to a stainless steel pressurizer weld.

Description

During the Unit 1 refueling outage, while performing UT examinations of the pressurizer Inconel alloy 82/182 welds, the licensee detected an axial flaw on pressurizer safety nozzle to safety valve inlet line weld 1-PRZ-23. By letter dated April 12, 2005, (as supplemented by letter dated April 15, 2005), the licensee requested relief to repair the weld using a weld overlay that was granted verbally on April 18, 2005.

On April 21, 2005, the licensee completed the Inconel 52 weld metal overlay repair on 1-PRZ-23. This repair extended over weld 1-RC-9-01F, which was a stainless steel safe end-to-elbow weld on the pressurizer safety relief valve line. However, the licensee did not obtain NRC approval through the relief request process to overlay this weld using techniques that deviated from the ASME Section XI Code and NRC approved Code Case N–504. Specifically, the licensee applied Inconel 52 weld metal instead of low carbon stainless steel weld metal identified in Code Case N–504, which was used as a basis for this overlay repair method. Consequently, the licensee did not measure delta ferrite levels as required for the stainless steel material discussed in Code Case N–504. Therefore, on May 3, 2005, the inspectors identified that the licensee had failed to obtain NRC approval as required by 10 CFR 50.55.a(a)(3)(I) for the weld metal overlay applied to weld 1-RC-9-01, which was not fabricated in accordance with the ASME Section XI Code or NRC approved Code Case N–504.

On June 29, 2005, the licensee staff stated that their relief request submitted for weld 1-PRZ-23 included weld 1-RC-9-01, and that they were in compliance with Code Case N–504, therefore no further actions were required. Pending NRC review of the licensee's written response establishing their basis for compliance, this issue is considered an Unresolved Item (URI 05000315/2005004-04).

.2 Pressurized Water Reactor Vessel Head Penetration ISI

a. Inspection Scope

The inspectors did not perform a review of this procedure Section (reduction in one inspection sample), because it was inspected through TI 2515/150, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles," as described in Section 4OA5.1.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) ISI

a. Inspection Scope

From March 26, 2005, through March 29, 2005, the inspectors reviewed the Unit 1 BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary."

The inspectors conducted a direct observation of BACC visual examination activities to evaluate compliance with licensee BACC program requirements and 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requirements. Specifically:

- On March 26, 2005, following shutdown, the inspectors reviewed a sample of BACC visual examination activities through direct observation. This walkdown was completed with Unit 1 in Mode 3 and included the lower containment building inner volume and annulus. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety significant components.

- The inspectors also reviewed the visual examination procedures and examination records for the BACC examination and verified that degraded or non-conforming conditions were properly identified in the licensee's corrective action system.

The inspectors reviewed the following boric acid leak corrective actions to confirm that they were consistent with the requirements of the ASME Code and 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions." The inspectors also reviewed the engineering evaluations performed for these same corrective action documents. The evaluations were verified, as applicable, to ensure that ASME Code wall thickness requirements were maintained:

- CR 04125013, component 1-QFI-220, "Reactor Coolant System," and

- CR 03291011, component 1-PP-42-2, "Reactor Coolant System."

The documents reviewed during this inspection are listed in the attachment to this report. The reviews represented one inspection sample.

b. Findings

No findings of significance were identified.

.4 Steam Generator (SG) Tube ISI

a. Inspection Scope

The inspectors did not perform a review of this procedure section (reduction in one inspection sample), because the licensee did not perform steam generator inspections during this outage due to recent replacement of the SGs.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

The inspectors completed one inspection sample of licensed operator requalification training by observing a crew of licensed operators during simulator training on May 24, 2005. The inspectors assessed the operator's response to the simulated events which included a steam generator tube leak that propagated to a steam generator tube rupture, a failed AFW flow indicator, a failed closed AFW valve, and a failed core exit thermocouple.

The inspectors verified that the operators were able to effectively mitigate the events through accurate and timely implementation of applicable alarm response procedures including Abnormal Operating Procedure OHP-4022-002-021, "Steam Generator Tube Leak;" and Emergency Operating Procedures E-0, "Reactor Trip or Safety Injection;" ES-0.1, "Reactor Trip Response;" and E-3, "Steam Generator Tube Rupture." The inspectors also observed the post-training critique to assess the licensee evaluators' and the operating crew's ability to self-identify performance deficiencies.

b. Findings

No findings of significance were identified.

.2 Annual Operating Test Results

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of Job Performance Measure operating tests and simulator operating tests required by 10 CFR 55.59(a)(2) that were administered by the licensee from February 8 through March 18, 2005. The overall results were compared with the significance determination process in accordance with NRC Manual Chapter 0609I, "Operator Requalification Human Performance Significance Determination Process (SDP)."

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Resident Inspector Quarterly Review

a. Inspection Scope

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

- Unit 1 West Centrifugal Charging Pump
- Unit 2 Pressurizer

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

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

In addition, the inspectors verified that problems associated with the effectiveness of plant maintenance were entered into the licensee's corrective action program with the appropriate characterization and significance.

b. Findings

(1) Inadequate Maintenance Procedure Led to Extended Time to Complete Unit 1 West Charging Pump Repair

Introduction

The inspectors identified a finding of very low safety significance and an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," when licensee personnel used an inadequate procedure to perform maintenance on the Unit 1 west centrifugal charging pump that resulted in an improper installation of the outboard bearing mechanical seal. The time required to correct the improper installation led to the unavailability of the west centrifugal charging pump beyond the originally planned 58-hour maintenance window. To preclude a plant shutdown, an emergency license amendment was granted to extend the 72-hour TS 3.1.2.4 and 3.5.2.a allowed outage time.

Discussion

On January 13, 2005, at 1:30 a.m., the Unit 1 west centrifugal charging pump was removed from service and declared inoperable due to indications of deteriorating pump performance that included lowering flow and elevated oscillating motor current. Upon disassembly, the licensee discovered a cracked pump shaft at the 11th stage impeller split ring.

The licensee assembled a Failure Investigation Team (FIT) to troubleshoot the pump shaft failure and develop a repair schedule to return the pump to an operable status well within the 72-hour TS allowed outage time. At the end of the day shift on January 13th, the responsibility for the repair schedule was turned over from the FIT to the Work Control organization to track replacement of the pump's rotating assembly (shaft and impellers). The licensee's original repair schedule projected returning the pump to service by mid-day on January 16th, with about 14 hours of margin before exceeding the 72-hour TS allowed outage time.

During pump reassembly, maintenance mechanics discovered that the shaft was binding with the outboard seal assembly sleeve key contacting the rotating element. The licensee subsequently determined that the shaft sleeve and key had not been installed correctly due to inadequate detail in the maintenance procedure work instructions. A FIT check for the shaft seal sleeve and key was not contained in the procedure, but had been performed in the past by experienced mechanics based on "skill of the craft" when the pump was assembled in the field.

A total of 15 hours were attributed to delays associated with the inadequate maintenance procedure. Additional delays contributing to the extended maintenance duration included: (1) mechanical maintenance department activities did not begin until about 3 hours after the FIT investigation ruled out a potential problem with the pump's motor; (2) cladding inspection issues resulted in about 2 hours of lost time; and, (3) problems with a faulty test gage during pump testing caused another 2 hours of delay.

On January 16th, at about 2:40 a.m., the licensee requested and was granted an emergency license amendment to extend the 72-hour TS allowed outage time for Unit 1 by an additional 24 hours to preclude a required entry into Mode 3 (Hot Standby) by 7:30 p.m. on January 16th and Mode 4 (Hot Shutdown) by 1:30 a.m. on January 17th.

On January 16th, at 10:10 a.m., the Unit 1 west charging pump was returned to service and declared operable after a total out of service time of 80 hours and 40 minutes. The inspectors reviewed the licensee's root cause evaluation, which examined the failures and delays associated with the execution of this emergent maintenance activity that delayed returning the charging pump to service. In addition, the inspectors reviewed an apparent cause evaluation performed to assess the actual equipment failure. The root cause evaluation concluded that inadequate procedure instructions for the assembly and installation of the charging pump mechanical seals caused the allowed outage time to be exceeded. The apparent cause evaluation concluded that the pump performance deterioration was due to high cycle fatigue of the tempered 414 stainless steel pump shaft at the 11th stage impeller split ring location.

The inspectors also reviewed the licensee's Maintenance Rule evaluation, which concluded that the Unit 1 charging pump failure was a repetitive maintenance preventable functional failure. The evaluation stated that the nuclear industry has had ample notification of this type of failure occurring at a number of different sites. Such notification reports also stated that installing a rotating element with a more robust design could prevent shaft cracking. The evaluation further concluded that the licensee had more than one opportunity to install an improved design in both units, but had not done so. The root cause evaluation also discussed a similar failure of the Unit 2 west charging pump in 1993 due to shaft cracking. A finding associated with the licensee's failure to take prompt corrective actions for a condition adverse to quality related to this issue is discussed in Section 1R12.b.2 of this report.

Based on their review of the circumstances surrounding this event, the inspectors concurred with the results of the root cause, apparent cause, and Maintenance Rule evaluations. The inspectors reviewed the licensee's corrective actions for the event and concluded that the actions were generally appropriate to address the causes discussed above. The actions included replacement of the entire pump assembly, a revision to the maintenance procedure to prevent a recurrence of problems with installation of the mechanical seals, and additional training on mechanical seal installation for mechanical maintenance craftsmen.

Analysis

The inspectors concluded that the failure to perform maintenance on the Unit 1 west centrifugal charging pump with adequate procedure instructions for the assembly and installation of the mechanical seals was a licensee performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting issue of human performance since the procedure resources were inadequate.

The inspectors assessed this finding using the SDP. The inspectors reviewed the samples of minor issues in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," and determined that there were no examples related to this issue. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," the inspectors determined that this finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences because the Unit 1 west charging pump was rendered unavailable for additional hours.

The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors determined that the additional outage time for the Unit 1 west charging pump was a degradation of the Mitigating System Cornerstone; however, this finding 1) was not a design deficiency or qualification deficiency confirmed to result in a loss of function per Generic Letter 91-18; 2) did not represent an actual loss of safety function of a system; 3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; 4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and 5) did not screen as potentially risk significant due to seismic, flooding, or a severe weather initiating event. Therefore, the finding screened as Green and 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 a procedure that was appropriate to the circumstances for performing maintenance on the Unit 1 west charging pump, which was an activity affecting quality. Specifically, 12-MHP-5021-003-001, "Centrifugal Charging Pump Maintenance," Revision 14, did not provide appropriate instructions for a fit check of the pump's shaft seal sleeve and key upon installation. This issue was self-revealed on January 15, 2005, when maintenance craftsmen found that the shaft was binding with the outboard seal assembly sleeve key contacting the rotating element. 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 05000315/2005004-05). The licensee entered this violation into its corrective action program as CR 05020013.

As part of the licensee's immediate corrective actions, the maintenance procedure was revised to include appropriate seal installation instructions.

(2) Failure to Take Prompt Corrective Actions for Conditions Adverse to Quality

Introduction

The inspectors identified a finding of very low safety significance (Green), with two examples. The licensee failed to take prompt and effective corrective actions to address conditions adverse to quality with asbestos-filled spiral wound gaskets subject to limited shelf life, which resulted in a steam leak from the Unit 2 pressurizer manway cover. The licensee also failed to take prompt and effective corrective actions to address conditions adverse to quality with tempered 414 stainless steel centrifugal charging pump shafts susceptible to high cycle fatigue cracking, which resulted in the failure of the Unit 1 west charging pump. In both of these examples, the reliability and availability of important safety-related plant components were adversely affected. The inspectors determined that this issue was a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action."

Discussion

Unit 2 Pressurizer Manway Cover Gasket Failure Example

As a preventative measure in response to a Unit 1 pressurizer manway cover gasket leak in March 2004, the licensee replaced the Unit 2 manway gasket during the Cycle 15 refueling outage that concluded on November 9, 2004. On November 22, Unit 2 was shut down due to increasing reactor coolant system (RCS) unidentified leakage. Following the unit shutdown, the licensee discovered a leak from the pressurizer manway cover due to a failed gasket.

The licensee's root cause evaluation found that the Unit 2 pressurizer manway cover gasket failed prematurely due to exceeding its defined shelf life. For these flexitallic gaskets, the asbestos binder material apparently degraded over time. The licensee had previously identified applicable operating experience involving shelf life/aging degradation of steam generator manway cover gaskets at the Seabrook Nuclear Plant in

May 2002. However, the inspector's review of the licensee's plant operating experience data base entry for the Seabrook report found that this operating experience was screened in error as "not applicable to D. C. Cook." The licensee stated in the root cause evaluation that the reason why this operating experience was not identified and used to prevent the Unit 2 pressurizer manway cover gasket failure was that it was not recognized as a significant event and did not meet the reviewer's criteria for further formal evaluation.

Unit 1 West Charging Pump Failure Example

On January 13, 2005, the Unit 1 west centrifugal charging pump was declared inoperable when Control Room operators removed it from service due to indications of an imminent failure. Upon disassembly, the licensee found a crack at the 11th stage impeller split ring location. The licensee subsequently discovered that the failure was due to high cycle fatigue failure of the tempered 414 stainless steel pump shaft. A related finding regarding the licensee's failure to perform maintenance on the Unit 1 west charging pump with a procedure that was appropriate to the circumstances is discussed above in Section 1R12.b.1 of this report.

The inspectors examined the licensee's root cause evaluation and an apparent cause evaluation performed to assess the actual equipment failure. Both of the evaluations identified multiple external operating experience documents, including NRC Information Notice 94-76, "Recent Failures of Charging/Safety Injection Pump Shafts," that discussed this issue affecting charging/safety injection pump shafts procured by Westinghouse from the Pacific Pump Division of Dresser Industries (later Ingersoll-Dresser Pump Company). The licensee's previous review of these operating experience documents identified that their charging pump shafts were constructed from the same material (i.e., tempered 414 stainless steel) that was susceptible to cracking as described in the operating experience. Although the licensee took action to upgrade safety injection pump shafts to a new design based on a 1980 Significant Event Report, action was not taken as a result of the licensee's previous operating experience review to replace all of the susceptible charging pump shafts.

Analysis

The inspectors determined that the licensee's failure to assure that prompt and effective corrective actions were taken to address known conditions adverse to quality with asbestos-filled spiral wound gaskets subject to limited shelf life and with tempered 414 stainless steel centrifugal charging pump shafts susceptible to cracking was a licensee performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting area of problem identification and resolution since previous operating experience was not adequately evaluated.

The inspectors assessed this finding using the SDP. The inspectors reviewed the samples of minor issues in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," and determined that there were no examples related to this issue. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," the inspectors determined that this finding was more than minor. Specifically, the pressurizer manway cover example was associated with the Equipment Performance attribute of the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions since the gasket failure resulted in leakage from the RCS that necessitated the reactor be shut down for repair. The charging pump example was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences since the pump was out of service for about 80 hours to correct the problem.

The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." The inspectors considered each of the two examples separately when completing the Phase 1 SDP review since each example occurred apart in time and neither one influenced the other.

Unit 2 Pressurizer Manway Cover Gasket Failure Example

In accordance with the "SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barrier Integrity] Cornerstones," the inspectors determined that this finding was a licensee performance deficiency of very low safety significance (Green) because assuming worst case degradation, the finding would not likely result in exceeding the TS limit for identified RCS leakage and would not likely affect other mitigation systems, resulting in a total loss of their safety function.

Unit 1 West Charging Pump Failure Example

In accordance with the "SDP Phase 1 Screening Worksheet for IE [Initiating Events], MS [Mitigating Systems], and B [Barrier Integrity] Cornerstones," the inspectors determined that the additional outage time for the Unit 1 west charging pump was a degradation of the Mitigating System Cornerstone. However, in answer to the five questions in the Mitigating System Cornerstone column of the Phase 1 screening worksheet, this finding 1) was not a design deficiency or qualification deficiency confirmed to result in a loss of function per Generic Letter 91-18; 2) did not represent an actual loss of safety function of a system; 3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; 4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and 5) did not screen as potentially risk significant due to seismic, flooding, or a severe weather initiating event. Therefore, the finding screened as Green and was considered to be of only very low safety significance.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. Contrary to the above, the licensee failed to take prompt and effective corrective action to address known conditions adverse to quality with asbestos gaskets subject to limited shelf-life and with

tempered 414 stainless steel centrifugal charging pump shafts that were susceptible to high cycle fatigue failure. Specifically, upon receipt of external operating experience documenting the above conditions adverse to quality, the licensee did not take prompt and effective corrective action to prevent: (1) the failure of a gasket installed under the Unit 2 pressurizer manway cover, which resulted in a steam leak and the subsequent shutdown of Unit 2 on November 22, 2004; and, (2) the failure of the Unit 1 west charging pump on January 13, 2005, which resulted in the unavailability of this important safety-related component. 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 05000315/316/2005004-06). The licensee entered this violation into its corrective action program as CR 05179010.

As part of the licensee's immediate corrective actions, both components were repaired.

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

a. Inspection Scope

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

- Unit 1 and Unit 2 Planned Dual Essential Service Water Train Outage Resulting in "Orange" Risk Status for Unit 2
- Planned Maintenance on Unit 2 East Centrifugal Charging Pump
- Planned Concurrent Maintenance on Unit 2 East Component Cooling Water Pump and Unit 2 East Residual Heat Removal Heat Exchanger
- Planned Maintenance on the Unit 1 'CD' Emergency Diesel Generator (deferred maintenance activity due to severe thunderstorm warning and high winds)
- Planned and emergent maintenance on Unit 2 'AB' Emergency Diesel Generator

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

The inspectors followed the licensee's activities concerning a planned maintenance outage on the Unit 1 'CD' EDG. During the preparations for the EDG outage, the licensee became aware that a severe thunderstorm warning with high winds had been issued for the immediate area. The inspectors observed the Control Room activities associated with the proposed EDG outage. The shift manager directed the EDG to be restored to service and deferred the EDG outage until the next day due to the potential risk associated with taking the EDG out of service during a severe thunderstorm.

In addition, the inspectors verified that maintenance risk-related problems were entered into the licensee's corrective action program with the appropriate characterization and significance.

b. Findings

No findings of significance were identified.

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

.1 Unit 1 Planned Maintenance Outage for Repair to Main Turbine Controls

a. Inspection Scope

On June 18, 2005, the licensee reduced Unit 1 power to about 8 percent to remove the main turbine from service to perform repairs to the main generator voltage regulator and load limiter. The voltage regulator would not control voltage in "automatic" and the load limiter response was erratic when adjusting power.

The inspectors reviewed the operational decision-making involved with this non-routine evolution, reviewed earlier troubleshooting efforts, reviewed the outage plan, and observed the conduct of operations while operators stabilized power at 8 percent and removed the main turbine from service.

b. Findings

No findings of significance were identified.

.2 Unit 1 Starting and Paralleling a Second Motor Generator Set

a. Inspection Scope

On June 20, 2005, the Unit 1 south control rod drive motor generator (MG) set received an unexpected trip. The MG set was found with its output breaker tripped open with an over current flag. This placed the unit in a position of not having a backup MG set online, as the loss of the remaining MG set would result in a unit trip. The licensee took steps to guard the operating MG set as a protected train. Licensee troubleshooting effort identified that a failed capacitor had caused the MG set to trip.

On June 22, 2005, following repairs to the MG set, the licensee made preparations to start and parallel the idle MG set to the operating MG set. The licensee took appropriate precautions in preparing to parallel the MG set. This consisted of pre-job briefings for both the maintenance crew and the operating crew, including contingency actions upon the loss of the operating MG set. The inspectors observed both the post maintenance testing activities, including voltage adjustments and paralleling operations for the second MG set. These activities were conducted without incident.

The inspectors reviewed the operational decision-making involved with this non-routine evolution, reviewed earlier troubleshooting efforts and observed the conduct of operations while operators paralleled the idle MG set.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- CR 04305061, "Momentary Air Binding of the East Charging Pump While Manually Making Up to the Volume Control Tank from the Refueling Water Storage Tank Via 2-IMO-910, Charging Pumps Suction From the Refueling Water Storage Tank"
- CR 05085047, CR 05085048, and CR 05085049, "Steam Generator Safety Valves Failed the As-found Set Point Test"
- CR 05091020, "Modes 1-4 Aggregate Operability Determination for Unit 1"
- CR 05117045, "Indication Identified in the Fusion Zone of Weld 1-RC-9-01F During Final Ultrasonic Testing of the Weld Overlay for 1-PRZ-23"
- CR 05136094, "Lower Containment Ventilation Unit Fan Motors Will Immediately Restart Following an Accident Upon Reset of the Containment Phase B Isolation Signal"
- CR 05124005, "Received Annunciator 207 Drop 42 RVLIS [Reactor Vessel Level Indication System] Train 'B' Hydraulic Isolator Fluid Abnormal"

In addition, the inspectors verified that problems related to the operability of safety-related plant equipment were entered into the licensee's corrective action program with the appropriate characterization and significance.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Semiannual Review of the Cumulative Effect of Operator Workarounds

a. Inspection Scope

The inspectors completed one baseline sample reviewing the cumulative effect of operator workarounds; control room deficiencies; and degraded conditions on equipment availability, initiating event frequency; and the ability of the operators to implement abnormal or emergency operating procedures. The inspectors observed the Work Around Review Board meetings on May 25 and June 30, 2005, to verify that potential workarounds were appropriately characterized in accordance with plant procedure PMP-4010-OWA-001, "Oversight and Control of Operator Workarounds."

During this review, the inspectors considered the cumulative effects of operator workarounds on the following:

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

In addition, the inspectors reviewed the issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for issues potentially affecting the functionality of mitigating systems or on the operators' response to initiating events that were documented in selected condition reports.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the engineering analyses, modification documents and design change information associated with the following two permanent plant modifications:

- 01-CMM-30061, "Install High Point Vent Valve in the Component Cooling Water Discharge Piping from the Spent Fuel Pool"
- 01-MOD-45517, "Install Check Valve in Reactor Coolant Pump Seal Return Line"

The first modification installed a high point vent on the component cooling water discharge piping from the spent fuel pit heat exchanger to allow proper air removal from the system following maintenance. The second modification installed a 4-inch check valve in the reactor coolant pump seal return flow path to the volume control tank, upstream of the isolation valve, to prevent gas intrusion into the charging pump.

During this inspection, the inspectors evaluated the implementation of the design modifications and verified that:

- the compatibility, functional properties, environmental qualifications, seismic qualification, and classification of materials and replacement components were acceptable;

- the affected operating procedures and training were identified and necessary changes were completed;
- the pressure boundary integrity was not compromised;
- the implementation of the modifications did not impair key safety functions;
- no unintended system interactions occurred;
- the system performance characteristics affected by the modification continued to meet the design basis; and
- the modification design assumptions were appropriate.

Completed activities associated with the implementation of the modifications were also inspected and the inspectors discussed the modifications with the responsible engineering, maintenance, performance verification and operations staff. In addition, the inspectors reviewed the applicable sections of the TS, UFSAR, and 10 CFR 50.59 safety evaluation associated with the design change packages.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

- Unit Control 2 Room Instrumentation Distribution (CRID) Synchronization Board and Oscillator Card Replacement
- Unit 1 Turbine Driven AFW Pump Trip and Throttle Valve Maintenance
- Unit 2 West Essential Service Water Pump Shaft Coupling Adjustment
- Unit 1 Turbine Driven AFW Pump Governor Replacement
- Unit 1 West Charging Pump Replacement
- Unit 1 'AB' EDG Governor Replacement

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

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Unit 1 Refueling Outage (U1C20)

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 TS and approved procedures. Other major outage activities evaluated included the licensee's control of the following:

- containment penetrations in accordance with the TS;
- SSC that could cause unexpected reactivity changes;
- flow paths, configurations, and alternate means for RCS inventory addition and control of SSC that 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 TS and shutdown risk plan; and
- SSC 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 TS. The inspectors observed operators drain the RCS to mid-loop conditions to accommodate vacuum fill of the RCS near the end of the refueling outage to verify that means of adding inventory to the RCS were available, sufficient indications of RCS water level were operable, and perturbations to the RCS were avoided. 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 and subsequent operational mode changes;
- verification that containment integrity was established prior to entry into Mode 4;
- inspection of the Containment Building, including the ice condenser, 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; and
- 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.

The inspectors also 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 condition reports.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- 01-EHP-4030-103-238, "Centrifugal Charging Pump Check Valve Leak Rate Test"
- 02-OHP-4030-STP-027AB, "AB Diesel Generator Operability Test (Train B)"
- 12-EHP-4030-002-356, "Low Power Physics Tests with Dynamic Rod Worth Measurement"
- 01-EHP-4030-134-203, "Unit 1 LLRT [Local Leak Rate Testing]"
- 12-MHP-4030-010-003, "Ice Condenser Lower Inlet Door Surveillance"
- 01-EHP-4030-103-208, "Unit 1 ECCS Flow Balance - Boron Injection"

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

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed two temporary modifications and verified that the installation was consistent with design modification documents and that the modifications did not adversely impact system operability or availability.

- 01-EHP-4030-134-204, "Unit 1 LLRT," Attachment 1, "Actions for Test Steps C068 and C069"
- 1-TM-05-16-R1, "Lift Lead for Main Generator Output Phase 1 Transformer 2-TR-MAIN-1 Cooling Oil Mechanical Relief Vent Line Pressure Alarm Switch"

The first temporary modification was installed to accommodate local leak rate testing of several residual heat removal system containment isolation valves during the Unit 1 refueling outage. The second temporary modification was installed to defeat one input to a common alarm switch for the Unit 1 main power transformer.

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

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed activities in the plant simulator and the Technical Support Center during two emergency preparedness training drills conducted on May 10, 2005, and on June 14, 2005. The first drill was an unannounced drill that was intended to exercise only the licensee's Emergency Response Organization. The second drill included operator participation in the plant simulator. The inspectors verified that the emergency classifications and notifications to offsite agencies were completed in an accurate and timely manner as required by the emergency plan implementing procedures. The inspectors also verified that the training drills were conducted in accordance with the prescribed sequence of events, drill objectives were satisfied and that the required prompts from the licensee drill controllers were appropriately communicated to the drill participants.

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

b. Findings

No findings of significance identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed the licensee's access controls and survey data for the following work areas located within radiation, high radiation, and locked high radiation areas in the plant to determine if radiological controls, postings and barricades were acceptable:

- Unit 1 Upper and Lower Containment;
- Unit 1 Reactor Vessel Head Temporary Storage/Staging Area; and
- Unit 1 and 2 Auxiliary Building (various areas).

The inspectors reviewed the radiation work permits (RWPs) that governed access to these areas and that defined the radiological conditions to ensure the work control instructions and control barriers that had been specified were adequate. The inspectors walked down selected areas in the Unit 1 Containment Building and in the Auxiliary Building to verify that licensee surveys and postings were complete and accurate and to assess the adequacy of physical barriers for high and locked high radiation areas.

These reviews represented one inspection sample.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance and an associated Non-Cited Violation of TS 6.12.2 when licensee personnel failed to provide an adequate physical barrier that prevented unauthorized entry into a locked high radiation area.

Description

During plant walkdowns on April 4, 2005, the inspectors identified that the physical barrier (a fence with a locked gate) that was provided to control entry into the Unit 1 reactor coolant drain tank area was not adequate to prevent unauthorized personnel entry into the area. Specifically, unauthorized entry into the Unit 1 reactor coolant drain tank area (located in the annulus of lower containment) was physically controlled by a lattice-style metal fence and locked gate. However, the fence did not extend the full length of the annulus region and an approximate 19-inch wide by 7-foot high opening existed between the outer annulus wall and the fence that allowed physical passage by an individual around the barrier. Other openings/gaps of approximately 12-inches wide existed between other structures (a vertical pipe, lateral cable tray, and a vertical support beam) that were interior to the 19-inch wide opening that would allow unauthorized personnel entry into the reactor coolant drain tank area and, therefore, access into areas with radiation levels that represented a locked high radiation area (LHRA) condition. Specifically, radiation levels in accessible areas around and under the reactor coolant drain tank and associated piping varied with plant condition, but had exceeded 1000 millirem per hour at a distance of 30 centimeters.

Based on the inspector's assessment, the openings/gaps were sufficiently large to allow personnel passage around the barrier and into the LHRA with only low to moderate physical effort. The size and configuration of the openings/gaps between the annulus wall and the fence and the other physical impediments interior to the fence would allow the locked high radiation area barrier to be circumvented without "exceptional measures" (use of ladders or mobile platforms or use of specialized tooling (to unbolt cover plates, etc.)), as discussed in Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

In early February 2005, the licensee identified the 19-inch opening between the annulus wall and the fence and generated a condition report. The licensee's condition evaluation that followed its initial problem identification concluded that the barrier was adequate and that TS requirements for unauthorized entry into the area were met because: (1) the area was physically difficult to enter through the gap; (2) the intended point of entry into the area was controlled by a locked gate and was posted as a LHRA; and (3) workers allowed access into radiologically controlled areas were provided training relative to radiological hazards, radiation postings and barricades. That conclusion, however, was inconsistent with the licensee's TS requirement to prevent unauthorized entry, Regulatory Guide 8.38, and performance indicator guidance and associated frequently asked question (FAQ) No. 368 contained in NEI 99-02, Revision 2, "Regulatory Assessment Performance Indicator Guideline." About 3 weeks following the licensee's initial identification of the problem, the licensee posted a sign on the fence indicating "Unauthorized Access Point Do Not Enter" to supplement the existing LHRA posting on the entry gate. At that time, the licensee also planned to extend the fence in the future to cover the 19-inch opening between the fence and the annulus wall as a means to address the barrier vulnerability.

<u>Analysis</u>

The inspectors determined that the openings/gaps between the annulus wall and the LHRA barrier (fence) and other interior structures would allow the barrier to be circumvented without exceptional measures. As a result, the barrier failed to satisfy TS requirements to prevent unauthorized access into a LHRA. Additionally, the cause of the problem was within the licensee's ability to foresee and correct and could have been prevented. Consequently, the issue represented a performance deficiency as defined in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening." The inspectors determined that the issue was associated with the Plant Facilities/Equipment attribute of the Occupational Radiation Safety Cornerstone and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. Therefore, the issue was more than minor and represented a finding that was evaluated using the SDP for the Occupational Radiation Safety Cornerstone.

Since the finding involved a radiological access control problem and the potential for unauthorized entry into a LHRA, the inspectors utilized IMC 0609, Appendix C, "Occupational Radiation Safety SDP," to assess its significance. The inspectors determined that the finding did not involve ALARA Planning or work controls. Since no unauthorized entry into the area occurred, there was no overexposure or substantial potential for an overexposure nor was the licensee's ability to assess worker dose compromised. Consequently, the inspectors concluded that this finding was of very low safety significance (Green).

As described above, the licensee identified the barrier problem about 2 months before identification by the inspectors. However, the licensee concluded the barrier was adequate to prevent unauthorized entry into the LHRA that impacted the extent and timeliness of the licensee's corrective actions. Consequently, the licensee's problem identification and resolution process, a cross-cutting area, was a contributing cause of the finding.

Enforcement

Technical Specification 6.12.2 required, in part, that areas in which the radiation level at 30 centimeters from any surface that radiation penetrates exceeds 1000 millirem in 1 hour be provided, when possible, with locked doors to prevent unauthorized entry into the area. The TS provided that in the event that it was not possible or practicable to provide locked doors due to area size or configuration, that the area be roped-off, posted and a flashing light be activated as a warning device. Contrary to this requirement, as of April 4, 2005, the physical barrier and corresponding locked/posted gate that controlled physical access into the Unit 1 reactor coolant drain tank area was not adequate to prevent unauthorized entry given the opening around the fence large enough to reasonably allow passage by an individual.

Corrective actions taken by the licensee following the identification of the problem by the inspectors included the installation of a flashing light and rope boundary across the annulus area that encompassed the opening in the fence. The licensee also planned to construct additional physical barriers to seal-off opening/gaps in the existing (fence) barrier. 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 05000315/316/2005004-07). The licensee entered this violation into its corrective action program as CR 05097036.

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

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective outage exposure history, Unit 1 outage exposure trends, and ongoing outage activities in order to assess current performance and exposure challenges. This included determining the plant's current 3-year rolling average for collective exposure in order to help establish resource allocations and to provide a perspective of significance for any resulting inspection finding assessment.

The inspectors reviewed the Unit 1 refueling outage (U1C20) work and the associated work activity exposure and time/labor estimates for the following work activities that were likely to result in the highest personnel collective exposures or were otherwise radiologically significant work activities:

- Under Reactor Vessel Inspections
- Refuel Cavity Decontamination Activities
- Insulation Activities in the Containment Building
- Scaffold Erection/Removal in the Containment Building
- Reactor Coolant Pump Seal Activities
- In-Service Tests and Inspections in Containment
- Containment and Annulus Sump Activities

The inspectors determined site specific trends in collective exposures based on plant historical exposure and source term data. The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and assessed those processes used to estimate and track work activity exposures.

These reviews represented four inspection samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning

a. Inspection Scope

The inspectors obtained the licensee's list of Unit 1 outage work activities ranked by estimated exposure that were in progress during the outage and reviewed the following 10 radiologically significant work activities:

- Refuel Cavity Decontamination Activities (RWP 05 - 1100);
- Insulation Activities in the Containment Building (RWP 05 - 1140);
- Scaffolding Erection/Removal in the Containment Building (RWP 05 - 1142);
- In-Service Tests and Inspections in Containment (RWP 05- 1143);
- Primary Valve Maintenance/Repair Activities in Containment (RWP 05 - 1145);
- Reactor Coolant Pump Seal Activities (RWP 05 - 1151);
- Containment and Annulus Sump Activities (RWP 05 - 1155);
- Reactor Nozzle Pits/Sandbox Activities (RWP 05 - 1164);
- Regenerative Heat Exchanger Activities (RWP 05 - 1176); and
- Under Reactor Vessel Inspections (RWP 05 - 1187).

For the activities listed above, the inspectors reviewed the ALARA Plan and associated total effective dose equivalent (TEDE) ALARA evaluations, exposure estimates, and exposure mitigation information in order to verify that the licensee had developed radiological engineering controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the exposure results achieved through the first 12 days of the scheduled 30-day outage including the dose rate reductions and person-rem expended with the dose projected in the licensee's ALARA planning for these 10 work activities. Reasons for inconsistencies between intended (projected) and actual work activity doses were evaluated to determine if the activities were planned reasonably well and to ensure the licensee identified any work interface/planning deficiencies.

The interfaces between operations, radiation protection, maintenance and scheduling groups were reviewed to varying degrees to identify potential interface problems that significantly affected outage dose. The extension of ALARA requirements into work procedures and/or RWP documents was also evaluated to verify that the licensee's radiological job planning was integrated into the work process.

The inspectors compared the person-hour estimates provided by maintenance planning and craft groups to the radiation protection ALARA staff with the actual work activity time expenditures in order to evaluate the accuracy of these time estimates.

The inspectors evaluated if work activity planning included consideration of the benefits of dose rate reduction activities such as shielding provided by water filled components/piping and system flushing, and sequencing of scaffold and shielding installation/removal along with logic-ties in the work scheduling process in order to maximize dose reduction. The licensee's work in progress reports were reviewed for selected outage jobs that accrued collective exposures of 50 and 80 percent of that projected to verify that the licensee could identify problems and address them as work progressed. RWP jobs or specific RWP tasks that accrued greater than one rem and exceed 125 percent of the projected dose were also reviewed to ensure that work was adequately evaluated and suspended, if warranted, and that identified problems were entered into the licensee's corrective action program consistent with the licensee's procedure.

These reviews represented seven inspection samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the licensee's assumptions and basis for its collective outage exposure estimate, and evaluated the methodology and practices for projecting work activity specific exposures. This included evaluating both dose rate and time/labor estimates for adequacy compared to historical station specific or industry data.

The inspectors reviewed the licensee's process for adjusting outage exposure estimates when unexpected changes in scope, emergent work, or other unanticipated problems were encountered that significantly impacted worker exposures. This included determining that adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to plan or control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning.

The licensee's exposure tracking system was evaluated to determine whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of collective exposures. RWPs were reviewed to determine if they covered too many work activities to allow work activity specific exposure trends to be detected and controlled. During the conduct of exposure significant work, the inspectors evaluated if licensee management was aware of the exposure status of the work and would intervene if exposure trends increased beyond exposure estimates. Additionally, the inspectors attended a station ALARA Committee meeting to assess the degree of oversight in outage dose management.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed (directly or remotely) portions of the following three jobs that were being performed in high or locked high radiation areas that potentially represented significant radiological risk to workers:

- Reactor Head/Upper Internals Lift and Set Activities
- Regenerative Heat Exchanger Activities
- Reactor Coolant Pump Motor Replacement

The licensee's use of ALARA controls for these work activities was evaluated using the following:

- The licensee's use of engineering controls to achieve dose reductions was evaluated to verify that procedures and controls were consistent with the licensee's ALARA reviews.

- Job sites were observed to determine if workers were cognizant of work area radiological conditions and utilized low dose waiting areas and were effective in maintaining their doses ALARA by moving to the low dose waiting area when subjected to temporary work delays.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.5 Source Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to understand historical trends and current status of plant source terms. The inspectors discussed the plant's source term with health physics staff to determine if the licensee had developed a good understanding of the input mechanisms and the methodologies and practices necessary to achieve reductions in source term. The inspectors discussed exposure reduction initiatives taken for U1C20 such as system flushing and use of shielding. Results of the licensee's controlled CRUD burst initiative was reviewed for the outage to assess the adequacy of the cooldown and reactor coolant system cleanup process relative to predictions and historical data.

The inspectors reviewed the licensee's Source Term Reduction 5 Year Plan and discussed its status with health physics staff. The inspectors determined if specific sources had been identified by the licensee for exposure reduction initiatives and if priorities were established or being considered for the implementation these actions.

These reviews represented two inspection samples.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

Radiation worker and radiation protection technician performance was observed during work activities being performed in radiation areas and locked high radiation areas, including work in the upper and lower Unit 1 Containment Building and in the Auxiliary Building. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and the tools to be used for the job, by utilizing low dose waiting areas, and by demonstrating knowledge of the radiological conditions and adhering to the ALARA requirements for the work activity. Job oversight, job support and the communications provided by the radiation protection staff were also evaluated by the inspectors.

This review represented one inspection sample.

b. Findings

No findings of significance were identified.

.7 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessment, audit and field observation reports related to the ALARA program since the last inspection, to assess the licensee's ability to identify and correct problems.

The inspectors assessed the adequacy of the licensee's problem identification processes and verified that identified problems were entered into the corrective action program for resolution. This included post-outage ALARA critiques/lessons learned for exposure performance from the licensee's previous refueling outage in October 2004.

Corrective action reports generated since the end of the licensee's Unit 2 outage in October 2004 and those generated during U1C20 that related to the ALARA program were selectively reviewed, and staff members were interviewed to verify that follow-up activities were being conducted in a timely manner commensurate with their importance to safety and risk using the following criteria:

1. Initial problem identification, characterization, and tracking;
2. Disposition of operability/reportability issues;
3. Evaluation of safety significance/risk and priority for resolution;
4. Identification of repetitive problems;
5. Identification of contributing causes; and
6. Identification and implementation of effective corrective actions.

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies in problem identification and resolution were being addressed.

These reviews represented three inspection samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Some minor issues were entered into the licensee's corrective action system as a result of inspectors' observations, but are not discussed in this report.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

.3 Annual Sample Review

a. Inspection Scope

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

- CR 04162065, "Apparent Level III Violation With A Potential For Civil Penalty of 10 CFR 50.9, Completeness and Accuracy of Information For Inaccurate Information Submitted to the NRC in 1999 License Renewal Application"
- CR 04296044, "Foreign Materials Exclusion Trend and Stop Work Order"

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

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

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

b. Findings

Review of Corrective Actions Associated With Previous Severity Level III Violation

(Open) VIO 05000315/316/2004007-01: "Inaccurate and Incomplete Information to the NRC Regarding the Medical Condition of One Senior Reactor Operator."

On March 24, 2004, the licensee provided information to the NRC regarding the medical status of a licensed senior reactor operator (SRO). That information indicated the SRO had a pre-existing medical condition since 1996 that was considered a potentially disqualifying condition in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS)-3.4 - 1983, "American National Standard Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants," and the SRO license should have required the presence of another qualified individual when the SRO was performing licensed duties. On December 28, 1999, the licensee provided information to the NRC regarding the medical status of the same individual in an application for renewal of the SRO's license and information provided in that renewal application did not describe the individual's pre-existing medical condition from 1996. The individual's license was renewed by the NRC on February 1, 2000, based on information provided by the licensee on December 28, 1999. Therefore, the information provided to the NRC on December 28, 1999, was material to an NRC licensing action. The failure to provide accurate and complete information to the NRC regarding a pre-existing medical condition of an SRO was a significant regulatory issue. If the information had been complete and accurate at the time provided, the NRC would have taken a different regulatory position and would not have renewed the license without requiring the presence of another qualified individual when the SRO was performing licensed duties.

A Severity Level III Notice of Violation was issued for the above event on September 29, 2004. In a letter to the NRC dated August 2, 2004, the licensee stated that in response to this violation, "A 100 percent review (self-assessment) of all operator medical records was performed in February and March of 2004." This response implied that there were no further problems identified with licensed operator medical records. On April 19, 2005, during a follow-up review of the corrective actions for the September 2004 violation, the inspectors identified that another licensed operator had a potentially disqualifying medical condition that had not been reported to the NRC, that existed prior to the licensee's review in March 2004. In response to the inspectors findings the licensee performed another medical record review and identified an additional example of a licensed operator who had a potentially disqualifying medical condition that had not been reported to the NRC and existed prior to the March 24, 2004, medical record review.

This Violation remains open pending further NRC review of the issues identified above.

.4 Cross-Cutting Aspects of Findings

Cross Reference to PI&R Related Findings Documented Elsewhere in the Report

Section 1R12.b.2 of this report describes a finding wherein the licensee's failure to assure that prompt and effective corrective actions were taken to address known conditions adverse to quality affected the reliability and availability of important safety-related plant components.

Section 2OS1 of this report describes a finding wherein the licensee's failure to adequately evaluate a self-identified condition resulted in inappropriate corrective actions for a posted locked high radiation area boundary.

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report (LER) 50-316/2000-003-00: "Containment Internal Concrete Structures Do Not Meet Design Load Margins."

(Closed) LER 50-316/2000-003-01: "Containment Internal Concrete Structures Do Not Meet Design Load Margins," Supplement 1.

On May 29, 2000, during an evaluation of concrete structures inside the Unit 2 containment, the licensee discovered that containment structures did not conform to their design and licensing basis requirements. At that time, both operating units were in an extended shutdown. The licensee reported this event as a condition that was outside the design basis of the plant in accordance with 10 CFR 50.73(a)(2)(ii)(B). Prior to restarting Unit 1 and Unit 2, the licensee committed to return the containment structures to their original design and licensing basis requirements.

The NRC staff performed a detailed review of the methods and calculations used to restore the original design and licensing basis requirements and margins to the containment structural components. In addition to this review, the NRC staff performed a design audit in January 2002. The audit reviewed structural calculations and other documents to verify conformance with design and licensing basis requirements for various structural components within the containment structure. Based on the results of the evaluation and audit, the NRC staff found that, with the exception of the upper reactor cavity area (control rod drive missile shield), the licensee used acceptable methods and appropriate assumptions and design parameters to restore the original design and licensing basis requirements and margins to the containment structural components. On March 11, 2005, the NRC approved a license amendment that revised the design basis as described in the UFSAR to allow the use in control rod drive missile shield structural calculations of a reinforcing bar (i.e., rebar) yield strength value based on measured material properties, as documented in licensee rebar acceptance tests. This resolved the last remaining issue associated with these LERs.

The licensee determined that the root cause for this event was the failure to adequately control the design basis. This root cause evaluation was reviewed by the NRC staff during restart inspection activities and corrective actions were incorporated in the Restart Action Matrix. This Restart Action Matrix item was reviewed and closed in NRC

Inspection Report 05000316/2000007. Since the licensee's operability determination documented in CR 00264095 and CR P-00-02506 ultimately determined that the containment internal concrete structures in both Units 1 and 2 were operable, the inspectors considered this to be a minor issue. These LERs are closed.

.2 (Closed) Licensee Event Report (LER) 50-316/1999-001-00: "Unit 2 Degraded Component Cooling Water Flow to Containment Main Steam Line Penetrations."

(Closed) Licensee Event Report (LER) 50-316/1999-001-01: "Unit 2 Degraded Component Cooling Water Flow to Containment Main Steam Line Penetrations," Supplement 1.

On February 26, 1999, during an engineering review of the Unit 2 containment system, the licensee identified that power operation was permitted in June 1996 with degraded component cooling water (CCW) flow to two containment penetrations, 2-CPN-3 and 2-CPN-4. The main steam lines for steam generators 22 and 23 passed through these penetrations. Operation with degraded CCW flow to these penetrations may have resulted in excessive thermal stresses on the penetration sleeves and liners. The licensee's 1996 operability determination for continued plant operation was based on a misinterpretation of the UFSAR and lack of review of the cooler configuration. The main steam line penetration coolers had two inner and two outer cooling coils. The licensee's initial evaluation concluded that the availability of any two of the four coolers would provide adequate penetration cooling. However, the design was such that each penetration cooler could fulfill its design function with the availability of at least one inner and one outer cooler. As a result, the licensee allowed plant operation with no CCW flow to the inner coolers for 2-CNP-3 and 2-CNP-4. Operation in this condition created the potential for excessive thermal stress on penetration sleeves and liners. Engineering analysis determined that although the concrete temperatures around the penetrations exceeded the 150 degrees Fahrenheit (EF) acceptance criteria (actual <179EF), localized concrete temperatures were acceptable up to 200EF. Subsequent analysis for the event determined that the resulting stresses in the penetration sleeves, anchors, welds, and liners were within allowable stresses.

The licensee determined that the root cause for this event was a less than adequate process for initiation, review, and approval of operability determinations. This led the preparer of the operability determination to reference the previous evaluation without verifying initial assumptions. This was further complicated by ambiguous wording in the UFSAR concerning the containment penetration coolers. Since the licensee's operability determination ultimately concluded that the penetrations were operable, the inspectors considered this to be a minor issue. These LERs are closed.

.3 Unit 2 Reactor Trip Response

a. Inspection Scope

On April 26, 2005, the inspectors responded to the Unit 2 control room after being notified that the reactor had automatically tripped during plant startup preparations to synchronize the main generator to the grid. The trip was caused by an unexpected intermediate range high flux reactor trip signal from channel –35 with indicated reactor power at 8 percent and stable. The cause for the trip was subsequently determined to be a failed bistable relay driver card. Following replacement of the driver card, the unit was synchronized to the grid later the same day. The inspectors assessed control room operator performance immediately following the reactor trip and reviewed the post trip report.

b. Findings

No findings of significance were identified

4OA4 Cross Cutting Aspect of Findings

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

Section 1R12.b.1 of this report describes a finding wherein an inadequate maintenance procedure contributed to human performance errors during maintenance on the Unit 1 west centrifugal charging pump (resources).

4OA5 Other Activities

.1 (Closed) Unresolved Item 50-315, 316/2003-007-02: "Bypassing Degraded Voltage Protection When Power Supplied by Unit Auxiliary Transformers"

On July 11, 2003, the NRC completed a safety system design and performance capability biennial baseline inspection at the D. C. Cook Nuclear Power Plant (IR 50-315, 316/2003007). During the inspection, the inspectors identified that the degraded voltage protection scheme was bypassed whenever the 4160V buses were not being supplied through the reserve auxiliary transformers. Based upon conflicting information, an Unresolved Item (URI 50-315, 316/2003007-02) was opened in the inspection report pending further NRC review to determine the current licensing basis for the D. C. Cook facility with respect to degraded voltage protection and whether the licensee was in conformance with TS 3.3.2.1.

On June 7, 2004, Region III requested assistance from the Office of Nuclear Reactor Regulation (NRR) to resolve the degraded voltage issue associated with the licensing basis and conformance with TS 3.3.2.1. at the Donald C. Cook Nuclear Power Plant under Task Interface Agreement (TIA) 2004-02 (ML041590273). The NRR staff responded by memorandum dated February 28, 2005, (ML043480350) concluding that the existing design was in conformance with the current licensing basis and, therefore, was in compliance with the TS requirements. Further resolution of this issue with the licensee will be pursued by the NRR staff in accordance with 10 CFR 50.109. This Unresolved Item is closed.

.2 (Closed) Temporary Instruction (TI) 2515/163: "Operational Readiness of Offsite Power."

The objective of TI 2515/163, "Operational Readiness of Offsite Power," was to confirm, through inspections and interviews, the operational readiness of offsite power (OSP) systems in accordance with NRC requirements. The inspectors reviewed licensee procedures and discussed the attributes identified in TI 2515/163 with licensee personnel. In accordance with the requirements of TI 2515/163, inspectors evaluated licensee procedures against the attributes discussed below.

The operating procedures that the control room operator used to assure the operability of the OSP have the following attributes:

1. Identify the required control room operator actions to take when notified by the transmission system operator (TSO) that post-trip voltage of the OSP will not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply.

2. Identify the compensatory actions the control room operator is required to perform if the TSO is not able to predict the post-trip voltage for the current grid conditions.

3. Identify the notifications required by 10 CFR 50.72 for an inoperable offsite power system when the site is either informed by its TSO or when an actual degraded voltage condition is identified.

The procedures to ensure compliance with 10 CFR 50.65(a)(4) had the following attributes:

1. Direct the plant staff to perform grid reliability evaluations as part of the required maintenance risk assessment before taking a risk-significant piece of equipment out-of-service to perform maintenance activities.

2. Direct the plant staff to ensure that the current status of the OSP system has been included in the risk management actions and compensatory actions to reduce the risk when performing risk-significant maintenance activities or when LOOP or SBO mitigating equipment are taken out-of-service.

3. Direct the control room staff to address degrading grid conditions that may emerge during a maintenance activity.

4. Direct the plant staff to notify the TSO of risk changes that emerge during ongoing maintenance at the nuclear power plant.

The procedures to ensure compliance with 10 CFR 50.63 had the following attribute:

1. Direct the control room operators on the steps to be taken to attempt to recover offsite power within the station blackout (SBO) coping time.

The results of the inspectors' review were forwarded to Office of Nuclear Reactor Regulation (NRR) for further review and evaluation.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. J. Jensen and other members of licensee management at the conclusion of the inspection on July 7, 2005. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was examined during this inspection, but is not specifically discussed in this report.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Licensed Operator Requalification with Mr. R. Sieber, on April 6, 2005, via telephone.

- Occupational Radiation Safety ALARA Program inspection during the Unit 1 refueling outage with Mr. M. Nazar on April 8, 2005.

- Temporary Instruction 2515/150, Temporary Instruction 2515/160, and Inservice Inspection during the Unit 1 refueling outage with Mr. J. Jensen on April 28, 2005

- Problem Identification and Resolution with Mr. M. Scarpello, on June 13, 2005, via telephone.

- Supplemental exit meeting for Inservice Inspection during the Unit 1 refueling outage with Mr. J. Zwolinski on July 7, 2005, via telephone.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Nazar, Senior Vice President, Chief Nuclear Officer
J. Jensen, Site Vice President
D. Fadel, Engineering Vice President
M. Finissi, Plant Manager
D. Garner, Reactor Vessel Project Manager
R. Gillespie, Operations Director
R. Hall, ISI Program Owner
M. Scarpello, Regulatory Affairs Manager
R. Serocke, Radiation Protection Superintendent
P. Schoepf, Design Engineering Manager
T. Summers, Chemistry Superintendent
L. Weber, Assistant Plant Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000315/316/2005004-01	URI	Potential External and Internal Flooding Impact on Safe Shutdown Equipment (Section 1R06.1)
05000315/2005004-02	NCV	Failure to Use a Code Qualified Weld Procedure for a Weld Overlay Repair Completed on a Pressurizer Nozzle-to-Safe End Weld (Section 1R08.1.b.1)
05000315/2005004-03	URI	Failure to Complete Code Repair or Flaw Evaluation for Pressurizer Safe End-to-Elbow Weld Crack Indication Prior to Returning Component to Service (Section 1R08.1.b.2)
05000315/2005004-04	URI	Failure to Obtain NRC Approval for a Non-Code Compliant Weld Metal Overlay Applied to a Stainless Steel Pressurizer Weld (Section 1R08.1.b.3)
05000316/2005004-05	NCV	Inadequate Maintenance Procedure Led to Extended Time to Complete Unit 1 West Charging Pump Repair (Section 1R12.b.1)
05000315/316/2005004-06	NCV	Failure to Take Prompt Corrective Actions for Conditions Adverse to Quality (Section 1R12.b.2)
05000315/2005004-07	NCV	Physical Barrier for a Locked High Radiation Area Was Not Adequate to Prevent Unauthorized Entry (Section 2OS1.1)

Closed

05000315/2005004-02	NCV	Failure to Use a Code Qualified Weld Procedure for a Weld Overlay Repair Completed on a Pressurizer Nozzle-to-Safe End Weld (Section 1R08.1.b.1)
05000316/2005004-05	NCV	Inadequate Maintenance Procedure Led to Extended Time to Complete Unit 1 West Charging Pump Repair (Section 1R12.1.b.1)
05000315/316/2005004-06	NCV	Failure to Take Prompt Corrective Actions for Conditions Adverse to Quality (Section 1R12.1.b.2)
05000315/2005004-07	NCV	Physical Barrier for a Locked High Radiation Area Was Not Adequate to Prevent Unauthorized Entry (Section 2OS1.1)

50-316/2000-003-00	LER	Containment Internal Concrete Structures Do Not Meet Design Load Margins (Section 4OA3.1)
50-316/2000-003-01	LER	Containment Internal Concrete Structures Do Not Meet Design Load Margins, Supplement 1 (Section 4OA3.1)
50-316/1999-001-00	LER	LER for Unit 2 Degraded Component Cooling Water Flow to Containment Main Steam Line Penetrations (Section 4OA3.2)
50-316/1999-001-01	LER	Supplemental LER for Unit 2 Degraded Component Cooling Water Flow to Containment Main Steam Line Penetrations (Section 4OA3.2)
05000315/316/2003007-02	URI	Bypassing Degraded Voltage Protection When Power Supplied by Unit Auxiliary Transformers (Section 4OA5.1)

Discussed

05000315/316/2004007-01	VIO	Failure to Provide Complete and Accurate Information to the NRC Which Impacted A Licensing Decision (Section 4OA2.3)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- PMP-2291- SCH-002; "Work Control Seasonal Readiness Process;" Revision 1
- PMI-5055; "Winterization/Summerization;" Revision 1
- PMP-5055-001-001; "Winterization/Summerization;" Revision 0
- 12-IHP-5040-EMP-004; "Plant Winterization and De-Winterization;" Revision 5
- PMP-5055-SWM-001; "Severe Weather Guidelines;" Revision 1
- 12-OHP-4022-001-010; "Severe Weather;" Revision 1
- CR 04112035; "Adjustable Hurricane Damper will not Rise to Open Position;" April 21,2004
- CR 04108014; "Transformer Basins on the South End of the Plant are Collecting Sand;" April 17, 2004

1R04 Equipment Alignment

- D. C. Cook Units 1 and 2 TSs and Bases
- D. C. Cook Updated Final Safety Analysis Report, Revision 19
- 01-OHP-4021-017-002; "Placing In Service The Residual Heat Removal System;" Revision 18
- 01-OHP-4021-017-001; "Operation of the Residual Heat Removal System;" Revision 17
- OP-1-5143-65; "Flow Diagram Emergency Core Cooling (RHR) Unit 1;" Revision 65
- 02-OHP-4021-032-008AB; "Operating DG2AB Subsystems;" Revision 7
- OP-2-5151A-51; "Flow Diagram Emergency Diesel Generator 'AB' Unit No. 2;" Revision 51
- OP-2-5151B-63; "Flow Diagram Emergency Diesel Generator 'AB' Unit No. 2;" Revision 63
- OP-2-5135-37; "Flow Diagram CCW Pumps and CCW Heat Exchangers;" Revision 37
- OP-2-5135A-37; "Flow Diagram CCW Safety Related Loads;" Revision 37
- 02-OHP-4021-016-001; "Filling and Venting the Component Cooling Water System;" Revision 13
- 02-OHP-4021-016-003; "Operation of the Component Cooling Water System During System Startup and Power Operation;" Revision 15
- CR 05151065; "Apparent Discrepancy in the CCW Valve Lineup that Potentially Affects Compliance with Tech Specs;" May 31, 2005

1R05 Fire Protection

- D. C. Cook Fire Hazards Analysis; Units 1 and 2; Revision 12 (Fire Zones 42C, 46C, 49, 50, 53 and 54)
- D. C. Cook UFSAR; Section 9.8.1; "Fire Protection System"; Revision 19

C D. C. Cook Fire Pre-Plan; Units 1 and 2, Revision 2 (Fire Areas C, MM, OO, PP and QQ)
• 12-5976-8; "Fire Hazard Analysis Turbine Building Main Floor Elevation 633 Foot;" Revision 8

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

C D. C. Cook Nuclear Plant Updated Final Safety Analysis Report; Revision 19
C Flooding Evaluation for AEP, D. C. Cook Unit #2, S&L Report No. SL-5369; Revision 0, AEP Report Number NED-2000-537-REP; May 19, 2000
C MD-12-SCRN-001-N; "Screen House Internal Flood Levels," Revision 0
C 12-EHP-5035-SMP-001;" Plant Structure Performance Evaluation and Monitoring Program," Revision 4
C DIT-No DIT-B-00305-00; "Safety Significance of Flooding Auxiliary Essential Service Water Electrical Equipment on the 591' Elevation of the Screen House"
C 12-OHP-4022-001-009; "Seiche," Revision 1
• Cook PRA Internal Flooding Analysis Notebook; Section 5; "Quantification of Internal Flooding to Core Damage;" Revision 0, April 29, 1992
• CR 04113028; "Discrepancies Found in Current Internal Flood Analysis;" April 17, 2004
• CR 05158029; "Questioned Whether 12-DR-129 should have Periodic Functional Test;" June 7, 2005

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

C Generic Letter 89-13; "Service Water System Problems Affecting Safety-Related Equipment," July 18, 1989
C Generic Letter 89-13; Supplement 1, "Service Water System Problems Affecting Safety-Related Equipment;" April 4, 1990
C MDS-607; "Heat Exchanger Tube Plugging," Revision 5
C Job Order R0257312-05; "1HE-47-CDN - Perform Visual examination of the Emergency Diesel Generator Combustion Air Aftercooler Heat Exchanger," April 13, 2005
C Job Order R0264015-05; "1HE-47-CDS - Perform Visual examination of the Emergency Diesel Generator Combustion Air Aftercooler Heat Exchanger," April 13, 2005
C CR 05100021; "Discovered CD Aftercoolers Left Unattended Without FME Covers," April 10, 2005

<u>1R08</u> <u>Inservice Inspection Activities</u>

C 12-QHP-5050-NDE-027; Visual Examination for Boric Acid and Condition of Component Surfaces; Revision 1; July 9, 2004
C CALC-SD-050406-001; Cook Nuclear Plant Unit 1 - Calculation of Effective Degradation Years (EDY) of Operation for Unit 1; Revision 0; April 6, 2005
C CR 00290026; Plant Event 37423 - Event Occurring at Another Nuclear Plant - Potential Condition of a Reactor Coolant System Boundary Degradation; October 16, 2000
C CR 01130647; NRC Information Notice 2001-05 - Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head CRDM Penetration Nozzles at Oconee Nuclear Station, Unit 3; May 16, 2001
C CR 04023051; Crack Found on Piping Nozzle Stub for Pressurizer Relief Valve; January 23, 2004

- CR 02136042; Inactive/Passive Boric Acid was Found on the Top of the Reactor Vessel Head; May 16, 2002
- PMP-5030-001-001; Boric Acid Corrosion of Ferritic Steel Components and Materials; Revision 8; July 30, 2004
- 01-OHP-4030-001-002; Containment Inspection Tours; Revision 19; June 7, 2004
- 54-ISI-240-41; Visible Solvent Removable Liquid Penetrant Examination Procedure; Revision 41; February 11, 2003
- CR 03300040; 1-PP-45-2, Evidence of Leakage at Pump/Flange Connection; October 25, 2003
- CR 03300041; 1-PP-45-3, Evidence of Leakage at Pump/Flange Connection; October 25, 2003
- CR 03328037; 1-PP-45-2, Boric Acid Leakage Found on RCP No. 12 Main Flange; November 24, 2003
- CR 04361013; 1-PP-26S, Unit 1 South Safety Injection Pump Has a Small, Unquantifiable Oil Leak at Inboard Pump Bearing Casing

Corrective Action Program Documents With Engineering Evaluations

- CR 04125013; Dry Boric Acid Leak on 1-QFI-220; May 4, 2004
- CR 03291011; Dry Boric Acid (Brown/White) on No. 2 RCP Flange; October 18, 2003

Documents Related to Code Pressure Boundary Welding

- PQR 757A; ASME IX Procedure Qualification Record (PQR); Revision 0; April 11, 2005
- PQR 757A; ASME IX Procedure Qualification Record (PQR); Revision 2; April 11, 2005
- PQR 760; ASME IX Procedure Qualification Record (PQR); Revision 0; April 21, 2005
- WPS 18-WBU/52 MC-GTAW; ASME IX Welding Procedure Specification; Revision 0; April 11, 2005
- WPS 18-WBU/52 MC-GTAW; ASME IX Welding Procedure Specification; Revision 2; April 21, 2005
- WPS 18-WBU/52 MC-GTAW; ASME IX Welding Procedure Specification; Revision 3, April 22, 2005
- Drawing 5D65245; Pressurizer Safety and Relief Nozzle Configurations; Revision 1
- Sketch-1006; Westinghouse Pressurizer Safety and Relief Nozzle (PCI-Pressurizer Safety and Relief Weld Overlay Assembly Illustration); Revision C; April 11, 2005
- Drawing 1097J51; Outline - Pressurizer (1800 Cubic Foot) 6-inch Safety and Relief Nozzles; March 25, 1969
- Certified Material Test Report No: 113056; February 6, 2002
- UT-05-065; UT Report, Nozzle to Safe-End Weld Overlay; April 25, 2005
- UT-05-066; UT Report, Nozzle to Safe-End Weld Overlay; April 25, 2005
- UT-05-067; UT Report, Nozzle to Safe-End Weld Overlay; April 25, 2005
- UT-05-068; UT Report, Nozzle to Safe-End Weld Overlay; April 25, 2005
- U2C15-UT-024; UT Report; October 25, 2004
- AEP-05-45; Pressurizer Safety Nozzle (SST Safe End Weld) Axial Flaw Evaluation; May 4, 2005
- CR 05117045; Indication Identified in the Fusion Zone of Weld 1-RC-9-01F During Final UT of the Weld Overlay for 1-PRZ-23; April 27, 2005
- CR 05118027; Untimely Notification of a Condition Adverse to Quality (CAQ); April 28, 2005

- CR 05099030; Rejectable Indication Found in Pressurizer Nozzle-to-Safe end Weld Joint that Requires Weld Repair; April 9, 2005
- Job Order 03304007; 1-IMO-315, Disassemble, Inspect, Repair Valve; October 31, 2003
- Job Order 03238024; ReplaceU1 SGBD Piping Under Flow Instruments; October 13, 2003

Documents Associated with ASME Code Nondestructive Testing

- 54-ISI-829; Manual Ultrasonic Examination of Dissimilar Metal Piping Welds; Revision 2; September 27, 2004
- 12-QHP-5050-NDE-001; Liquid Penetrant Examination; Revision 4; August 8, 2003

Documents Associated with Disposition of Relevant Indications

- CR 04057001; Suspected Weld Crack at Weld Joint Upstream of 1-CTS-140W; February 26, 2004
- CR 05014039; Liquid Penetrant Inspection of the Unit 1 West Centrifugal Charging Revealed Linear Indications in the Cladding; January 14, 2005

Corrective Action Documents As A Result of NRC Inspection

- CR 05090045; Focal Length Changed Without Re-calibration of Camera per Procedure 12-QHP-5050-NDE-027; April 1, 2005
- CR 05102021; Larger Transducer than Permitted by Procedure was Being Calibrated for Use in the Field; April 8, 2005
- CR 05109030; Rewrite of CR05102021 Which Accepts the Use of PDI Procedure to Gather Data; April 12, 2005
- CR 05111034; Weld Overlay Repair Should Have Included the Requirements for Charpy Impact Tests; April 21, 2005
- CR 05109041; Discrepancy Between Argon Flow Rate on Pressurizer Nozzle Repair Traveler and Repair Procedure; April 19, 2005
- CR 05017016; Information Requested for the In-Office Preparation Week; January 21, 2005
- CR 05111051; Methodologies Utilized to Document and Accept Repairs to the Pressurizer Nozzle-to-Safe End Weld Repair Are Not Clearly Described in CR 09099030; April 21, 2005

1R11 Licensed Operator Requalification

- RQ-E-3020; D. C. Cook Nuclear Plant Dynamic Simulator Evaluation Guide; Period 2 Evaluation - SGTR, Revision 0

1R12 Maintenance Effectiveness

- PMI-5035; "Maintenance Rule Program;" Revision 11
- Maintenance Rule Evaluation Desktop Guide; Revision 1
- Root Cause Evaluation; "Unit 1 West Charging Pump Exceeded 72-hour TS Limiting Condition for Operation;" June 1, 2005

C Root Cause Evaluation; "Unit 2 Pressurizer Manway Leak;" May 27, 2005
C NRC Information Notice 94-76; "Recent Failures of Charging/Safety Injection Pump Shafts;" October 26, 1994
C Letter from Carl F. Lyon, U. S. Nuclear Regulatory Commission, to Mano K. Nazar, American Electric Power; "Subject: Donald C. Cook Nuclear Plant, Unit 1 - Issuance of Emergency Amendment Regarding One-Time Allowed Outage Time Extension for West Centrifugal Charging Pump (TAC No. MC3377);" January 16, 2005
C D. C. Cook Nuclear Station Plant Operating Experience Number 02-001610; "OE 14191 - Aging Gasket Identified As the Cause of a Steam Generator Manway Leak;" July 1, 2002
C CR 05020013; "Delays in the Restoration of the Unit 1 West Centrifugal Charging Pump Including the Need to Rework the Outboard Pump Mechanical Seal Resulted in Exceeding the 72-hour Allowable TS Time;" January 20, 2005
C CR 05013003; "Unit 1 West Charging Pump Had Abnormal Flows and Amps During Operation;" January 13, 2005
C CR 04305061; "Momentary Air Binding of the East Charging Pump While Manually Making Up to the Volume Control Tank from the Refueling Water Storage Tank Via 2-IMO-910, Charging Pumps Suction From the Refueling Water Storage Tan;," October 31, 2004
C CR 03014038; "Operating Experience (OE) 15224 Was Reviewed and Is Applicable to D. C. Cook. This OE discusses the Issue of Centrifugal Charging Pump Shaft Failure at Byron Station. The Cook Plant Has Similar Charging Pumps," January 14, 2003
C CR 03336012; "OE 17355 - Failure of Rotating Element on Centrifugal Charging Pump (Update to OE 16977). This Condition Was Identified at Another Facility." December 2, 2003
C CR 04328019; "The Unit 2 Pressurizer Manway Was Found Leaking During the Walkdown with Unit 2 in Mode 3 to Determine Source of RCS Leakage,; November 23, 2004
C CR 05179010; "Evaluation of OE 17355 Documented in CR 03336012 and Initial Screening of OE 14191 Failed to Take Corrective Action for Conditions Adverse to Quality;" June 28, 2005

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

C D. C. Cook Units 1 and 2 TSs and Bases
C D. C. Cook Updated Final Safety Analysis Report; Revision 19
• PMP-2291-OLR-001; "On-Line Risk Management;" Revisions 5 and 6
C PMP-2291-OLR-001; "On-Line Risk Management," Data Sheet 1; "Work Schedule Review and Approval Form," Cycle 54, Week 6, May 8 through 14, 2005
• PMP-2291-WAR-001; "Work Activity Risk Management Process;" Revision 4
• 12-OHP-5030-057-001; "Screenhouse Vulnerability Determination;" Revision 7
• OHI-4101; "Operations Aggregate Risk Review Process;" Revision 2
• OHI-4101-1; "Operations Aggregate Risk Review Process;" Revision 2
C CR 05076022; "The Unit 2 On-line Risk Would Be in a Red Condition (CDF exceeds 1E-3 and LERF Exceeds 1E-4) Due to the Proposed Work to Clean the Unit 1 ESW Forebay Area;" March 17, 2005
• CR 05171017; "Unit 1 South Rod Drive Motor Generator Set Output Breaker Trip;" June 21, 2005

- EVAL-PA-01-06; "D. C. Cook Unit 1 and Unit 2 Safety Monitor 3.0 Reference Database Development;" Revision 0
- EVAL-PA-01-02; "Unit 1 and Unit 2 Core Damage and Large Early Release Quantification;" Revision 1
- Unit 1 and Unit 2 Control Room Logs; May 11, 2005
- Shift Manager's Logs; May 11, 2005
- Unit 1 and Unit 2 Control Room Logs; June 21-23, 2005
- Daily maintenance schedule; June 21-23, 2005
- Shift Manager's Logs; June 21-23, 2005

1R14 Personnel Performance During Non-Routine Plant Evolutions

- Unit 1 Planned Maintenance Outage Schedule (U1M05A); June 22, 2005
- JOA 05171017-02, 1-67-G1-C, S MG Set; Output Breaker Overcurrent Relay Calibration and Maintenance; June 22, 2005
- 01-OHP-4021-012-001; "Operation of the Control Rod Drive System;" Revision 13
- U1M05A Power Maneuver Reactivity Plan, Revision 1

1R15 Operability Evaluations

- D. C. Cook Units 1 and 2 TSs and Bases
- PMP 7030-OPR-001; "Operability Determinations;" Revision 8
- D. C. Cook Nuclear Plant Updated Final Safety Analysis Report; Revision 18
- CR 05091020; "Modes 1-4 Aggregate Operability Determination for Unit 1;" April 1, 2005
- CR 05131070; "The Operability Review for Condition Report 05124005 relied on Condition Report 03186008 Evaluation Which Assumed a Greater Hydraulic Isolator Range than Is Certified. It Was Assumed 2.5 Cubic Inches Verse Certified of 2.3 Cubic Inches;" May 11, 2005
- CR 05085047; "Safety Valve 1-SV-2B-3 (Steam Generator OME-3-3 Safety Valve 2B) Failed the As-Found Set Point;" March 26, 2005
- CR 05085048; "Safety Valve 1-SV-1B-3 (Steam Generator OME-3-3 Safety Valve 1B) Opened at 1115.54 Psig, Which is Out of Tolerance;" March 26, 2005
- CR 05085049; "Safety Valve 1-SV-1B-1 (Steam Generator OME-3-1 Safety Valve 1B) Opened at 1096.72 Psig, Which Is Out of Tolerance;" March 26, 2005
- CR 05124005; "Received Annunciator 207 Drop 42 RVLIS Train B Hydraulic Isolator Fluid Abnormal;" May 4, 2005
- CR 03186008; "Unit 1 RVLIS Annunciator 107 Drops 41 and 42 Repeatedly Come In. Hydraulic Isolator Replacement May Be Necessary;" July 5, 2003
- SOD 00200; Reactor Coolant System, Revision 4
- CR 04305061; "Momentary Air Binding of the East Charging Pump While Manually Making up to the VCT from the RWST via 2-IMO-910, Charging Pumps Suction from the RWST;" October 31, 2004
- CR 05118027; "Untimely Notification of a Condition Adverse to Quality;" April 28, 2005
- CR 05117045; "Indication Identified in the Fusion Zone of Weld 1-RC-9-01F During Final Ultrasonic Testing of the Weld Overlay for 1-PRZ-23;" April 29, 2005

1R16 Operator Workarounds

- Work Around Review Board Meeting Agenda; May 25 and June 30, 2005
- Unit 1 and 2 Contingency and Compensation Actions; June 28, 2005
- List of Identified Control Room Deficiencies; June 30, 2005
- CR 05136094; "Diesel Generator Overload Concerns When Resetting Phase B"
- CR 04105021; "DRV-407 Caused Cooldown Following Reactor Trip"

1R17 Permanent Plant Modifications

- 1-MOD-45517-R0; "Install Check Valve and Vent Valve in Unit 1 Reactor Coolant Pump Seal Return Line;" Revision 0
- 1-MOD-45517-TP-1; "VCT Check Valve 1-CS-605 Leak Test;" Revision 0
- 01-CMM-30061; "Install High Point Vent Valve in the Component Cooling Water Discharge Piping from the Spent Fuel Pool;" Revision 0
- Job Order 04182058-02; "1-MOD-45517 Perform PMT Leak Inspection;" April 10, 2005
- Job Order 04182058-05; "Perform VT-2 System Leakage Test in Accordance with 12-QHP-5070-NDE-002, Code Case –416-1, and Repair Plan;" April 22, 2005
- Job Order 03268016-01; "1-CMM-30061, Install a High Point Vent to CCW Discharge Line from the Spent Fuel Pool Heat Exchanger 12-HE-16N;" April 14, 2005
- SOD-00300; Chemical Volume and Control System; Revision 3
- OP-1-5129-47; "Flow Diagram CVCS Reactor Letdown and Charging Unit No.1;" Revision 47
- OP-1-5129A-31; "Flow Diagram CVCS Reactor Letdown and Charging;" Revision 31
- DNA History Plot, Volume Control Tank Pressure (P0139A) and Volume Control Tank Temperature (T0140A); April 10, 2005
- DNA History Plot, Volume Control Tank Pressure (P0139A) and Volume Control Tank Temperature (T0140A); April 22, 2005
- Elevation Diagram INT-1-CCW-6-30061; Revision 0
- CR 05131073; "NRC identified that prerequisites were not met prior to performing PMT;" May 11, 2005

1R19 Post Maintenance Testing

- D. C. Cook Updated Final Safety Analysis Report; Revision 19
- 1-MOD-35181-TP-1AB; "Emergency Diesel Generator 1 AB Governor Replacement Modification Test;" Revision 1
- CR 05089028; "During Course of Performing 1-MOD-35181-TP-1AB Section 4.3 Tuning of Diesel Governor 2301A the Diesel Did Not Respond as Expected;" March 30, 2005
- 12-IHP-6030-IMP-355; "Check of Control Room Instrumentation Distribution (CRID) Power Supply Before Returning to Normal Power Source;" Revision 5
- 01-OHP-4021-082-008; "Operation of CRID Power Supplies;" Revision 16
- 12-MHP-5021-056-010; "Turbine Driven Auxiliary Feed Pump Overspeed Trip Test;" Revision 12
- 02-OHP-4030-219-022W; West Essential Service Water System Test
- Job Order R026214603; West Essential Service Water Pump, 2-PP-7W, Check and Adjust Coupling Gap; June 14, 2005
- Job Order R025418501; 1-PP-4, 1-QT-507, "Electronic Overspeed Test;" April 16, 2005

- CR 05131073; "NRC Identified that Prerequisites Not Met Prior to Performing PMT;" May 11, 2005
- Job Order R0231319-05; "1-QT-507, Perform Leak Inspection/Adjust Governor;" April 25, 2005
- 01-OHP-4030-STP-052W; "West Centrifugal Charging Pump Operability Test;" Revision 14
- 01-OHP-4030-156-017T; "Turbine Driven Auxiliary Feedwater System Test;" Revision 0, performed April 24, 2005
- 01-OHP-4030-STP-107R; "Auxiliary Feedwater Pump Response Time," Revision 11; performed April 25, 2005
- PMP-4030-TRT-001; "Time Response and Verification of Engineered Safety Features;" Revision 8
- Design Information Transmittal B-01872-03; "Accuracy of AFW Flow As Read at the Output of FFI-210, 220, 230, 240;" Revision 3
- Design Information Transmittal B-02827-00 and 01; "Unit 1 TDAFP T&TV Time Response Testing;" Revisions 0 and 1
- Design Information Transmittal B-02806-02; "1-OHP-4030-156-017CS Att1 TDAFP Check Valve Test;" Revision 2
- Technical Data Book Figure 1-15.1; "Safety Related Pump Inservice Test Hydraulic Reference;" Revisions 81 and 82
- Technical Data Book Figure 1-15.2; "Safety Related Pump Inservice Test Vibration Reference;" Revision 75
- Technical Data Book Figure 1-19.1; "Power Operated Valve Stroke Time Limits;" Revision 73
- CR 05114021; "Turbine Driven Aux Feed Pump Failed Time to Flow Test per 01-OHP-4030-STP-017R;" April 24, 2005
- CR 05119058; "1-TDAFP T&TV Was Stroke Timed Per OHP-4030-156-017T and the Results Were Faster Than the IST MIN Time. An Immediate Retest Was Not Performed and the Surveillance Was Signed as Satisfactory;" April 29, 2005

1R20 Refueling and Outage Activities

- D. C. Cook Unit 2 TS
- D. C. Cook UFSAR; Revision 19
- 01-OHP-4021-001-001; "Plant Heatup From Cold Shutdown to Hot Standby;" Revision 36
- 01-OHP-4021-001-004; "Plant Cooldown From Hot Standby to Cold Shutdown;" Revision 41
- 01 OHP 4021-017-002; "Placing In Service the Residual Heat Removal System; Revision 18
- 01-OHP-4021-002-005; "Reactor Coolant System Draining;" Revision 30
- 01-OHP-4021-002-013; "Reactor Coolant System Vacuum Fill; Revision 7
- 01-OHP-4030-114-030; Daily and Shiftly Surveillance Checks;" Revision 2
- 01-OHP-4021-001-002; "Reactor Startup;" Revision 32
- 12-OHP-4050-FHP-001; Refueling Procedure Guidelines; Revision 6
- 12-OHP-4050-FHP-005; "Core Unload/Reload and Incore Shuffle;" Revision 6
- 01-OHP-4030-227-041; Refueling Integrity; Revision 5
- 12-EHP-4030-002-356; "Low Power Physics Tests with Dynamic Rod Worth Measurement;" Revision 5

- OP-1-5106A-58; "Flow Diagram Aux-Feedwater Unit 1;" Revision 58
- Clearance Tag List; Clearance Group 1-05, Clearance: R-AFW- AFPW-0246, "Disassemble and Inspect Suction Check Valve;" March 30, 2005
- PMP-4100-SDR-001; "Plant Shutdown Safety and Risk Management;" Revision 7
- PMP 4100-SDR-001; "Plant Shutdown Safety and Risk Management;" Revision 10
- Shift Manager's Logs; March 26 through April 28, 2005
- U1C20 Refueling Outage Schedule Shutdown Risk Review; March 2005
- Drawing OP-1-5143-66; "Flow Diagram Emergency Core Cooling (RHR) Unit No. 1;" Revision 66
- Drawing 1-5663-8; "Unit 1 Reactor Coolant System Loop Details;" Revision 8
- Drawing OP-1-5128-23; "Flow Diagram Reactor Coolant Unit No. 1;" Revision 23
- Drawing OP-1-5128A-48; "Flow Diagram Reactor Coolant Unit No. 1;" Revision 48
- Clearance Tag List; Clearance R-CCW-CCWM-0204, "CCW to North Spent Fuel Pit Heat Exchanger 12-HE-16N Inlet Valve;" March 29, 2005.
- OP-1-5135-41; "Flow Diagram CCW Pump and CCW Heat Exchangers;" Revision 41

1R22 Surveillance Testing

- D. C. Cook Unit 2 TS
- 01-EHP-4030-103-238; "Centrifugal Charging Pump Check Valves Leak Rate Test;" Revision 2
- 01-EHP-4030-103-208; "Unit 1 ECCS Flow Balance - Boron Injection;" Revision 2
- 02-OHP-4030-STP-027AB; "AB Diesel Generator Operability Test (Train B);" Revision 21
- 02-OHP-4021-032-008AB; "AB Operating DG2AB Subsystems;" Revision 7
- 12-MHP-4030-010-003; "Ice Condenser Lower Inlet Door Surveillance;" Revision 5
- Design Information Transmittal S-00105-04; "Ice Condenser Lower Inlet Doors Surveillance Requirements and Bases;" January 24, 2003, performed April 2-11, 2005
- 12-EHP-4030-002-356; "Low Power Physics Tests with Dynamic Rod Worth Measurement;" Revision 5
- 01-EHP-4030-134-203; "Unit 1 LLRT;" Revision 2
- Technical Data Book Figure 2-15.1; "Safety Related Pump Inservice Test Hydraulic Reference;" Revision 67
- Technical Data Book Figure 2-15.2; "Safety Related Pump Inservice Test Vibration Reference;" Revision 56

1R23 Temporary Modifications

- 12-EHP-5040-MOD-001; "Temporary Modifications;" Revision 11
- PMP-2350-SES-001; "10 CFR 50.59 Reviews;" Revision 3
- PMP-2350-SES-001; "10 CFR 50.59 Reviews;" Revision 4
- 2-TM-05-16-R1; "Lift Lead for Main Generator Output Phase 1 Transformer 2-TR-MAIN-1 Cooling Oil Mechanical Relief Vent Line Pressure Alarm Switch;" Revision 1
- Job Order 05157015-01; "Install Temporary Modification 2-TM-05-16-R1;" June 10, 2005
- 01-EHP-4030-134-204; "Unit 1 LLRT," Attachment 1, "Actions for Test Steps C068 and C069;" Revision 2
- PS-1-95236-10; "Containment Spray Water Valves Wiring Diagram;" Revision 10

C PS-1-92479-2; "Aux Relay Cabinet ARB-2 Wiring Diagram;" Revision 2
C PS-1-92475-7; "Aux Relay Cabinet ARA-2 Wiring Diagram;" Revision 7
C PS-1-95240-8; "Emergency Core Cooling Water Valves SH-3 Wiring Diagram;" Revision 8
C PS-1-95234-18; "Emergency Core Cooling Water Valves SH-1 Wiring Diagram;" Revision 18

1EP6 Drill Evaluation

C PMP-2080-EPP-101; "Emergency Classification;" Revision 4
C PMP-2080-EPP-107; "Notification;" Revision 18
C RMT-2080-TSC-001; "Activation and Operation of the Technical Support Center;" Revision 4
C RMA-2080-EPA-008; "Emergency Plan Management;" Revision 0
C RMT-2080-EOF-001; "Activation and Operation of the EOF;" Revision 7
C Timeline With Initial Actions; Emergency Response Drill; June 14, 2005
C Emergency Response Drill Exercise Messages; June 14, 2005
C EMD-32A; "Nuclear Plant Event Notification;" Drill Messages for Declared Unusual Event, Alert and Site Area Emergency; June 14, 2005
C CR 05131072; "The TSC Activated 5 Minutes Late During 5/10/05 Unannounced Off-hours ERO Drill;" May 11, 2005

2OS1 Access Control to Radiologically Significant Areas

C RWP 05-1118; U1C20 Tours and Inspections; Revision 0
C RWP 05-1102; U1C20 Remove/Replace Reactor Head and Upper Internals; Revision 0
C RWP 05-1175; U1C20 Regenerative Heat Exchanger LHRA; Revision 0
C CR 05034002; Request RP Management Inspection of U1 Reactor Coolant Drain Tank LHRA Gate; February 3, 2005
C CR 05097036; NRC-Identified Concern with Adequacy of the Personnel Access Barrier at the Entrance to the Reactor Coolant Drain Tank Room; April 7, 2005

2OS2 ALARA Planning and Controls

C U1C20 RWP Summary and Associated Time/Dose Estimates; undated
C Historical Outage Dose Information for U1C18, U1C19, U2C13, U2C14, and U2C15
C U-1C20 RWP Daily Dose Total Reports for April 4 - 8, 2005
C PMP-6010-ALA-001; ALARA Program - Review of Plant Work Activities; Revision 14
C 12-THP-6010-RPP-014; Total Effective Dose Equivalent Evaluation; Revision 7
C PMP-6010-RPP-006; RWP Program; Revision 8
C RWP 05-1175 and Associated ALARA Plan and TEDE Evaluation; Regenerative Heat Exchanger Activities; Revision 0
C RWP 05-1187 and Associated ALARA Plan; Under Reactor Vessel Inspections; Revision 0
C RWP 05-1155 and Associated ALARA Plan and TEDE Evaluation; Containment and Annulus Sump Activities; Revision 0
C RWP 05-1165 and Associated ALARA Plan and TEDE Evaluation; Containment Reactor Nozzle Pit Activities; Revision 0

- RWP 05-1100 and Associated ALARA Plan and TEDE Evaluation; Refuel Cavity Decontamination Activities; Revision 0
- RWP 05-1151 and Associated ALARA Plan and TEDE Evaluation; Reactor Coolant Pump Seal Maintenance Activities; Revision 0
- RWP 05-1140 and Associated ALARA Plan and TEDE Evaluation; Remove, Reinstall, and Modify Insulation in Containment; Revision 0
- RWP 05-1142 and Associated ALARA Plan; Install, Modify, and Remove Scaffolding from Containment; Revision 1
- RWP 051143 and Associated ALARA Plan; Perform In-Service Inspection Activities in Containment; Revision 0
- RWP 05-1145 and Associated ALARA Plan and TEDE Evaluation; Valve Maintenance and Repair; Revision 1
- ALARA Work In-Progress Review for RWP 05-1101; Refuel Preparation Activities and Reactor Disassembly; March 30, 2005
- ALARA Work In-Progress Reviews for RWP 05-1102; Remove/Replace Reactor Head, Upper Internals, Reactor Head 'O' Ring; April 5 and 7, 2005
- ALARA Work In-Progress Review for RWP 05-1103; Fuel Shuffle and Support Work; April 4, 2005
- ALARA Work In-Progress Review for RWP 051106; Control Rod Drive Mechanism Inspections; April 3, 2005
- ALARA Work In-Progress Review for RWP 05-1114; Operations Activities in Auxiliary and Containment Buildings; April 6, 2005
- ALARA Work In-Progress Review for RWP 05-1123; Temporary Shielding Activities in the Auxiliary and Containment Buildings; April 1, 2005
- ALARA Work In-Progress Reviews for RWP 05-1142; Containment Scaffolding Activities; April 1, 2, and 5, 2005
  ALARA Work In-Progress Review for RWP 05-1145; Valve Maintenance and Repair; April 6, 2005
- ALARA Work In-Progress Reviews for RWP 05-1151; Reactor Coolant Pump Seal Maintenance Activities; April 4 and 6, 2005
- ALARA Work In-Progress Review for RWP 05-1175; Regenerative Heat Exchanger; April 7, 2005
- D. C. Cook Nuclear Power Plant Dose Reduction 5 Year Proposed Plan - 2004; September 24, 2004
- Radiation Protection Self-Assessment SA-2004-RPS-004-F; ALARA Program; December 30, 2004
- Performance Assurance Audit PA-05-01; Radiation Protection; March 10, 2005
- U2C15 Outage Final Report; October 2004
- Performance Assurance Field Observation; Oversight of Work Associated with U2 Containment Recirculation Pump; November 5, 2004
- Performance Assurance Field Observation; Seal Water Injection Filter Change-Out; December 3, 2004
- Performance Assurance Field Observation; Auxiliary Building Walkdown; December 14, 2004
- CR 0400066; Reactor Head Weld Repairs Expended All of its Dose Estimate But Only 50 Percent Complete; October 26, 2004
- CR 05087013; RCS Activity Increase Following Forced Oxidation During U1C20 Shutdown; March 28, 2005

C CR 05096017; Dose Estimate for U1C20 Scaffold RWP Calculated in Error; April 6, 2005

4OA1 Performance Indicator Verification

C Summary of Monthly Dose Calculations and Dose Projections from Liquid & Gaseous Effluents for 2004

4OA2 Problem Identification and Resolution

C 10 License Reactor Operator and Senior Reactor Operator Medical Records
C CR 04162065; "Apparent Level III Violation With A Potential For Civil Penalty of 10 CFR 50.9, 'Completeness and Accuracy of Information For Inaccurate Information Submitted to the NRC in 1999 License Renewal Application;'" June 12, 2004
C CR 04296044; "Weaknesses and Implementation of Standards and Procedures for Foreign Material Exclusion (FME) Have Allowed Foreign Materials to Enter Systems and Components Susceptible to Damage;" October 22, 2004
C CR 04298002; "Issued Stop Work Order on All Contractor and Maintenance FME Activities in the Screen House;" October 23, 2004
C Root Cause Evaluation for Condition Reports 04296044 and 04298002; "Foreign Material Exclusion Trend and Stop Work Order;" January 21, 2005

4OA3 Event Follow-up

C LER 50-316/1999-001; "LER for Unit 2 Degraded Component Cooling Water Flow to Containment Main Steam Line Penetrations,"
C LER 50-316/1999-001-01; "Supplemental LER for Unit 2 Degraded Component Cooling Water Flow to Containment Main Steam Line Penetrations;" Supplement 1
C LER 50-316/2000-003-00; "Containment Internal Concrete Structures Do Not Meet Design Load Margins;" June 28, 2000
C LER 50-316/2000-003-01; "Containment Internal Concrete Structures Do Not Meet Design Load Margins;" Supplement 1, November 20, 2000
C AEP Report No. NED-2000-517-REP; "Large Bore Piping Assessment Report, SL-5366;" Rev. OA, March 17, 2000
• UFSAR Change Request; UCR No. 0381, "Clarification Changes to UFSAR;" May 21, 2001
• MD-12-CCW-818-N; "Main Steam Penetration Thermal Quantification (CNP-2,3,4,5);" November 26, 2002
C CR P99-12563; "Design Basis Thermal Analysis of Containment Hot Penetrations not Retrievable;" May 19,1999
C CR 99-03641; "Unit 2 Operation was Allowed to Continue Based on Technically Incorrect Evaluation of the Loss of the Penetrations Inner Coolers;" February 26, 1999
• CR 98-6832; "Results of Initial AEP Finite Element Nodal Analysis of Penetrations;" November 10, 1998
• CR 97-1374; "Safety Review Memo was Written to Evaluate a Temporary Modification (02-TM-0729);" April 29, 1997
• CR 96-0937; "No Fluid was Drained through the 2-CCW-377-72 Drain Line on CCW to CNP 3 & 4 Cooling Coil;" June 13, 1996

- CR 94-1682; "1-CCR-441, West Main Steam Cooling Penetration for CCW went Closed for No Apparent Reason;" August 28, 1994
- CR P-99-00594; "Design Basis Integrity Not Controlled, Maintained, and Respected by the Cook Team;" January 11, 1999
- CR 00264095; "Physical Non-conformances Found on the #4 Accumulator Room End Wall [Unit 1];" September 20, 2000
- CR P-00-02506; "Poor Quality of Grout/concrete Found near the Top at the 18" [Inch] Thick Concrete Wall Between the Accumulator #22 Room and the CEQ Fan Room;" February 10, 2000
- Letter from Mr. J. Dyer, Region 3, Regional Administrator, to Mr. R. Powers, Senior Vice President Nuclear Generation Group, American Electric Power Company; "Closure of NRC Inspection Manual Chapter 0350 Restart Action Plan For Restart of the Donald C. Cook Nuclear Plant - Unit 2;" June 13, 2000
- "Safety Evaluation by The Office of Nuclear Reactor Regulation Related to Amendment No. 286 to Facility Operating License No. DPR-58 and Amendment No. 268 to Facility Operating License No. DPR-74 Indiana Michigan Power Company Donald C. Cook Nuclear Plant; Units 1 and 2 Docket Nos. 50-315 and 50-316;" March 11, 2005

4OA5 Other Activities

- D. C. Nuclear Plant; Units 1 & 2 - Response To Task Interface Agreement (TIA 2004-02), Request for Technical Assistance Regarding Degraded Voltage Protection (TAC Nos. MC3428 and MC3429)
- 01-OHL-4030-SOM-031; "Unit 1 Tours - Unit 1 CR M1&2 Shift Checks;" Revision 5
- 02-OHL-4030-SOM-041; "Unit 2 Tours - Unit 2 CR M1&2 Shift Checks;" Revision 7
- 01-OHP-4030-114-031; "Operations Weekly Surveillance Checks;" Revision 4
- 02-OHP-4030-214-031; "Operations Weekly Surveillance Checks;" Revision 4
- PMP-3100-IOA-001; "Inter-organizational Agreement Between the AEP Energy Delivery and the AEP Nuclear Generation Group for Assistance to Cook Nuclear Plant;" Revision 0
- 12-OHP-4022-082-004; "Degraded Offsite AC Voltage Response;" Revision 4
- D. C. Cook Unit 1(2) Plant TS
- Donald C. Cook Nuclear Plant Emergency Plan; Revision 20
- PMP-2080-EPP-101; "Emergency Classification;" Revision 5
- PMP-2291-OLR-001; "On-line Risk Management;" Revision 7
- PMP-2291-EXE-001; "Work Control Activity Execution Process;" Revision 12
- PMP-2291-WAR-001; "Work Activity Risk Management Process;" Revision 4
- PMP-4100-SDR-001; "Plant Shutdown Safety and Risk Management;" Revision 10
- 01-OHP-ECA-0.0; "Loss of All AC Power;" Revision 14
- 01-OHP-4023-SUP-002; "Restoration of Reserve Power to 4kV Buses;" Revision 5
- 02-OHP-ECA-0.0; "Loss of All AC Power;" Revision 13
- 02-OHP-4023-SUP-002; "Restoration of Reserve Power to 4kV Buses;" Revision 5a
- 01-OHP-4022-001-005; "Loss of Offsite Power with Reactor Shutdown;" Revision 5
- 02-OHP-4022-001-005; "Loss of Offsite Power with Reactor Shutdown;" Revision 6

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
AFW	Auxiliary Feedwater
ALARA	As Low As Is Reasonably Achievable
ANSI/ANS	American National Standard Institute/American Nuclear Society
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRID	Control Room Instrumentation Distribution
CRDM	Control Rod Drive Mechanism
U1C20	D. C. Cook's 20th Unit 1 Refueling Outage
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EHP	Engineering Head Procedure
FIT	Failure Investigation Team
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
KV	Kilovolt
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LLRT	Local Leak Rate Test
LOOP	Loss of Offsite Power
MG	Motor generator
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
OHP	Operations Head Procedure
OSP	Offsite Power
OWA	Operator Workaround
PARS	Publically Available Records
PDI	Performance Demonstration Initiative
PI	Performance Indicator
PMI	Plant Manager's Instruction
PMP	Plant Manager's Procedure
PQR	Procedure Qualification Record
psig	Pounds Per Square Inch Gage
PT	Penetrant Test (Dye Penetrant Examination)
RCS	Reactor Coolant System
RWP	Radiation Work Permit
SBO	Station Blackout
SDP	Significance Determination Process
SG	Steam Generator
SRO	Senior Reactor Operator
SSC	Structures, Systems, and Components
TI	Temporary Instruction

TIA	Task Interface Agreement
TS	Technical Specification
TSO	Transmission System Operator
TEDE	Total Effective Dose Equivalent
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Examination
VT	Visual Examination
WPS	Welding Procedure Specification