April 29, 2003

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

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

Dear Mr. Bakken:

On, March 31, 2003, 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 April 4, 2003, with Mr. M. Finissi 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, two findings of very low safety significance (Green) were identified which involved violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC Enforcement Policy. If you contest the Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the D. C. Cook facility.

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

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

Sincerely,

/RA/

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

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

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

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

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-315; 50-316
License Nos:	DPR-58; DPR-74
Report No:	50-315/03-02; 50-316/03-02
Licensee:	Indiana Michigan Power Company
Facility:	D. C. Cook Nuclear Power Plant, Units 1 and 2
Location:	1 Cook Place Bridgman, MI 49106
Dates:	December 29, 2002 through March 31, 2003
Inspectors:	B. Kemker, Senior Resident Inspector I. Netzel, Resident Inspector A. Dunlop, Reactor Engineer G. O'Dwyer, Reactor Engineer
Approved by:	E. Duncan, Chief Branch 6 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000315-03-02, IR 05000316-03-02; Indiana Michigan Power Company; on 12/29/2002-03/31/2003; D. C. Cook Nuclear Power Plant, Units 1 and 2; Surveillance Testing; Identification and Resolution of Problems.

This report covers a 13-week period of inspection by resident and regional based inspectors. The inspectors identified two 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). Findings for which the SDP does not apply may be “Green” or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. **Inspector Identified Findings**

Cornerstone: Mitigating Systems

- Green. Licensee personnel failed to promptly evaluate operability of number 23 steam generator power operated relief valve (PORV) 2-MRV-233 following inservice testing failures on two occasions.

  This issue was of very low safety significance since the redundant steam generator PORVs were available and therefore no actual loss of safety function occurred. One Non-Cited Violation of Technical Specification 4.0.5.a was identified. (Section 1R22)

- Green. The licensee failed to promptly evaluate operability of the Unit 1 normal Reactor Coolant System letdown isolation valve 1-QRV-112 on two occasions when its ability to satisfy inservice testing program requirements could not be demonstrated.

  This issue was of very low safety significance since the redundant letdown isolation valve, 1-QRV-111, was available during the period that 1-QRV-112 was inoperable and therefore no actual loss of safety function occurred. One Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” was identified. (Section 4OA2.1)

B. **Licensee Identified Violations**

None.

REPORT DETAILS

Summary of Plant Status

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

- On January 15, 2003, a fault occurred in the Unit 1 main transformer resulting in a fire. The fault caused an automatic main generator trip and reactor trip. The Emergency Plan was activated at the Unusual Event level due to a fire within the protected area not being extinguished within 15 minutes. Following replacement of the transformer, Unit 1 was restarted and returned to full power on February 6, 2003.

- On February 23, 2003, the licensee reduced power to about 90 percent to flush the main condenser waterboxes due to intrusion of silt, shells, and ice. Unit 1 returned to full power on February 24, 2003.

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

- On January 26, 2003, the licensee performed a reactor shutdown as required by Technical Specification (TS) 3.8.1.1.b for an inoperable emergency diesel generator (EDG). After replacing the governor on the engine and completing post maintenance testing, Unit 2 was restarted and synchronized to the grid on January 29, 2003.

- On February 5, 2003, Unit 2 experienced an automatic reactor trip due to a power supply failure that caused the number 23 steam generator feedwater regulating valve to fail closed. The main steam isolation valves were shut following the trip to arrest an excessive reactor coolant system (RCS) cooldown, resulting in a loss of the normal heat sink. Following necessary repairs, Unit 2 was restarted and returned to full power on February 16, 2003.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the licensee's procedures and preparations for cold weather conditions. During post-winterization walkdowns conducted the week of January 13, 2003, the inspectors toured plant areas to monitor the physical condition of cold weather protection features following a period of extended freezing temperatures. The inspectors observed insulation, heat trace circuits, space heater operation, and weatherized enclosures to ensure operability of affected systems.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

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

- Unit 1 East and West Motor Driven Auxiliary Feedwater Pump (MDAFP) sub-systems with the Unit 1 Turbine Driven Auxiliary Feedwater Pump (TDAFP) out-of-service for planned maintenance.

- Unit 2 CD EDG with the Unit 2 AB EDG out-of-service for planned maintenance.

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

b. Findings

No findings of significance were identified.

.2 Complete System Walkdown

a. Inspection Scope

The inspectors performed a complete system walkdown of the Unit 1 and Unit 2 Control Air System to verify system operability. This system was selected because it was considered risk-significant in the licensee’s probabilistic risk assessment and impacted the Initiating Event cornerstone.

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

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

- Auxiliary Building Basement 573 Foot Elevation (Zone 1)
- Unit 1 East and West Containment Spray Pump Rooms (Zones 1A and 1B)
- Unit 1 East and West Residual Heat Removal Pump Rooms (Zones 1C and 1D)
- Unit 2 East and West Containment Spray Pump Rooms (Zones 1E and 1F)
- Unit 2 East and West Residual Heat Removal Pump Rooms (Zones 1G and 1H)
- Auxiliary Building 587 Foot Elevation East End (Zone 5)
- Auxiliary Building 587 Foot Elevation Middle Section of West End (Zone 6M)
- Auxiliary Building 587 Foot Elevation North Section of West End (Zone 6N)
- Auxiliary Building 587 Foot Elevation South Section of West End (Zone 6S)

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.

1R07 Heat Sink Performance - Biennial Review (71111.07)

a. Inspection Scope

The inspectors reviewed documents associated with inspection, cleaning, and performance trending of heat exchangers primarily focusing on the 1E Component Cooling Water (CCW) heat exchanger. This heat exchanger was chosen based upon its importance in supporting required safety functions as well as a relatively high risk achievement worth in the plant specific risk assessment. The licensee did not perform thermal performance testing of this heat exchanger. During the inspection, the inspectors reviewed calculations, and performed independent calculations to verify that these activities adequately ensured proper heat transfer. The inspectors reviewed documentation to confirm that the inspection methodology was consistent with accepted industry and scientific practices, based on review of heat transfer texts and Electrical Power Research Institute (EPRI) standards (EPRI NP-7552, Heat Exchanger Performance Monitoring Guidelines, December 1991 and EPRI TR-107397, Service Water Heat Exchanger Testing Guidelines, March 1998) and Mark's Engineering Handbook.

The inspectors reviewed condition reports concerning heat exchanger and ultimate heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and entering them in the corrective action program. The inspectors also evaluated the effectiveness of the corrective actions for identified issues, including the engineering justification for operability, if applicable.

b. Findings

.1 Essential Service Water (ESW) System Water Hammer Load Calculation Concern

Introduction

The inspectors identified an unresolved item concerning whether estimated water hammer load calculations on the ESW system were valid. The inspectors challenged several assumptions and conclusions in the calculation. This is considered an unresolved item pending review of additional documentation and revised analyses.

Description

The inspectors reviewed condition report P-00-10960 initiated August 6, 2000, which documented that water hammers had occurred in the ESW system and were expected to occur during certain potential events in the future (e.g., during a loss of offsite power on one unit with the Unit ESW crosstie valve closed). The condition report documented that the water hammers were due to water column separation and rejoining. The inspectors were concerned that the condition report had not properly identified and addressed all potential causes of water hammer (e.g., water hammer will occur after a loss of offsite power on both units regardless of whether the unit ESW crosstie valve is open or closed). As a part of corrective action 5 generated by the condition report the licensee performed calculation EVAL-MD-02-ESW-092-N, revision 0, to determine the maximum pressures that could potentially result from column separation/rejoining in the ESW system. The calculation also addressed the water hammer's potential adverse impacts on the ESW system and the many components it services (e.g., the CCW and Containment Spray heat exchangers). The inspectors were concerned that the licensee had underestimated the magnitude of the water hammer loads because the inspectors identified the following potential deficiencies or concerns with the calculation, which were non-conservative and applicable to both units:

> The purpose of the calculation stated that only the pressure spikes would be determined and the calculation would not include pipe reaction forces resulting from acceleration/deceleration of water from column rejoining. The calculation did not provide a basis for not considering the pipe reaction forces. The inspectors were concerned because pipe support damage is far more probable than piping damage. For example, this was documented in NUREG/CR-5220, Creare TM-1189, "Diagnosis of Condensation-Induced Waterhammer," October 1988, particularly case study 5 of volume 2. Case study 5 documented that severe damage to multiple mechanical snubbers resulted from a water hammer even though there was no observable piping damage.
>
> The calculation did not appear to adequately consider the impacts from momentary loads, specifically on ductile materials. Conclusion 8.8 of the calculation stated that some of the water hammer pressures exceeded the valve and flange acceptance criteria for the 2E auxiliary feedwater (AFW)

pump and 2E AFW pump room cooler during the ESW flow balance and for the CCW heat exchangers during the injection phase and station blackout events. The calculation stated that the valve and flange acceptance criteria were from ASME/ANSI B16.5-1988 “Pipe Flanges and Flanged Fittings,” ASME/ANSI B16.34-1988 “Valves-flanged, Threaded, and Welding End,” and NAVCO piping datalog, by National Valve and Manufacturing Company, Pittsburgh, PA. Conclusion 8.8 stated there was no cause for concern because “these calculated loads are spike loads and could be considered momentary peak pressures which should not be detrimental to a ductile material.” The licensee did not provide a basis for this conclusion. The inspectors noted this conclusion did not consider potential excessive plastic deformation of the ductile materials of components. Cases where spike loads or momentary peak pressures caused enough plastic deformation of ductile materials for the materials to fail were documented in NRC Information Notice No. 91-50, Supplement 1: Water Hammer Events Since 1991, dated July 17, 1997.

The calculation did not appear to adequately consider potential water hammer damage to more limiting equipment. For example, the licensee had previously identified bowed (plastic deformation) CCW heat exchanger divider plates and cracked attachment welds for those divider plates. The design differential pressure across the divider plates was approximately 15 psig. The licensee believed that the deformation and cracked welds resulted from the normal operating differential pressure across the divider plate of 14.7 psid. The licensee had specified to the Framaton company that the maximum differential pressure across the divider plate would be 15 psid and requested Framaton company to design a repair to prevent future damage to the divider plate. However, this direction did not properly consider that during water hammer events the maximum differential pressure was and will likely be significantly greater than 15 psid. For example, the differential pressure certainly exceeded 15 psig during the June 8, 2000, and August 6, 2000 water hammers which lifted a relief valve set at about 115 psig located about 30 feet downstream of the CCW divider plate. The outlet side of the divider plate was at approximately atmospheric pressure (due to voiding in the CCW heat exchanger outlet piping) before the pressure pulse hit the inlet side of the divider plate. The calculation also did not address the potential for water hammer damage to the tubes of the CCW heat exchangers and baffles of the Containment Spray heat exchangers.

The calculation did not provide a basis for the assumption in Conclusion 8.8 that the ESW fluid was entraining sufficient air to allow a reduction in the sonic velocity. This assumption was used to demonstrate that all the water hammer pressures were within the piping code allowables. However, the inspectors noted that a properly functioning ESW system should not be entraining air. The inspectors were concerned that entrained air might indicate a mechanism detrimental to the system (e.g., lack of proper submergence for the ESW pump impellers indicated by air-entraining vortexing).

The inspectors questioned an assumption in the methodology section that the velocity of the water in the inlet piping to the CCW heat exchanger during water hammer events was equal to the velocity of the water during safety actuation flow (5270 gpm). The inspectors were concerned because the velocity of the water during the water hammer might have been and might in the future be significantly higher because portions of the ESW piping void due to draining while the flow is stopped. Therefore, when the pumps restart they are pumping against a significantly lower system resistance than when the system is completely filled. The pumps would pump a greater volume which would result in a higher velocity. The flow rate of the pumps might even be near runout.

The licensee indicated that they would provide additional documentation and analyses to address the above concerns. This is considered an unresolved item (URI 50-315/316-03-02-01(DRS)) pending receipt and inspector review of this information.

.2 Estimation of Tube Blockage in the CCW Heat Exchangers

Introduction

The inspectors identified an unresolved item regarding the licensee's methodology in determining the number of CCW heat exchanger tubes that were considered blocked. This is an unresolved item pending receipt and inspector review of a revised methodology and impact on the previous as-found conditions of the CCW heat exchanger tubes and assessment of the frequency of inspections and cleanings.

Description

The licensee did not perform thermal performance testing on any heat exchangers. The licensee relied on periodically opening and cleaning the heat exchangers to ensure their ability to perform their safety function versus performing thermal performance testing. The licensee considered the CCW heat exchanger tubes to be blocked only if they could not be cleared with air pressurized to 65 psig. The inspectors questioned that criteria and the licensee agreed that there was insufficient basis to consider a tube not blocked if it can be cleaned by 65 psig air. The as-found condition of the tubes was important in determining the correct frequency of cleanings. As a result of the inspectors' concerns, the licensee was developing other criteria to determine tube blockage. The licensee indicated that they would provide this revised criteria, an analysis that provides a justification for its use, and the projected impact on the previous as-found condition of the CCW heat exchangers including necessary frequency of cleaning.

This is considered an unresolved item (URI 50-315/316-03-02-02(DRS)) pending receipt and inspector review of the licensee's analysis.

.3 Questionable Data Regarding CCW Heat Exchanger As-Built Specification Sheet

Introduction

The inspectors identified an unresolved item in that the as-built specification sheet for the CCW heat exchangers may overestimate the heat transfer capability of the heat exchangers. The item is considered an unresolved item pending inspector receipt and review of additional justification of the as-built specification sheet.

Description

During a previous Safety System Design and Operational Performance Inspection (SSDI) documented in report 50-315/316-01-15, the inspectors identified that calculation ENSM 990305AF, revision 0, “Determine CCW Heat Exchanger UA Value During the Recirculation Operation,” assumed on page 4 of attachment 1 that the velocity through the shell side of the heat exchanger was 15.95 feet/second which resulted in a heat transfer rate of 65,000,000 British Thermal Units/hour (BTUs/hr). The inspector’s preliminary calculations determined that the shell side velocity was actually approximately 2.7 ft/sec. The inspectors also determined that the PROTO-HX heat exchanger computer program that the licensee used to model the CCW heat exchanger only modeled single segmental baffled heat exchangers. The inspectors found that the shell side drawings indicated that the CCW heat exchanger had a triple segmental baffled shell. On October 17, 2001, the licensee had Proto-Power Corporation perform for these discrepancies an evaluation, “Investigation into the Effects of Lower Shell Side Velocities on Calculated Heat Transfer Rates for DC Cook CCW Heat Exchanger.” On page 4, this evaluation stated that if a shell side velocity of 3.62 ft/sec was assumed, then the manufacturer’s heat exchanger specification sheet required the outside heat transfer coefficient correction factor to be equal to 0.8.

During the current inspection, the inspectors questioned the reasonableness of the 0.8 outside heat transfer coefficient correction factor. The Electric Power Research Institute Service Water Heat Exchanger Testing Course Manual, edition 1, page 34 of 44 in learning module 6, indicated that the outside heat transfer coefficient correction factor for most well-designed single segmental heat exchangers is about 0.6. On page 733, the “Heat Transfer - Professional Version,” 2nd edition by Lindon Thomas also stated that the outside heat transfer coefficient correction factor for well-designed single segmental baffle heat exchangers under turbulent flow conditions was around 0.6. Page 643 of “Heat Transfer-Professional Version,” 2nd edition by Lindon Thomas indicated that triple segmental baffles are used to reduce the pressure drop across the shell by a factor of 3 compared to single segmental baffles at the sacrifice of some velocity which sacrifices some heat transfer in the shell. The optimum heating-to-pumping power performance has been found to occur with a single segmental baffle cut of about 20 to 25 percent. The inspectors were concerned that the specification sheet might have overestimated the CCW heat exchangers’ heat transfer capabilities and therefore the impact on their ability to perform their safety function. The licensee indicated that they would obtain additional verification or justification that the specification sheet was correct and provide that information for inspector review. This is considered an unresolved item (URI 50-315/316-03-02-03(DRS)) pending inspector review of this information.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspectors assessed licensed operator performance and the training evaluators' critique during licensed operator annual re-qualification evaluations in the D. C. Cook Plant operations training simulator on March 19, 2003. The inspectors focused on alarm response, command and control of crew activities, communication practices, procedural adherence, and implementation of emergency plan requirements.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Resident Inspector Review

a. Inspection Scope

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

- Radiation Monitoring for RCS Leakage Detection
- Safety Injection System

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

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

b. Findings

No findings of significance were identified.

.2 Periodic Maintenance Rule Evaluation

a. Inspection Scope

The objective of the inspection was to:

- verify that the periodic evaluation was completed within the time restraints defined in 10 CFR 50.65, the Maintenance Rule (once per refueling cycle, not to exceed two years), ensuring that the licensee reviewed its goals, monitoring, preventive maintenance activities, industry operating experience, and made appropriate adjustments as a result of that review;

- verify that the licensee balanced reliability and unavailability during the previous refueling cycle, including a review of safety significant SSCs;

- verify that (a)(1) goals were met, corrective actions were appropriate to correct the defective condition including the use of industry operating experience, and (a)(1) activities and related goals were adjusted as needed; and

- verify that the licensee has established (a)(2) performance criteria, examined any SSCs that failed to meet their performance criteria, or reviewed any SSCs that have suffered repetitive maintenance preventable functional failures including a verification that failed SSCs were considered for (a)(1).

The inspectors examined the last periodic evaluation report for November 1, 1999 through October 5, 2001. To evaluate the effectiveness of (a)(1) and (a)(2) activities, the inspectors examined (a)(1) action plans, justifications for returning SSCs from (a)(1) to (a)(2), and a number of CRs (contained in the list of documents at the end of this report). In addition, the CRs were reviewed to verify that the threshold for identification of problems was at an appropriate level and the associated corrective actions were appropriate. The inspectors focused the inspection on the following systems:

- Containment Spray System
- Residual Heat Removal System
- Chemical and Volume Control System

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 affecting the following equipment:

- Unit 1 Rod Control System
- Unit 2 Containment Lower Airlock Door
- Unit 1 Turbine Driven Auxiliary Feedwater Pump
- Unit 2 AB EDG

These activities were selected based on their potential risk significance relative to the reactor safety cornerstones. The maintenance associated with the Unit 1 rod control system was emergent work to correct a control malfunction that prevented manual insertion of control bank "B" that was identified during a scheduled surveillance test. There was also emergent work associated with the Unit 2 containment lower airlock to allow its restoration to an operable status when maintenance personnel were unable to open the inner door during testing.

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

b. Findings

No findings of significance were identified.

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

.1 Unit 1 Main Power Transformer Fire and Reactor Trip

a. Inspection Scope

On January 15, 2003, a fault occurred in the Unit 1 main transformer resulting in a fire. The fault caused an automatic main generator trip and reactor trip. The Emergency Plan was activated at the Unusual Event level due to a fire within the protected area not being extinguished within 15 minutes. The licensee returned the unit to full power on February 6, 2003, after replacing the transformer and accomplishing some additional maintenance activities. The inspectors assessed control room operator performance immediately following the reactor trip and reviewed the post trip report.

b. Findings

No findings of significance were identified.

.2 Unit 2 Reactor Trip Due to Control Group Power Supply Failures

a. Inspection Scope

On February 5, 2003, Unit 2 experienced an automatic reactor trip due to a power supply failure that caused the number 23 steam generator feedwater regulating valve to fail closed. The main steam isolation valves were shut after the trip to arrest an excessive RCS cooldown due to auxiliary steam loads, causing a loss of the normal heat sink. The licensee returned the unit to full power on February 16, 2003, after replacing the failed power supplies and accomplishing some additional maintenance activities. The inspectors assessed control room operator performance immediately following the reactor trip and reviewed the post trip report.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- CR 03013023 — Unit 2 Containment Hydrogen Recombiner Maintenance Scheduled During 1East ESW Pump Replacement
- CR 03009038 — Control Switch Placed in Stop Rendering Both Auxiliary Building Engineered Safety Features Fans Inoperable
- CR 03011014 — 2-QFI-420 Output Intermittently Spiking High
- CR 03011017 — 2-QT-117-AB Pump Coupling Disengaged from the Motor
- CR 03011027 — Output Frequency of 2-CRID-4-INV Is High and "In-Sync" Light Not Lit
- CR 03028032 — Horizontal Missile Blocks Over Unit 1 Reactor Head Found With Numerous Hold Down Nuts Loose

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

- Unit 1 TDAFWP
- Unit 2 AB EDG

The inspectors selected these post maintenance testing activities because the systems were identified as risk significant in the licensee's risk analysis. The inspectors reviewed the scope of the work performed and evaluated the adequacy of the specified post maintenance testing. The inspectors verified that the post maintenance testing was performed in accordance with approved procedures, that the procedures clearly stated acceptance criteria, and that the acceptance criteria were met. During this inspection, the inspectors interviewed operations, maintenance, and engineering department personnel and reviewed the completed post maintenance testing documentation.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Unit 1 Forced Outage (Main Transformer Fire)

a. Inspection Scope

On January 15, 2003, the licensee entered a forced outage on Unit 1 following a reactor trip caused by the loss of the main power transformer due to a fire. The licensee entered Mode 5 (Cold Shutdown) to replace the transformer and perform additional maintenance work, including the repair of an oil leak on a reactor coolant pump motor. The licensee performed a reactor startup and synchronized the unit to the grid on February 5, 2003.

The inspectors evaluated the licensee's conduct of forced outage activities to assess the 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 and reviewed outage work activities to ensure that correct system lineups were maintained for key mitigating systems. Other major outage activities evaluated included the licensee's control of the following:

- SSCs which could cause unexpected reactivity changes;
- switchyard activities and the configuration of electrical power systems in accordance with the TSs and shutdown risk plan; and
- SSCs required for decay heat removal.

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

- verification that RCS boundary leakage requirements were met prior to entry into Mode 4 (Hot Shutdown) and subsequent operational mode changes;
- verification that containment integrity was established prior to entry into Mode 4;
- inspection of the Containment Building to assess material condition and search for loose debris, which if present could be transported to the containment recirculation sumps and cause restriction of flow to the emergency core cooling system (ECCS) pump suctions during loss-of-coolant accident conditions.

b. Findings

No findings of significance were identified.

.2 Unit 2 Forced Outage Due to Inoperable CD EDG

a. Inspection Scope

On January 26, 2003, the licensee performed a reactor shutdown as required by TS 3.8.1.1.b to troubleshoot and repair the Unit 2 CD EDG governor. The licensee successfully completed repairs and post maintenance testing of the engine on January 27, 2003. The licensee performed a reactor startup and synchronized Unit 2 to the grid on January 29, 2003.

The inspectors evaluated the licensee's conduct of forced outage activities to assess the licensee's control of plant configuration and risk management actions. The inspectors reviewed the apparent cause for the governor failure as well as the extent of condition to other EDG governors. The inspectors also observed portions of the restart activities to verify that requirements of the TS and administrative procedure requirements were met prior to changing operational modes or plant configurations.

b. Findings

No findings of significance were identified.

.3 Unit 2 Forced Outage Due to Control Group Power Supply Failures

a. Inspection Scope

On February 5, 2003, the licensee entered a forced outage on Unit 2 following a reactor trip caused by the failure of redundant 24-volt direct current power supplies in reactor control instrumentation cabinet 2-PS-CGC-21. The licensee maintained Unit 2 in Mode 3 (Hot Standby) to replace the failed power supplies and perform additional maintenance work. The licensee performed a reactor startup and synchronized the unit to the grid on February 14, 2003.

The inspectors evaluated the licensee's conduct of forced outage activities to assess the licensee's control of plant configuration and risk management actions. The inspectors observed portions of the restart activities to verify that requirements of the TS and administrative procedure requirements were met prior to changing operational modes or plant configurations.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- 01-OHP-4030-STP-016, "Reactor Coolant System Leak Test"
- 01-IHP-4030-STP-089, "Pressurizer Power Operated Relief Valve [PORV] Cold Over-Pressurization Bistable and Backup Air Pressure System Functional Test"
- 01-OHP-4030-STP-011, "Containment Isolation and IST Valve Operability Test," Attachment 4, "Post Accident Sample System Valves Test"
- 02-OHP-4030-214-049, "Hot Shutdown Panel Operability Test," Attachment 14, "Steam Generator PORV Operability Test"

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.

b. Findings

The inspectors identified a finding of very low safety significance (Green) associated with the licensee's failure to correctly evaluate the operability of the number 23 Steam Generator PORV, 2-MRV-233, following inservice testing (IST) failures of the valve on two separate occasions. One Non-Cited Violation of TS 4.0.5 was identified.

Description

On November 30, 2001, 2-MRV-233 failed to fully open when it was stroked open for testing. Operators concluded that the valve was operable because there was no specific TS Limiting Condition for Operation (LCO) governing operability of the steam generator PORVs. Operators wrote CR 01334058 to address the valve's failure to fully open and a job order was prepared to correct the condition. In reviewing CR 01334058, the inspectors identified that operators did not enter TS 4.0.5.a and consider the valve to be inoperable as a result of its inability to satisfy IST program requirements. In addition, the inspectors found no entry in the Shift Manager's Log stating that 2-MRV-233 was inoperable. Since the valve would not fully stroke open, the inspectors determined that it should have been declared inoperable in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code contained in OMa-1988, "Inservice Testing of Valves in Light Water Reactor Power Plants," paragraph 4.2.1.9.a. The Code as well as OHI-4016, "Conduct of Operations Guidelines," both required that the valve be considered inoperable if it failed to attain the expected status (e.g., full open or full closed) upon demand. The licensee later determined that the valve positioner was out of calibration allowing air pressure to remain on top of the operating piston, which would not allow the valve to fully open. The licensee evaluated this condition and concluded that at 65 percent open, 2-MRV-233 was still able to perform its function of cooling the RCS prior to overfilling a ruptured steam generator during a steam generator tube rupture event. Subsequent testing on the valve determined that it actually opened to about 74 percent.

A failure of 2-MRV-233 also occurred on May 17, 2002. The valve failed its initial and immediate retest stroke time testing because it stroked too fast. According to the Shift Manager's Log, operators concluded that the valve was inoperable in accordance with

TS 4.0.5.a and the IST program; however, operators did not enter TS 4.0.5.a and did not enter the valve in the Open Items Log to track its status. Paragraph 4.2.1.9.b of OMa-1988 states that if the valve is retested and the second set of data also does not meet the acceptance criteria, the data shall be analyzed within 96 hours to verify that the new stroke time represents acceptable valve operation, or the valve shall be declared inoperable. Operators wrote CR 02137011 to address the valve stroke testing failure and a job order was prepared to correct the condition. However, contrary to the Shift Manager's Log entry, operations review of the CR concluded that the valve remained operable. The IST engineer was also not informed of the valve test failure to evaluate its performance in accordance with the Code requirements. The IST engineer identified this problem about 2 months later when he reviewed CR 02137011. On July 5, 2002, the IST engineer wrote CR 02186022 to address the incorrect actions taken for the inoperable valve.

On July 9, 2002, 2-MRV-233 again failed its routine quarterly stroke time testing. At that time, no actions had yet been taken to address the valve test failure on May 17, 2002. Subsequent inspection of the valve determined that the valve handwheel was not in the appropriate neutral position and restricted valve movement. This problem was subsequently resolved and the valve stroked properly within IST limits.

Analysis

The inspectors concluded that the licensee's failure to correctly evaluate conditions affecting the operability of 2-MRV-233 in accordance with IST program requirements was a performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting issue of problem identification and resolution.

The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors concluded that this issue, if left uncorrected, would become a more significant safety concern with operators' continued failure to correctly evaluate conditions affecting the operability of power operated valves in accordance with IST program requirements and therefore concluded that this issue was of more than minor concern. The inspectors also concluded that this finding was associated with the mitigating systems cornerstone and adversely affected the cornerstone objective. Specifically, the active and passive valves covered under the scope of the IST program are those which affect the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was a licensee performance deficiency of very low safety significance because the finding: (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; (4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

Enforcement

Technical Specification 4.0.5.a states, in part, that inservice inspection of ASME Code Class 1, 2 and 3 valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a. For inservice testing of valves, the applicable Code requirements are found in OMa-1988, "Inservice Testing of Valves in Light-Water Reactor Power Plants." Paragraph 4.2.1.9.a of OMa-1988 states, in part, that if a valve fails to exhibit the required change of position, the valve shall be immediately declared inoperable. Contrary to the above, the licensee failed to immediately declare 2-MRV-233 inoperable when the valve failed to exhibit the required change of position when it was stroked closed during testing on November 30, 2001. In addition, Paragraph 4.2.1.9.b of OMa-1988 states, in part, that if the valve is retested and the second set of data also does not meet the acceptance criteria, the data shall be analyzed within 96 hours to verify that the new stroke time represents acceptable valve operation, or the valve shall be declared inoperable. Contrary to the above, the licensee failed to analyze the retest failure of 2-MRV-233 within 96 hours or declare the valve inoperable when the valve failed its initial and immediate retest stroke time testing on May 17, 2002. This is a violation of TS 4.0.5.a. However, 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 50-316-03-02-04(DRP)). The licensee entered this violation into its corrective action program as CR 03062038 and CR 02186022.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

.1 Failures to Adequately Address 1-QRV-112 Stroke Test Failures

a. Inspection Scope

A failure of Unit 1 normal RCS letdown isolation valve 1-QRV-112 to satisfy IST requirements occurred on January 20, 2003. The inspectors reviewed this test failure and a previous test failure of 1-QRV-112 on April 28, 2001, to evaluate the licensee's resolution of these conditions adverse to quality.

The inspectors reviewed condition evaluations completed for the following CRs:

- CR 02067033, "On April 28, 2001, 1-QRV-112 Indicated Intermediate Position When Stroked in the Closed Direction. On September 3, 2001, Actual Travel Closed Was Measured to Be 50 Percent of Valve Stroke."

- CR 03031047, "Incorrect Operability Call Made on CR 0301819 for Valve 1-QRV-112. Screening Also Failed to Recognize Program Impact Resulting from the Leaking Solenoid Valve."

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

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

The inspectors discussed the corrective actions and associated CR evaluations with engineering and operations personnel.

b. Findings

The inspectors identified a finding of very low safety significance (Green) associated with the licensee's failure to correctly evaluate the operability of the Unit 1 normal RCS letdown isolation valve 1-QRV-112 on two occasions when its ability to satisfy IST program requirements could not be demonstrated. One Non-Cited Violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified.

Description

On April 28, 2001, 1-QRV-112 indicated an intermediate position when it was stroked closed for testing. Valve 1-QRV-112 is an air-operated valve located in the regenerative heat exchanger room and is not readily accessible due to very high dose rates in the room. Although the valve's full closure capability was not verified, operators assumed that the intermediate position indication was due to a limit switch problem and did not consider the valve inoperable. The licensee wrote electronic single action tracking (eSAT) 01118005 to document the condition; however, only a work request was issued to correct the assumed limit switch problem. On September 3, 2001, maintenance craftsmen found that the limit switches functioned correctly and that the valve would not fully close due to solenoid valve 1-XSO-502 failing to return to its rest position when power was removed. This was due to hardening of elastomer seals within the solenoid valve; a known age/heat related degradation problem for this type of solenoid valve. The valve was inoperable from April 28, 2001 to September 6, 2001, because it was not able to satisfy IST program requirements. During this time, the redundant train letdown isolation valve, 1-QRV-111, was available to satisfy the letdown isolation function.

A similar failure of 1-QRV-112 occurred on January 20, 2003. In this case, the valve failed stroke time testing because it stroked too slowly. A job order was written 2 days prior to the test failure because solenoid valve 1-XSO-502 was continuously venting air with 1-QRV-112 closed. Operators concluded that air venting from the solenoid valve did not affect the operability of 1-QRV-112 since its "fail-safe" position was closed. The eSAT written for the solenoid valve problem was closed to a job order and a CR was not generated for engineering evaluation. The inspectors determined that operators failed to recognize that the solenoid valve problem would potentially affect the ability of 1-QRV-112 to satisfy IST program requirements since they did not enter TS 4.0.5.a and consider the valve inoperable until operability of the valve could be demonstrated. The IST engineer learned of the solenoid valve problem only after the valve stroke test failure. Following the valve stroke test failure, operators declared 1-QRV-112 inoperable and entered TS 4.0.5.a.

Analysis

The inspectors concluded that the licensee's failure to correctly evaluate conditions affecting the operability of 1-QRV-112 in accordance with IST program requirements and to correct those conditions adverse to quality was a performance deficiency warranting a significance evaluation. The inspectors also concluded that this finding affected the cross-cutting issue of problem identification and resolution.

The inspectors assessed this finding using the SDP. The inspectors concluded that this issue, if left uncorrected, would become a more significant safety concern with operators' continued failure to correctly evaluate conditions affecting the operability of power operated valves in accordance with IST program requirements and therefore concluded that this issue was of more than minor concern. The inspectors also concluded that this finding was associated with the mitigating systems cornerstone and adversely affected the cornerstone objective. Specifically, the active and passive valves covered under the scope of the IST program are those which affect the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in NRC Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that this finding was a licensee performance deficiency of very low safety significance because the finding: (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; (4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

Enforcement

10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, deficiencies, and defective material and equipment are promptly identified and corrected. Contrary to the above, the licensee failed to promptly identify that 1-QRV-112 was inoperable on April 28, 2001 and January 20, 2003 and failed to take prompt corrective action to address these problems. This is a violation of 10 CFR 50, Appendix B, Criterion XVI. However, because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-315-03-02-05(DRP)). The licensee entered this violation into its corrective action program as CR 03090074.

4OA5 Other

On March 8, 2003, the licensee received a Notice of Enforcement Discretion (NOED) to extend the 72-hour allowed action time of TS 3.7.1.2.a to preclude shutting down Unit 2 until the West motor driven auxiliary feedwater pump (MDAFWP) could be restored to an operable status.

4OA6 Meetings

.1 Interim Exit Meetings

The results of the Biennial Maintenance Rule Program Inspection were presented to Mr. M. Finissi and other members of licensee management at the conclusion of the inspection on January 24, 2003. The results of the Heat Sink Inspection were presented to Mr. J. Pollock, Site Vice President and Mr. M. Finissi, Plant Manager on March 20, 2003. The licensee acknowledged the findings presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Resident Inspector Exit Meeting

The inspectors presented the inspection results to Mr. M. Finissi and other members of licensee management at the conclusion of the inspection on April 4, 2003. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee

P. Cowan, System Engineering Manager
M. Finissi, Plant Manager
J. Giessner, Regulatory Affairs/Nuclear Technical Services Director
E. Larson, Operations Director
T. Noonan, Performance Assurance Director
J. Pollock, Site Vice President

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315/03-02-01 50-316/03-02-01	URI	Water Hammer Analysis Deficiencies (Section 1R07)
50-315/03-02-02 50-315/03-02-02	URI	CCW Heat Exchanger Blockage Analysis (Section 1R07)
50-315/03-02-03 50-316/03-02-03	URI	CCW Heat Exchanger Specification Sheet (Section 1R07)
50-316/03-02-04	NCV	Deficient 23 Steam Generator PORV Operability Evaluation
50-315/03-02-05	NCV	Deficient Operability Evaluation of Letdown Valve 1-QRV-112

Closed

50-316/03-02-04	NCV	Deficient 23 Steam Generator PORV Operability Evaluation
50-315/03-02-05	NCV	Deficient Operability Evaluation of Letdown Valve 1-QRV-112

Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
AFW	Auxiliary Feedwater
ALARA	As-Low-As-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
ANS	Alert and Notification System
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CTS	Containment Spray
CVCS	Chemical Volume Control System
CY	Calender Year
DC	Direct Current
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EHP	Electrical Maintenance Head Procedure
EP	Emergency Preparedness
EPRI	Electrical Power Research Institute
eSAT	Electronic Single Action Tracking
ESW	Essential Service Water
IHP	Instrument Maintenance Head Procedure
IMC	Inspection Manual Chapter
IST	Inservice Testing
KW	Kilowatts
LER	Licensee Event Report
LCO	Limiting Condition For Operation
MHP	Maintenance Head Procedure
MDAFP	Motor Driven Auxiliary Feedwater Pump
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NOED	Notice of Enforcement Discretion
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OA	Other Activities
OHP	Operations Head Procedure
PARS	Publically Available Records
PI	Performance Indicator
PMI	Plant Manager's Instruction
PMP	Plant Manager's Procedure
PORV	Power Operated Relief Valve
psig	pounds per square inch gauge
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SSDI	Safety System Design and Operational Performance Inspection
SDP	Significance Determination Process

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

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

PMP 5055-001-001	Winterization/Summerization	Revision 0
12-OHP 4022.001.010	Severe Weather	Revision 1
12-IHP 4022.001.009	Plant Winterization and De-Winterization	Revision 0

1R04 Equipment Alignment

1R04.1 Partial System Walkdowns

Unit 1 East and West Motor Driven Auxiliary Feedwater Pump (MDAFWP) Trains

01-OHP-4021.056.001 Lineup Sheet 2	Filling and Venting Auxiliary Feedwater System: East MDAFWP Lineup	Revision 20
01-OHP-4021.056.001 Lineup Sheet 3	Filling and Venting Auxiliary Feedwater System: West MDAFWP Lineup	Revision 20
OP-1-5106A-55	Flow Diagram: Auxiliary Feedwater Unit 1	Revision 55
OP-1-5105D-9	Flow Diagram: Steam Generator System Unit No. 1	Revision 9
OP-1-5105E-1	Flow Diagram: Main Steam Unit No. 1	Revision 1

Unit 2 CD Emergency Diesel Generator (EDG)

OP-2-5151C-47	Flow Diagram: Emergency Diesel Generator "CD" Unit No. 2	Revision 47
OP-2-5151D-56	Flow Diagram: Emergency Diesel Generator "CD" Unit No. 2	Revision 56
02-OHP-4021-032-008CD	Operating DB2CD Subsystems, Attachments 1 - 7, Lineup Sheets 1 - 4	Revision 3

1R04.2 Complete System Walkdown

Unit 1 and 2 Control Air System

01-OHP-4021-064-001	Operation of Plant and Control Air Systems: Line Up Sheets 2 - 5	Revision 13
02-OHP-4021-064-001	Operation of Plant and Control Air Systems: Line Up Sheets 2 - 5	Revision 12
OP-1-5120C-32	Flow Diagram: Compressed Air System (Arrangement Of Control Air Equipment) Unit 1	Revision 32
OP-2-5120C-38	Flow Diagram: Compressed Air System (Arrangement Of Control Air Equipment) Unit 2	Revision 38

Miscellaneous Condition Reports

CR 03020049	Containment Isolation Valve B&C Test Procedure Failed to Maintain the Desired Mode 1-4 Configuration of Some Valves Which Were Manipulated to Establish the Desired Test Configuration Upon System Restoration.	January 20, 2003
CR 03030003	Unexplained Loss of Reactor Coolant System.	January 30, 2003
CR 03064050 (1)	1-IPX-211-V1 Has an Active Packing Leak. Area is Moist. Leakage Is Too Small to Quantify.	March 5, 2003
CR 03064051 (1)	2-IPX-221-V1 Leak-by and Pipe Plug Leakage. Replace as Necessary.	March 5, 2003
CR 03081008 (1)	Dry Boric Acid Found on 1-QRV-200 (Charging to Regenerative Heat Exchanger Flow Control Valve)	March 22, 2003
CR 03081009 (1)	Dry Boric Acid Found on 1-QMO-201 (CVCS [Chemical and Volume Control System] Charging to Regenerative Heat Exchanger Train 'B' Shutoff Valve)	March 22, 2003
CR 03081010 (1)	Dry Boric Acid Found on 1-QMO-200 (CVCS Charging to Regenerative Heat Exchanger Train 'A' Shutoff Valve)	March 22, 2003

CR 03081011 (1)	Dry Boric Acid Found on 1-QRV-251 (CVCS Centrifugal Charging Pumps Discharge Flow Control Valve)	March 22, 2003
CR 03081012 (1)	Dry Boric Acid Found on 1- IMO-910 (Refueling Water Storage Tank to CVCS Charging Pumps Suction Header Train 'A' Shutoff Valve)	March 22, 2003
CR 03083034 (1)	Inadequate Tours of Positive Displacement Charging Pump Area for Boric Acid Leak Identification	March 24, 2003

1R05 Fire Protection

1R05.1 Routine Resident Inspector Tours

	D. C. Cook Nuclear Plant UFSAR, Section 9.8.1, "Fire Protection System"	Revision 18
	D. C. Cook Nuclear Plant Fire Hazards Analysis, Units 1 and 2	Revision 9
	D. C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook	February 1995
	D. C. Cook Nuclear Plant Administrative Technical Requirements Manual, Sections 1-FP-7 and 2-FP-7, "Fire Rated Assemblies"	Revision 25
PMP 2270.CCM.001	Control of Combustible Materials	Revision 1
PMP 2270.WBG.001	Welding, Burning and Grinding Activities	Revision 0
PMP 5020.RTM.001	Restraint of Transient Material	Revision 1
PMI 2270	Fire Protection	Revision 26
12-PPP-2270-066-001	Portable Fire Extinguisher Inspections	Revision 0
12-PPP-4030-066-021	Inspection of Fire Dampers Protecting Safety-Related Areas	Revision 1

Miscellaneous Condition Reports

CR 0221018	Fire Penetration Seal Configuration Control Is Weak and Design Information Is Lacking in Detail or Missing	August 9, 2002

CR 03064054 [1]	On March 4, 2003, During a Routine NRC Tour of the Auxiliary Building, Several Items Were Questioned and Identified by the External Organization	March 5, 2003

1R07 Biennial Review of Heat Sink Performance

	Order number 31760; MLW-Worthington CCW Heat Exchangers' Specification Sheets; dated February 15, 1967	February 15, 1967
CR 0128246	NRC identified that it is not appropriate to consider as fully functional tubes that could be cleaned with 105 psig air	October 9, 2001
CR 03085009[1]	NRC identified that it is not appropriate to consider as fully functional tubes that could be cleaned with 65 psig air	March 12, 2003

1R11 Licensed Operator Requalification

1R11.1 Resident Inspector Quarterly Review

	Licensed Operator Requalification Training Annual Simulator Evaluation Scenarios for March 19, 2003	

1R12 Maintenance Effectiveness

1R12.1 Resident Inspector Quarterly Review

PMP-5035-MRP-001	Maintenance Rule Program Administration	Revision 4
PMI-5035	Maintenance Rule Program	Revision 10

Radiation Monitoring for Reactor Coolant System (RCS) Leakage Detection

	Maintenance Rule (a)(1) Action Plan Radiation Monitoring System	Revision 5
	Maintenance Rule Scoping Document Radiation Monitoring System	July 19, 2002
01-OHP-4030-STP-016	Reactor Coolant System Leak Test	Revision 14a
Unit 1 Technical Specification (TS) 3/4.4.6.2	Reactor Coolant System Leakage: Operational Leakage	Amendment 215

Safety Injection System

	Maintenance Rule Scoping Document Emergency Core Cooling System	Revision 2
	Shift Manager's Logs	November 2, 2002 through November 3, 2002
Job Order 02089024	Investigate Oil Leak at Bottom of Sight Glass on 1-PP-26N	November 2, 2002
Job Order C0044857	Disassemble Sight Glass/Repair Oil Leak on 1-PP-26N	November 21, 1998
Job Order C0017484	Install New Sight Glass Assembly on 1-PP-26N	November 10, 1993
Job Order C0015584	Disassemble Sight Glass on 1-PP-26N and Repair	May 26, 1993
CR 01260026	Operability Determinations for Safety Related Equipment Oil Leaks Time Consuming and Inconsistent.	September 17, 2001
CR 02089024	Oil Leak on Level Indicator for Oil Reservoir.	March 30, 2002
CR 02307003	1-PP-26N Oil Reservoir Sight Glass Found to Have Wrong and Missing Parts.	November 3, 2002
CR 02307020	The North Safety Injection Pump Oil Sump Sight Glass Had a Leak from the Bottom. This Leak Was Already Documented but Increased in Leakrate.	November 3, 2002
CR 03016037	Found Open Coil on Time Delay Relay for North Safety Injection Pump.	January 16, 2003

Miscellaneous Condition Reports

CR 03017025	1-QRV-160 Will Not Close (Stuck 100% Open). Attempted to Remove Orifice from Operation per Procedure 01-OHP-4021-003-001 and Valve Failed to Move.	January 17, 2003

1R12.2 Periodic Maintenance Rule Evaluation

PMI-5035	Maintenance Rule Program	Revision 10

PMP-5035-MRP-001	Maintenance Rule Program Administration	Revision 4
12-EHP-5035-MRP-001	Maintenance Rule Program Administration	Revision 5
	D. C. Cook Nuclear Power Plant Periodic Assessment of Maintenance Effectiveness Report	October 30, 2001
	Maintenance Rule Scoping Document - Chemical Volume Control System	Revision 1 April 19, 2001
	Maintenance Rule Scoping Document - Emergency Core Cooling and Residual Heat Removal System	Revision 2 August 22, 2001
	Maintenance Rule Scoping Document - Containment Spray	Revision 2 February 14, 2002
	Maintenance Rule a(1) Consideration - 250 VDC Battery Charger	October 24, 2001
	Maintenance Rule a(1) Action Plan - 250 VDC Control Power Fuse Blocks	March 1, 2001
	Maintenance Rule a(1) Action Plan - 120V Vital/CRID Inverters System	January 25, 2001
	Maintenance Rule a(1) Action Plan - Auxiliary Building Ventilation System	January 26, 2001
	Maintenance Rule a(1) Action Plan - Chemical Volume Control System	February 21, 2001
	Maintenance Rule a(1) Action Plan Supplement - Chemical Volume Control System	Revision 0 November, 14, 2001
	Maintenance Rule a(1) Action Plan Supplement - Chemical Volume Control System	Revision 1 April 10, 2002
	Maintenance Rule a(1) Action Plan [(a)(1) Consideration] - Chemical Volume Control System	July 19, 2002
	Maintenance Rule a(1) Action Plan - Chemical Volume Control System	Revision 1 July 30, 2002

	Maintenance Rule a(1) Action Plan Supplement - Chemical Volume Control System	Revision 3 September 30, 2002
	Maintenance Rule a(1) Action Plan - Unit 2 Containment Spray Train 'A'	May 23, 2001
	Maintenance Rule a(1) Action Plan - 250 VDC Battery Charger	January 26, 2002
	Maintenance Rule a(1) Action Plan - Unit 2 Containment Spray Train 'A'	January 30, 2002
	System Health Report - Containment Spray	Third Quarter 2002
	System Health Report - Chemical Volume Control System	Third Quarter 2002
	System Health Report - Emergency Core Cooling and Residual Heat Removal	Third Quarter 2002
CR 0010503	Ice Buildup on Intermediate Deck Door 6C Caused Failure of As-found Force Test.	July 26, 2000
CR 0011002	Ice Buildup on Intermediate Deck Door 6C to Become Inoperable.	August 7, 2000
CR 00341075	Unit 1 Centrifugal Charging Pump Auxiliary Lube Oil Pump Needs to Be Run While the East Centrifugal Charging Pump Is Running.	December 6, 2000
CR 00345029	Breaker for Unit 1 Refueling Water Storage Tank Heat Trace Has Tripped Twice.	December 10, 2000
CR 01023004	West Main Feedwater Pump Operating Device Not Working.	January 23, 2001
CR 00312088	Failure of 1-HV-AES-2 Unit to Pass Charcoal Filter Leak Test.	January 26, 2001
CR 01047033	Perform Self-assessment SA-2001-ENP-018, Maintenance Rule Implementation.	February 16, 2001
CR 01109112	Maintenance Rule Evaluation Screen Was Not Turned On for CR 01012036.	April 19, 2001

CR 01116074	Component Cooling Water Flow from North Safety Injection Pump Was at 23.75 gallons-per-minute.	April 26, 2001
CR 01198002	1-QRV-251 Was 100% Open and Pressurizer Level Was Rising Above Setpoint.	July 17, 2001
CR 01311007	Disc on Valve 2-CS-342 Was Found in the Open Position During Nonintrusive Check Valve Examination.	November 7, 2001

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

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

Unit 1 Rod Control System

	Shift Manager's Logs	March 21, 2003 through March 22, 2003
PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 45, Week 2, with Revisions	March 16, 2003 through March 22, 2003
	D. C. Cook Nuclear Plant Unit 1 Technical Specifications	

Unit 2 Containment Lower Airlock Door

	Shift Manager's Logs	March 25, 2003 through March 26, 2003
PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 45, Week 3, with Revisions	March 23, 2003 through March 29, 2003
	D. C. Cook Nuclear Plant Unit 2 Technical Specifications	

12-EHP-4030-046-204	Unit 1 and Unit 2 Personnel Airlock Leak Rate Surveillance	Revision 0
CR 03084046	Unable to Open Inner Door of Unit 2 612 Foot Elevation Airlock.	March 25, 2003

Unit 1 Turbine Driven Auxiliary Feedwater Pump (TDAFWP)

	Shift Manager's Logs	March 12, 2003 through March 14, 2003
PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 45, Week 1, with Revisions	March 9, 2003 through March 15, 2003
	D. C. Cook Nuclear Plant Unit 1 Technical Specifications	

Unit 2 AB EDG

	Shift Manager's Logs	March 6, 2003 through March 7, 2003
PMP-2291-OLR-001 Data Sheet 1	On-Line Risk Management Work Schedule Review and Approval Form Cycle 44, Week 12, with Revisions	March 2, 2003 through March 8, 2003
	D. C. Cook Nuclear Plant Unit 2 Technical Specifications	

Miscellaneous Condition Reports

CR 03023012	Tracking Condition Report Is Requested for Implementation of Two Regulatory Commitments Cook Plant Response to NRC Conditions and Limitations on the Reactor Trip System and Engineered Safety Feature Actuation System.	January 23, 2003

1R14 Personnel Performance During Non-routine Plant Evolutions

1R14.1 Unit 1 Main Power Transformer Fire (Unusual Event) and Reactor Trip

Event Notification 39513	Event Notification Worksheet	January 15, 2003

	Shift Manager's Logs	January 15, 2003
PMP 4010-TRP-001	Reactor Trip Review	January 16, 2003
CR 03016003	Unit 1 Main Steam Stop Valves Drifted in the Closed Direction When Unit 1 Tripped.	January 16, 2003
CR 03016007	Unit 1 Reactor Trip Due to Fire in Main Transformer.	January 16, 2003
CR 03016032	Unit 1 Oscillograph Failed to Function Following Automatic Reactor/Turbine Trip Due to Main Transformer Fault.	January 16, 2003

1R14.2 Unit 2 Reactor Trip (Control Group Power Supply Failures)

Event Notification 39564	Event Notification Worksheet	February 5, 2003
	Shift Manager's Logs	February 5, 2003
PMP 4010-TRP-001	Reactor Trip Review	February 6, 2003
CR 03037028	Excessive Unit 2 RCS Cooldown After Trip from 100 Percent Power	February 6, 2003

1R15 Operability Evaluations

	D. C. Cook Nuclear Plant Unit 1 and 2 Technical Specifications	
	D. C. Cook Nuclear Plant UFSAR	Revision 18
Generic Letter 91-18	Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions	Revision 1
OHI-4016	Conduct of Operations Guidelines	Revision 4
PMP-7030-ORP-001	Operability Determinations	Revision 9
02-OHP 4023.FR-S.1	Response to Nuclear Power Generation Anticipated Transient Without Scram	Revision 9
CR 00339065	Action Pack Reset Values to Far from Trip Point.	December 4, 2000

CR 02318061	Request Licensing Perform Bases Review of TS 3.4.1.1, Reactor Cooling System Relief Valves Concerning Power-Operated Relief Valve Operability Requirements for Procedure Enhancement.	November 14, 2002
CR 02344016	EDG Fuel Oil Receipt Inspection Samples Collected During Two Fuel Deliveries Made on November 5, 2002 and November 6, 2002 Failed to Be Analyzed by the Vendor Testing Laboratory Within the TS Required 31-Day Window.	December 10, 2002
CR 03009038	While Performing 2-EHP-4030-228-228B 2-HV-AES-1 Control Switch Was Placed in Stop While 2-HV-AES-2 Was Previously Inoperable Making Both Auxiliary Building Engineered Safety Features Fans Inoperable.	January 9, 2003
CR 03011014	2-QFI-420 Output Is Intermittently Spiking High.	January 11, 2003
CR 03011017	The Pump Coupling Is Disengaged From the Motor on 2-QT-117-AB.	January 11, 2003
CR 03011027	Output Frequency of 2-CRID-4-INV Is High at 60.4 Hertz and the "In-Sync" Light Is Not Lit.	January 11, 2003
CR 03013023	Unit 2 Containment Hydrogen Recombiner Number 2 Maintenance Was Scheduled During the Unit 1 East ESW [Essential Service Water] Pump Replacement.	January 13, 2003
CR 03014006	Auxiliary Building and Turbine Building Is Eroding Away at the Intersecting Joint.	January 14, 2003
CR 03019009	2-CRID-1-INV, 120VAC Control Room Instrument Distribution System Channel Inverter Was Found With the In-Sync Light Not Lit.	January 19, 2003
CR 03019017	Plant Process Computer Point for 2-QFI-420 Is Programmed Improperly.	January 19, 2003
CR 03023004	1-RH-128W Was Found to Be in the Closed Position.	January 23, 2003

CR 03026040	Received the Control Room Instrumentation Distribution 2 Inverter Abnormal Alarm. In-Sync Light Is Not Lit. The Isolimiter Transformer Has Failed.	January 26, 2003
CR 03028005	Investigate Cause for the Rise in Required Containment Pressure Reliefs.	January 28, 2003
CR 03028032	Horizontal Missile Blocks Over Unit 1 Reactor Head Found Numerous Hold Down Nuts Loose.	January 28, 2003
CR 03030003	Unexplained Loss of Reactor Coolant System.	January 39, 2003
CR 03040142	2-TR21PHC, 480V 2-21PHC Supply Transformer, Has Acid Odor, Is Running Warm, and Has a Humming Noise.	February 10, 2003
CR 03041011	Refueling Water Storage Tank Vent and Over Flow Pipe Heat Trace.	February 10, 2003
CR 03044042	Motor Very Dirty and Packed with Old Dried up Grease.	February 13, 2003
CR 03048041	Technical Specifications Interpreted Differently for Two Similar Situations.	February 17, 2003
CR 03064040	The Three Reoccurring Tasks Could not Be Done in Week Zero Due to Inability to Keep CD Battery Room Temperatures Above the Required Temperature. This Was Due to the Fact that 2-HV-EH-20-3 Heater is Broke.	March 5, 2003
CR 03076036	1-HV-ACR-DA-2 Did Not Reposition to the Part Open Position When the Local Control Switch on 1-ACRA-2 Was Taken to the Part Open Position.	March 17, 2003
CR 03078001	Operations Prompt Operability Determination on eSAT [Electronic Single Action Tracking] 03076036 Concerning the Unit 1 Control Room Ventilation Damper 1-HV-ACR-DA-2 Although Correct, Lacked Sufficient Justification.	March 18, 2003

1R19 Post Maintenance Testing

Unit 1 TDAFWP Maintenance

01-OHP-4030-STP-017T	Turbine Driven Auxiliary Feedwater System Test	Revision 15a
01-OHP-4030-STP-017T Attachment 1	Turbine Driven Auxiliary Feedwater System Test	Revision 15a
Job Order R0090119-02	Perform Full Preventive Maintenance at 1-FMO-221	
Job Order R0225215-02	1-FMO-241-ACT, Perform MOV Preventive Maintenance	
Job Order R0238995-01	1-OHP-4030-STP-017T, "Turbine Driven Auxiliary Feedwater System Test Surveillance"	

Unit 2 AB EDG

Job Order 02177034-02	2-QT-100-AB, Investigate/Repair Air Intake Filter
Job Order 01270050-03	Repair Oil Leak at Crankcase Breathers 2-QT-140-AB
Job Order 01261028-02	2-QT-141-AB1, Repair Tank Interior Coating
Job Order 00356039-02	2-QT-117-AB. Repair Mechanical Seal Oil Leak
Job Order 00356039-03	2-QT-117-AB. Repair Mechanical Seal Oil Leak
Job Order 02240002-02	2-PP-120-AB-CLPG Replace Rubber Spider (Insert)
Job Order R0240645-02	STP027AB Diesel Generator 2AB Slow Start
Job Order R0080221-01	Calibrate Pressure Switch 2-LPS-120
Job Order 01263008-02	2-PRV-1-AB, Replace Tubing and Fittings
Job Order R0089621-01	Calibrate Pressure Indicator 2-XPI-211
Job Order 02156043-01	2-SV-78-AB1 Return Tail Piece to Design Configuration

Job Order 01263066-03	2-OME-150-AB, Repair #2FB and #4FB Return Fuel Oil Hose Rub	
Job Order 01263033-02	2-OME-150-AB, Repair #2FB and #4FB Return Fuel Oil Hose Rub	
Job Order R0099729-02	Calibrate Time Delay Relays for AB Diesel	
Job Order R0099729-03	Calibrate Time Delay Relays for AB Diesel	
Job Order R0097773-02	EDG Volt-Reg Cleaning and PM of Circuit Boards	
Job Order R0221279-02	2-WMO-722-ACT, Perform External Preventive Maintenance	
Job Order 02240002-03	2-PP-120-AB-CLPG Replace Rubber Spider (Insert)	

Miscellaneous Condition Reports

CR 02350023	Refueling Water Storage Tank Vent Pipe Heat Trace Temporary Modifications Has Been in Place to Long. The Vent Piping Needs this Heat Tracing to be a Permanent Configuration.	December 16,2002
CR 03025026	Whiled Performing Megger Test of Cables for 1-TR-MAIN Found That There Were Clearance Red Tags on All Leads for Cables 1-31400-R-1 and 1-31402-R-1.	January 25, 2003
CR 03039027	Unit 1 West ESW Pump Failed its Surveillance Test (01-OHP-4030-119-022W) Due to Low Pump Differential Pressure.	February 8, 2003

1R20 Refueling and Outage Activities

1R20.1 Unit 1 Forced Outage

	D. C. Cook Nuclear Plant Unit 1 Technical Specifications	
	D. C. Cook Nuclear Plant UFSAR	Revision 18

01-OHP-4021-001-004	Plant Cooldown From Hot Standby to Cold Shutdown	Revision 37a
01 OHP 4021-017-002	Placing In Service The Residual Heat Removal System	Revision 16
01-OHP-4021-001-002	Reactor Startup	Revision 27a
01-OHP-4030-114-030	Daily and Shiftly Surveillance Checks	Revision 2
PMP 4100-SDR-001	Plant Shutdown Safety and Risk Management	Revision 5
	Shift Manager's Logs	January 15, 2003 through February 5, 2003

<u>1R20.2</u> <u>Unit 2 Forced Outage</u>

	D. C. Cook Nuclear Plant Unit 2 Technical Specifications	
	D. C. Cook Nuclear Plant UFSAR	Revision 18
02-OHP-4021-001-002	Reactor Startup	Revision 22a
PMP-2291-TRS-001	Troubleshooting	Revision 1a
PMP-2291-OLR-001	On-Line Risk Management	Revision 3
NUMARC 93-01	Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Section 11, "Assessment of Risk Resulting From Performance of Maintenance Activities"	Revision 2
	Shift Manager's Logs	February 5, 2003 through February 15, 2003

<u>1R22</u> <u>Surveillance Testing</u>

<u>Reactor Coolant System Leak Test</u>

01-OHP-4030-STP-016	Reactor Coolant System Leak Test	Revision 14a
Unit 1 Technical Specification 3/4.4.6.2	Reactor Coolant System Leakage: Operational Leakage	Amendment 215

Pressurizer Power Operated Relief Valve (PORV) Cold Over-pressurization Bistable and Backup Air Pressure System Functional Test

ASME/ANSI OMa-1988	Inservice Testing of Valves in Light-Water Reactor Power Plants	1988
OHI-4016	Conduct of Operations Guidelines	Revision 4
NRC Information Notice 89-32	Surveillance Testing of Low-Temperature Overpressure-Protection Systems	March 23, 1989
NRC Information Notice 89-32, Supplement 1	Surveillance Testing of Low-Temperature Overpressure-Protection Systems	February 12, 1991
01-IHP-4030-STP-089	Pressurizer PORV Cold Over-pressurization Bistable and Backup Air Pressure System Functional Test	Revision 5
01-OHP-4030-102-060	Pressurizer Relief Valve Testing	Revision 0
Engineering Programs Technical Data Book Figure 1-19.1	Power Operated Valve Stroke Time Limits	Revision 67
	Shift Manager's Logs	January 16, 2003
Job Order R0204016	2-NRV-153 Replace Actuator Diaphragm	February 8, 2002
Job Order R0229793	Perform 01-IHP-4030-STP-089	January 17, 2003
CR 02144038	When Valve Was Disassembled, It Was Discovered that the Old Valve Stem Was Shorter than the Approved Stem that Was to Be Installed.	May 23, 2002
CR 02147055	1-NRV-152 Failed Its Open Stroke Time.	May 27, 2002
CR 02212062	Pressurizer PORV 2-NRV-153 Stroked Too Fast in the Open Direction Using Backup and Normal Control Air During the Performance of 02-OHP-4030-202-060, "Pressurizer Relief Valve Testing." Retest Was Performed Satisfactorily.	July 31, 2002
CR 03016066	1-NRV-153, Pressurizer PORV #3, Stroked Too Fast During 01-OHP-4030-102-060 Testing 1-NRV-153.	January 16, 2003

CR 03062038 [1]	Based on Their Review of a Number of CRs, the Resident Team Determined the Cook Nuclear Power Was Not Meeting All the Requirements of the Corrective Action Program and the IST Program for Valve Testing.	March 3, 2003

<u>Containment Isolation and IST Valve Operability Test</u>

ASME/ANSI OMa-1988	Inservice Testing of Valves in Light-Water Reactor Power Plants	1988
OHI-4016	Conduct of Operations Guidelines	Revision 4
01-OHP-4030-STP-011	Containment Isolation and IST Valve Operability Test, Attachment 4, "Post Accident Sample System Valves Test"	Revision 20
NRC Information Notice 97-16	Preconditioning of Plant Structures, Systems, and Components Before ASME Code Inservice Testing or Technical Specification Surveillance Testing	April 4, 1997
Engineering Programs Technical Data Book Figure 1-19.1	Power Operated Valve Stroke Time Limits	Revision 67
CR 03021006	Upon Performance of 01-OHP-4030-STP-011, Attachment 4, Step 4.4.3, the Physical Timing Open of Valves 1-ECR-11 and 1-ECR-21 Was Not Performed.	January 21, 2003
CR 03062038 [1]	Based on Their Review of a Number of CRs, the Resident Team Determined the Cook Nuclear Power Was Not Meeting All the Requirements of the Corrective Action Program and the IST Program for Valve Testing.	March 3, 2003
CR 03070029 [1]	Procedure Enhancement Is Needed for OHI-4016, "Conduct of Operations Guidelines," Attachment 3, IST Test Criteria, to Provide Guidance on Preventing Preconditioning.	March 11, 2003

<u>Steam Generator PORV Operability Test</u>

ASME/ANSI OMa-1988	Inservice Testing of Valves in Light-Water Reactor Power Plants	1988
OHI-4016	Conduct of Operations Guidelines	Revision 4
02-OHP-4030-214-049	Hot Shutdown Panel Operability Test, Attachment 14, "Steam Generator Power Operated Relief Valve Operability Test"	Revision 1
Engineering Programs Technical Data Book Figure 2-19.1	Power Operated Valve Stroke Time Limits	Revision 56
OP-2-5120R-4	Control Air System - Auxiliary Building Tapoffs - Unit 2	Revision 4
CR 01334058	2-MRV-233 Failed to Fully Open	November 30, 2001
CR 02037011	2-MRV-233 Stroked Faster Than its IST Stroke Time of 81.2 Seconds. As Found Stroke Was 76.55 Seconds and the Confirmatory OHI-4016 Retest Was 79.88 Seconds.	May 17, 2002
CR 02186022	Procedure Non-compliance on Call Made on CR 02137011 for IST Stroke Time Failure of 2-MRV-233 Steam Generator PORV.	July 5, 2002
CR 02305080	2-MRV-233 Stroked Faster Than Its IST Minimum Stroke Time of 81.2 Seconds. As Found Stroke Was 76.0 Seconds and the Confirmatory OHI-4016 Retest Was 81.5 Seconds.	November 1, 2002
CR 03058030 (1)	Review Again the Lack of a Periodic Functional Stroke of the SG PORVs Using the Reactor Nitrogen Backup System.	February 27, 2003
CR 03062038 (1)	Based on Their Review of a Number of CRs, the Resident Team Determined the Cook Nuclear Power Was Not Meeting All the Requirements of the Corrective Action Program and the IST Program for Valve Testing.	March 3, 2003

<u>Miscellaneous Condition Reports</u>

CR 01284013	The Pre-conditioning Statements Contained Within PMI-4030 and PMP-4030-BD-001 May Be More Limiting Than Required.	October 11, 2001
CR 02019055	Serious Consideration Needs to be Given to Differentiating Between Pressurizer PORV Air Supply Setup and IST Stroke Time Limitations, for Normal Operating Conditions, and for Shutdown Low Temperature Overpressure Protection Conditions.	January 19, 2002
CR 02147035	During Investigation of CR 02144038 Extent of Condition, It Was Discovered that the Wrong Valve Stem (M & E 30-042250) Was Used in Repair of 2-NRV-152 Under Job Order C0036209-04.	May 27, 2002
CR 02211018	2-NRV-153 Stem Nut Was Found to Be Loose.	July 30, 2002
CR 03017025	1-QRV-160 will not Close (Stuck 100% Open). Attempted to Remove Orifice from Operation per Procedure 01-OHP-4021-003-001 and Valve Failed to Move.	January 17, 2003
CR 03022005	Control Valve 'B' Opened Approximately 4% (from 57% to 61%) Resulting in a Reactor Power Rise from 99.92% to 100.19% on the 10 Minute Average (1137 MW to 1140 MW). The Load Limiter Was Manipulated Several Times During the Transient with Little Effect.	January 22, 2003
CR 03023005	1-QMO-452 Closed Stroke Time Exceeded the Limit Which Makes it Inoperable.	January 23, 2003
CR 03028065	During the Performance of 2-IHP-4030-STP-180 Attachment 9 and 10, it Was Discovered that Both Manual Safety Injection Switches Would Be Made Inoperable by this Procedure.	January 28, 2003
CR 03033015	1-MRV-223 IST Retest	February 2, 2003
CR 03033028	1-FRV-240 SG# 14 Feed Control Valved Failed to Close Within its IST MAX Time With an As-Found Time of 6.72 Seconds for Only Train 'B' (1-RPSX-B Fuse 7).	February 2, 2003

CR 03023005	1-QMO-452 Closed Stroke Time Exceeded the Limit Which Makes it Inoperable.	January 23, 2003
CR 03048069	Definition of Staggered Test Basis for Listed Components Has Not Been Literally Completed with Required Interval Was Not Tested in Subinterval Time-Requirements per Definition 1.20 of Technical Specifications.	February 17, 2003

4OA2 Identification and Resolution of Problems

4OA2.1 Recurring Failures to Correctly Address Inoperability of 1-QRV-112 (Normal RCS Letdown Train 'B' Isolation Valve) Following Stroke Test Failures

ASME/ANSI OMa-1988	Inservice Testing of Valves in Light-Water Reactor Power Plants	1988
OHI-4016	Conduct of Operations Guidelines	Revision 4
	Maintenance Rule Scoping Document Chemical Volume Control System	Revision 2
NRC Information Notice 84-23	Results of the NRC-Sponsored Qualification Methodology Research Test on ASCO Solenoid Valves	April 5, 1984
Job Order 03018019	Replace Solenoid 1-XSO-502 Valve Body	January 24, 2003
Job Order 03024030	1-QRV-112 Failed Stroke Time. 1-AV-23 Is Suspect.	January 25, 2003
CR 02067033	On April 28, 2001, 1-QRV-112 Indicated Intermediate Position When Stroked in the Closed Direction. On September 3, 2001, Actual Travel Closed Was Measured to Be 50% of Valve Stroke.	March 8, 2002
CR 03018019	Solenoid 1-XSO-502 for 1-QRV-112 Is Continuously Venting Control Air with 1-QRV-112 Closed.	January 18, 2003
CR 03020045	1-QRV-112 Failed IST Valve Testing. Closed Too Slowly.	January 20, 2003
CR 03024030	1-QRV-112, Reactor Coolant Normal Letdown Train 'B' Shutoff Valve Stroked Closed Too Quickly When Performing IST Timing Following Valve Repair.	January 24, 2003

CR 03031047	Incorrect Operability Call Made on CR 0301819 for Valve 1-QRV-112. Screening Also Failed to Recognize Program Impact Resulting from the Leaking Solenoid Valve.	January 31, 2003
CR 03062038 [(1)]	Based on Their Review of a Number of CRs, the Resident Team Determined the Cook Nuclear Power Was Not Meeting All the Requirements of the Corrective Action Program and the IST Program for Valve Testing.	March 3, 2003
CR 03090074 [(1)]	Two Examples of Cook Nuclear Power's Failure to Adequately Assess Component Operability.	March 31, 2003

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