February 28, 2002

EA-02-021

Mr. A. C. Bakken III
Senior Vice President
Nuclear Generation Group
American Electric Power Company
One Cook Place
Bridgman MI 49106

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 50-315/01-20(DRP); 50-316/01-20(DRP)

Dear Mr. Bakken:

On February 9, 2002, the NRC completed an inspection at your D. C. Cook Nuclear Power Plant, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on February 13, 2002 with you 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, seven issues of very low safety significance (Green) were identified. These issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC Enforcement Policy. If you contest the Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the D. C. Cook facility.

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

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

Sincerely,

/RA/

Anton Vegel, Chief
Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 50-315/01-20(DRP);
50-316/01-20(DRP)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/01-20(DRP); 50-316/01-20(DRP)
Licensee:	American Electric Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	1 Cook Place Bridgman, MI 49106
Dates:	December 30, 2001 through February 9, 2002
Inspectors:	B. Kemker, Senior Resident Inspector K. Coyne, Resident Inspector J. Maynen, Resident Inspector M. Holmberg, Senior Reactor Inspector
Approved by:	A. Vegel, Chief Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000315-01-20(DRP), IR 05000316-01-20(DRP), on 12/30/2001-02/09/2002, Indiana Michigan Power Company, D. C. Cook Nuclear Power Plant, Units 1 and 2. Inservice Inspection Activities, Post Maintenance Testing, Refueling and Outage Activities, Surveillance Testing.

The baseline inspection was conducted by resident and region based inspectors. The inspectors identified seven Green findings. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, “Significance Determination Process” (SDP). The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html. Findings for which the SDP does not apply are indicated by “No Color” or by the severity level of the applicable violations.

A. Inspector Identified Findings

Cornerstones: Barrier Integrity

- Green. A Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion IX was identified for an inadequate calibration of an ultrasonic transducer used to size flaws in the pressurizer girth weld.

  This finding had the potential to affect the initiating events and barrier integrity cornerstones and was more than minor because it had a credible impact on safety, in that, errors in the ultrasonic testing calibration invalidated the flaw sizes recorded. Because this finding did not result in degradation of the reactor coolant system pressure boundary, the risk significance was very low (Green) as determined by the Reactor Safety Significance Determination Process. (Section 1R08)

- Green. A Non-Cited Violation of Technical Specification 4.0.5.a was identified for application of incorrect acceptance criteria to flaws in the pressurizer vessel welds.

  This finding had the potential to affect the initiating events and barrier integrity cornerstones and was more than minor because these types of errors, if left uncorrected, could result in acceptance of a flaw size greater than that allowed by the American Society of Mechanical Engineers Code. Because this finding did not result in degradation of the reactor coolant system pressure boundary, the risk significance was very low (Green) as determined by the Reactor Safety Significance Determination Process. (Section 1R08)

- Green. A Non-Cited Violation of 10 CFR 50.55a(g)(5)(iii) was identified for failure to obtain NRC concurrence (Code relief) associated with incomplete weld examinations.

This finding had the potential to affect the barrier integrity and initiating events cornerstones and was more than minor because, the reduced examination of welds was left uncorrected, which could result in operation with undetected flaws affecting the reactor coolant system pressure boundary. Because this finding did not result in degradation of the reactor coolant system pressure boundary, the risk significance was very low (Green) as determined by the Reactor Safety Significance Determination Process. (Section 1R08)

- Green. A Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the licensee's failure to adequately correct a failure of containment isolation valve 2-CCR-440 during routine inservice testing on April 11, 2001. Specifically, the licensee adjusted the 2-CCR-440 position indication limit switch mechanism to obtain indication of valve closure without verifying that 2-CCR-440 was capable of fully closing. Subsequently, on January 20, 2002, 2-CCR-440 failed a 10 CFR 50, Appendix J, leak rate test due to the valve not being fully closed.

  The inspectors assessed this finding using the Significance Determination Process. The inspectors concluded that this issue represented an actual degradation in the redundancy of a containment penetration barrier and had a credible impact on safety and was more than a minor concern. The inspectors determined that the failure of a containment isolation valve was associated with the containment barrier and was within the barrier integrity cornerstone. As described in Updated Final Safety Analysis Report, Table 5.4-1, "Unit 2 Containment Penetration Isolation Barriers," 2-CCR-440 was a barrier for containment penetration CPN-25. The second barrier for CPN-25 was composed of the closed component cooling water (CCW) system piping loop inside containment. Based on satisfactory Appendix J, Type C, leak rate test results for CPN-25 obtained on January 29, 2002, the inspectors determined that the CCW piping inside containment was intact and that failure of 2-CCR-440 to fully close did not represent an actual open pathway in the physical integrity of the reactor containment. Consequently, the inspectors determined that the failure to properly evaluate and correct the cause for the inservice stroke time test in April 2001, did not result in an open leak path from the Unit 2 containment. (Section 1R19)

- Green. A Non-Cited Violation of Technical Specification (TS) 6.8.1 was identified for the licensee's failure to adequately implement the requirements of 12-MHP 4030.010.003, "Ice Condenser Lower Inlet Door Surveillance." Specifically, the licensee failed to adequately perform the following: (1) install a protective end tip on the spring scale to protect the lower ice condenser doors from damage as required by step 4.2.6, (2) ensure that installation of the TE-132 test fixture met the moment arm and degree of opening requirements in accordance with steps 4.2.3 and 4.2.5, and (3) accurately record surveillance test data for lower inlet door limit switch checks as required by steps 4.1.8.d and 4.1.9.d.

  The inspectors assessed this finding using the Significance Determination Process. The inspectors determined that the failure to correct these procedural implementation inadequacies could become a more significant safety concern if left uncorrected and was therefore more than a minor concern. Specifically, the failure to adequately perform surveillance testing could result in the failure to identify degraded or inoperable safety related equipment. Because the ice condenser was primarily associated with containment heat removal following certain design basis accidents, the inspectors concluded that this issue was associated with the barrier integrity cornerstone. Because the Unit 2 ice condenser was not required to be capable of performing a safety related function immediately following the inadequate surveillance testing on January 24, 2002, the inspectors concluded that this issue did not result in an actual loss or degradation of the heat removal function performed by the ice condenser. (Section 1R22)

- Green. The inspectors identified a Non-Cited Violation of Unit 2 Technical Specification (TS) 4.6.5.3.1.b.3, 4.6.5.3.1.b.4, and 4.6.5.3.1.b.5 requirements associated with testing of the ice condenser lower inlet doors. Contrary to the TS requirements, previous TS 4.6.5.3.1.b surveillance testing performed in Unit 2 on April 21, April 22, and May 4, 2000 failed to adequately measure the door opening torque and the door closing torque in accordance with the TS requirements. Specifically, the methodology used by the licensee to perform TS 4.6.5.3.1.b.3 and 4.6.5.3.1.b.4 testing resulted in door closing torques that were greater in magnitude than the door opening torques, contrary to the TS description of these torque values. The inspectors identified that the measured opening torque values for all the Unit 2 lower inlet doors were less than the associated door closing torque values. Because calculation of the door frictional torque required accurate measurement of the door opening and closing torques, the licensee was unable to demonstrate compliance with the requirements of TS 4.6.5.3.1.b.5.

  The inspectors assessed this finding using the Significance Determination Process. The inspectors determined that the failure to adequately implement TS 4.5.6.3.b requirements for testing of the Unit 2 lower inlet doors had a credible impact on safety and was more than a minor concern. As stated in the TS 3.6.5 bases, operability of the ice condenser doors ensures that reactor coolant fluid released during a loss of coolant accident (LOCA) will be diverted through the ice condenser bays for heat removal. The ice condenser also augmented the containment recirculation sump water inventory in the event of certain small break LOCAs and limited ice maldistributions within the ice condenser. Because the proper functioning of the ice condenser lower inlet doors was primarily associated with the heat removal function of the ice condenser, the inspectors determined that this issue was associated with the barrier integrity cornerstone. Based on a review of additional testing results for the Unit 2 lower inlet doors performed on February 3 and 4, 2002, the inspectors concluded that there was no actual reduction in the atmospheric pressure control function of the reactor containment nor a loss of capability to provide additional recirculation sump inventory during certain small break LOCAs. (Section 1R22)

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

The inspectors evaluated this failure to meet Unit 1 TS 4.6.5.3.1.b requirements using the Significance Determination Process. However, at the conclusion of this inspection period, the licensee had not been able to test the Unit 1 ice condenser lower inlet doors using a methodology consistent with TS 4.6.5.3.1.b requirements. Due to the unavailability of Unit 1 lower inlet door performance data, the inspectors identified the licensee's failure to adequately perform Unit 1 lower inlet door testing as required by TS 4.6.5.3.1.b as an URI pending a final safety significance evaluation. Consequently, the risk significance of this issue will be characterized as to be determined (TBD) until the performance of the Unit 1 ice condenser lower inlet doors can be assessed. (Section 1R22)

Cornerstone: Mitigating Systems

- Green. A Non-Cited Violation of 10 CFR 50.65(a)(4) was identified for the licensee's failure to assess the risk associated with maintenance activities affecting both Unit 2 safety injection (SI) system pumps. Operators deviated from the licensee's outage schedule and prematurely vented and drained both Unit 2 SI system pumps without assessing the increase in shutdown risk during a period of reduced reactor coolant system (RCS) inventory. This resulted in the inadvertent entry into a higher shutdown risk configuration, for which the licensee had not implemented additional risk management actions to protect available equipment and to maintain an adequate level of defense as required by the licensee's plant shutdown safety and risk management procedure.

  The inspectors assessed this finding using the Significance Determination Process. The inspectors concluded that this issue had a credible impact on safety because the SI pumps were made unavailable for core cooling in the event of a loss of RCS inventory. At the time, Unit 2 was in Mode 5 (Cold Shutdown) with the RCS loops not filled and vented, and only one of the two Unit 2 centrifugal charging pumps was available. The inspectors reviewed the guidance in Inspection Manual Chapter (IMC) 0609, Appendix G, "Shutdown Operations Significance Determination Process," including the checklist for "Pressurized Water Reactor Cold Shutdown and Refueling Operation - Reactor Coolant System Closed and No Inventory in Pressurizer." Although having both SI pumps unavailable degraded the licensee's ability to add inventory to the

RCS, the inspectors determined that sufficient plant equipment existed to keep the core covered because the capability existed for operators to cross-tie the Unit 1 and Unit 2 charging systems to make an additional centrifugal charging pump available. The inspectors concluded that this issue was of very low safety significance because there was no challenge to RCS inventory control during the time that the SI pumps were unavailable. (Section 1R20)

B. Licensee Identified Violations

No violations of significance were identified.

Report Details

Summary of Plant Status:

Unit 1 operated at or near full power for the duration of the inspection period.

Unit 2 operated at or near full power until January 19, 2002, when the licensee conducted a reactor shutdown for refueling outage U2C13. Unit 2 was defueled at the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

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

Mitigating Systems Cornerstone

- Unit 2 Residual Heat Removal System

The inspectors selected this system based on its risk significance relative to the mitigating systems cornerstone. The inspectors reviewed operating procedures, Technical Specification (TS) requirements, Administrative Technical Requirements (ATRs), system diagrams, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the system incapable of performing its intended functions.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

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

Mitigating Systems Cornerstone

- Unit 2 Emergency Core Cooling System Pump Rooms (Fire Zones 1E, 1F, 1G, 1H, 63A, 63B, 63C, 65A, and 65B)

- Unit 2 Containment Building Piping Annulus and Lower Volume (Fire Zones 74 and 75)
- Unit 1 East Centrifugal Charging Pump Room (Fire Zone 62B)
- Unit 1 West Centrifugal Charging Pump Room (Fire Zone 62C)

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

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's inservice inspection program for monitoring degradation of the reactor coolant system boundary and risk significant piping system boundaries, based on review of records and in-process observation of nondestructive examinations. From January 22, 2001, through February 6, 2002, the inspectors performed the following activities:

- Reviewed repair and replacement records required by the American Society of Mechanical Engineers (ASME) Code for the following components:

    - Job Order C003595402, "2-FW-132-4 Replace Valve by Welding"
    - Job Order C004223902 ,"2-CTS-105E Remove Flanges & Weld into System"
    - Job Order C004083002, "2-OME-3-2 Repair Manway Bolt Holes as Required;"

- Reviewed licensee corrective actions for Code recordable indications for five weld examinations identified during the first period of the third Code interval;
- Reviewed nine weld examination records (of ultrasonic (UT), magnetic particle (MT), or dye penetrant (PT)) on Class 1 and 2 components from the first period of the third Code interval; and
- Observed acquisition and evaluation of eddy current (ET) data on the Unit 2 steam generators.

The records reviewed and activities observed were evaluated for conformance with requirements in the ASME Code, Section III, Section V, Section IX, and Section XI.

The inspectors also reviewed a sample of inservice inspection related problems documented in the licensee's corrective action program, to assess conformance with 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Action," requirements.

b. Findings

b.1 Inadequate Sizing of Flaws Identified in the Pressurizer Girth Weld

The inspectors identified a finding of very low safety significance (Green) associated with a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion IX for an inadequate calibration of an ultrasonic transducer used to size flaws in the pressurizer girth weld.

Description

On November 5, 1997, the licensee identified three indications on the pressurizer girth weld (2-PRZ-11) while performing the examination required by the Code. The licensee considered these indications detected by UT to be of acceptable size and returned these welds to service. On February 5, 2002, the inspectors identified errors in the calibration of the transducer used to size these indications. On the UT calibration sheet U2R97-330 completed on November 14, 1997, an 11.25 inch metal path and 60 degree angle were recorded. This information placed the notch location at a depth greater than the thickness of the calibration block. Additionally, the far side notch amplitude response of 80 percent was equivalent to the 1/4T side drilled hole response, which was not possible with the 25 decibel gain setting recorded. These errors indicated that the calibration for this transducer was done incorrectly. Because this transducer was used to size the indications in the girth weld, the inspectors questioned the accuracy of the measured flaw location and sizes.

The licensee subsequently demonstrated a new calibration using the same transducer and calibration block. This re-calibration identified that the correct angle for the transducer used was actually 64 degrees and that the correct gain setting was 80 decibels. The licensee reevaluated the flaw data recorded based on this revised calibration data and determined that these flaws remained acceptable per the Code.

Analysis

This finding had the potential to affect the barrier integrity cornerstone. This finding was more than minor because it had a credible impact on safety, in that, errors in the UT calibration invalidated the flaw sizes recorded. Fortuitously, after re-calibration and reevaluation, the size of these flaws remained within the Code acceptance criteria and no degradation of the reactor coolant system pressure boundary occurred. Therefore, the risk significance was very low (Green) as determined by the Reactor Safety Significance Determination Process.

Enforcement

10 CFR Part 50, Appendix B, Criterion IX, “Control of Special Processes,” requires, in part, that measures shall be established to assure that special processes, including nondestructive testing are controlled by personnel using qualified procedures. Section 6.6.4 of procedure 54-ISI-130-33, “Ultrasonic Examination of Ferritic Vessel Welds Greater Than Two Inches Thick,” required setting the amplitude and measuring the location of the transducer relative to the opposite surface notch. Contrary to these

requirements, on November 14, 1997, as recorded in calibration sheet U2R97-33, the licensee failed to correctly establish the location and amplitude of the opposite surface notch on the calibration block. This is considered a violation of 10 CFR Part 50, Appendix B, Criterion IX. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-316-01-20-01(DRS)) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the licensee's corrective action program in condition report (CR) 02035046.

b.2 Errors in Application of Code Flaw Acceptance Criterion

The inspectors identified a finding of very low safety significance (Green) associated with a Non-Cited Violation of TS 4.0.5.a. for application of incorrect acceptance criteria to flaws in the pressurizer vessel welds.

Description

On November 5, 1997, the licensee identified three indications on the pressurizer girth weld (2-PRZ-11) while performing the Code required volumetric examination. The licensee considered these indications detected by UT to be of acceptable size and returned these welds to service. On January 24, 2002, the inspector identified the following errors in the flaw evaluation as documented in U2R97-023:

- The acceptance criteria applied to indication 1, for the flaw depth to thickness ratio was documented as 6.6 percent. This was not the correct acceptance criteria. From ASME Code Section XI Table IWB 3510-1, the correct acceptance criteria to use was 3.8Y, which for this subsurface flaw in 4" material was 2.8 percent.

- The acceptance criteria for indication 2 and 3 was not calculated or definitively selected. The licensee had documented 14.3Y or 7.6Y as the acceptance criteria for the flaw depth to thickness ratio. From ASME Code Section XI Table IWB 3512-1, the appropriate acceptance criteria was 7.6 percent.

- For indication 2 and 3, from the ASME Code Section XI, IWA-3320-1 the minimum required flaw length to use was equal to twice the depth. However, the licensee used a flaw length below this minimum value and calculated flaw aspect ratios of 0.68 and 0.605. These aspect ratios were greater than the ASME Code Section XI, IWA-3300(a)(3), maximum allowable ratio of 0.5.

Fortuitously, these errors did not result in acceptance of a Code rejectable indication.

On October 31, 1997, the licensee identified an indication on the pressurizer nozzle to shell weld (6"-2-RC-25) while performing the Code required examination as documented in U2R97-010. On January 24, 2002, the inspectors identified that this indication was accepted based on criteria from Article 4 and 5 of Section V of the ASME Code. However, these Code Sections do not contain flaw acceptance criteria. The correct acceptance criteria for flaws identified during inservice inspection are found in Table IWB-3412-1 of Section XI. Fortuitously, this error did not result in acceptance of a Code rejectable indication.

Analysis

This finding had the potential to affect the barrier integrity cornerstone. This finding was more than minor because, these types of errors, if left uncorrected could result in acceptance of a flaw size greater than that allowed by the Code. Fortuitously, these errors did not directly result in accepting a Code rejectable flaw and no degradation of the reactor coolant system pressure boundary occurred. Therefore, the risk significance was very low (Green) as determined by the Reactor Safety Significance Determination Process.

Enforcement

TS 4.0.5.a requires that for Class 1, 2 and 3 components that Inservice Inspection requirements of the ASME Code Section XI are followed. For flaws identified in pressurizer welds 6"-2-RC-25 and 2-PRZ-11 on October 31, 1997, the applicable acceptance criteria are found in Section XI Table IWB 3410-1 and Table IWB-3412-1. Contrary to the above, for flaws identified in examinations U2R97-023 and U2R97-010, the licensee failed to use the applicable acceptance criteria from these tables. This is considered a violation of TS 4.0.5.a. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-316-01-20-02(DRS)) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the licensee's corrective action program in CR 02035044 and CR 02035046.

b.3 NRC Approval of Limited Code Weld Examinations Not Obtained

The inspectors identified a finding of very low safety significance (Green) associated with a Non-Cited Violation of 10 CFR 50.55a(g)(5)(iii) for failure to obtain NRC concurrence (Code relief) associated with incomplete weld examinations.

Description

On January 23, 2002, the inspectors identified several ultrasonic examinations of ASME Code components for which the required weld and base metal volume was not completely scanned. These examinations typically had access limitations, which precluded obtaining the required Code examination volume. However, the licensee failed to request NRC concurrence to deviate from the Code requirements for these limited examinations which were completed in the Second Code interval for Unit 1 and 2. Examples of Unit 2 welds with limited examinations are identified below:

- 316540, Containment Spray, PT & UT Exam Report 2-CTS-10-12F, April 5, 1996
- 011400, Steam Generator 24, UT Exam Report STM-24-I-IRS, April 8, 1996
- 011600, Steam Generator 24, PT & UT Exam Report STM-24-02, April 10, 1996
- 011700,Steam Generator 24, PT & UT Exam Report STM-24-03, April 10, 1996
- 009500, Steam Generator 22, UT Exam Report STM-22-01, April 11, 1996

Some of these limited examinations lacked information to determine the amount of the Code required volume examined. For example the UT examinations of the elbow to inlet and outlet nozzle welds for steam generator 24 did not quantify the volume examined. For a valve to pipe weld (2-CTS-10-12F), the examination, coverage could have been increased with additional examinations using other types of transducers. For examinations of the steam generator welds (STM-24-02 and STM-24-03), the licensee identified nozzle surface taper as a limitation which did not exist during the pre-service UT. Therefore, a technical basis beyond what was documented would be needed to justify the inservice limitations.

Based on these examples, the inspectors concluded that additional nondestructive examinations of welds would likely be needed prior to obtaining NRC approval for deviation from the Code volumetric weld inspection requirements. The limited examination problem was applicable to both Unit 1 and Unit 2. At the conclusion of this inspection the licensee had identified 48 Unit 1 inservice examinations from the second Code interval which were also limited.

The NRC issued information notice IN 98-42, "Implementation of 10 CFR 50.55a(g) Inservice Inspection Requirements," which described a failure of other NRC licensees to submit for Code relief for limited examinations. The licensee entered this information notice into the corrective action system on June 9, 1999 (Condition Report P-99-14983), and documented that Cook had limited examinations. However, as of January 23, 2002, the licensee had not taken actions to correct this condition.

Analysis

This finding had the potential to affect the barrier integrity cornerstone. This finding was more than minor because, the reduced examination of welds was left uncorrected, which could result in operation with undetected flaws affecting the reactor coolant pressure boundary. Subsequently, the licensee performed an operability determination and found that the affected systems were operable because the limited examinations were completed to the extent practical. Because this finding did not result in degradation of the reactor coolant system pressure boundary, the risk significance was very low (Green) as determined by the Reactor Safety Significance Determination Process.

Enforcement

10 CFR 50.55a(g)(5)(iii) requires that requests for relief from limited examinations be submitted when it is impractical to complete the examination coverage requirements of Section XI of the ASME Code. Further, 10 CFR 50.55a(g)(5)(iv) requires that relief requests be submitted to the NRC within one year of the end of the Code interval. The second Code interval ended on June 30, 1996. Contrary to the above, as of January 23, 2002, the licensee failed to request relief from the ASME Code for the limited examinations of welds identified during the Second Code interval on Units 1 and 2. This finding is considered a violation of 10 CFR 50.55a(g)(5)(iii). Because of the very low safety significance, this violation is being treated as a Non-Cited Violation (NCV 50-315/316-01-20-03(DRS)) consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is documented in the licensee's corrective action program in CR 02023050.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

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

Mitigating Systems Cornerstone

- 250 Volt Direct Current Fuse Holders

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

Initiating Events Cornerstone

- Unit 2 Dual Train Essential Service Water (ESW) Pump Outage

Mitigating Systems Cornerstone

- Unit 2 West Motor Driven Auxiliary Feedwater Pump
- Unit 1 West ESW Pump Replacement

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

b. Findings

A finding related to an inadequate maintenance risk assessment is discussed below in Section 1R20, "Refueling and Outage Activities."

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors evaluated the licensee's basis that the issues identified in the following condition reports did not render the involved equipment inoperable or result in an unrecognized increase in plant risk:

Barrier Integrity Cornerstone

- CR 02032016 Operability of Unit 1 Ice Condenser Lower Inlet Doors

Mitigating Systems Cornerstone

- CR 02017002 1-ESW-115, Essential Service Water to Turbine Driven Auxiliary Feedwater Pump Shutoff Valve Will Not Open
- CR 02019039 2-NRV-153 Is Inoperable Due to Too Fast Stroke Time

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

b. Findings

A finding associated with the surveillance testing and operability of the Unit 1 ice condenser lower inlet doors is discussed below in Section 1R22, "Surveillance Testing."

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

Barrier Integrity Cornerstone

- Job Order 02021004 Repair 2-CCR-440 to Correct Seat Leakage

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

b. Findings

The inspectors identified a finding of very low safety significance (Green) associated with the corrective actions taken for the failure of containment isolation valve 2-CCR-440 (containment penetration CPN-2 and CPN-5 inner cooling coils component cooling water (CCW) outlet isolation valve) to fully close during quarterly inservice testing (IST). This issue was self-revealed on January 20, 2002, following the failure of 2-CCR-440 to meet the leakage rate testing requirements of 12-EHP 4030.234.203, “Unit 2 B & C Leak Rate,” during 10 CFR 50, Appendix J, Type C, leak rate testing. This finding was dispositioned as a Non-Cited Violation.

Description

On April 11, 2001, valve 2-CCR-440 failed to indicate closed during routine IST stroke time testing. Following this valve failure, the licensee declared valve 2-CCR-440 inoperable and initiated CR 01101073. Job Order 01101073 was issued to investigate and repair the cause of the testing failure. The licensee determined that rotation of the limit switch striker plate resulted in the failure of the closed limit switch to actuate during the testing. The licensee aligned the limit switch mechanism to obtain closed indication and declared valve 2-CCR-440 operable on April 12, 2001. Based on a review of Job Order 1101073, the inspectors determined that the corrective actions taken for the IST testing failure of 2-CCR-440 were not adequate to ensure that 2-CCR-440 was capable of performing its safety function. Specifically, maintenance workers adjusted the 2-CCR-440 limit switch mechanism without verifying that the valve was capable of fully closing. Job Order 01101073 originally included steps to verify that valve 2-CCR-440 was making hard seat contact, but these steps were closed out by the maintenance supervisor with no work performed. On January 20, 2002, 2-CCR-440 failed to meet the 12-EHP 4030.234.203 acceptance criteria with a leak rate in excess of the maximum allowable containment leak rate. The licensee determined that the cause of the excessive leak rate was the failure of 2-CCR-440 to fully close. The licensee rebuilt the 2-CCR-440 actuator in accordance with Job Order 02021004 and satisfactorily retested the valve on January 29, 2002.

Analysis

The inspectors assessed this issue using the Significance Determination Process. The inspectors concluded that this issue represented an actual degradation in the redundancy of a containment penetration barrier and had a credible impact on safety and was more than a minor concern. The inspectors determined that the failure of a containment isolation valve was associated with the barrier integrity cornerstone. As described in Updated Final Safety Analysis Report Table 5.4-1, “Unit 2 Containment Penetration Isolation Barriers,” 2-CCR-440 was a barrier for containment penetration CPN-25. The second barrier for CPN-25 was composed of the closed CCW piping loop inside containment. Based on satisfactory Appendix J, Type C, leak rate test results for CPN-25 obtained on January 29, 2002, the inspectors determined that the CCW piping inside containment was intact and that failure of 2-CCR-440 to fully close did not represent an actual open pathway in the physical integrity of the reactor containment. Consequently, the inspectors concluded that this issue was of very low safety significance (Green) because the failure to properly evaluate and correct the cause for the IST stroke time test in April 2001 did not result in an open leak path from the Unit 2 containment.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” required, in part, that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, the licensee failed to adequately correct the failure of valve 2-CCR-440 during a routine stroke time test on April 11, 2001, a condition adverse to quality. Specifically, the licensee adjusted the 2-CCR-440 limit switch mechanism to obtain indication of valve closure without verifying that 2-CCR-440 was capable of fully closing. Subsequently, on January 20, 2002, 2-CCR-440 failed a 10 CFR 50, Appendix J, Type C, leak rate test due to the valve not being fully closed. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-316-01-20-04(DRP)). This violation is in the licensee’s corrective action program as CR 02043052 and CR 02037089.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated the licensee’s conduct of Unit 2 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
- Systems, structures, and components (SSCs) which could cause unexpected reactivity changes
- Flow paths, configurations, and alternate means for reactor coolant system (RCS) inventory addition and control of SSCs which could cause a loss of inventory
- RCS pressure, level, and temperature instrumentation
- Spent fuel pool cooling during and after core offload
- Switchyard activities and the configuration of electrical power systems in accordance with the TS and shutdown risk plan
- SSCs required for decay heat removal

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

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

Outage activities were still in progress at the end of this inspection period. Additional findings, if any, will be documented at the close of the inspection in a subsequent inspection report.

b. Findings

A finding of very low safety significance (Green) was self-revealed. With Unit 2 in Mode 5 (Cold Shutdown) operators vented and drained both Unit 2 safety injection (SI) system pumps making them unavailable for core cooling in the event of a loss of RCS inventory earlier in the outage than intended and failed to perform an assessment of the increase in risk for the maintenance activity as required by 10 CFR 50.65(a)(4). This resulted in the plant being in a higher risk configuration than that planned by the licensee. This finding was dispositioned as a Non-Cited Violation.

Description

On January 23, 2002, operators deviated from the licensee's planned outage schedule and prematurely vented and drained both Unit 2 SI system pumps without assessing the increase in shutdown risk during a period of reduced RCS inventory. An operator discovered the premature draining of the SI pumps during a review of the prerequisites for a scheduled surveillance test procedure that required the SI pumps to be filled and vented. The licensee's original outage risk evaluation recognized the risk significance of these pumps in Mode 5 with respect to inventory control and the outage schedule was appropriately established to maintain the pumps available but administratively out-of-service to comply with the TS requirements for low temperature over pressure protection. According to the licensee's outage schedule, the SI pumps were not to have been made unavailable by venting and draining until 6 days later when Unit 2 entered Mode 6 (Refueling) and the refueling cavity was filled to greater than 23 feet above the reactor vessel flange. The SI pumps would not be needed after that time due to the increased RCS inventory. The licensee's original outage risk evaluation reflected a "yellow" risk configuration (i.e., acceptable but reduced level of defense) by maintaining the two SI pumps available. By not maintaining the SI pumps available, the licensee inadvertently entered a higher "orange" risk configuration (i.e., minimum acceptable level of defense). The licensee's plant shutdown safety and risk management procedure, PMP 4100-SDR-001, "Plant Shutdown Safety and Risk Management," required the implementation of additional risk management actions to protect available equipment and to maintain an adequate level of defense which were not taken for the unplanned entry into the "orange" risk configuration.

Analysis

The inspectors concluded that this issue had a credible impact on safety because the SI pumps were made unavailable for core cooling in the event of a loss of RCS inventory.

At the time, Unit 2 was in Mode 5 with the RCS loops not filled and vented and only one of the two Unit 2 centrifugal charging pumps was available. The inspectors reviewed the guidance in Inspection Manual Chapter (IMC) 0609, Appendix G, "Shutdown Operations Significance Determination Process," including the checklist for "Pressurized Water Reactor Cold Shutdown and Refueling Operation - Reactor Coolant System Closed and No Inventory in Pressurizer." Based on this guidance, a minimum of one high pressure injection pump train (after breaker racked-in) and one other pump capable of keeping the core covered, in addition to the residual heat removal system pumps (low pressure injection), were required for the plant conditions that existed. Although having both SI pumps unavailable degraded the licensee's ability to add inventory to the RCS, the inspectors concluded that sufficient plant equipment existed to keep the core covered because the capability existed for operators to cross-tie the Unit 1 and Unit 2 charging systems to make an additional centrifugal charging pump available. The inspectors noted, however, that while this cross-tie capability existed, operators did not brief on the procedure as a contingency action prior to draining the SI pumps. The SI pumps were unavailable for approximately 6 hours, and although RCS draining was in progress during that time there was no challenge to RCS inventory control. This event did not meet the criteria in IMC 0609, Appendix G, Table 1 to be considered a loss of control and therefore was determined to be of very low safety significance (Green).

Enforcement

10 CFR 50.65(a)(4) requires, in part, that before performing maintenance activities (including but not limited to surveillances, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. Contrary to the above, on January 23, 2002, the licensee failed to assess the risk associated with maintenance activities affecting both Unit 2 SI system pumps which made them unavailable for core cooling in the event of a loss of RCS inventory. This resulted in the inadvertent entry into a higher shutdown risk configuration, for which the licensee had not implemented additional risk management actions to protect available equipment and maintain an adequate level of defense. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-316-01-20-05(DRP)). The licensee entered this violation into its corrective action program as CR 02023077.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

Barrier Integrity Cornerstone

- 12-MHP-4030.010.003, "Ice Condenser Lower Inlet Door Surveillance"

The inspectors reviewed the test methodology and test results in order to verify that equipment performance was consistent with safety analysis and design basis assumptions. The inspectors also reviewed condition reports concerning surveillance testing activities to verify that identified problems were appropriately characterized.

b. Findings

While observing Unit 2 ice condenser lower inlet door surveillance testing, the inspectors identified two findings of very low safety significance (Green) associated with the failure to adequately implement surveillance test procedure requirements and the failure to establish an ice condenser lower inlet door testing methodology capable of demonstrating compliance with TS surveillance requirements. These findings were dispositioned as Non-Cited Violations.

b.1 TS 4.6.5.3.1.b Ice Condenser Lower Inlet Door Testing Requirements

Technical Specifications 4.6.5.3.1.b.3 and 4.6.5.3.1.b.4 required the licensee to measure the opening and closing torque for each of the ice condenser lower inlet doors, with the door opened 40 degrees from the closed position, at least once every 18 months. The purpose of the torque testing was to verify that the opening torque was less 195 inch-pounds and that the closing torque was greater than 78 inch-pounds. Additionally, TS 4.6.5.3.1.b.5 required the licensee to calculate the door frictional torque based on the results of the door opening and door closing torque measurements. The licensee implemented the requirements of TS 4.6.5.3.1.b by performing surveillance procedure 12-MHP 4030.010.003, “Ice Condenser Lower Inlet Door Surveillance.”

b.2 Failure to Adequately Implement the Requirements of the Lower Ice Condenser Door Surveillance Test Procedure

On January 24, 2002, the inspectors observed portions of the lower inlet door testing, including the 40 degrees torque tests and door limit switch checks. The inspectors identified several instances where the licensee failed to adequately implement procedural requirements, including the following:

- Inconsistent Labeling and Identification of Lower Inlet Doors and Failure to Accurately Record Surveillance Test Data: During lower inlet door limit switch checks, the inspectors noted that the identification of lower ice condenser doors was inconsistent with the door labeling at the containment auxiliary sub-panel (CAS). Each of the two lower inlet doors in each ice condenser bay was designated as either the left or the right door. The door labeling convention contained in 12-MHP 4030.010.003 for identification of the left and right lower inlet doors in each ice condenser bay was opposite to that used for labeling the open door indicator lights at the CAS (e.g., opening the Bay 12 right side door resulted in illumination of light 4-1L vice 4-1R). The inspectors noted that personnel performing the lower inlet door limit switch checks in accordance with procedure steps 4.1.8.d and 4.1.9.d compensated for the labeling inconsistency by initialing the test data sheet for the door light that matched the procedural door identification convention rather than annotating the data sheet for the light that actually illuminated (e.g., the test performer would initial the test data sheet for door 4-4L when the Bay 9 left hand door was opened even though the light for door 4-4R actually illuminated). The inspectors discussed the failure to correctly record surveillance test data with the maintenance supervisor. The licensee initiated CR 02024066 and CR 02025084 to document this issue.

- Failure to Perform Procedurally Required Step: Personnel performing the surveillance test procedure failed to perform step 4.2.6 of 12-MHP 4030.010.003, which required placement of a protective tip on the spring scale to protect the ice condenser door. When the inspectors questioned the failure to perform the step, the maintenance personnel performing the testing stated that they intentionally did not perform the step to limit the amount of foreign material brought into the ice condenser. The inspectors discussed the failure to follow the procedural requirement with the maintenance supervisor. The licensee then complied with the procedural requirement and initiated CR 02025075 to document the issue.

- Failure to Establish Required Initial Conditions for Lower Inlet Door Testing: In order to establish the door angle and moment-arm position for the 40 degrees opening and closing force tests, the licensee clamped a test rig (the TE-132 spring scale bracket) to the ice condenser door frame. Test procedure step 4.2.5 required that the use of the TE-132 test fixture result in an applied moment arm of 27 ± 1/8 inches. Additionally, step 4.2.3 required that the door be opened to the 40 degrees position, which the procedure stated could be obtained by proper installation of the TE-132 test fixture. The inspectors noted that the licensee did not verify that the door opening angle or moment-arm length, two critical parameters for the surveillance test, prior to testing each door. Maintenance personnel performing the test stated that, if the test rig were installed correctly, the door opening angle and moment-arm would be correct. Following additional questioning by the inspectors, the licensee measured the moment-arm applied by the test rig and determined that the test rig did not establish consistent conditions for door testing. Specifically, following measurement on ten lower ice condenser inlet doors, the licensee determined that the applied moment-arm varied from 25 inches to 27 inches. The inspectors noted that variations in the applied moment-arm could also result in deviations from the 40 degrees opening initial test conditions. Following discovery of this issue, the licensee stopped ice condenser lower inlet door testing and initiated CR 02025024 and CR 02025084.

Based on the preceding observations, the licensee stopped lower inlet door testing and placed an administrative hold on procedure 12-MHP 4030.010.003. The licensee revised the testing methodology to eliminate use of the TE-132 test fixture, retrained personnel performing the testing, and clarified the procedural guidance for spring scale thermal soaking.

Analysis

The inspectors assessed the failure to adequately implement the requirements of 12-MHP 4030.010.003 using the Significance Determination Process. The inspectors determined that the failure to correct these procedural implementation inadequacies could become a more significant safety concern if left uncorrected and was therefore more than a minor concern. Specifically, the failure to adequately perform surveillance testing could reasonably result in the failure to identify degraded or inoperable safety related equipment. Additionally, the inspectors determined that, had the inspectors not questioned the implementation of the ice condenser surveillance testing, the licensee would not have identified these procedure implementation inadequacies. Because the ice condenser was primarily associated with containment heat removal following certain design basis accidents, the inspectors concluded that this issue was associated with the barrier integrity cornerstone. Because the Unit 2 ice condenser was not required to be capable of performing a safety related function immediately following the inadequate surveillance testing on January 24, 2002, the inspectors concluded that this issue did not result in an actual loss or degradation of the heat removal function performed by the ice condenser. Consequently, the inspectors concluded this issue was of very low safety significance (Green).

Enforcement

Technical Specification 6.8.1.a requires that written procedures shall be established, implemented and maintained covering the activities recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A of Regulatory Guide 1.33, Revision 2, February 1978 recommends, in part, procedures for surveillance tests. Contrary to the above, on January 24, 2002, the licensee failed to adequately implement the requirements of 12-MHP 4030.010.003, a surveillance test procedure written to cover an activity referenced in Appendix A of Regulatory Guide 1.33. Specifically, the licensee failed to adequately perform the following: (1) install a protective end tip on the spring scale to protect the lower ice condenser doors from damage as required by step 4.2.6, (2) ensure that installation of the TE-132 test fixture met the moment arm and degree of opening requirements in accordance with steps 4.2.3 and 4.2.5, and (3) accurately record surveillance test data for lower inlet door limit switch checks required by steps 4.1.8.d and 4.1.9.d. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-316-01-20-06(DRP)). This violation is in the licensee's corrective action program as CR 02025084.

b.3 Test Procedure Methodology for Lower Inlet Door Testing Failed to Meet Technical Specification Requirements

In addition to the licensee's failure to adequately implement existing procedural guidance for lower inlet door testing (described in Section 1R22.1.b.2 above), the inspectors identified that the licensee's testing methodology was not capable of demonstrating compliance with the requirements of TS 4.6.5.3.1.b. Specifically, the door closing torque (described in TS 4.6.5.3.1.b.4) should always be less than the door opening torque (described in TS 4.6.5.3.1.b.3) by an amount proportional to the door friction. However, during a review of the completed test data from lower inlet door testing prior to the Unit 1 and Unit 2 restarts in the year 2000, the inspectors noted that the licensee recorded door opening torques that were less than the associated door closing torque. For example, the licensee recorded the following torque and frictional values for the Unit 1, Bay 10, left side door in November 2000:

Opening Torque TS 4.6.5.3.1.b.3	Closing Torque TS 4.6.5.3.1.b.4
68.6 inch-pounds	128.3 inch-pounds

The inspectors noted that, although these measured values met the associated TS acceptance criteria, the opening torque was actually less than the minimum allowable closing torque of 78 inch-pounds. Additionally, the opening torque was less than the closing torque, implying that the door frictional torque component was less than 0 inch-pounds. The inspectors concluded that the licensee's testing results were not physically possible, and that the testing methodology was not capable of measuring the door torques as described in TS 4.6.5.3.1.b.

Based on a review of the test procedure guidance and direct observation of actual door testing, the inspectors noted several concerns with the testing methodology specified in procedure 12-MHP 4030.010.003. For example, step 4.2.9 of the testing procedure required that the door be opened approximately 1 inch beyond the 40 degrees open position and then released in order to measure the closing force. After release, the door impacted the spring scale installed on the TE-132 test fixture and bounced until the door settled at a force level capable of holding the door in a static position. The inspectors determined that this test method was not adequate in that the measured closing force was not the minimum force required to hold the door in position by TS 4.6.5.3.1.b.4. Additionally, the licensee determined that the opening force measurements were highly dependent on the technique used by the individual performing the test and therefore did not reliably measure door opening force. Based on these findings, the licensee declared the Unit 1 lower ice condenser doors inoperable on January 31, 2002 and entered the associated 14 day limiting condition for operation action statement (the Unit 2 ice condenser was already inoperable to support maintenance activities). On February 3 and 4, 2002, the licensee re-performed lower inlet door testing in accordance with special testing procedure 12-MHP-SP-LID, "Data Acquisition For Ice Condenser Lower Inlet Doors," Revision 1. The methodology used in 12-MHP-SP-LID provided reasonable assurance that the door torque values were measured consistently with TS 4.6.5.3.1.b requirements. Based on a review of the Unit 2 lower inlet door test data obtained by the special procedure, the licensee concluded that all 48 Unit 2 lower inlet doors met TS requirements. On February 8, 2002, the licensee submitted an emergency license amendment request to the NRC in order to obtain a one-time limited duration exemption from the Unit 1 TS 4.6.5.3.1.b lower inlet door testing requirements. On February 14, 2002, the NRC issued Unit 1 license amendment 265 which allowed the licensee to defer TS 4.6.5.3.1.b ice condenser door testing until the next refueling outage. The licensee exited TS action statement 3.6.5.3 on February 14, 2002.

Analysis

The inspectors evaluated this failure to meet TS 4.6.5.3.1.b requirements using the Significance Determination Process. The inspectors determined that the failure to adequately implement TS 4.5.6.3.1.b requirements for testing of lower inlet doors had a credible impact on safety and was more than a minor concern. As stated in the TS 3.6.5 bases, operability of the ice condenser doors ensures that reactor coolant fluid released during a loss of coolant accident (LOCA) will be diverted through the ice condenser bays for heat removal. The ice condenser also augmented the containment recirculation sump water inventory in the event of certain small break LOCAs and limited ice maldistributions within the ice condenser. Because the proper functioning of the ice condenser lower inlet doors was primarily associated with the heat removal function of the ice condenser, the inspectors determined that this issue was associated with the barrier integrity cornerstone. Based on a review of the testing methodology and results from the Unit 2 lower inlet door testing performed on February 3 and 4, 2002, the inspectors concluded that there was no reduction in the atmospheric pressure control function of the reactor containment nor a loss of capability to provide additional recirculation sump inventory during certain small break LOCAs. Consequently, the inspectors concluded that this issue was of very low safety significance (Green).

Enforcement

Technical Specifications 4.6.5.3.1.b.3, 4.6.5.3.1.b.4, and 4.6.5.3.1.b.5 require testing of the ice condenser lower inlet doors at least once per 18 months in order to measure the torque required to open the door, the torque required to keep the door from closing, and the door frictional torque. Technical Specification 4.6.5.3.1.b.3 stated that the door opening torque was equal to the nominal door torque plus a frictional torque component. Technical Specification 4.6.5.3.1.b.4 stated that the door closing torque was equal to the nominal door torque minus a frictional torque component. Contrary to the above, previous TS 4.6.5.3.1.b surveillance testing performed in Unit 2 on April 21, April 22, and May 4, 2000 and in Unit 1 on November 21, 2000, failed to adequately measure the door opening torque and the door closing torque in accordance with TS requirements. Specifically, the methodology used by the licensee to perform TS 4.6.5.3.1.b.3 and 4.6.5.3.1.b.4 resulted in door closing torques that were greater in magnitude than the door opening torques, contrary to the TS description of these torque values. The inspectors identified that the measured opening torque values for all 48 of the Unit 2 lower inlet doors and 36 Unit 1 lower inlet doors were less than the associated door closing torque. Because calculation of the door frictional torque required accurate measurement of the door opening and closing torques, the licensee was unable to demonstrate compliance with the requirements of TS 4.6.5.3.1.b.5. Because of the very low safety significance, the licensee's failure to comply with TS 4.6.5.3.1.b.3, 4.6.5.3.1.b.4, and 4.6.5.3.1.b.5 is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-316-01-20-07(DRP)). This violation is in the licensee's corrective action program as CR 02032016.

At the conclusion of this inspection, the licensee had not been able to test the Unit 1 ice condenser lower inlet doors using a methodology consistent with TS 4.6.5.3.1.b requirements. As required by Unit 1 licensee amendment 265, the licensee will complete ice condenser lower inlet door testing prior to mode ascension following the upcoming cycle 18 refueling outage or if the unit enters Mode 5 for sufficient duration prior to the cycle 18 refueling outage. Consequently, due to the unavailability of door performance data, the inspectors identified the licensee's failure to adequately perform Unit 1 lower inlet door testing as required by TS 4.6.5.3.1.b as an Unresolved Item (URI 50-315-01-20-08) pending a final safety significance evaluation.

4. OTHER ACTIVITIES (OA)

4OA3 Event Follow-up (71153)

(Closed) Licensee Event Report (LER) 50-315-01-05-00: "RCCA [Rod Control Cluster Assembly] Tool Over Spent Fuel Pool Racks TS Violation." The inspectors previously reviewed this event and issued Non-Cited Violation 50-315-01-19-06 for the licensee's failure to verify that the impact energy limit specified in TS 3.9.7 for the movement of an RCCA change out tool over spent fuel pool storage racks containing fuel was met as required by TS 4.3.9.7. The inspectors determined that the information provided in LER 50-315-01-05-00 did not raise any new issues or change the conclusions of the initial review, which was documented in NRC Inspection Report 50-315/01-19(DRP); 50-316/01-19(DRP). This LER is closed.

4OA5 Other

.1 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (TI 2515/145)

a. Inspection Scope

D.C. Cook Unit 2 Reactor Facility is in the sub-population of plants (Bin 1) that have experienced head penetration cracking. The licensee responded to NRC Bulletin 2001-01 by performing a direct visual examination of the reactor vessel head and under head examinations of 78 control rod drive mechanism (CRDM) penetrations using Eddy Current Examination (ET) and Ultrasonic Examination (UT) methods. The inspectors interviewed inspection personnel, reviewed procedures and inspection reports including photographic documentation to assess the licensee's efforts in conducting an "effective" visual examination of the reactor vessel head.

b. Evaluation of Inspection Requirements

a. Was the examination:

1. Performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

Top of Vessel Head Visual Examinations

Yes. The remote visual examination of the head was performed by knowledgeable licensee personnel certified to Level II or III as VT-2 examiners in accordance with programs meeting the American Society for Nondestructive Testing (ASNT) Recommended Practice SNT-TC-1A.

Under Vessel Head Ultrasonic and Eddy Current Examinations

Yes. The ultrasonic and eddy current examinations were performed by contract personnel certified to Level II and III in accordance with programs meeting ASNT

Recommended Practice SNT-TC-1A and CP 189. A portion of the ET personnel had additional qualifications from Electric Power Research Institute (EPRI) as qualified data analysts. A portion of the UT personnel also had EPRI Performance Demonstration Initiative qualifications which met ASME Code Section XI, Appendix VIII requirements.

2. Performed in accordance with approved and adequate procedures?

Top of Vessel Head Visual Examinations

Yes. The visual examinations were conducted in accordance with MRS-SSP-1269, “Head Penetration Visual Inspection.” This inspection included all vessel head penetrations and was intended to meet visual quality standards established for remote VT-2 examinations as defined in Section XI of the ASME Code.

Under Vessel Head Ultrasonic and Eddy Current Examinations

Yes. The ultrasonic and eddy current inspections were performed in accordance with procedures WDI-ET-003, “Intraspect Eddy Current Imaging Procedure for Inspection Procedure for Inspection,” WDI-UT-008, “Intraspect Time of Flight Ultrasonic Procedure for Inspection of RX [Reactor] Vessel Head Penetrations,” and WDI-UT-009, “Intraspect Ultrasonic Procedure for the Detection of Circumferential Indications.” These procedures provided for documentation of equipment setup, calibration and sizing of indications.

3. Adequately able to identify, disposition, and resolve deficiencies?

Top of Vessel Head Visual Examinations

Yes. The visual inspection procedure was qualified by demonstrating the ability to resolve characters of dimensions identified in ASME Code Section XI. The remote visual inspection was conducted with a set of cameras mounted to a robot crawler. Inspectors reviewed the tape of the qualification of the visual system and considered the lighting and picture quality to be excellent. Results of each penetration inspection were recorded in Appendix C of procedure MRS-SSP-1269. The crawler mounted camera visual inspection was supplemented with a camera mounted to a fiber optic scope to obtain 100 percent coverage of all penetrations.

A brick of insulation material was found resting between the reactor vessel head and the insulation support structure near penetration 2. The licensee documented this condition in CR 02027032.

Under Vessel Head UT and ET

Yes. The UT and ET system calibrations were performed at 12 hour intervals on calibration standards which contained inside and outside diameter notches. The ET and UT examinations were conducted from the inside of the penetration and data was recorded in a downward direction from 2 inches above the J-weld to the end of the penetration. The licensee used a rotating head probe for 9 of the 18 unsleeved penetrations. This probe contained an X-wound ET coil, 0-degree UT transducer, 20 degree UT transducer and UT transducers setup for time-of-flight-diffraction. For the remaining penetrations a sword probe was used. This probe contained UT transducers setup for time-of-flight-diffraction oriented such that it would provide maximum sensitivity for circumferentially oriented cracks near the outside diameter of the tube.

For 9 penetrations near the center of the head, complete UT coverage for the penetrations could not be achieved due to interference with the thermal sleeve centering tabs. To ensure that 100 percent of the wetted surface was examined for these penetrations the licensee performed scans of the penetration inner diameter and J-weld using an X-wound ET coil. Additionally, for penetration 75, the licensee had not achieved complete UT coverage (70 degrees of data not available) due to loss of couplant. The licensee was evaluating options such as performing a dye penetrant exam on this penetration to attain complete coverage.

4. Capable of identifying the primary water stress corrosion cracking (PWSCC) phenomenon described in the bulletin?

<u>Top of Vessel Head Visual Examinations</u>

Yes, for 17 of the penetrations located on the periphery of the head. For the remaining head penetrations, the licensee could not positively identify that a leak path would exist through the annulus gap around each penetration. Each of the 78 head penetrations and head vent were accessible for complete visual examination.

<u>Under Vessel Head Ultrasonic and Eddy Current Examinations</u>

Yes, for penetration tube material. The UT search units were designed to detect flaws in the penetration tube and this equipment had reportedly been used to detect cracking in CRDM penetration tubes removed from the Oconee Nuclear Power Station. The licensee had performed a demonstration of this technique in accordance with EPRI guidelines which were not yet available for the inspectors' review. The calibration standard used was of similar material and dimensions as the head penetration tubes. This standard contained both axial and circumferential oriented notches located at the inside and outside surface. The inspectors concluded that the UT method used would likely detect PWSCC in the penetration tube.

No, for the J-weld region. The inspectors concluded that the UT technique used would not be effective for detection of PWSCC, that was entirely within the J-weld attaching the penetration tubing to the vessel head. However, the licensee had completed dye penetrant testing on one J-weld and was planning on performing ET of nine other J-welds, which would be capable of detecting PWSCC in the J-weld for these penetrations.

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

Top of Vessel Head Visual Examinations

The reactor head has reflective metal insulation panels which are installed on a support structure over the top of the reactor head. The remote camera visual inspection was conducted under the insulation support structure and the as-found head condition was generally clean (free of debris, insulation, dirt). The uphill side of the annulus gap on several penetrations contained loose debris, which did not hinder the licensee's evaluation of the penetration. Some quadrants of penetrations near insulation support structures were obstructed from the crawler mounted camera. These locations were reinspected with a fiber scope mounted camera such that 100 percent visual examination of the penetration head interface was achieved for all 78 head penetrations.

Under Vessel Head Ultrasonic and Eddy Current Examinations

The surface of the inner bore of the CRDM penetrations was sufficiently smooth, such that the UT and ET examinations were not affected. An exception to this was the surface condition of the weld for the embedded flaw repaired in CRDM penetration 75. The surface condition precluded a meaningful UT examination of the embedded flaw below the repair weld.

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

Top of Vessel Head Visual Examinations

Yes. Small boron deposits as described in the bulletin could be identified due to the cleanliness of the head and full access for visual inspection. However, the licensee had not conclusively demonstrated that a leakage path would exist for any penetrations other than 17 CRDM penetrations at the periphery of the head. Therefore, the licensee was relying on the effectiveness of the under vessel UT and ET examinations to detect cracking. No indications of boron deposits (indicative of penetration leakage) were found on the 78 penetrations.

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

None. Only one indication was identified in penetration 74, by the ET probe. This indication included a cluster of axial oriented indications approximately 2 inches in length and within a 1 inch wide band, located just below the J-weld at the inside diameter of the tube. The licensee considered these indications to be surface cracking and did not intend to remove this group of shallow flaws. The licensee used the time-of-flight-diffraction UT method to confirm that these flaws were shallow. However, this method had not been demonstrated/qualified for sizing indications. The inspectors confirmed that the licensee could detect a break in the lateral wave of the UT search unit caused by an axial oriented inside diameter 1 millimeter deep notch in the calibration standard.

Further, the penetration 74 flaw response was similar to the 1 millimeter deep calibration standard notch response.

Eleven indications were detected with the UT time of flight transducers scans. These indications were subsequently evaluated with a 45 degree shear wave and 0 degree transducers. The licensee considered these indications to be associated with weld geometry or original construction defects and not inservice flaws (e.g. PWSCC). To confirm this, the licensee performed a surface dye penetrant examination of the J-weld on penetration 32 that was considered representative of the larger UT weld anomalies. This examination identified three very small (1/32, 1/16 and 3/32) rounded indications in the J-weld, which the licensee considered weld porosity and acceptable under the original Code of construction. The inspectors reviewed pictures of the dye penetrant results from PWSCC indications found in the J-welds at Oconee. The Oconee dye penetrant exams, clearly showed a larger dye penetrant bleed-out and linear crack like features which were not present in the D. C. Cook penetration 32 examination that was observed by the inspectors.

e. What, if any, significant items that could impede effective examinations and/or As-Low-As-Reasonably-Achievable (ALARA) issues were encountered?

No significant impediments to the examination were identified. ALARA projected dose for all head examinations was 11.1 Rem. The actual dose received was substantially under the projected dose at the conclusion of this inspection.

4OA6 Meetings

.1 Interim Exits

The results of the Unit 2 Biennial Inservice Inspection and TI-145 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (NRC Bulletin 2001-01) Inspection were presented to Mr. M. Rencheck and other members of licensee management at the conclusion of the inspection on February 6, 2002. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was examined during the inspection but is not specifically discussed in this report.

.2 Resident Inspector's Exit

The inspectors presented the inspection results to Mr. C. Bakken and other members of licensee management at the conclusion of the inspection on February 13, 2002. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was examined during this inspection but is not specifically discussed in this report.

KEY POINTS OF CONTACT

Licensee

G. Arent, Manger, Regulatory Affairs
C. Bakken, Senior Vice President, Nuclear Generation
G. Bourlodan, Plant Programs Manager
P. Cowan, Licensing Supervisor, Regulatory Affairs
R. Gaston, Regulatory Affairs Compliance Supervisor
J. Gebbie, System Engineering Manager
S. Greenlee, Director, Nuclear Technical Services
R. Hall, Inservice Inspection Program
N. Jackiw, Regulatory Affairs
E. Lamoureut, Westinghouse Project Manager
C. Lane, Inservice Inspection Supervisor
E. Larson, Manager, Operations
R. Meister, Regulatory Affairs
D. Moul, Assistant Manager, Operations
D. Noble, Radiation Protection Manager
T. Noonan, Director, Performance Assurance
J. Pollock, Plant Manager
M. Rencheck, Vice President, Strategic Business Improvement
R. Smith, Assistant Director, Plant Engineering
C. Vanderniet, Project Manager
L. Weber, Performance Assurance
D. Wood, RadChem Environmental Manager
T. Woods, Regulatory Affairs

NRC
A. Vegel, Chief, Reactor Projects Branch 6

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-316-01-20-01	NCV	Violation of 10 CFR Part 50 Appendix B, Criterion IX for failure to correctly calibrate an ultrasonic transducer (Section 1R08)
50-316-01-20-02	NCV	Violation of Technical Specification 4.0.5.a for application of incorrect acceptance criteria applied to flaws in the pressurizer welds (Section 1R08)
50-315/316-01-20-03	NCV	Violation of 10 CFR Part 50.55a(g)(5)iii for failure to obtain NRC concurrence associated with incomplete nondestructive weld examinations (Section 1R08)
50-316-01-20-04	NCV	Failure to properly evaluate and correct the cause for an inservice stroke time test failure in April 2001 (Section 1R19)
50-316-01-20-05	NCV	Failure to assess the risk associated with maintenance activities affecting both Unit 2 safety injection system pumps (Section 1R20)
50-316-01-20-06	NCV	Failure to adequately implement the requirements of surveillance procedure 12-MHP 4030.010.003 (Section 1R22)
50-316-01-20-07	NCV	Failure to adequately measure the ice condenser lower inlet door opening torque and closing torque in accordance with Technical Specification requirements (Section 1R22)
50-315-01-20-08	URI	Failure to adequately measure the ice condenser lower inlet door opening torque and closing torque in accordance with Technical Specification requirements (Section 1R22)

Closed

50-316-01-20-01	NCV	Violation of 10 CFR Part 50 Appendix B, Criterion IX for failure to correctly calibrate an ultrasonic transducer (Section 1R08)
50-316-01-20-02	NCV	Violation of Technical Specification 4.0.5.a for application of incorrect acceptance criteria applied to flaws in the pressurizer welds (Section 1R08)
50-315/316-01-20-03	NCV	Violation of 10 CFR Part 50.55a(g)(5)iii for failure to obtain NRC concurrence associated with incomplete nondestructive weld examinations (Section 1R08)
50-316-01-20-04	NCV	Failure to properly evaluate and correct the cause for an inservice stroke time test failure in April 2000 (Section 1R19)

50-316-01-20-05	NCV	Failure to assess the risk associated with maintenance activities affecting both Unit 2 safety injection system pumps (Section 1R20)
50-316-01-20-06	NCV	Failure to adequately implement the requirements of surveillance procedure 12-MHP 4030.010.003 (Section 1R22)
50-316-01-20-07	NCV	Failure to adequately measure the ice condenser lower inlet door opening torque and closing torque in accordance with Technical Specification requirements (Section 1R22)
50-315-01-05-00	LER	RCCA [rod control cluster assembly] tool over spent fuel pool racks Technical Specification violation (Section 4OA3)

Discussed

50-315-01-19-06	NCV	Failure to maintain load carried over spent fuel within impact energy requirements of Technical Specification 3.9.7

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
AEP	American Electric Power
ALARA	As Low As Is Reasonably Achievable
ANST	American Society of Non-Destructive Testing
ATR	Administrative Technical Requirement
ASME	American Society of Mechanical Engineers
CAS	Containment Auxiliary Sub-panel
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CRDM	Control Rod Drive Mechanism
DIT	Design Information Transmittal
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EPRI	Electric Power Research Institute
ESW	Essential Service Water
ET	Eddy Current Examination
IMC	Inspection Manual Chapter
ISI	Inservice Inspection
IST	Inservice Testing
LER	Licensee Event Report
LOCA	Loss-of-Coolant Accident
MHP	Maintenance Head Procedure
MT	Magnetic Particle Examination
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
OHP	Operations Head Procedure
PARS	Publically Available Records
PMI	Plant Manager's Instruction
PMP	Plant Manager's Procedure
PT	Die Penetrant Examination
PWSCC	Primary Water Stress Corrosion Cracking
RCCA	Rod Control Cluster Assembly
RCS	Reactor Coolant System
SDP	Significance Determination Process
SG	Steam Generator
SI	Safety Injection
SSC	Structures, Systems, and Components
STP	Surveillance Test Procedure
TBD	To Be Determined
TS	Technical Specification
URI	Unresolved Item
UT	Ultrasonic Examination

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

Plant Manager's Procedure (PMP) 5020.RTM.001	Restraint of Transient Material	Revision 1
12-MHP-5021.SCF.001	Scaffolding Guidelines	Revision 0b
Flow Diagram OP 2-5143	Emergency Core Cooling Residual Heat Removal Unit No. 2	
Condition Report (CR) 02044022 (1)	A Secondary Swing Gate Style Posting for a High Radiation Area Was Found in an Unacceptable Rotated Position That Allowed Passage Past the Gate	February 3, 2002
CR 02044062 (1)	Failure to Comply with Cook Nuclear Plant Corrective Action Program and Management Expectations	February 13, 2002

1R05 Fire Protection

Updated Final Safety Analysis Report, Section 9.8.1	Fire Protection System	
	D. C. Cook Nuclear Plant Fire Hazards Analysis, Units 1 and 2	Revision 8
	D. C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook	February 1995
PMP 2270.CCM.001	Control of Combustible Materials	Revision 1
PMP 2270.FIRE.002	Responsibilities for Cook Plant Fire Protection Program Document Updates	Revision 0
PMP 2270.WBG.001	Welding, Burning and Grinding Activities	Revision 0
Plant Manager's Instruction (PMI) 2270	Fire Protection	Revision 26
Technical Specification (TS) 3.1.2.3	Charging Pump - Shutdown	

1R08 Inservice Inspection Activities

Audit

PA-00-011	Work Management Process and Special Process Control Programs	November 10, 2000

Condition Reports

CR P-99-29740	Rejectable Indications in 2-10" Welds and 1-6" Weld FW-1R1	December 24, 1999
CR P-00-00026	Rejectable Indications Found During Radiography	January 02, 2000
CR P-99-29419	Rejectable Indications Found During Radiography	December 19, 1999
CR P-00-00883	Rejectable Indications Found During Radiography	January 18, 2000
CR P-00-00674	Rejectable Indications Found During Radiography	January 13, 2000
CR 01012034	No Leak Test on Emergency Diesel Generator Air Relief Valves	January 12, 2001
CR 00252027	Required VT-2 Inspection Was Not Performed After Bonnet Was Replaced	September 8, 2000
CR P-95-01557	Notification of Potential Defect Per 10 CFR Part 21 Rupture of Plugged Steam Generator (SG) Tubes	September 27, 1995
CR P-99-07966	No Record of American Electric Power Response to Generic Letter, Dated February 10, 1978	April 7, 1999
CR P-00-00445	Insufficient Weld Was Discovered by Quality Control on New SG 4	April 7, 2000
CR P-00-01572	Cracks Found During Inspection of Support Ring on SG 2	January 27, 2000
CR P-00-00320	During Pipe Modification 2DCP-648 It Was Noted That the Existing Pipe Did Not Have Bearing Contact with Support	April 7, 2000
CR 02023050 (1)	Unit 2 Third Period Second Interval Exam Code Relief Requests Had Not Been Submitted as Required by 10 CFR 50.55a	January 23, 2002

CR 02033031 [(1)]	The Inservice Inspection (ISI) Program Was Not Given an Audit Frequency	February 2, 2002
CR 02035044 [(1)]	NRC Inspector Raised Following Questions During 2002 ISI	February 4, 2002
CR 02035044 [(1)]	NRC Inspector Raised Following Questions on Inaccurate Application of Weld Acceptance Criteria	February 4, 2002

Code Replacement/Repair Activities

Job Order C003595402	2-FW-132-4 Replace Valve by Welding
Job Order C004223902	2-CTS-105E Remove Flanges & Weld into System
Job Order C004083002	2-OME-3-2 Repair Manway Bolt Holes as Required

Drawings

2SI-7	Safety Injection	Revision 4
2-CTS-10	Containment Spray	Revision 17
2-FW-55	Feedwater Containment Vicinity	Revision 9

Nondestructive Examination Reports

#316540	Containment Spray, PT [Die Penetrant] & UT [Ultrasonic] Exam Report 2-CTS-10-12F	April 3, 1996
#011400	Steam Generator 24, UT Exam Report STM-24-I-IRS	April 5, 1996
#011600	Steam Generator 24, PT & UT Exam Report STM-24-02	April 9, 1996
#011700	Steam Generator 24, PT & UT Exam Report STM-24-03	April 9, 1996
#009500	Steam Generator 22, UT Exam Report STM-22-01	April 4, 1996
#006100	Pressurizer, UT Exam Report 2-PRZ-11	November 5, 1997 and November 14, 1997
#006930	Pressurizer, UT Exam Report 6"-2-RC-22	October 31,1997
#006940	Pressurizer, UT Exam Report 6"-2-RC-25	October 31, 1997

#014000	Reactor Coolant, UT Exam Report 2-RC-17-01N	November 7, 1997
#027700	Reactor Coolant, UT Exam Report 2-RC-23-03	October 22, 1997
#300450	Regenerative Heat Exchanger, UT Exam Report RHE-2-03	November 4, 1997
#300540	Regenerative Heat Exchanger, UT Exam Report RHE-2-25	November 4, 1997
#312590	Emergency Core Cooling, PT and UT Exam Report 2-SI-77-15F	October 31, 1997
#318380	Feedwater System, MT [Magnetic Particle] and UT Exam Report 2-FW-71-04S	November 8, 1997

<u>Procedures</u>

54-ISI-130-33	Ultrasonic Examination of Ferritic Vessel Welds Greater Than Two Inches Thick	April 25, 1997
WDI-ET-003	Intraspect Eddy Current Imaging Procedure for Inspection Procedure for Inspection	Revision 0
WDI-UT-008	Intraspect Time of Flight Ultrasonic Procedure for Inspection of Reactor Vessel Head Penetrations	Revision 0
WDI-UT-009	Intraspect Ultrasonic Procedure for the Detection of Circumferential Indications	Revision 0
12-QHP-5050-NDE-010	Radiographic Examination of Welds	Revision 2

<u>Miscellaneous Documents</u>

MRS-SSP-1269, Appendix C&D	Cook Unit 2- RV Head Penetration Inspection Records	January 27, 2002
WPS 1-8.1TS	Welding Procedure Specification Manual Gas Tungsten Arc and Shielded Metal Ark Welding	Revision 3
WPS 1-2TS	Welding Procedure Specification Manual Gas Tungsten Arc and Shielded Metal Ark Welding	Revision 2
PQR 232	American Society of Mechanical Engineers (ASME) Procedure Qualification Record	Revision 2
PQR 234	ASME Procedure Qualification Record	Revision 2

PQR 235	ASME Procedure Qualification Record	Revision 2
PQR 255	ASME Procedure Qualification Record	Revision 2
ETSS 96004.1	Eddy Current Examination Technique Specification Sheet	Revision 7
ETSS 96001.1	Eddy Current Examination Technique Specification Sheet	Revision 6
ETSS 96004.1	Eddy Current Examination Technique Specification Sheet	Revision 8
ETSS 20409.1	Eddy Current Examination Technique Specification Sheet	Revision 3
ETSS 20511.1	Eddy Current Examination Technique Specification Sheet	Revision 1
ETSS 20510.1	Eddy Current Examination Technique Specification Sheet	Revision 1
ETSS 20511.2	Eddy Current Examination Technique Specification Sheet	Revision 10
SGP-DA-U2-C13	Steam Generator Degradation Assessment - Unit 2 Cycle 13	Revision 2
R-4025-00-1	D. C. Cook Cycle 12 Operational Assessment	Revision 1
	Site Specific Eddy Current Data Analysis Guidelines D. C. Cook Nuclear Plant Unit 2	Revision 1

<u>Welding Radiographic Records</u>

Weld FW-1	Line 2-CTS-10
Weld FW-2	Line 2-SI-7
Weld OW-1	Line 2-FW-55

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

PMP 2110-CPS-001	Clearance Permit System	Revision 5
Job Order 01054065	Inspect Fuse Holders on Bus 2A	

CR 00238033	Condition Report 99-20499 Had a Maintenance Rule Functional Failure Identified. Further Research Indicates That the Condition Should Also Be a Maintenance Preventible Functional Failure	August 25, 2000
CR 00266029	Allowable Functional Failures for the 250 Volts Direct Current System Have Been Exceeded	September 22, 2000
Action Plan 00-525	Fuse Block Failures	
System Health Report	250 Volts Direct Current Distribution	July 1, 2001 through September 30, 2001

1R13 Maintenance and Emergent Work Control

CR 02010001	Unit 2 West Motor Driven Auxiliary Feedwater Pump Outboard Pump Bearing Recorded High Vibrations	January 9, 2002
2OHP-4030-STP-017W	West Motor Driven Auxiliary Feedwater System Test	Revision 11
CR 02010026	Oil Sampling Method and Instructions for All of the Auxiliary Feedwater Pumps Are Suspected to Be Inadequate for Returning the Component Oil Level to the As-found Condition	January 10, 2002
CR 02020014	Unit 2 West Motor Driven Auxiliary Feedwater Pump Coupling Has Thrown a Substantial Amount of Coupling Grease on the Inside of the Coupling Guard and on the Skid Below the Coupling	January 20, 2002
VTD-INDR-0045	Installation, Operation, and Maintenance of Ingersoll-Dresser Auxiliary Feedwater Pumps	Revision 2
	Clearance Permit Log	January 14, 2002
PMP 2291.OLR.001 Data Sheet 1	Work Schedule Review and Approval Form, Cycle 40, Week 1	January 14, 2002

1R15 Operability Evaluations

	D.C. Cook Nuclear Plant Unit 2 Technical Specifications	
	D. C. Cook Nuclear Plant Updated Final Safety Analysis Report	
Design Information Transmittal (DIT) B-02308-00	Power Operated Relief Valve (NRV-152, -153) Back-up Air Bottle Pressure	January 19, 2002
DIT B-02308-03	Review of 02-OHP-4030-202-060, Revision 0, “Pressurizer Relief Valve Testing,” Results for NRV-152, -153	January 21, 2002
DIT B-02327-00	Stroke Time Acceptance Criteria for ½-NRV-152, -153	February 1, 2002
Calculation MD-12-CA-004-S	Determination of Available Pressurizer Power Operated Relief Valve Strokes Using the Auxiliary Air Supply	Revision 1
Engineering Programs Technical Data Book, Figure 2-19.1	Power Operated Relief Valve Stroke Time Limits	Revision 52
Memorandum to John A. Grobe, Director Division of Reactor Safety, Region III	Response to the Task Interface Agreement Regarding Compliance With Generic Letter 90-06 Back-up Air Supplies to the Power Operated Relief Valves at D.C. Cook Units 1 and 2, AITS 97-02 (TAC Nos. M97886 and M97887)	October 1, 1998
02-OHP-4030-202-060	Pressurizer Relief Valve Testing	Revision 0
	Daily Shift Manager’s Logs	January 17, 2002 through January 19, 2002
CR 02017002	1-ESW-115, Essential Service Water to Turbine Driven Auxiliary Feedwater Pump Shutoff Valve Will Not Open	January 17, 2002
CR 02019039	2-NRV-153 Is Inoperable Due to Too Fast Stroke Time	January 19, 2002
CR 02019040	As Found Data Out of Specification for 2-AV-152 and 2-XRV-153 While Performing STP-189	January 19, 2002

CR 02019055	Serious Consideration Needs to be Given to Differentiating Between Pressurizer Power Operated Relief Valve Air Supply Setup and Inservice Test Stroke Time Limitations for Normal Operating Conditions and for Shutdown Low Temperature Over-pressure Protection Conditions	January 19, 2002
CR 02026029	Design Information Transmittal SGRP-99035-00, Revision 0, Incorrectly Added Unit 2 Reactor Coolant System (RCS) Volumes to Get a Total RCS Volume	January 26, 2002
CR 02032016 (1)	Unit 1 and Unit 2 Ice Condenser Lower Inlet Doors Were Improperly Tested Prior to Unit Startup	January 31, 2002
Letter AEP:NRC 2591	Emergency License Amendment Request for One-Time Limited Duration Exemption from Ice Condenser Lower Inlet Door Testing	February 8, 2002
Letter AEP:NRC 2591-01	Emergency License Amendment Request for One-Time Limited Duration Exemption from Ice Condenser Lower Inlet Door Testing	February 10, 2002
Job Order R0087658	Perform Unit 1 Lower Inlet Door Surveillance Testing 12-MHP 4030.010.003	
	Unit 1 License Amendment 265 to DPR-58 and associated NRC Safety Evaluation Report	February 14, 2002

1R19 Post Maintenance Testing

CR 02037089	2-CCR-440 Failed an IST Stroke Time Test in April 2000. The Closing Limit Switch Was Reset Without Adequately Verifying That the Valve Had Traveled Fully Closed	February 6, 2002
CR 02021004	2-CCR-440 Leaked at 44,000 Standard Cubic Centimeters Per Minute With a Supply Pressure of 1.5 Pounds Per Square Inch During Local Leak Rate Testing	January 21, 2002

CR 01101073	2-CCR-440 Failed to Indicate Closed During Cycling of the Valve for Surveillance	April 11, 2001
Job Order 02021004	B & C Leak Rate Retest of 2-CCR-440	
12-MHP 5021.001.143	Hammel-Dahl Series A40 Actuator Maintenance	Revision 0
Job Order 01101073	Readjust/tighten Actuator Arm 2-CCR-440	
DCC NEMP 306 QCN	Containment Penetration Isolation Barrier	Revision 1
02-EHP 4030.234.203	Unit 2 B & C Leak Rate	Revision 0

1R20 Refueling and Outage Activities

	D.C. Cook Nuclear Plant Unit 2 Technical Specifications	
	D. C. Cook Nuclear Plant Updated Final Safety Analysis Report	
12-OHP-4050-FHP-001	Refueling Procedure Guidelines	Revision 3
12-OHP-4050-FHP-005	Core Unload/Reload and Incore Shuffle	Revision 3
12-OHP-4050-FHP-023	Reactor Vessel Head Removal With Fuel in the Vessel	Revision 0
12-OHP-4050-FHP-026	Upper Internals Removal With Fuel in the Vessel	Revision 1
2-OHP-4030-STP-041	Refueling Integrity	Revision 8
PMP 4100-SDR-001	Plant Shutdown Safety and Risk Management	Revision 5, C1
	Daily Shift Manager’s Logs	January 19, 2002 through February 9, 2002
Memo From R.W. Hennen to Shift Technical Advisors	Unit 2 Time to 200°F and Time to Boil Figures for the Refueling Outage	January 4, 2002
	U2C13 Outage Schedule Shutdown Risk Review	
Rapid Event Response Report	Electrical Flash During Diesel Generator Sequence Testing	January 21, 2002

Rapid Event Response Report	Personnel Contaminated in Unit 2 Lower Containment While Performing 2-EHP-4030-202-226	January 21, 2002
Rapid Event Response Report	Loss of 110 Volt Power During Dive Operations	January 22, 2002
CR 02023077	Unit 2 Entered an Unintended Shutdown Risk Orange Path on Inventory Control Due to a Clearance Hung Prematurely (by 6 Days) on Both Safety Injection Pumps	January 23, 2002
CR 02023092	Procedure Conflict With Engineering and Operations Procedures Caused an Isolation of the RCS Drain Down and Water Spill in the North Chemical and Volume Control System Holdup Tank Room From 12-CS-469	January 23, 2002
CR 02024018	Service Penetration Not Installed Properly	January 24, 2002
CR 02024058	Valve Found Partially Open Which Was Causing a Loss of RCS Inventory While RCS Was Drained to 619.3 Feet	January 24, 2002
CR 02025012	Errors Identified on Guarded Equipment Poster in the OCC and on OCC Shift Turnover Sheet	January 25, 2002
CR 02026033	Contractor Crew Removed the Steam Generator Manway Bolting From the Wrong Steam Generator	January 26, 2002
CR 02026054	The Guarded Equipment Sign Intended for the Unit 2 East Component Cooling Water Pump Was Discovered Affixed to the Permanent Platform Adjacent to the Unit 1 East Component Cooling Water Pump by Plant Assurance Observation	January 26, 2002
CR 02026060	Unit 2 Safety Related 600 Volt Bus Voltages Exceeded 645 Volts for Greater Than Two Hours	January 26, 2002
CR 02026077	Entered Guarded Equipment Room to Perform Work Prior to Receiving Approval	January 26, 2002

CR 02027039	Activity to Replace Cell 61 on 2-AB Battery Did Not Receive a Safety Assessment Review for Outage Scope Add Consistent With the Scope of Work in Accordance With PMP-2291-OUT-001	January 27, 2002
CR 02029013	Worker Received a Dose Rate Alarm of His Electronic Dosimeter and Failed to Leave the Area Immediately and Notify Radiological Protection	January 29, 2002
CR 02029016	Individual Disregarded Radiological Protection Technician Directive and Then Disregarded His Electronic Dosimeter Alarm While Working in Unit 2 Lower Containment	January 29, 2002
CR 02029044	Unit 2 "Z" Fuel Assemblies With Top Nozzle Gaps Greater Than 0.025 Inches	January 29, 2002
CR 02036014	Adjustable Wrench Inadvertently Dropped in Spent Fuel Pit	February 5, 2002
CR 02037107	Recommended Guarded Equipment Was Not Properly Posted	February 6, 2002

1R22 Surveillance Testing

12-MHP 4030.010.003	Ice Condenser Lower Inlet Door Surveillance	Revision 0
Calc Note FAI/99-77	Containment Sump Level Evaluations for the D.C. Cook Plant in Support of License Amendment s 234/217	September 1999
CR 02020024	Ice Condenser Lower Inlet Doors Were Inadvertently Forced Open at Least Twice after Entry into Mode 5	January 20, 2002
CR 02022012	Eleven out of Forty-eight Lower Inlet Doors Failed Opening Force Testing	January 21, 2002
CR 02024066	Labeling of Lower Inlet Doors on Containment Auxiliary Sub-panel Is Inconsistent with Surveillance Procedure	January 24, 2002
CR 02025024 (1)	Test Device Used to Perform Lower Inlet Door Testing Failed to Meet Procedural Requirements	January 25, 2002

CR 02025074 [1]	Inconsistencies in Thermal Soak Requirements for Spring Scales Used for Lower Inlet Door Testing	January 25, 2002
CR 02025075 [1]	Apparent Procedural Violation During Performance of Ice Condenser Lower Inlet Door Testing	January 25, 2002
CR 02025084 [1]	Several Performance Problems, Errors, and Violations Related to Performance of 12-MHP 4030.010.003, Ice Condenser Lower Inlet Door Surveillance	January 25, 2002
CR 02032016 [1]	Ice Condenser Lower Inlet Door Testing Performed Prior to Unit 1 and Unit 2 Restart in 2000 Was Inadequate	January 31, 2002
DIT B-000312-01	Instrument Uncertainty Review of Ice Condenser Surveillance Procedures	July 7, 2000
DIT S-00105-01	Ice Condenser Lower Inlet Door Surveillance Requirements	March 15, 2000
Job Order R0087658	Perform Ice Condenser Lower Inlet Door Surveillance Testing on Unit 1	
Job Order R0087666	Perform Ice Condenser Lower Inlet Door Surveillance Testing on Unit 2	
Job Order R0102475	Unit 2 - Perform Lower Inlet Door Surveillance	
WCAP-7689	Design and Performance Evaluation of Ice Condenser Inlet Doors	

4OA3 Event Follow-up

LER 50-315-2001-005-00	RCCA [Rod Control Cluster Assembly] Tool Over Spent Fuel Pool Racks Technical Specification Violation	January 11, 2002

(1) Condition Reports Issued As a Result of Inspection Activities

INFORMATION REQUESTED ON DECEMBER 18, 2001 BY E-MAIL (To R. Gaston)

A. Please provide the following information to Melvin Holmberg at the Region III NRC office located at 801 Warrenville Rd, Lisle IL 60532, no later than January 14, 2002, to support the NRC Inservice Inspection (IP 71111.08 and TI-145) scheduled for January 22, 2002 - February 15, 2002 at the D.C. Cook Unit 2 site.

1) A detailed schedule of nondestructive examinations planned for Class 1 & 2 systems and containment, performed as part of your ASME Code ISI Program during the scheduled inspection weeks. Provide a detailed schedule of vessel head examinations which fulfill NRC commitments made in response to NRC Bulletin 2001-01. Provide a detailed schedule of SG tube inspection and repair activities for the upcoming outage.
2) A copy of the procedures used to perform the examinations identified in A.1. For ultrasonic examination procedures qualified in accordance with Appendix VIII, of Section XI of the ASME Code, provide documentation supporting the procedure qualification (e.g. the EPRI performance demonstration qualification summary sheets). Also, include documentation of the specific equipment to be used (e.g. ultrasonic unit, cables, and transducers including serial numbers).
3) A copy of any ASME Section XI, Code Relief Requests applicable to the examinations identified in A(1).
4) A list identifying nondestructive examination reports (ultrasonic, radiography, magnetic particle, dye penetrant, visual (VT-1, VT-2, VT-3)) which have identified relevant indications on Code Class 1 & 2 systems in the past two refueling outages (both Units).
5) List of welds in Code Class 1, and 2 systems which have been completed since the beginning of the last refueling outage (both Units and identify system, weld number and reference applicable documentation).
6) For reactor vessel weld examinations required by the ASME Code, that are scheduled during the inspection, provide a detailed description of the welds to be examined, extent of the planned examination and a copy of your responses to the NRC, associated with Generic Letter 83-15.
7) Provide a list with description of ISI and steam generator related issues entered into your corrective action system beginning with the date of the last refueling outage (both Units).
8) Copy of any part 21 reports submitted beginning with the date of the last refueling outage.
9) Copy of SG history documentation given to vendors performing eddy current (ET) testing of the SGs during the upcoming outage.
10) Copy of procedure containing screening criteria used for selecting tubes for in-situ pressure testing and the procedure to be used for in-situ pressure testing.
11) Copy of previous outage SG tube operational assessment completed following ET of the SGs.
12) Copy of the document defining the planned ET scope for the SGs and the scope expansion criteria which will be used.
13) Copy of the document describing the ET probe types, and ET acquisition equipment to be used, including which areas of the SG (e.g. top of tube sheet, U-bends) each probe will be used in. Also, provide your response letter(s) to generic letters 95-03, 95-05, 97-05, and 97-06.
14) Copy of document describing actions to be taken if a new SG tube degradation mechanism is identified.

15) Identify the types of SG tube repair processes which will be implemented for defective SG tubes. Provide the flaw depth sizing criteria to be applied for ET indications identified in the SG tubes.
16) If tube leakage was identified during the previous operating cycle, provide documentation identifying which SG was leaking and planned corrective actions.
17) Provide a copy of the EPRI Technique Specification Sheets which support qualification of the ET probes to be used during the upcoming SG tube inspections.
18) Provide a copy of the guidance to be followed if a loose part or foreign material is identified in the SGs.
19) Detailed scope of the planned nondestructive examinations (NDE) of the vessel head which identifies the types of NDE methods to be used on each specific part of the vessel head to fulfill NRC commitments made in response to NRC Bulletin 2001-01. Also include examination scope expansion criteria and planned expansion sample sizes if relevant indications are identified.
20) Copy of NDE procedures to be used for performing vessel head inspections that fulfill NRC commitments in response to NRC Bulletin 2001-01.
21) Identify what standards or requirements will be used to evaluate indications identified during NDE examinations of the vessel head.

B. Information to be provided on-site to the inspector at the entrance meeting:

1) For welds selected by the inspector from A.5 above, provide copies of the following documents:
    a) Document of the weld number and location (e.g. system, train, branch).
    b) Document with a detail of the weld construction.
    c) Applicable Code Edition and Addenda for weldment.
    d) Applicable Code Edition and Addenda for welding procedures.
    e) Applicable weld procedures (WPS) used to fabricate the welds.
    f) Copies of procedure qualification records (PQRs) supporting the WPS on selected welds.
    g) Copies of mechanical test reports identified in the PQRs above.
    h) Copies of the nonconformance reports for the selected welds.
    i) Radiographs of the selected welds and access to equipment to allow viewing radiographs.
    j) Copies of the pre-service examination records for the selected welds.

2) For the replacement activities selected by the inspector provide a copy of the records of the repair or replacement required by the ASME Code Section XI Articles IWA -4000 or IWA 7000.
3) Provide a list of NDE personnel performing inspections of the vessel head and the qualification records for these personnel.
4) Copies of commitments made to the NRC for performing vessel head examinations.
5) Copy of the most recent quality assurance department audit, which included the ISI program and activities. Copies of documents resolving findings in this audit.
6) Updated schedules for item A.1.