January 23, 2002

EA-02-010

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

SUBJECT: D. C. COOK NUCLEAR POWER PLANT, UNITS 1 AND 2
NRC INSPECTION REPORT 50-315/01-19(DRP); 50-316/01-19(DRP)

Dear Mr. Bakken:

On December 29, 2001, 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 December 28, 2001 and January 23, 2002 with Mr. Joseph Pollock and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified two issues of very low safety significance (Green) and one No Color finding which 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. The inspectors also identified one Green finding associated with the human performance cross-cutting area. 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.

Immediately following the terrorist attacks on the World Trade Center and the Pentagon, the NRC issued an advisory recommending that nuclear power plant licensees go to the highest level of security, and all promptly did so. With continued uncertainty about the possibility of additional terrorist activities, the Nation's nuclear power plants remain at the highest level of security and the NRC continues to monitor the situation. This advisory was followed by additional advisories over the coming weeks, and although the specific actions are not releasable to the public, they generally include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with law enforcement and military authorities, and more limited access of personnel and vehicles to the sites. The NRC has conducted various audits of your response to these advisories and your ability to respond to terrorist attacks with the capabilities of the current design basis threat (DBT). From these audits, the NRC has concluded that your security programs are adequate at this time.

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).

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 50-315/01-19(DRP);
50-316/01-19(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:	50-315/01-19(DRP); 50-316/01-19(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:	November 18, 2001 through December 29, 2001
Inspectors:	J. Maynen, Acting Senior Resident Inspector K. Coyne, Resident Inspector Z. Falevits, Senior Reactor Inspector B. Kemker, Resident Inspector - Byron R. Krsek, Resident Inspector - Palisades J. Lennartz, Senior Resident Inspector - Palisades T. Madeda, Physical Security Inspector D. Passehl, Senior Project Engineer W. Slawinski, Senior Radiation Specialist R. Schmitt, Radiation Specialist R. Winter, Reactor Engineer
Approved by:	A. Vegel, Chief Branch 6 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000315-01-19(DRP), IR 05000316-01-19(DRP), on 11/18-12/29/2001; Indiana Michigan Power Company, D. C. Cook Nuclear Power Plant, Units 1 and 2. Fire Protection, Radiological Environmental and Radioactive Material Control Program, Performance Indicator Verification, Identification and Resolution of Problems, Cross-Cutting Issues.

This report covers a 6-week routine inspection. The inspection was conducted by resident and Region III inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described 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

Cornerstone: Mitigating Systems

- No Color. The inspectors identified a Non-Cited Violation (NCV) for failure to ensure that coordination and selective tripping was provided in accordance with the Safe Shutdown Capability Assessment. The Current Transformers (CT) for protective relaying at the 4.16 kV level were undersized and could reach saturation conditions if a bolted fault were to occur on the associated cabling. This condition could result in inadvertent tripping of 4.16 kV circuit breakers supplying safe shutdown equipment. The failure to ensure coordination and selective tripping is a violation of the D. C. Cook Operating license Section 2.C.(4) for Unit 1 and Section 2.C.(3)(0) for Unit 2.

  The finding was determined to be No Color because the finding was not suitable for SDP evaluation because it did not involve the impairment or degradation of a fire protection feature. Because the finding was of very low safety significance and the finding was captured in the licensee's corrective action system, this finding is being treated as a NCV consistent with Section VI.A1 of the NRC Enforcement Policy (Section 1R05).

- TBD. The inspectors identified an apparent violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to include appropriate quantitative acceptance criteria in maintenance procedure 12-MHP 5021.056.007, "Auxiliary Feed Pump Trip and Throttle Valve Linkage Adjustment," Revision 2. Specifically, the procedure specified a trip throttle valve contact alignment criteria that was less conservative than the contact alignment specified in the vendor's turbine driven auxiliary feedwater pump (TDAFWP) trip throttle valve test instructions. Alignment of the trip throttle valve using a less conservative contact acceptance criteria could result in less latch engagement than required by a surface contact area acceptance criteria and a greater potential for inadvertent disengagement of the trip throttle latching mechanism. On June 14, 2000, the Unit 2 TDAFWP trip throttle valve was adjusted in accordance with 12-MHP 5021.056.007. Subsequently, on August 10, 2001, the Unit 2 TDAFWP trip throttle valve failed to adequately engage during three successive start attempts. The licensee determined that the apparent cause of the August 2001 failure was insufficient engagement of the trip throttle valve latching mechanism.

The staff's significance determination of this finding was not complete at the time of issuance of this report; therefore, this issue is considered an unresolved item. The safety significance of this issue has been characterized as "To Be Determined (TBD)" pending the completion of additional risk analysis. (Section 4OA1)

- Green. The inspectors determined that the licensee failed to address a design deficiency on the Unit 1 and the Unit 2 safety-related 4.16 kV circuit breakers in a timely manner. This design deficiency could result in exceeding the 4.16 kV circuit breaker's momentary interrupting rating capability during a 3-phase bolted fault condition. This concern was initially noted by the licensee in 1988, was identified again by the NRC during a Safety System Functional Inspection in 1990, and during an Electrical Distribution Safety Functional Inspection in 1992. The failure to properly evaluate and correct this degraded condition is a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI.

  The inspectors evaluated the risk significance of this issue using the Significance Determination Process. Because no actual loss of safety function occurred, the low probability of failure, and system redundancy, this issue screened as Green (very low risk significance) after a Phase 1 Significance Determination Process review. (Section 4A02)

Cornerstone: Public Radiation Safety

- Green. A Non-Cited Violation of Technical Specification 6.8 was identified for the failure to meet Offsite Dose Calculation Manual (ODCM) required radioanalytical detection capabilities for some environmental samples collected during the third and fourth quarters of 2000, and the first quarter of 2001. This finding included a cross-cutting element as a contributing factor related to the timeliness of the licensee's corrective actions, since the sample analytical problems were known but not effectively corrected for an extended period.

  Although the licensee's ability to evaluate the environmental impact from some exposure pathways was impaired, this finding was determined to be of very low safety significance because the majority of sample analyses satisfied detection requirements to enable the overall impact on the environment from actual plant effluents to be assessed. (Section 2PS3)

Cross-Cutting Issues: Human Performance

- Green. The inspectors identified a Finding of very low safety significance associated with recent licensee human performance weaknesses. Specifically, two licensee identified violations of NRC requirements occurred during this period which indicated weaknesses in the human performance cross-cutting area. The violations involved inadequate control of the impact energy of loads carried over the spent fuel pool contrary to Technical Specification requirements and the failure to adequately align the Unit 1 "B" Train diesel generator (D/G) voltage regulator for standby service. The human performance aspects of these issues are related to failures to follow procedural guidance, inadequate self checking, and the failure to perform adequate independent verifications.

  The inspectors assessed the safety significance of this issue using the Significance Determination Process (SDP). The inspectors concluded that these human performance weaknesses had a credible impact on safety and could become a more significant safety concern if left uncorrected. Specifically, the failure to limit the impact energy of loads carried over spent fuel could result in fuel barrier damage greater than assumed in the safety analysis following a postulated crane failure. The inspectors determined that the failure to adequately control impact energy was associated with the fuel barrier; therefore, this issue was determined to be of very low safety significance following a Phase 1 SDP. Additionally, the failure to align the diesel generator voltage regulation system for standby service could result in the failure of the diesel generator to adequately provide power to supported equipment. The inspectors determined that, based on the as-found voltage regulator settings, the Unit 1 "B" Train D/G would have been able to perform its associated safety function. Because the failure to adequately align the Unit 1 "B" Train D/G did not result in an actual loss of safety function, this issue was also determined to be of very low safety significance. Therefore, the inspectors concluded that these human performance weaknesses constituted a finding of very low risk significance based on the safety significance of the resultant issues and their impact to the cornerstones of reactor safety. (Section 4OA4)

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

Violations of very low safety significance, which had been identified by the licensee, were reviewed by the inspectors. Corrective actions taken or planned by the licensee are reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status:

Unit 1 and Unit 2 both began the inspection period at full power. Both units operated at or near full power throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's winterization program in preparation for the cold weather season. The inspectors walked down the screenhouse area of the plant, which houses the essential service water (ESW) system pumps, and the main steam valve enclosures, which house the main steam isolation valves and the main steam safety valves. In addition, the inspectors walked down the safety-related spare parts storage areas. The inspectors verified the design features and implementation of the licensee's procedures protected these systems and components from cold weather effects. The inspectors also reviewed a selection of previous condition reports (CRs) regarding winterization to verify that conditions adverse to quality were properly addressed.

b. Findings

No findings of significance were identified.

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of Evaluations and Screenings for Changes, Tests, or Experiments

a. Inspection Scope

The inspector reviewed eleven full evaluations performed pursuant to Federal Regulations 10 CFR 50.59. The full evaluations were related to temporary and permanent plant modifications, set-point changes, procedure changes, potential conditions adverse to quality, and changes to the licensee's updated safety analysis report. The inspector confirmed that the full evaluations were thorough and that prior NRC approval was obtained when appropriate. The inspector also reviewed eleven screenings, where the licensee had determined that a 10 CFR 50.59 full evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 full evaluation was performed, the inspector reviewed the changes to verify that they did not meet the threshold requiring a 10 CFR 50.59 full evaluation. These 10 CFR 50.59 evaluations and screenings were chosen based on risk significance of samples from the different cornerstones.

b. Findings

No findings of significance were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed the licensee's Condition Reports concerning 10 CFR 50.59 evaluations and screenings to verify that the licensee had an appropriate threshold for identifying issues. The inspector evaluated the effectiveness of the corrective actions for the identified issues.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Equipment Walkdowns

a. Inspection Scope

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

Mitigating Systems Cornerstone

- Alignment of Unit 1 "A" Train emergency diesel generator (D/G) for standby service

Barrier Integrity Cornerstone

- Placing Unit 2 "A" Train containment spray system in standby readiness

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Fire Protection Safe Shutdown Analysis

a. Inspection Scope

The Fire Protection Safe Shutdown Analysis (SSA) for D. C. Cook assumes that coordination and selective tripping is provided for all circuits on the emergency power system. The inspectors examined the licensee's existing coordination design against the assumptions made in the SSA.

b. Findings

The inspectors identified a Non-Cited Violation for failure to ensure that coordination and selective tripping was provided. The existing current transformers (CTs) are undersized and are not suitable for their present application. The licensee documented in CR 00-9424, dated June 29, 2000, that under certain severe conditions, the CTs that feed the phase instantaneous current (PJC) relays may saturate and impact the timing of the PJC relays. The licensee stated that spurious tripping of safety-related equipment due to this phenomenon was highly unlikely since the instantaneous units were set at a high value (1.75 percent locked rotor amps) such that sufficient margin was provided to account for any error introduced by CT saturation effects. However, relay coordination problems were introduced by the identification of the CT saturation. In response the licensee stated, "Due to the potential for CT saturation, a postulated bolted fault may not result in a trip of the circuit breaker nearest the fault." The inspectors noted that this condition could result in the inadvertent trip of an entire 4.16 kV bus due to a load fault. If the downstream breaker were properly coordinated, only the affected load would be tripped. Since redundant trains exist, the loss of a single 4.16 kV bus is already bounded by the existing D. C. Cook Safety Analysis.

The SSA for D. C. Cook assumes "that coordination and selective tripping is provided for all circuits on the emergency power system." The licensee has recognized that these CT saturation concerns present a condition that "is inconsistent with the coordination assumptions in the SSA." Following NRC questioning, the licensee issued Condition Report (CR) 01208057 to evaluate and address this non-conformance. While a CR was written and the licensee plans to study the issue, no action plan appears to exist for completion and resolution. In the interim, the licensee has determined that the worst-case situation (i.e., a single fire induces severe faults in both trains of redundant 4.16 kV motors and results in loss of both trains of electrical power in the fire affected unit) is bounded by the analysis for a fire in the 4.16 kV switchgear room. The licensee informed the NRC that the issue has been addressed to ensure that the plant can be safely shut down.

Operating License Section 2.C.(4) for Unit 1 Docket No. 50-315, Operating License Number DPR-58, and Operating License Section 2.C.(3)(o) for Unit 2 Docket No. 50-316, Operating License Number DPR-74, requires D. C. Cook plant "to implement and maintain, in effect, all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility." UFSAR Section 9.8.1 incorporates the Safe Shutdown Capability Assessment (SSCA) by reference. SSCA Section 2.7.2 states, "The electrical distribution system was reviewed to ensure that coordination and selective tripping is provided for all circuits on the emergency power system." It further states that "a fuse/circuit breaker coordination study and a multiple high impedance fault study are maintained and reviewed for design changes to assure coordination and to remove this potential for functional loss of safe shutdown components." Contrary to the SSCA, undersized CTs could result in inadvertent tripping of 4.16 kV circuit breakers. This is considered a violation of the D. C. Cook Operating License. This violation is not suitable for SDP evaluation because it did not involve the impairment or degradation of a fire protection feature, and is therefore considered a No Color finding. Because the licensee entered the finding into the corrective action program as CR 01208057, this violation is being treated as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy (NCV 50-315/01-19-01(DRP); 50-316/01-19-01(DRP)).

.2 Routine Fire Zone Tours

a. Inspection Scope

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

Mitigating Systems Cornerstone

- Unit 1 Quadrant 1 Cable Tunnel (Fire Zone 7)
- Auxiliary Building - Elevation 650' (Fire Zone 69)
- Unit 1 Turbine Room - Elevation 609' (Fire Zones 91, 92, 93, 94)
- Security Diesel Generator and Switchgear Room

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

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On November 20, 2001, the inspectors observed Operations "C" Shift during licensed operator training. The training consisted of an evaluated simulator scenario that required the operators to respond to and mitigate a steam generator tube rupture event concurrent with a loss of reserve power. The training scenario also required the licensed operators to implement the emergency plan. The inspectors verified that the training was effective and assessed the operator's ability to mitigate the event and to implement the emergency plan. The inspectors observed the post-scenario critique of operator performance to assess the licensee evaluators' ability to identify and assess operator performance deficiencies.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

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

Mitigating Systems Cornerstone

- Annunciator System
- Reactor Protection System
- Emergency Diesel Generators

Initiating Events Cornerstone

- Compressed Air System

b. Findings

No findings of significance were identified.

1R13 Maintenance Planning and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed the risk assessment and risk management for the following risk significant maintenance activities:

Mitigating Systems Cornerstone

- Unit 1 "A" Train motor driven auxiliary feedwater pump maintenance, November 21, 2001
- Installation of design change on Unit 1 "A" Train containment spray heat exchanger ESW outlet valve, December 1, 2001

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

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following operability determinations and evaluations affecting the reactor safety cornerstones to determine whether operability was properly justified and that no unrecognized risk increase had occurred.

Mitigating Systems Cornerstone

- CR 01347067 Internal degradation found on cells of Unit 2 station battery 2-BATT-AB during performance of surveillance
- CR 01332066 Operability of the Unit 1 accumulator level instrument, 1-ILA-111

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

.1 Review of the Cumulative Effect of Operator Workarounds (Unit 1)

a. Inspection Scope

The inspectors reviewed the cumulative effect of Operator Workarounds, control room deficiencies, and degraded conditions on equipment availability, initiating event frequency, and the ability of the operators to implement abnormal or emergency operating procedures.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

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

Mitigating Systems Cornerstone

- 1-DCP-744 Replace Unit 1 D/G high pressure fuel lines
- 2-DCP-526 Replace Unit 2 D/G high pressure fuel lines
- 1-DCP-5173 Provide ESW minimum flow path via Unit 1 containment spray heat exchanger
- 2-DCP-5174 Provide ESW minimum flow path via Unit 2 containment spray heat exchanger

The inspectors verified the design adequacy of the modifications and focused the inspection activities on the following parameters associated with the design changes: heat removal, control signals, equipment protection, operations, flowpaths, process media, licensing basis, and failure modes.

Completed activities associated with the implementation of the modification were also inspected and the inspectors discussed the modifications with the responsible engineers, and operations staff. In addition, the inspectors reviewed the applicable sections of the Technical Specifications, Updated Final Safety Analysis Report, and condition reports associated with the design change packages and installation of the modification.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

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

Mitigating Systems Cornerstone

- JO 01296060 Post modification wiring check for 2-DCP-5174, Alternate flow path for ESW
- JO 01320005 Install Temporary Modification 2-TM-00-54-R1 for accumulator level instruments in Unit 2
- JO 01341004 Replace failed undervoltage relay on 1-CD-BC2, Unit 1 “A” Train 250 VDC station battery charger
- JO 01355003 Replace failed control air regulating valve, 1-XRV-237, on Unit 1 “A” Train D/G

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

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

For each of the surveillance test procedures 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:

Mitigating Systems Cornerstone

- 01-IHP 4030.SMP.131, "Power Range Nuclear Instrumentation Functional Test and Calibration," Revision 0
- 01-OHP 4030.STP.018, "Steam Generator Stop Valve Dump Valve Surveillance Test," Revision 14
- 02-OHP 4030.STP.030, "Daily and Shiftly Surveillance Checks," Revision 38

Barrier Integrity Cornerstone

- 01-OHP 4030.STP.030, "Daily and Shiftly Surveillance Checks," Data Sheet 19, "Ice Condenser Tour Data Sheet," Revision 38

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

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

On December 18, 2001, Operations "B" Shift performed an emergency planning drill in conjunction with licensed operator training. The drill involved a steam generator tube leak and recovery actions. The inspectors reviewed the drill scenario, observed the licensed operators perform the drill in the simulator, and discussed the drill with members of the licensee's training staff.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Controls for Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiological Boundary Verifications

a. Inspection Scope

The inspectors conducted walkdowns of the radiologically protected area to verify the adequacy of radiological area boundaries and postings. Specifically, the inspectors walked-down numerous radiologically significant work area boundaries (high and locked high radiation areas) in the Unit 1 and 2 Auxiliary Buildings. Confirmatory radiation measurements were taken to verify that these areas and selected radiation areas were properly posted and controlled in accordance with 10 CFR Part 20, licensee procedures, and Technical Specifications. The inspectors also examined the radiological conditions of work areas within those radiation and high radiation areas walked-down, to assess the radiological housekeeping and contamination controls.

b. Findings

No findings of significance were identified.

.2 High Risk Significant, High Radiation Area, and Very High Radiation Area Access Controls

a. Inspection Scope

The inspectors reviewed the licensee's procedures and practices for the control of access to radiologically significant areas (high, locked high, and very high radiation areas) to verify compliance with Technical Specifications, procedures and the requirements of 10 CFR 20.1601 and 20.1602. Specifically, the inspectors evaluated the licensee's latest revisions to their procedures and the current practices for the control/inventory of keys to locked high radiation areas (LHRAs), and the licensee's methods for independently verifying proper closure and latching of LHRA doors upon area egress. Additionally, the inspectors reviewed radiological postings and challenged access control boundaries to determine if LHRAs and very high radiation areas were properly controlled.

b. Findings

No findings of significance were identified.

.3 Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed several radiation work permits (RWPs) for work in radiologically significant areas, including the RWPs for routine plant tours, removal of test coupons from the spent fuel pool, and for a dive into the fuel transfer canal. The RWPs were evaluated for protective clothing requirements and contamination controls. Electronic dosimeter alarm set points for both dose rate and integrated dose were evaluated to verify conformity with work area radiological conditions given the work activity and survey indications. The inspectors also reviewed work instructions specified in the RWPs, associated work packages, and pre-job briefing information in order to verify access control restrictions for compliance with Technical Specifications.

b. Findings

No findings of significance were identified.

.4 Review of Radiologically Significant Work

a. Inspection Scope

The inspectors monitored the following high exposure or high radiation area work activities performed during the inspection:

- Retrieval of test coupons from the Spent Fuel Pool
- Dive in the fuel transfer canal, to repair fuel handling equipment

The inspectors attended pre-job briefings for both of the aforementioned activities and evaluated the radiological job requirements for each. The inspectors also reviewed the licensee's procedure and practices for dosimetry placement, including the use of multiple dosimetry for work in high radiation areas having significant dose gradients, for compliance with the requirements of 10 CFR 20.1201 and applicable Regulatory Guides. The inspectors examined the as-low-as-is-reasonably-achievable (ALARA) plan for the work in the spent fuel pool to determine if it contained adequate information to safely control radiological work. The inspectors observed the work evolution to retrieve the test coupons and for the transfer canal dive to verify adherence to the ALARA plan. The inspectors reviewed those radiological surveys completed prior to and during the fuel pool work, and assessed the radiation protection job coverage and the overall work activities, to verify that the work was completed safely and consistent with work plans. The inspectors also reviewed completed surveys and applicable postings and barricades associated with this work to verify their adequacy.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors evaluated the licensee's calender year 2000-2001 condition report (CR) database and a variety of individual CRs relating to problems with access controls to radiologically significant areas, as well as radiation worker performance and work practices in or around those areas. The inspectors also reviewed Performance Assurance Department Assessment Report No. PA-01-014, "Radiation Protection," and several field observation reports, to verify the licensee's ability to identify and correct problems and to evaluate the effectiveness of the licensee's self-assessment process.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (71121.03)

.1 Operability and Testing of Post Accident Sampling System

a. Inspection Scope

The inspectors evaluated accident monitoring instrumentation associated with the Post Accident Sampling System (PASS) used for emergency plant assessment. The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and reviewed surveillance test records to verify that the system was capable of obtaining representative samples of the containment sump, containment atmosphere, and reactor coolant system. The inspectors reviewed the licensee's procedure for testing the PASS and reviewed surveillance records completed in 2000 and 2001, to verify that calibrations were conducted consistent with industry standards and in accordance with the station procedure. The inspectors performed a walkdown of the PASS to verify that equipment was in good material condition and reviewed training records for those station personnel qualified to operate the PASS.

b. Findings

No findings of significance were identified.

.2 Calibration of Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors examined the electronic dosimeters (EDs) maintained in the licensee's instrument calibration facilities and access control areas. The inspectors evaluated the EDs to verify that these instruments were source checked and had current calibrations consistent with station procedures and industry standards. The inspectors reviewed the EDs to verify that an adequate number of those instruments were designated "ready for use" were operable and were in good physical condition. The inspectors observed radiation protection staff source check and calibrate a number of EDs, to verify that those activities were completed using appropriate radiation sources. The inspectors also reviewed the calibration procedures and selected calendar year 2001 calibration records to verify that the ED instruments had been properly calibrated.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed calendar year 2001 CRs that addressed radiation instrument (i.e. PASS or EDs) deficiencies to determine if any significant radiological incidents involving radiation instruments had occurred. Additionally, these CRs were examined to verify the licensee's ability to identify repetitive problems, contributing causes and the extent of condition, and to implement corrective actions to achieve lasting results. The inspectors examined "closed" CR P-99-25781 and CR P-99-29165 related to prior deficiencies with some of the area and process radiation monitors, to verify that corrective actions taken by the licensee had adequately addressed UFSAR, Technical Specification and instrument drawing issues.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs (71122.03)

.1 Review of Radiological Environmental Monitoring Reports and Data

a. Inspection Scope

The inspectors reviewed the Annual Radiological Environmental Operating Reports for calendar year 1999 and 2000, and results of monthly radiological environmental monitoring analyses for the first half of 2001. The inspectors also reviewed the land use census for 2000 and 2001, results of the inter-laboratory comparison program for 1999 and 2000, and changes made to the Offsite Dose Calculation Manual (ODCM) in 2000 and 2001 relative to the environmental monitoring program. These reviews were conducted to verify that the radiological environmental monitoring program (REMP) was implemented as required by Technical Specifications and the ODCM, and that any changes did not affect the licensee's ability to monitor the impacts of radioactive effluents on the environment. Additionally, the inspectors evaluated the locations of the environmental monitoring stations and the type of samples collected as part of the REMP, to determine if they were consistent with the UFSAR and NRC guidance.

b. Findings

No findings of significance were identified.

.2 Walkdowns of Radiological Environmental Monitoring Stations and Meteorological Towers

a. Inspection Scope

The inspectors walked-down all six onsite environmental air sampling and thermoluminescent dosimeter (TLD) monitoring stations to determine whether they were located as described in the ODCM, and to assess equipment material condition and operability. The inspectors discussed tree growth in the vicinity of the air sampling stations with the REMP Coordinator, to verify that its potential impact on sample representativeness was recognized and to review those actions the licensee was contemplating to address this issue with the State of Michigan. Both the primary and backup meteorological towers were also walked-down by the inspectors, and data readouts in the control room were observed to verify Technical Specification required meteorological instruments were operable and that current meteorological conditions were available. In addition, the inspectors visited one of the two municipal drinking water sampling stations and discussed sampling practices with one of the sample collectors to determine if adequate methods were used to collect the sample and ensure its integrity.

b. Findings

No findings of significance were identified.

.3 Review of Radiological Environmental Monitoring Equipment Maintenance and Testing

a. Inspection Scope

The inspectors reviewed the most recent air sample pump calibration records and associated procedures, and meteorological tower equipment calibration and maintenance records for calendar year 2000 through October 2001, to verify that the maintenance and testing program for this equipment was implemented consistent with Technical Specifications and procedural requirements. The most recent calibration of the air sample pump rotameter standard used by the licensee was also reviewed to verify that its certification met industry standards and had traceability to the National Institute of Standards and Technology. The inspectors discussed air sample pump calibration and maintenance activities with an instrument technician and the REMP Coordinator to assess the adequacy of the calibration methods, and to review actions being considered for a routine preventative maintenance program for associated equipment.

b. Findings

No findings of significance were identified.

.4 Review of REMP Sample Collection and Analyses

a. Inspection Scope

The inspectors accompanied a REMP technician and observed the collection and change-out of air particulate filters and charcoal cartridges at each of the licensee's six onsite environmental stations, to determine whether samples were collected consistent with procedures and if good practices were used. The inspectors observed the technician complete air sample pump field checks upon sample change-out to determine whether the checks were conducted in accordance with procedure. The inspectors assessed the analytical detection capabilities of the contract laboratory used by the licensee to analyze its environmental samples, and reviewed licensee identified problems with the vendor's sample analyses. The inspectors' assessment was conducted to determine if the radiological environmental sample analysis program was implemented consistent with the ODCM, and to verify that the vendor was capable of making adequate radiological measurements.

b. Findings

The inspectors identified a Green Finding and an associated Non-Cited Violation concerning the failure to routinely meet ODCM required radioanalytical detection capabilities for a variety of environmental samples collected over an approximate 5 month period. The inspectors identified that the Finding also included a cross-cutting element in the area of problem resolution, because the licensee's actions to effectively correct known problems were not timely.

The licensee utilized the services of a vendor laboratory to analyze the environmental samples collected by its staff. A variety of samples were collected to monitor each exposure pathway and included well, surface and municipal drinking water, air particulate and charcoal, vegetation, milk and other food products, which were all analyzed for their radioactivity content specific to each sample type. Analytical detection criteria for each sample type were specified in the ODCM in the form of lower limits of detection (LLD), which were consistent with industry standards and NRC guidelines for routine environmental measurements.

Beginning the second quarter of 2000, the licensee identified that the vendor laboratory failed to meet ODCM specified LLDs for several water samples. These problems were attributed to laboratory equipment failure and were corrected by the vendor laboratory. Subsequently, the vendor moved its laboratory operation, as planned, to a new facility; however, the move affected its analytical capabilities because samples were not analyzed in a timely manner to meet the LLDs for shorter lived radionuclides. A fire occurred in one of the laboratory facilities about the same time the move took place, which exacerbated the analytical problems. As a result, between approximately September 2000 and January 2001, numerous (more than 50) environmental samples analyzed from several exposure pathways did not meet REMP sample LLD criteria specified by the ODCM. Specifically, numerous drinking water samples and several milk samples were not analyzed in a sufficiently timely manner to achieve ODCM required LLDs for certain isotopes, including some LLDs that were not achieved by several orders of magnitude. The licensee recognized the problem and regularly communicated with the vendor to resolve the analysis difficulties; however, the problems continued for approximately 5 months until the vendor's new laboratory operations stabilized in approximately February 2001.

This issue, if not corrected, would become a more significant concern because it could impact the licensee's ability to assess the effect of plant effluents on the environment. Therefore, the issue represents a Finding which the inspectors evaluated using the significance determination process (SDP) for the public radiation safety cornerstone. Since the sample analysis problems related primarily to certain shorter lived isotopes that were not released in plant effluents during the affected time periods (other than a few samples that did not meet LLDs for iodine-131 and iron-59), a failure to assess the overall impact of plant operations on the environment for a given pathway did not occur. Consequently, the inspectors concluded that the problem was of very low safety significance (Green).

Technical Specification 6.8.4(b) requires, in part, that a program be established, implemented, and maintained to monitor the radiation and radionuclides in the environs of the plant. The program shall be contained in the ODCM, and include sampling and analyses in accordance with the methodology and parameters in the ODCM. The ODCM (station procedure PMP-6010.OSD.001), Section 3.5, requires that sample analysis for the REMP be conducted in accordance with Attachment 3.20, "Maximum Values for Lower Limits of Detection - REMP." The REMP bases specifies that analyses be performed in such a manner that the stated LLDs be achieved under routine analysis conditions. The failure to meet ODCM specified LLDs for numerous samples collected over an approximate 5 month period is a violation of Technical Specification 6.8.4. However, because of the very low safety significance of the violation and because the licensee included this item in its corrective action program (CR 01110029 and CR 00243086), this violation is being treated as a Non-Cited Violation (NCV 50-315/01-19-02; 50-316/01-19-02).

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP4 <u>Security Plan Changes</u> (71130.04)

a. <u>Inspection Scope</u>

The inspector reviewed Revision 1 to the Donald C. Cook Nuclear Plant Security Training and Qualification Plan to verify that the changes did not decrease the effectiveness of the submitted document. The referenced revision was submitted in accordance with 10 CFR 50-54(p)(2) requirements by licensee letter dated November 16, 2001.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

.1 Unit 2 Turbine Driven Auxiliary Feedwater Pump Fault Exposure

a. Inspection Scope

The licensee estimated that approximately 1007 hours of fault exposure hours for the Unit 2 turbine driven auxiliary feedwater pump (TDAFWP) were accumulated during the second and third quarters of 2000. The inspectors reviewed the circumstances associated with this fault exposure time to assess the safety significance of this issue. Because the licensee did not monitor system unavailability during the extended dual unit outage that began in September 1997, the licensee has reported safety system unavailability data only since the second quarter of 2000. Consequently, the licensee lacks sufficient data to calculate the final value of the system unavailability performance indicator; therefore, the safety system unavailability indicator was considered to be "Not Applicable" at the time of the inspection.

The licensee submitted frequently asked question (FAQ) 291 to the Nuclear Energy Institute to address calculation of the safety system unavailability performance indicator with less than twelve quarters of system performance data. This FAQ was answered in Revision 2 to NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," with the recommendation to zero sum unavailability time prior to the second quarter of 2000 to enable calculation of the performance indicator. Additionally, the FAQ response stated that T/2 fault exposure time accumulated prior to obtaining twelve quarters of performance data would not be included in the performance indicator calculation but instead be evaluated within the inspection and significance determination processes. Therefore, the inspectors reviewed T/2 safety system fault exposure time accumulated during the performance indicator reporting period using the SDP process.

b. Findings

The inspectors identified a potential violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to include appropriate quantitative acceptance criteria in maintenance procedure 12-MHP 5021.056.007, "Auxiliary Feed Pump Trip and Throttle Valve Linkage Adjustment," Revision 2. The safety significance of this issue has been characterized as "To Be Determined (TBD)" pending the completion of additional risk analysis.

On August 9, 2001, the licensee removed the Unit 2 TDAFWP from service to perform several pre-planned maintenance activities. Following completion of these activities on August 10, the licensee performed two unsuccessful TDAFWP start attempts in accordance with 02-OHP 4021.056.001, "Filling and Venting of the Auxiliary Feedwater

System." A subsequent TDAFWP start attempt for troubleshooting on August 10, 2001 was also unsuccessful. The licensee investigated the failure and determined that the cause of the failure to start was insufficient engagement of the trip throttle valve latching mechanism. The licensee repaired the trip throttle valve under Job Order (JO) 01222001 and returned the Unit 2 TDAFWP to an operable status on August 11, 2001.

The inspectors reviewed the licensee's apparent cause evaluation for the TDAFWP trip throttle valve failure performed for CR 01222001. The licensee determined that the trip throttle valve alignment criteria specified in maintenance procedure 12-MHP 5021.056.007, "Turbine Driven Auxiliary Feed Pump Trip and Throttle Valve Linkage Adjustment," Revision 2, was inconsistent with guidance used by the valve vendor for trip throttle valve alignment. Specifically, Procedure 12-MHP 5021.056.007 specified a trip throttle valve contact alignment of a minimum of 75 percent contact line from side to side on the trip hook as measured by blue check. However, the vendor trip throttle test procedure (Schutte & Koerting Co. Drawing 77S-0048V), written in 1977, specified a blue check latch face contact acceptance criteria of a minimum of 75 percent of the surface area. Alignment of the trip throttle valve using a line contact acceptance criteria could result in less latch engagement than required by a surface contact area acceptance criteria and a greater potential for inadvertent disengagement of the trip throttle latching mechanism. Procedure 12-MHP 5021.056.007 originally required a minimum 75 percent contact on the trip hook latch as determined by blue check, but did not specify if the contact criteria referred to a line or area blue check. In January 1997, the licensee evaluated the 12-MHP 5021.056.007 blue check acceptance criteria under an engineering evaluation supporting work request (WR) A0107471 in order to clarify the contact blue check criteria. This evaluation incorrectly concluded that the blue check acceptance criteria applied to line contact as measured from side to side rather than area contact. Consequently, Procedure 12-MHP 5021.056.007 was revised on June 11, 1997, to specify a trip throttle valve trip hook blue check criteria of 75 percent contact line. The licensee later determined that the contact line blue check acceptance criteria was applicable to a type of trip throttle valve not used at D. C. Cook.

During the apparent cause evaluation for the TDAFWP pump failure, the licensee identified that the trip throttle failed during testing in June 2000. During testing following a design change to the TDAFWP governor control system, the Unit 2 TDAFWP failed to start. The licensee determined that the cause of the failure was due to excessive wear of the trip hook latching mechanism. The trip hook latch mechanism was replaced under JO C0052930, "2-DCP-617, Rework TDAFWP Governor," and adjusted to at least a 75 percent line contact in accordance with 12-MHP 5021.056.007. The inspectors determined that the licensee failed to initiate a condition report to document and evaluate this previous failure. Initiation of a condition report for the June 2000 failure would have been appropriate since the trip throttle valve failure was unrelated to the original governor testing activities and trip hook latch assembly replacement was not within the original scope of the JO C0052930. The inspectors concluded that the failure to document the June 2000 failure of the Unit 2 TDAFWP trip throttle valve within the corrective action system potentially delayed adequate evaluation of the trip throttle valve failure mechanism and contributed to the August 2001 failure. The licensee initiated CR 01362027 to document this issue.

10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures and Drawings," required in part that activities affecting quality be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, Step 8.G of Procedure 12-MHP 5021.056.007, "Turbine Driven Auxiliary Feed Pump Trip and Throttle Valve Linkage Adjustment," Revision 2, did not include appropriate acceptance criteria for determining that alignment of the trip throttle linkage, an activity affecting quality, was satisfactorily accomplished. Specifically, Step 8.G of the procedure specified an alignment acceptance criteria of 75 percent contact line from side to side on the trip hook by blue check. The vendor test procedure for trip throttle valves specified a latch face alignment of face contact over 75 percent of the surface area of the latch face. Alignment of the trip throttle valve using a line contact acceptance criteria could result in less latch engagement than required by a surface contact area acceptance criteria and a greater potential for inadvertent disengagement of the trip throttle latching mechanism. The Unit 2 TDAFWP trip throttle valve was adjusted in accordance with 12-MHP 5021.056.007 on June 14, 2000. Subsequently, the Unit 2 TDAFWP trip throttle valve failed to engage during three successive start attempts on August 10, 2001. The licensee determined that the apparent cause of the pump start failure was due to insufficient engagement of the trip throttle valve latching mechanism. This issue is considered an apparent violation of 10 CFR 50 Appendix B, Criterion V. The licensee entered this issue into its corrective action program as CR 01222001.

The staff's significance determination of this finding was not complete at the time of issuance of this report; therefore, this issue is considered an Unresolved Item (50-316/01-19-03(DRP)). The safety significance has been characterized as "TBD" pending the completion of additional risk analysis.

.2 Safety System Unavailability Performance Indicators

a. Inspection Scope

Mitigating Systems Cornerstone

The inspectors verified the following performance indicators for both units:

- Safety System Unavailability - Emergency AC [Alternating Current] Power
- Safety System Unavailability - Auxiliary Feedwater
- Safety System Unavailability - High Pressure Safety Injection
- Safety System Unavailability - Residual Heat Removal

The inspectors reviewed operating logs, maintenance history and surveillance test history for unavailability information for these systems from October 2000 to September 2001. The inspectors also verified the licensee's calculation of required hours for both units and evaluated applicable safety system equipment unavailability against the performance indicator definition.

The inspectors noted that both units were returned to operation in 2000 following extended outages. The licensee has not yet had sufficient operational service to calculate the safety system performance indicators. It is expected that these indicators will be calculated starting with the first quarter of 2002.

b. Findings

No findings of significance were identified. However, the inspectors identified several issues related to the inaccurate reporting of performance indicator data.

During review of performance indicator data for the emergency AC power system, the inspectors identified that the licensee had not accounted for unavailability time for the D/Gs during the performance of periodic carbon dioxide fire suppression system "puff" testing consistent with the guidance in NEI [Nuclear Energy Institute] 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 1. The licensee entered this reporting discrepancy into its corrective action program as Condition Report (CR) 01355064.

During review of performance indicator data for the auxiliary feedwater system, the inspectors identified that the licensee had not included hours when an opposite unit's auxiliary feedwater system train is required to be available to perform its intended safety function per the Technical Specifications (TS) in its calculation of hours required. The inspectors noted that TS 3.7.1.2.b required at least one auxiliary feedwater system flowpath in support of the opposite unit's safe shutdown functions to be available whenever the opposite unit is in Modes 1, 2, or 3. Although the licensee entered the TS limiting condition for operation (LCO) during these times, it did not believe that unavailability monitoring was expected because the TS LCO was written for an Appendix R based safety function. The inspectors reviewed the definitions of "hours required" and "off-normal events or accidents" in NEI 99-02, Revision 1, and determined that unavailability monitoring for Appendix R based safety functions is consistent with the current guidance. The licensee entered this reporting discrepancy into its corrective action program as CR 01355072.

In addition, the inspectors identified multiple minor reporting discrepancies involving the reporting of unavailable hours for the wrong train of several mitigating systems on each unit and the inconsistent tracking of unavailable hours under the licensee's Maintenance Rule Program. The licensee entered these reporting discrepancies into its corrective action program as CR 01355058 and CR 01355071.

None of the performance indicator reporting discrepancies noted above would lead to a performance indicator crossing a threshold. See Section 4OA1.1 for discussion of a finding related to the Unit 2 turbine driven auxiliary feedwater pump.

.3 Occupational Exposure Control Effectiveness and Radiological Effluent Technical Specification (RETS)/ODCM Radiological Effluent Occurrence PIs

a. Inspection Scope

The inspectors reviewed data associated with the Occupational Exposure Control Effectiveness PI and the RETS/ODCM PI, to determine if these indicators were accurately assessed and reported since last reviewed in December 2000. To evaluate the PI data, the inspectors reviewed the licensee's CR database and selected CRs generated between December 2000 and November 15, 2001, to identify any potential occurrences that were not recognized by the licensee. For the occupational radiation safety PI, the inspectors also selectively reviewed RCA egress transaction dose information and ED alarm reports generated in 2001 to determine if any potential unintended dose occurrences took place. For the public radiation safety PI, the inspectors selectively reviewed gaseous and liquid effluent release data and associated offsite dose information for December 2000 through October 2001.

The inspectors also reviewed quarterly PI verification records generated as required by station Procedure PMP 7110.PIP.001, "Regulatory Oversight Program Performance Indicators," for the fourth quarter of 2000 and the first three quarters of 2001. Additionally, PI data collection and analyses were discussed with involved staff to determine if the program and processes were implemented consistent with industry guidance in Nuclear Energy Institute 99-02, Revision 1, "Regulatory Assessment Performance Indicator Guideline."

b. <u>Findings</u>

No findings of significance were identified.

4OA2 <u>Identification and Resolution of Problems</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the capability of 4.16 kV breakers to function properly during severe accident conditions.

b. <u>Findings</u>

The inspectors identified a Non-Cited Violation for failure to address a long-standing design deficiency with 4.16 kV air circuit breakers. The inspectors noted that a potential safety concern exists with the capability of the 4.16 kV breakers to function properly during a severe fault condition. The fault current available on 4.16 kV load feeders could exceed the circuit breaker's momentary interrupting capacity rating of 250 MVA during a 3-phase bolted fault condition. The momentary rating is used to measure the circuit breaker's ability to safely close during a fault condition and carry the fault current. Consequently, the affected circuit breaker could fail to trip and the upstream bus supply circuit breaker would potentially trip the entire 4.16 kV bus. This condition exists on all four independent 4.16 kV auxiliary buses of Unit 1 and 2, however, the redundant bus should remain available to perform the affected safety function.

This design deficiency was initially noted by the licensee in 1988. This issue was identified again by the NRC during the Essential Service Water (ESW) inspection in August 1990, during the Safety Systems Functional Inspection (SSFI) in March 1992, and was documented as an open item in NRC Inspection Report 50-315/316/92003-01(DRS). The NRC identified that the 4.16 kV switchgear short-circuit momentary duty exceeded the circuit breaker capability by 21 percent for the worst-case condition. The open item was subsequently closed out in NRC

Inspection Report 50-315/316/94022(DRS) based on licensee's commitment to review this issue and perform detailed short-circuit calculations to address the concern noted by the Electrical Distribution Safety Functional Inspection team.

During the 1997 extended plant shutdown, the NRC issued a violation to D. C. Cook for a corrective action program breakdown. The licensee made significant improvements in the corrective action program; however, the inspectors determined that from 1988 to 1999, little progress had been made to address this particular design issue. On April 5, 1999, the licensee initiated CR 99-07602. The CR stated that the 4.16 kV breakers were operable in all modes of plant operations and that the short-circuit fault duty of each 4.16 kV load feeder was required to be limited to the interrupting capability of its 250 MVA air circuit breakers, even for a 3-phase bolted fault. The CR concluded that the worst-case short-circuit overduty for the Unit 1 4.16 kV switchgear was 11 percent over the tested breaker capability for momentary duty and 12.8 percent for the symmetrical interrupting breaker rating which represent significant overduty. Calculations 1-E-N-ELCP-4 kV-001 and 2-E-N-ELCP-4 kV -001, dated October 31, 2000, also confirmed that the potential fault current available on 4.16 kV load feeders could exceed the circuit breaker's momentary interrupting capacity rating of 250 MVA.

The licensee opened Corrective Action Item No. 8 in CR 99-07602 to address this design deficiency. Corrective Action Item No. 8 had a due date of July 31, 2001, and required that an engineering study be performed to address this issue. Sargent and Lundy (S&L) performed an engineering evaluation and on March 27, 2001, issued a report which included actions needed to resolve this issue. The report revealed that a retrofit of the 4.16 kV switchgear to a 350 MVA rating was the most feasible solution and recommended a breaker upgrade. Subsequently, the licensee informed the NRC that the scope of the S&L study was too narrow and that the licensee had decided to expand the scope of the study to identify and evaluate other options beyond breaker upgrades. On May 9, 2001, the licensee initiated CR 01129088 to expand the S&L study and evaluate more options for resolution of the 4.16 kV breaker short circuit overduty concerns. Corrective Action Item No. 8 was still open in October 2001.

The inspectors noted that the condition of the 4.16 kV system was contrary to UFSAR Section 8.1.2.d which states "the 4160 volt transformer secondary feeds four independent 4160 volt auxiliary buses of each unit. The short-circuit fault duty on each bus is limited to within the interrupting capability of the 250 MVA air circuit breakers."

The inspectors assessed these findings relative to the problem identification and resolution cross-cutting area. The inspectors informed the licensee that failure to correct a design deficiency which was noted in 1988 and which could result in exceeding the 4.16 kV breaker's momentary interrupting rating capability during a severe fault condition, constituted a Violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because the licensee entered the finding into the corrective action program as CR 99-07602, this violation is being treated as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-315/01-19-04(DRP), 50-316/01-19-04(DRP)). This violation is in the licensee's corrective action system as CR 99-07602, dated April 5, 1999. The inspectors determined that the failure to adequately resolve this design deficiency could have a credible impact on safety if left uncorrected. This issue affects the mitigating systems cornerstone. This issue screened as GREEN during the Phase 1 Significance Determination Process review because it did not present an actual loss of safety function and it did not result in an actual loss of Technical Specification related equipment. Also, the redundant electrical train which would not be affected by a common mode fault should be available.

4OA3 Event Follow-Up (71153)

.1 Licensee Event Reports

a. Inspection Scope

The inspectors reviewed the corrective actions associated with the following licensee event reports.

b. Findings

(Closed) Licensee Event Report 50-315/99011-01: Air system for emergency diesel generators may not support long term operability due to original design error. This LER was discussed and closed in NRC Inspection Report 50-315/00-03; 50-316/00-03 as Restart Action Matrix Item 1.38. The licensee documented the error in Condition Report 99-3087. The supplement to the LER described the corrective actions taken to correct the problem, but the supplement did not identify any new issues. Therefore, this LER is closed.

(Closed) Licensee Event Report 50-316/00006-00: Failure to comply with requirements of Technical Specifications for nuclear instrumentation. On June 22, 2000, the licensee commenced low power physics testing on Unit 2, using the special test exception of Technical Specification (TS) 3.10.3, "Physics Test". This TS required that the thermal power not exceed 5 percent of rated thermal power (RTP), and the reactor trip setpoints for the operable intermediate range neutron flux and the power range neutron flux low setpoints are set at less than or equal to 25 percent RTP. The power range instruments were found to have a setpoint greater than 25 percent RTP. This represented a failure to meet the requirements of TS 3.10.3. Additionally, the requirements of TSs 2.2.1 and 3.3.1.1 which govern the setpoints and operability requirements during Modes 1 and 2, were not met, resulting in an unrecognized entry into TS 3.0.3. The inspectors reviewed this issue of power range trip setpoints above the TS limit in NRC Inspection Report 50-315/00-16(DRP); 50-316/00-16(DRP). The inspection report discussed the licensee's failure to set the power range NIs to less than or equal to the values required in TS 2.2.1 and identified a Non-Cited Violation 50-316/00-16-05. Details of this event and the corrective actions performed by the licensee are documented in Condition Report P-00-09197. This LER is closed.

(Closed) Licensee Event Report 50-315/00007-00, -01: ESF (engineered safety feature) ventilation system inoperable due to Technical Specification surveillance test methodology. This licensee identified issue was entered into the corrective action program as Condition Report P-00-11175. During an evaluation of industry operating experience information, OE11256, "Control Room Emergency Filtration Inoperable Due to Testing Method," systems engineering personnel determined that the issue was applicable to D. C. Cook Nuclear Plant. Specifically, licensee personnel determined that Technical Specification flow requirements for the ESF ventilation system could not be met during testing if the system automatically started from an accident signal. Consequently, both trains of the ESF ventilation system would be inoperable when aligned for testing per the plant procedures while the plant was in Modes 1-4 when technical specification required both trains to be operable. The event was appropriately reported to the NRC as a condition prohibited by technical specifications.

Licensee personnel analyzed the event and determined that inadequate test procedures caused the technical specification non-compliance. However, licensee personnel concluded that the inadequate test procedures would not have adversely impacted the plant's ability to mitigate the consequences of an accident and therefore had minimal safety significance. The inspectors reviewed the licensee's analysis and did not identify and findings of significance. Consequently, this technical specification non-compliance constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The inspectors also verified that the corrective actions documented in Condition Report P-00-11175 were reasonable and that the corrective actions had been completed. This licensee event report is closed.

<u>(Closed) Licensee Event Report 50-315/01-01-00</u>: Reactor trip due to loss of main feedwater pump. On February 15, 2001, with reactor power at approximately 100 percent, a low vacuum trip on the Unit 1 east main feedwater pump turbine occurred. Plant operators manually tripped the reactor in accordance with plant procedures. The licensee identified the cause to be a loss of condenser vacuum as the result of corrosion debris, a condition that lead to an elevated condenser backpressure and low vacuum trip of the pump. The licensee's corrective actions were reviewed and considered adequate. The licensee's corrective actions included the cleaning of both east and west main feed pump condensers. Details of this event are documented in licensee Condition Report 01046054. This LER is closed.

<u>(Closed) Licensee Event Report 50-316/01-01-00</u>: Plant shutdown due to control rod shutdown bank misalignment. On January 22, 2001, the licensee was performing a routine surveillance test of the Unit 2 rod control system. During the surveillance test, Shutdown Bank "C" would not respond to movement commands. The licensee entered TS action statement 3.1.3.1.b, which required that the plant be placed in Mode 3 (Hot Standby) within 6 hours. Additional testing identified that Shutdown Bank "D" also would not respond to movement commands. Subsequently, the licensee performed an operability review and decided that the shutdown banks remained operable and that TS action statement 3.1.3.1.b should be exited. The licensee identified the cause to be an inadequate cleaning and inspection program that failed to ensure the proper tightening of terminal connection. The licensee's corrective actions were reviewed and considered adequate. Corrective actions included tightening the loose connections and inspecting all terminal board connections. The inspectors discussed this event in NRC Inspection Report 50-315/01-02(DRP); 50-316/01-02(DRP). Details of this event are documented in Condition Report 01029009. This LER is closed.

<u>(Closed) Licensee Event Report 50-315/01-04-00</u>: Unit 1 entered Mode 3 with the remote shutdown panel pressurizer level instrument channel inoperable. On

September 27, 2001, during Unit 1 startup activities, Unit 1 was taken from Mode 4 to Mode 3 with the remote shutdown pressurizer level instrument 1-NLP-151 inoperable. Although the licensee identified the instrument as inoperable in Mode 4, Unit 1 was taken to Mode 3 in violation of Technical Specifications (TS) 3.0.4. The licensee identified the cause to be human error. Plant operators improperly used the operability requirements for the reactor protection instrumentation Technical Specification TS 3.3.1.1, instead of the remote shutdown instrumentation Technical Specification TS 3.3.3.5. The licensee's corrective actions were reviewed and considered satisfactory. A proposed amendment to the Unit 1 TS 3.3.3.5 has been submitted to the NRC. Details of this event and the corrective actions performed by the licensee are documented in licensee Condition Report 01270063. Although this issue was corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

(Closed) Licensee Event Report 50-316/01-04-00: Reactor protection system (RPS) actuation initiated by power range, neutron flux, high negative rate. On October 7, 2001, Unit 2 reactor tripped from 8 percent power as a result of a loss of rod control system voltage. The licensee identified the cause to be a failed resistor at the input to the north control rod drive motor generator set. The failed resistor was replaced. The licensee's corrective actions were reviewed and considered satisfactory. The corrective actions included the replacement of similar series resistors in the Unit 2 south control rod drive motor generator set. Details of this event are documented in licensee Condition Report 01280017. This LER is closed.

4OA4 Cross-Cutting Issues

.1 Human Performance Issues

a. Inspection Scope

The inspectors assessed licensee performance relative to the human performance cross cutting issue. As documented in Section 4OA7 below, the licensee identified two violations of NRC requirements during this inspection report period: (1) a violation of TS 3.9.7 requirements associated with inappropriate movement of loads over the spent fuel pool, and (2) failure to adequately implement procedural requirements for placing the Unit 1 "A" Train D/G in standby. The inspectors assessed the circumstances and causes of these issues relative to the human performance cross-cutting area.

b. Findings

The inspectors identified a Finding of very low safety significance related to human performance weaknesses that contributed to the licensee identified violations documented in Section 4OA7. The human performance aspects of these issues were related to failures to follow procedural guidance, inadequate self checking, and the failure to perform adequate independent verifications. The inspectors considered the following in the assessment of this issue:

Failure to Adequately Control Movement of Loads of the Spent Fuel Pool

On November 19, 2001, the licensee moved the rod control cluster assembly (RCCA) change out tool over racks containing spent fuel with the crane height interlock bypassed and the crane carrying a load above the interlock setpoint height limit. The crane height interlock setpoint was intended to limit the impact energy of a postulated dropped load to less than the maximum impact energy specified in TS 3.9.7. The licensee bypassed the crane height interlock in accordance with plant procedures to lift the RCCA change out tool above the interlock height limit to perform modifications to support the upcoming Unit 2 refueling outage. When the spent fuel pool crane height interlock was initially bypassed, the crane was positioned in the fuel transfer canal in an area away from spent fuel assemblies. Prior to lowering the RCCA change out to a height below the interlock setpoint and removing the interlock from bypass, the crane operator moved the spent fuel pool crane over spent fuel assemblies at a height which exceeded the TS 3.9.7 maximum impact energy limit. Although the licensee immediately identified and corrected this condition, the inspectors determined that several human performance errors led to this occurrence. Fuel handling procedure 12-OHP 4050.FHP.046, "Control of Loads over the Spent Fuel Pool," step 2.2 required that a qualified spent fuel area supervisor (SFPAS) supervise the handling of loads over the spent fuel pool. Additionally, step 4.2 of 12-OHP 4050.FHP.046 required performance of an impact energy calculation to determine the height at which loads may be carried over the spent fuel pool. The inspectors determined that the crane operator failed ensure that TS 3.9.7 impact energy limitations were met prior to movement of the RCCA tool over spent fuel. Additionally, the SFPAS failed to provide adequate oversight of crane operation during the period of time that the crane height interlock was bypassed. The inspectors concluded that administrative controls intended to limit the probability of a fuel handling accident failed due to these human performance weaknesses.

The inspectors assessed the safety significance of the violation of the impact energy requirement of TS 3.9.7 using the SDP. Updated Final Safety Analysis Report (UFSAR) Section 14.2.1, "Fuel Handling Accident," analyzed the consequences of a load drop over spent fuel pool racks containing spent fuel. Because the maximum TS 3.9.7 impact energy was intended to bound the fuel assembly damage following a postulated crane failure, the inspectors determined that this issue was associated with the barrier integrity cornerstone. The inspectors concluded that this issue had a credible impact on safety and was more than a minor concern. Movement of loads over spent fuel with an impact energy greater than the TS limits could result in damage to spent fuel greater than analyzed in the event of a credible crane failure. Because this issue was determined to affect the fuel integrity barrier, this issue was determined to be of very low safety significance (GREEN) following a Phase 1 SDP.

Failure to Adequately Align the Unit 1 "B" Train D/G for Standby Service

During a shift turnover walkdown on December 9, 2001, the oncoming shift manager noted that the manual and automatic voltage regulator settings for the

Unit 1 "B" Train D/G failed to match the Technical Data Book (TDB) required settings. The licensee's investigation determined that following routine D/G surveillance testing on December 8, 2001, the operations crew failed to align the voltage regulator controls for standby service and failed to perform an adequate independent verification of the D/G alignment. Following surveillance testing, the D/G was aligned in standby in accordance with Procedure 01-OHP 4021.008AB, "Operating D/G Unit 1 "B" Train Subsystems." Procedure 01-OHP 4021.008AB required an operator to initially position the automatic and manual voltage regulator potentiometers to the TDB required setting. After the initial positioning, the procedure required a second verification of potentiometer settings by a different operator. The licensee stated that the initial positioner adjusted the manual voltage potentiometer to the required automatic potentiometer setting and failed to adjust the automatic potentiometer back to its normal standby position. (The automatic voltage regulator potentiometer was adjusted during the previous surveillance test to minimize generator circulating currents.) The second reactor operator performing the independent verification failed to identify that neither the manual nor the automatic voltage regulator potentiometers were set to their required TDB positions. The inspectors concluded that the failure to adequately identify safety related equipment prior to manipulation, the failure to adequately follow procedural requirements, and the failure to adequately perform an independent verification constituted weaknesses in the human performance cross-cutting area.

The inspectors assessed the safety significance of this human performance issue using the SDP. The failure align the diesel generator voltage regulation system for standby service could result in the failure of the diesel generator to adequately provide power to supported equipment and therefore impacted the mitigating systems cornerstone. The inspectors determined that this was more than a minor concern because the failure adequately align the D/G for standby service and adequately perform an independent verification of D/G alignment could result in a more serious safety concern if left uncorrected. Specifically, the failure to adequately identify system components prior to manipulation and the failure to perform an adequate independent verification of D/G system alignments could credibly result in the failure of the D/G to perform its associated safety function. In this case, although the automatic voltage regulator potentiometer was set inconsistently with TDB requirements, the as-found potentiometer settings would not have prevented the D/G from performing its safety function. Because the failure to adequately align the Unit 1 "B" Train D/G did not result in an actual loss of safety function, this issue was also determined to be of very low safety significance (GREEN).

The inspectors assessed the safety significance of this cross-cutting issue using the Significance Determination Process (SDP) assessments for the resultant issues. The inspectors concluded that these human performance weaknesses had a credible impact on safety and could become a more significant safety concern if left uncorrected; therefore, these human performance weaknesses were more than a minor concern. Therefore, the inspectors concluded that these human performance weaknesses constituted a finding of very low risk significance based on the safety significance of the resultant issues and their impact to multiple cornerstones of reactor safety. (Section 4OA4)

4OA6 Management Meetings

The inspectors presented the Occupational Radiation Safety - Access Controls for Radiologically Significant Areas and Radiation Monitoring Instrumentation and Public Radiation Safety - Radiological Environmental Monitoring Program inspection results (Report Section 2) on November 15, 2001. The baseline inspection results for Changes, Tests or Experiments (Report Section 1R02) was presented on November 30, 2001. The inspectors presented the Security, Training and Qualification Plan inspection results (Report Section 3) on December 5, 2001. The inspectors presented the remaining inspection results to licensee management listed below on December 28, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following findings of very low safety significance (GREEN) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section IV of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as Non-Cited Violations (NCV).

NCV Tracking Number	Requirement Licensee Failed to Meet
50-315/01-19-06 50-316/01-19-06	TS 4.9.7.2, "Crane Travel - Spent Fuel Storage Pool Building," requires, in part, that the potential impact energy due to dropping a crane's load be determined to be less than or equal to 24,240 in-lbs prior to moving each load over racks containing fuel. Contrary to this requirement, on November 19, 2001, the licensee moved the rod control cluster assembly (RCCA) change out tool over storage racks containing fuel without determining the impact energy of the load. The impact energy associated with the RCCA change out tool movement exceeded the TS limit of 24,240 in-lbs. This issue is in the licensee's corrective action system as CR 01323024 and is being treated as a Non-Cited Violation.
50-315/01-19-07	TS 6.8.1 requires, in part, that procedures shall be established, implemented and maintained covering the activities recommended in Appendix "A" of Regulatory Guide 1.33, Rev 2, February 1978. Operations Procedure 01-OHP-4021-032-008AB, "Operating D/G Unit 1 "B" Train Subsystems," was written to cover activities recommended by RG 1.33. Steps 4.1.6 and 4.1.9 of 01-OHP-4021-032-008AB required that the control room panel diesel generator voltage regulator potentiometer settings be verified to match the required settings specified in the

Technical Data Book. Contrary to the above, on December 8, 2001, the licensee failed verify that the Unit 1 "B" Train D/G control room panel diesel generator voltage regulator potentiometer settings matched the required settings. This issue is in the licensee's corrective action system as CR 01343015 and is being treated as a Non-Cited Violation.

KEY POINTS OF CONTACT

Licensee

G. Arent, Manger, Regulatory Affairs
C. Bakken, Senior Vice President, Nuclear Generation
M. Barfelz, Regulatory Affairs
J. Carlson, Environmental Superintendent
P. Cowan, Licensing Supervisor, Regulatory Affairs
R. Gaston, Regulatory Affairs Compliance Supervisor
J. Gebbie, System Engineering Manager
S. Greenlee, Director, Nuclear Technical Services
J. Harner, REMP Coordinator
R. LaBurn, General Supervisor, Radiation Protection Production
E. Larson, Manager, Operations
R. Meister, Regulatory Affairs
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T. Noonan, Director, Performance Assurance
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K. Steinmetz, Licensing 50.59 Program Owner
L. Weber, Performance Assurance
D. Wood, RadChem Environmental Manager

NRC
A. Vegel, Chief, Reactor Projects Branch 6
H. Gonzalez, Reactor Engineer
D. Rivera-Martinez, Reactor Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-315/01-19-01 50-316/01-19-01	NCV	Failure to ensure that breaker coordination and selective tripping was provided at the 4.16kV system (Section 1R05)
50-315/01-19-02 50-316/01-19-02	NCV	Failure to meet analytical detection capabilities for numerous radiological environmental samples collected between the third quarter of 2000 and the first quarter of 2001 (Section 2PS3)
50-316/01-19-03	URI	Apparent violation of 10 CFR Appendix B, Criterion V for the failure to incorporate adequate quantitative acceptance criteria in turbine driven auxiliary feedwater pump maintenance instructions (Section 4OA1)
50-315/01-19-04 50-316/01-19-04	NCV	Failure to correct a long standing design deficiency associated with 4.16 kV breakers momentary interrupting rating capability (Section 4OA2)
50-315/01-19-05 50-316/01-19-05	FIN	Human performance weaknesses related to procedural adherence and independent verification (Section 4OA4)
50-315/01-19-06 50-316/01-19-06	NCV	Failure to maintain load carried over spent fuel within impact energy requirements of TS 3.9.7 (Section 4OA7)
50-315/01-19-07	NCV	Failure to appropriately align Unit 1 "B" Train D/G for standby following testing (Section 4OA7)

Closed

50-315/99011-01	LER	Air system for emergency diesel generators may not support long term operability due to original design error (Section 4OA3)
50-316/00006-00	LER	Failure to comply with requirements of Technical Specifications for nuclear instrumentation (Section 4OA3)
50-315/00007-00 50-315/00007-01	LER	ESF ventilation system inoperable due to TS surveillance test methodology (Section 4OA3)
50-315/01-01-00	LER	Reactor trip due to loss of main feedwater pump (Section 4OA3)
50-316/01-01-00	LER	Plant shutdown due to control rod shutdown bank misalignment (Section 4OA3)
50-315/01-04-00	LER	Unit 1 entered Mode 3 with the remote shutdown panel pressurizer level instrument channel inoperable (Section 4OA3)
50-316/01-04-00	LER	Reactor protection system (RPS) actuation initiated by power range, neutron flux, high negative rate (Section 4OA3)

50-315/01-19-01 50-316/01-19-01	NCV	Failure to ensure that breaker coordination and selective tripping was provided at the 4.16kV system (Section 1R05)
50-315/01-19-02 50-316/01-19-02	NCV	Failure to meet analytical detection capabilities for numerous radiological environmental samples collected between the third quarter of 2000 and the first quarter of 2001 (Section 2PS3)
50-315/01-19-04 50-316/01-19-04	NCV	Failure to correct a long standing design deficiency associated with 4.16 kV breakers momentary interrupting rating capability (Section 4OA2)
50-315/01-19-05 50-316/01-19-05	FIN	Human performance weaknesses related to procedural adherence and independent verification (Section 4OA4)
50-315/01-19-06 50-316/01-19-06	NCV	Failure to maintain load carried over spent fuel within impact energy requirements of TS 3.9.7 (Section 4OA7)
50-315/01-19-07	NCV	Failure to appropriately align Unit 1 "B" Train D/G for standby following testing (Section 4OA7)

<u>Discussed</u>

None

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
AEP	American Electric Power
ALARA	As Low As Is Reasonably Achievable
ATR	Administrative Technical Requirement
AV	Apparent Violation
CFR	Code of Federal Regulations
CR	Condition Report
CT	Current Transformer
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ED	Electronic Dosimetry
EP	Emergency Preparedness
ESW	Essential Service Water
FIN	Finding
IMC	Inspection Manual Chapter
LERF	Large Early Release Frequency
LHRA	Locked High Radiation Area
LLD	Lower Limits of Detection
LOOP	Loss of Offsite Power
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OA	Other Activities
ODCM	Offsite Dose Calculation Manual
OHP	Operations Head Procedure
PARS	Publically Available Records
PASS	Post Accident Sampling System
PDR	Public Document Room
PI	Performance Indicator
PJC	Phase Instantaneous Current
PMP	Plant Manager’s Procedure
PMT	Post-maintenance Testing
RCA	Radiologically Controlled Area
RCCA	Rod Control Cluster Assembly
REMP	Radiological Environmental Monitoring Program
RHR	Residual Heat Removal
RPS	Reactor Protection System
RTP	Rated Thermal Power
RWP	Radiation Work Permit
RP	Radiation Protection
SDP	Significance Determination Process
SRO	Senior Reactor Operator
SSA	Safe Shutdown Analysis
SSC	Structures, Systems, and Components
SSCA	Safe Shutdown Capability Assessment
SSPS	Solid State Protection System
STP	Surveillance Test Procedure

TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TLD	Thermoluminescent Dosimeter
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
VIO	Violation

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather Protection

12-TM-00-61-R2	Winterization/De-Winterization TM to Support 12-IHP 5040.EMP.004	Revision 2
PMI 5055	Winterization/Summerization	Revision 0
12-IHP 5040.EMP.004	Plant Winterization and De-Winterization	Revision 3

1R02 Evaluation of Changes, Tests, or Experiments

10 CFR 50.59 Evaluations

2000-1069-01	Lake Temperature Project - CCW and ESW 12-DCP-174	July 7, 2000
2000-1140-00	Feeding 600 Volt Buses Through Bus Breakers 2-OHP 4021.082.003	June 3, 2000
2000-1143-01	Addition of Administrative Technical Requirements for Unit 2 EDGs ATR2-EDG-1	May 27, 2000
2000-1649-00	ESS Thermal Overload List	August 18, 2000
2000-2063-01	Unit 1 Boric Acid Concentration Reduction Modification /UCR 99-UFSAR-1343 and 13 47 1-DCP 120	December 2, 2000
2000-2077-02	Winterization and De-Winterization 12 TM-00-61	December 29, 2000
2000-2319-00	Unit 1 Core Reload 1-DCP- 4872	November 6, 2000
2000-2446-00	Change of Safety Analysis UCR-1540	November 22, 2000
2001-0223-00	Add New Evaluation Results Pertaining to SBO Coping UCR-1458	May 4, 2001
2001-1008-00	Removal of Auto-Open Feature on Diesel Start for EDG Coolers Alternate ESW Valves Temp Mod 12-TM-01-52-RO, ODE CR-1242013	September 12, 2001

2001-1197-00	Allow ESW Flow Normally Through CTS HXs to Meet ESW Flow Requirements during Low Lake Water 1-DCP-5173 and 2-DCP-5174 (includes TS Bases 3/4.1.2 and 3/4.5.5 Change and UCR-1609	November 1, 2001

10 CFR 50.59 Screenings

2000-1963-01	Unit 1Motor Operated Valve (MOV) Setpoint Control Data Sheets - Component Cooling Water VDS-1 ccm-430/431/432/433.Revision 0 and VDS-1-CMO-410/411/412/413	November 30, 2000
2000-2061-00	Unit 2 Feed Pump Room Cooler ESW Return Valves Installed Backwards CR 00-09639 (Use-As-Is)	October 9, 2000
2000-2129-00	Safety Related Pump Inservice Test Hydraulic Reference Tech Databook Figure 1-15.1	October 12, 2000,
2000-2263-01	Removal of Inner Debris Screens From EDG Intake Ventilation System Duct Work 1-LDCP-4889	November 8, 2000
2000-2490-00	Power Operated Valve Stroke Time Limits Technical Data Book Figure 1-19.1	November 25, 2000
2001-0013-00	Loss of All Offsite Power 01-OHP 4023. ECA-0.0,	January 17, 2001
2001-0378-00	Under 2-DCP-4908, the Unit 2 ECCS MOVs 2-IMO-255, 256 and 2.ICM-250, 251 will be Modified to Reduce the Stem Diameter in Order to Improve Valve Operation 2-DCP-4908	May 16, 2001
2001-0519-00	Annunciator #134 Response: Spent Fuel Pit 12-OHP 4024-134	April 1, 2000
2001-0575-00	Comp Action for Degraded 1-1A5 Breaker 1-1A5	July 5, 2001
2001-0626-00	Locating 250 VDC Grounds 12-OHP 4021-005-012,	August 2, 2001
2001-1033-00	Increase Structural Integrity of the Unit 2 ESW Strainers 2-LDCP-5147	September 12, 2001

2001-1214-00	Fuel Transfer Pump HELB Protection 1-DCP-5021	October 30, 2001
2001-1266-00	Revise Setting of Differential Relays for 4kV/600V Transformers Relay Setting Sheets RSC1-4072, Etc.	October 26, 2001
2001-1278-00	Operation of the Boric Acid Reserve Tank 12-OHP 4021-005-008	October 23, 2001

<u>Condition Reports</u>

CR 00293063	12 4021.006.002 Allows Deenergization of Conductivity Cell Which May Not Have Been Evaluated in 10 CFR 50.59	October 19, 2000
CR 00318068	Unit Technical Specifications Bases Change for Spray Additive Test Parameters Did Not Have a Complete Safety Evaluation	November 13, 2000
CR 01039036	Potential 10 CFR 50.59 Bypass in PMP-7030-OPR-001, Operability Determination in Providing Guidance for the SS/SM to Implement Required Compensatory Measures PRIOR to Completing a 10 CFR 50.59 Review	February 8,2001
CR 01114018	Calculations for Spent Fuel Pool Performed Using Methodology Not in Compliance With the CNP Current Licensing Basis Issued as Unrestricted Without a 10 CFR 50.59 Review	April 24, 2001
CR 01221046	The Validation to Use Safety Screening/Safety Evaluation (SS/SE) 1999-1608-01(2-DCP-4247) for SS/SE 2000-1468-00(1-DCP-4247) Did Not Address Effects of the LOOP in the Winter on Unit 1 "B" Train Battery	August 9, 2001
CR 01265020	Instrument Change Package ICP-00758, Revision 0, Does Not Contain All Relevant Data	September 22, 2001
CR 01284045	A 10 CFR 50.59 Screen Was Determined to Be Inadequate (10 CFR 50.59 Tracking Number 2001-0729-00)	October 11, 2001

CR P-00-09957	DIT S-00625-00 Changed the AFW Room Cooler Setpoint Without an SE or SS and Without Evaluation of Potential Cooler Freeze Conditions at Higher ESW Flowrates than Test Qualification. The Temperature Switch is Not on Appropriate Plant Control Lists	July 14, 2000

1R04 Equipment Alignment (71111.04)

TS 3.6.2.1	Containment Spray System	Amendment 188
OP-2-5144	Flow Diagram - Unit 2 Containment Spray	
01-OHP 402.032.008AB	Operating DGUnit 1 "B" Train Subsystems	Revision 2
01-OHP 5030.001.001	Operations Plant Tours	Revision 19a
02-OHP 4021.009.001	Placing the Containment Spray System in Standby Readiness	Revision 6b
Technical Data Book 1-Figure 19.9	Diesel Generator Pot Settings	Revision 20
CR 01339050	The door between the control rooms, 12-DR-AUX415, was found in the open position	December 5, 2001
CR 01216057	Received low control air pressure annunciator during surveillance testing	August 4, 2001

1R05 Fire Protection

.1 Fire Protection Safe Shutdown Analysis

Calculations

1-E-N-PROT-RLY-002	4kV SR Motors Phase Instantaneous Relay (PJC) Setting Calculation, U1	Revision 0
1-E-N-PROT-BKR-007	U1 600V SWGR Breaker 11A6, 11A7, 11B3, 11C3, 11C9, 11C8, and 11D9 Settings	August 14, 2000
1-E-N-ELCP-4 kV-001	U1 4.16 kV/600V Load Control Calc’s	October 31, 2000
2-E-N-ELCP-4 kV-001	U2 4.16 kV/600V Load Control Calc’s	January 14, 2000
2-E-N-PROT-RLY-002	4.16 kV SR Motors Phase Instantaneous Relay (PJC) Setting Calculation, U2	February 15, 2000

Drawings

OP-2-12003	25O VDC Main One-Line ESF Train A, B, and N	Revision 23
1-1412-27,1-1421-80, 1-1428-32, 1-1431-34, 1-1433-23,1-1435-81, 1-2074-34,1-2037-49	Conduit Routing	

Condition Reports

CR P-00-03109	This CR Documents Superceded Calculations, Uninstalled DCPs, Limitation, Equipment Not Meeting Acceptance Criteria and Recommendations in Calc. 2-E-N-ELCP-4.16 kV-001, Revision 1	February 23, 2000
CR P-99-18634	Discrepancy in Electrical Protection Calculations	July 16, 1999
CR 01129088	S&L Study to Resolve 4.16 kV Switchgear Short Circuit Overduty Concerns was OAR'd with Comments	May 9, 2001
CR P-99-07602	Calculation PS-4.16 kVD-002 Shows that the Momentary Ratings on the 4.16 kV Circuit Breakers are Exceeded for Fault Conditions	April 5, 1999
CR P-00-01627	Discrepancy with FSAR Q&A 40.7	January 28, 2000
CR P-00-09424	Instrument Overcurrent Settings for Several 4.16 kV ESS Pump Motors May Require Revision	June 29, 2000
CR P-00-02519	Instantaneous Overcurrent Relay Settings for the AFW-2W, AFW-2E, CTS-2W and ESW-2W Pump Motors May Require Revision	February 11, 2000

Procedures

EHI-2070	Engineering Support Personnel (ESP) Training and Qualification	Revision 0a
PMI-1030	Personnel Selections and Administrative Controls	Revision 4

Miscellaneous

	AEP Engineering Position Description Matrix	January 1, 1997
ANSI N18.1-1971	Selection and Training of Nuclear Power Plant Personnel	
	AEP Exempt Summary Job Description	
VTD-GENE-1188	General Electric Instructions for Instantaneous Current Relays Type PJC (Pub. #GEH=1790B)	May 27, 1996
GEH-1753	Time Overcurrent Relays	
PS-EPCS-001	Electrical Protection Coordination Study	

CRs Initiated as a Result of NRC Questions

CR 01129088	S&L Study to Resolve 4.16 kV Switchgear Short Circuit Overduty Concerns to be Expanded Options Other than Replacement of Existing Overduty Breakers	May 9, 2001
CR 01208057	The Impact Assm't for Calculations 1-E-N-PROT-RLY-002 and 2-E-N-PROT-RLY-002 Fail to Identify the Impact on the Appendix "R" Program	July 27, 2001

.2 Routine Fire Zone Tours

UFSAR Section 7.7.6	Control Room Fire Prevention Design	
UFSAR Section 9.8.1	Fire Protection System	
	D. C. Cook Nuclear Plant Fire Hazards Analysis, Units 1 and 2	Revision 8
	D. C. Cook Nuclear Plant Units 1 and 2 Probabilistic Risk Assessment, Fire Analysis Notebook	February 1995
Fire Hazards Analysis	Fire Zone 7, Quadrant 1 Cable Tunnel	
ESAT 01352053	NRC identified a 3' rope hanging from the bottom of ventilation louver	December 18, 2001
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

PMI 2270	Fire Protection	Revision 26
PA-01-10	Performance Assurance Audit, "Fire Protection"	November 13, 2001

1R11 Licensed Operator Requalification

RQ-E-1717	Cook Nuclear Plant Simulator Evaluation Guide, Steam Generator Tube Rupture with Loss of Reserve Power	Revision 4
	Desktop Guide For Emergency Planning Performance Indicators	Revision 1
	Simulator Crew Evaluation Standards	
	Operating crew performance evaluation comments	

1R12 Maintenance Rule Implementation

.1 Annunciator System

	Maintenance Rule Scoping Document Annunciator System	August 28, 2001
	Unit 1 and Unit 2 Blocked Alarm Index	November 27, 2001
CR 00345028	Source range level trip bypass annunciator came in and cleared with no alarm or operator action	December 10, 2000
ESAT 00353035	Annunciator 1-30cd-cdap-8 does not annunciate when tested	December 18, 2000
CR 01107036	Annunciator maintenance rule scoping document does not address cumulative failures	April 17, 2001
ESAT 01117050	Fire panel Unit 1 30-RS-RSAP annunciator ground when tested	April 27, 2001
CR 01143065	Annunciator 122 drop 29 came in and cleared with no audible tone	May 23, 2001
CR 01226026	Licensee identified that maintenance rule evaluation for CR 00-10013 was inadequate	August 14, 2001

CR 01226027	Licensee identified that maintenance rule review for annunciator system was inadequate	August 14, 2001
CR 01249091	Annunciator bus ground alarm drop 40 panel 121 illuminated	September 6, 2001
CR 01289053	Control room annunciator panel 101, drop 22, failed to light during performance of 12-PPP-2270-066-019	October 16, 2001
CR 01323025	Evaluate all abnormal positions and blocked alarms in place for greater than 30 days to determine if 50.59 evaluation is required	November 19, 2001
CR 01325007	Ice condenser door open annunciator did not alarm when personnel entered ice condenser	November 21, 2001
CR 01332080	Annunciator 204 drop 4 reflashed several times while clearance 2013443 was in effect. The annunciator should not have reflashed	November 28, 2001

.2 Reactor Protection System

	Maintenance Rule Scoping Document Reactor Protection System	May 11, 2001
CR 00350032	1-BLP-140 reading at the 6 percent notification limit	December 15, 2000
CR 01009034	Integrated results of the Maintenance Rule recovery project for the reactor protection system	January 9, 2001
CR 01018035	1-NTI-22 did not return to normal due to faulty test injection switch (1-PS-456Q)	January 18, 2001
CR 01018038	1-NTI-42 did not return to normal due to faulty test injection switch	January 18, 2001
CR 01040013	During replacement of the Unit 1 Train B logic power supply, the 15 V power supply failed and caused the B train reactor trip breaker to open	February 8, 2001
CR 01140002	2-FFC-241 #4 S/G flow control transmitter Channel 2 partially failed	May 20, 2001

CR 01196010	Train A solid state protection system PS2 breaker tripped unexpectedly during fuse removal for replacement of 48 volt power supply PS1 causing a loss of all Train A 15 volt power	July 14, 2001
CR 01212017	During replacement of power supply 1 in Unit 1 Train A SSPS, status lights and annunciators flashed unexpectedly when the input error inhibit switch was placed in inhibit	July 31, 2001
CR 01220032	During a historical review of preventative maintenance items, it was determined that four PMs were completed with out of specification conditions and a new ESAT was not initiated.	August 8, 2001
CR 01236037	There have been a significant number of electronic DC power supply failures in the past 24 months	August 24, 2001
CR 01282031	2-MPP-212 was found out of tolerance during as found calibration check	September 19, 2001
CR 01296002	2-NTI-12 (Loop 1 overtemperature delta T) indicator became erratic	October 23, 2001
CR 01341105	NRC identified that MR evaluation for a failure of 2-FFC-241 failed to consider functions associated with reactor protection system and RG 1.97	December 7, 2001
CR 01341104	NRC identified that there was no MR evaluation for out of calibration condition for 1-BLP-140.	December 7, 2001

.3 Emergency Diesel Generators

	Maintenance Rule Scoping Document - Emergency Diesel Generators	Revision 2
	Emergency Diesel Generator Performance Monitoring Plan	
	System Health Report - Emergency Diesel Generators	July 1, 2001 through September 30, 2001

TS 3.8.1	AC Sources - Operating	Amendment 183 (Unit 1) Amendment 168 (Unit 2)
Regulatory Guide 1.9	Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants	Revision 3
RG 1.155	Station Blackout	Revision 0
UFSAR Section 8.4	Emergency Power System	Revision 17
PMI 6080	Emergency Diesel Generator (EDG) Reliability Monitoring Program	Revision 3
12-MHP 4030.032.046	Emergency Diesel Generator System 18 Month Inspection	Revision 2
CR 01136042	Presentation to Maintenance Rule Expert Panel for Unit 1 emergency diesel generators to be considered for (a)(1)	May 16, 2001
CR 01257072	When running STP.027 (under full load), the output of the diesel generator was fluctuating	September 14, 2001
CR 01258009	Attempted start of DG2CD failed when DG2CD Stop/Run control switch was taken to RUN	September 15, 2001

.4 Compressed Air System

	Maintenance Rule Scoping Document - Compressed Air System	Revision 1
	System Health Report - Compressed Air	July 1, 2001 through September 30, 2001

1R13 Maintenance and Emergent Work (71111.13)

NUMARC 93-01	Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	Revision 2
	Operations Night Orders	November 20, 2001

PMP 2291.OLR.001 Data Sheet 1	On-Line Risk Management, Work Schedule Review and Approval Form, Cycle 39, Week 5	November 16, 2001

1R15 Operability Evaluations

	Unit 1 Control Room Logs	November 27 -28, 2001
12-IHP-4030-082-003	AB, CD and N-Train Battery Discharge Test and 18-Month Surveillance Requirements	
12 QHP.SP.001	Determination of Accumulator Water Level Utilizing Ultrasonic Measurement	Revision 0
01-OHP 4030.STP.030	Daily and Shiftly Surveillance Checks	Revision 34
Technical Data Book Figure 12- Figure 18.6	Accumulator Level Conversion	May 18, 1992
ECP 12-I1-02	Accumulator Tank Level and Pressure Transmitter Calibration	Revision 9
VTD-CDBA-0001	Vendor Technical Data - C&D Charter Power Systems Standby Battery Vented Cell Installation and Operating Instructions	
EPRI TR-100248-R1	EPRI - Stationary Battery Guide Design, Application, and Maintenance	
JO R0221335	Job Order - Perform 2-BATT-AB, 92-day surveillance	
CR 01332066	1-ILA-111 Unit 1, Accumulator 1 level is oscillating between 934 and 940 cubic feet	November 28, 2001
CR 01347067	Internal Degradation found on cells of Unit 2 Battery 2-BATT-AB during performance of surveillance R221335-01, 92 day surveillance of 2-BATT-AB	December 13, 2001
CR 01353053	The accumulator volume calculations may not have accounted for cladding thickness. This could result in non-conservative results for ultrasonic level measurement	December 19, 2001

1R16 Operator Workarounds (71111.16)

	Unit 1 Operations Daily Status Report	December 18, 2001
	Unit 1 Control Room Deficiency Report	November 28, 2001
	Unit 1 Caution Tag and Abnormal Position Logs	November 28, 2001
CR 01264048	Unit aggregate operability determination for restart	September 21, 2001

1R17 Permanent Plant Modifications

.1 Emergency Diesel Generator High Pressure Fuel Injection Lines

12-EHP 5040.DES.001	Control of Design Input	Revision 1
12-EHP 5040.MOD.006	Design Change Packages	Revision 5a
12-MHP 5021.032.018	Emergency Diesel Engine Fuel Injection Maintenance	Revision 5a
12-MHP 5021.032.051	Nova Swiss Fuel Injector Line Maintenance	Revision 0
MPR-2011	Root Cause Investigation of Diesel Engine High Pressure Fuel Injection Line Failures	Revision 0 February, 1999
1-DCP-744	Upgrade of EDG High Pressure Fuel Injection Lines	
2-DCP-526	Upgrade of EDG High Pressure Fuel Injection Lines	
Drawing INT-1025-040-01	Worthington SWB-12 High Pressure Fuel Injection Lines	Revision A
JO 01046018	Install 2-DCP-526 on 2 CD emergency diesel generator	September 14, 2001
CR 98-6950	In house and third party reviews of EDG fuel line failure root cause analysis have identified weaknesses in the analysis	November 13, 1998
CR 01200015	DRB review of 2-DCP-526 noted inadequate supporting calculation	July 19, 2001

.2 <u>Provide Essential Service Water Flow Path via the Containment Spray Heat Exchangers for Units 1 and 2</u>

2-DCP-5174	Design Change Package - Unit 2 Provide Essential Service Water Minimum Flow Path via Containment Spray Heat Exchanger	November 2, 2001
1-DCP-5173	Design Change Package - Unit 1 Provide Essential Service Water Minimum Flow Path via Containment Spray Heat Exchanger	November 2, 2001
12-OHP-4021-019-001	Operation of the Essential Service Water System	Revision 24
01-DCP-5173-TP1	Functional Test of 1-WMO-713 and 1-WMO-717	Revision 0
DIT-B-00011-06	Accident Analysis Input Assumptions for Containment Sump Water Level Analysis	
DIT-B-02219-00	Evaluation of the Effect of Open Containment Spray Heat Exchanger Essential Service Water Shutoff Valves (WMO-713, -717, -714, -718) on the Hydrogen Sub-compartment Analyses for DBA LOCA	
DIT-B-00069-08	Design Input for D.C. Cook Offsite and Control Room Dose Analyses	
Unit 1 UFSAR Chapter 14	Unit 1 Updated Final Safety Analysis Report - Accident Analysis	
Unit 2 UFSAR Chapter 14	Unit 2 Updated Final Safety Analysis Report - Accident Analysis	
	NRC Safety Evaluation Report for Amendment No. 252 to DPR-58	March 19, 2001
RG 1.187	NRC Regulatory Guide - Guidance for Implementation of 10 CFR 50.59, Changes, Tests and Experiments	November 2000
CR 01263055	Condition Report - Review of Westinghouse Letter AEP-01-119 identifies issues requiring at least tracking attention	September 20, 2001

CR 01353051	Condition Report - Questions to the EQ Checklist for 2-DCP-5174 and 1-DCP-5173 were incorrectly answered leading to the conclusion that further EQ review was not necessary	December 19, 2001
CR 01354092	Condition Report - Need to define approach to UFSAR updating for LOCA peak clad temperature changes and associated evaluations	December 20, 2001
CR 01355076	NRC identified that DCP-5173/5174 (Attachment 5) indicates the maximum combined CCW and CTS HX flow should not exceed 5000 gpm - the normal operating procedure does not reflect this limit	December 21, 2001

1R19 Post Maintenance Testing

.1 Unit 1 Accumulator Level Alarm Temporary Modification

CR 01296004	2-ILA-111 indicated level fluctuations of 10 cubic feet in #1 accumulator which brought in the low level alarm	October 23, 2001
JO 01320005	2-ILA-111, Install 2-TM-00-54-R1	November 17, 2001
2-TM-00-54-R1	Alleviate unstable indication and spurious alarms from 2-ILA-111.	Revision 1 November 16, 2001

.2 Unit 1 "A" Train Battery Charger Repair

01-OHP 4021.082.006	Operation of 1AB and 1CD Battery Chargers	Revision 9
JO 01341004	1-BC-CD2, Replace K301 relay	December 7, 2001

.3 Unit 2 Containment Spray Heat Exchangers Essential Service Water Outlet Valves

JO 01296060	Implement 2-DCP-5174, Alternate Flow Path for Essential Service Water	November 30, 2001
02-DCP-5174-TP1	Completed Functional Tests of 2-WMO-714 and 2-WMO-718	Revision 0
CR 01333071	Condition Report - 2-WMO-714 Did Not Meet Acceptance Criteria for 02-DCP-5174-TP1	November 29, 2001

2-FCN-5174-R0-01	Field Change Notice - Revise Step 7.2.2 of Procedure 02-DCP-5174-TP1	November 29, 2001
2-FCN-5174-R0-02	Field Change Notice - Valve Control Circuits were not Designed to Support Referenced Test Statement in Acceptance Criteria for 02-DCP-5174-TP1	November 29, 2001
2-DCP-5174	Design Change Package - Unit 2 Provide Essential Service Water Minimum Flow Path via Containment Spray Heat Exchanger	November 2, 2001
DB-12ESW	Design Basis Document - Essential Service Water System	Revision 0

.4 Unit 1 “A” Train Emergency Diesel Generator Control Air Regulating Valve

TS 3.8.1	AC Power Sources - Operating	Amendment 183
01-OHP 4030.STP.027CD	CD Diesel Generator Operability Test (Train A)	Revision 17
JO 01355003	Remove and replace 1-XRV-237	December 21, 2001

1R22 Surveillance Testing

.1 Steam Generator Stop Valve Dump Valve Surveillance Test

01-OHP 4024.113	Annunciator #113 Response: Steam Generator 1 and 2	Revision 6
01-OHP 4030.STP.018	Steam Generator Stop Valve Dump Valve Surveillance Test	Revision 14
01-OHP 4030.STP.019F	Steam Generator Stop Valve Operability Test	Revision 3
Technical Data Book Figure 19.1-1	Stroke Times by Valve	Revision 60
UFSAR Section 14.3.4.4.2.1	Pipe Break Blowdown Spectra and Assumptions	Revision 17.1
UFSAR Table 14.2.5-2	Time Sequence of Events Double Ended Rupture Inside Containment With Offsite Power Available	Revision 16.4
JO R0071578	Perform **12-EHP 4030.STP.257, Steam Generator Stop Valve ESF Test	December 16, 2000

.2 Unit 2 Daily and Shiftly Surveillances

	D. C. Cook Nuclear Plant Unit 1 and Unit 2 Technical Specifications	
02-OHP 4030.STP.030	Daily and Shiftly Surveillance Checks	Revision 38

.3 Unit 1 Nuclear Instrumentation Functional Checks

TS 3.3.1.1	Reactor Trip System Instrumentation	Amendment 202
01-IHP 4030.SMP.131	Power Range Nuclear Instrumentation Functional Test and Calibration	Revision 0

.4 Unit 1 Ice Condenser Tour

TS 3.6.5.3	Ice Condenser Doors	Amendment 144
PMP 4010.CAC.001	Containment Access Control	Revision 1
02-OHP 4030.STP.030 Data Sheet 19	Ice Condenser Tour Data Sheet	Revision 38

20S1 Access Control to Radiologically Significant Areas

Condition Reports

CR 01003029	Declining Trend in High Radiation Area Controls	January 3, 2001
CR 01009041	Exposure of Personnel to Unanticipated High Radiation Area	January 5, 2001
CR 01147002	Posting for High Radiation Area Found Missing	May 5, 2001
CR 1278044	High Radiation Area Found During Surveillance	October 5, 2001

Procedures and Surveillance Records

PMI 4090	Criteria for Conducting Infrequently Performed Tests or Evolutions	Revision 6
PMI 6010	Radiation Protection Plan	Revision 11b
PM -6010. ALA.001	ALARA Program - Review of Plant Work Activities	Revision 11
PM -601.RPP.-003	High, Locked High, and Very High Radiation Area access	Revision 10

RP-014-01	Total Effective Dose Equivalents, Calculation Data Sheet, 2-FTPL-Upender Re-work	Revision 0, C1
THG.015	RP Job Coverage Coordinator (JCC) Expectations	Revision 1
12-THP 6010.RPP.006, Data Sheet 1	Radiation Work Permit (RWP) Processing, Task 01 and 02, Pre-job ALARA Briefing Checklist	Revision 17
12-THP 6010.RPP.018	Controls for Radiological Risk Significant Work Activities	Revision 0
12-THP-6010.RPP.018, Data Sheet 1	Radiological Risk Significant Work Brief Checklist	Revision 0
12-THP 6010.RPP.018, Data Sheet 3	ALARA Plan Template, Dive Repair of U-2 Upender Clevis	Revision 0
12-THP 6010.RPP.018, Data Sheet 5	Pre-Dive Checklist	Revision 3
12-THP 6010.RPP.413	Radiological Controls for Nuclear Diving Operations	Revision 3a
12-THP 6010.RPP.413, Data Sheet 1	Radiological Controls for Nuclear Diving Operations, Pre-Dive Planning and Setup Checklist	Revision 3a
12-THP 6010.RPP.413, Data Sheet 1	Radiological Controls for Nuclear Diving Operations, RP Pre-Dive Checklist	Revision 3
12-THP 6010.RPP.703	Monitor Alarm Response and Personnel Decontamination, Log Sheets for CY 2001	Revision 10
12-THP 6020.CSP.203	BORAL Surveillance Program	Revision 1
RWP 01-1047	Perform Dive Activities in the Fuel Transfer Canal	Revision 1
	Radiation Protection ALARA Plan, Fuel Transfer Canal Dive, Re-work Upender Cables/Clips	Revision 0

Miscellaneous Data

TS 6.12	High Radiation Area	Amendment 245
	BORAL Coupon Tree Sampling, IPTE Briefing Guide	November 14, 2001

	Operations Night Orders	November 14, 2001
	Radiation Protection Department Key Logs, Previous Twelve Month Records (December 2001 to November 2001)	November 15, 2001
	Spent Fuel Pool Surveys (Pre-job, During, and upon Completion of Dive)	
<u>Self-Assessments</u>		
PA-01-14	Radiation Protection	March 16, 2001
	Field Observation Logs	January through October 2001

2OS3 <u>Radiation Monitoring Instrumentation</u>

<u>Condition Reports</u>		
CR P-99-25781	Errors in USAR/Tech Specs Documentation	October 21, 1999
CR P-99-29165	USAR Contains Inconsistent Alarm Values	December 15, 1999
CR 01143016	Inaccurate Test Results from PASS Hydrogen Analyzer	May 23, 2001
<u>Procedures</u>		
CH-O-706A	PAS Sampling (PH, 02,Count., ATM) Training Qualification Matrix	November 14, 2001
CH-O-706B	PAS Sampling (H2, TG, B) Training Qualification Matrix	November 14, 2001
CH-O-706C	PAS Sampling (Back-up PAS Sampling) Training Qualification Matrix	November 14, 2001
12-THP 6010.RPC.552	Calibration of the DMC-2000 Electronic Dosimeter	Revision 1
12-THP 6010.RPC.552, Data sheet 1	Calibration of the DMC-2000 Electronic Dosimeter, EDs #165618 and #162674	Revision 1
12-THP 6020.PASS.612	PASS Dilute Liquid Sampling	Revision 0
<u>Miscellaneous Data</u>		
TS 6.12	High Radiation Area	Amendment 245
TS 6.8.3	PASS Requirements	Amendment 210

UFSAR Section 7.8	Post-Accident Monitoring Instrumentation	July 1992
UFSAR Section 11.3.3	Radiation Monitoring, PASS Instrumentation	July 1997
TS 3/4.3.3	Monitoring Instrumentation	Amendment 60

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs

Condition Reports

CR 00243086	Vendor Having Limited Capability to Analyze REMP Samples	August 29, 2000
CR01110029	Vendor for Analyzing REMP Samples Still Having Limited Capability to Analyze Samples	April 20, 2001
CR 01235021	REMP Air Sampler Exhaust Needs to Be Re-routed	
CR 01312052	Vegetation Around REMP Air Sampling Location Affecting Sample Results	
CR01136059	Potentially Contaminated "Out of Calibration Period" Gauge, Sent to Vendor	May 8, 2001

Procedures and Surveillance Records

PMP 6010 OSD .001	Off-site Dose Calculation Manual	Revision 16
PMP-6010-RPP-301	Control of Material in a Restricted Area	Revision 14
RP-TB-001	Evaluation of the Use of the Bicron NE Small Article Monitor (SAM-11) for Unconditional Release of Material from a Restricted Area	Revision 0
12IHP6030.IMP.333	Meteorological Instrumentation Calibration	Revision 3 CS-1
12-THP-6010-RPP-301	Radiation Protection Actions for Restricted Area Material Control	Revision 0
12-THP-6010-RPP-514	Calibration of the AVS-28A with the AVT-100 Air Volume Totalizer	Revision 2
12-THP-6010-RPP-630	Collection of REMP Surface Water Samples	Revision 2b

12-THP-6010-RPP-632	Collection of Environmental Samples	Revision 4a
12-THP-6010-RPP-642	Collection of Drinking Water Samples	Revision 2
Miscellaneous Data		
D.C. Cook Nuclear Plant Sample collection data sheets	REMP Air Sample Pump Calibrations	CY 2001
12IHP6030.IMP.333, data sheets	Meteorological Instrumentation Calibration, Primary/Backup Instrumentation	July 17, 2000 to October 10, 2001
D.C. Technical Specifications, Administrative controls Paragraph 6.0	Radiological Environmental Monitoring Program	Amendment 245
Self-Assessments and Field Observations		
PA-99-06/NSDRC #266	Radiological Environmental Monitoring Program (REMP)/Off-site Dose Calculation Manual	June 2, 1999
PA-00-07/NSDRC 277	Radiological Environmental Monitoring Program(REMP)/Off-site Dose Calculation Manual (ODCM)	May 26, 2000

3PP4 Security Plan Changes

Revision 1	Cook Nuclear Plant Security Training and Qualification Plan	October 31, 2001

4OA1 Performance Indicator Verification

.1 Unit 2 Turbine Driven Auxiliary Feedwater Pump Fault Exposure

12-MHP 5021.056.007	Turbine Driven Auxiliary Feed Pump Trip and Throttle Valve Linkage	Revision 2, CS 4 Revision 2, CS 5
AR 0107471	Adjust trip and throttle valve on Unit 2 TDAFP	January 8, 1997
JO C0052930	2-DCP-617, Rework TDAFP Turbine Governor	June 14, 2000
CR 01222001	Unit 2 TDAFP failed to start on two consecutive start attempts	August 10, 2001

CR 01354104	Document prompt operability determination for Unit 1 and 2 TDAFWP trip throttle valve latch faces	December 20, 2001
CR 01362027	NRC identified that a condition report was not written to document the June 2000 failure of the Unit 2 TDAFWP trip throttle valve	December 28, 2001
VTD-SKIN-0001	Schutte and Koerting Installation and Operating Instructions for Motor Operated Trip Throttle Valve	
	Unit 2 Control Room Logs	
EPRI TR 105874	Terry Turbine Maintenance and Troubleshooting Guide	

.2 Safety System Unavailability

	D. C. Cook Nuclear Plant Unit 1 and Unit 2 Technical Specifications	
NEI [Nuclear Energy Institute] 99-02	Regulatory Assessment Performance Indicator Guideline	Revision 1
Plant Manager's Procedure 7110.PIP.001	Regulatory Oversight Program Performance Indicators	Revision 1
	D. C. Cook Maintenance Rule Database Two-Year Unavailability Report for the Emergency Diesel Generators System	December 7, 2001
	D. C. Cook Maintenance Rule Database Two-Year Unavailability Report for the Emergency Core Cooling and Residual Heat Removal Systems	December 7, 2001
	D. C. Cook Maintenance Rule Database Two-Year Unavailability Report for the Auxiliary Feedwater System	December 7, 2001
	Daily Shift Manager's Logs	October 1, 2000 through September 1, 2001

Condition Report (CR) 01029040	Action Request Generated to Document Basis for Not Counting Unavailability Time When Rolling an Engine Over to Check for Moisture in the Cylinders	January 29, 2001
CR 01355058	NRC Identified Inconsistent Reporting of Unavailable Hours for the Maintenance Rule and the Reactor Oversight Process for the Same Conditions	December 21, 2001
CR 01355064	NRC Identified Emergency Diesel Generator Unavailable Hours Are Not Being Reported During Carbon Dioxide Fire Suppression Testing	December 21, 2001
CR 01355071	NRC Identified Safety System Unavailable Hours Reported in the Reactor Oversight Process for the 4th Quarter 2000 and 1st Quarter 2001 Were Reported for the Wrong Train	December 21, 2001
CR 01355072	NRC Identified Hours Reported in the Reactor Oversight Process for the Auxiliary Feedwater System Did Not Account for the Appendix R Safety Function When the Opposite Unit Was in Mode 3 or Above	December 21, 2001

.3 Occupational Exposure Control Effectiveness and Radiological Effluent Technical Specification (RETS)/ODCM Radiological Effluent Occurrence PIs

PMP 7110.PIP.001	Regulatory Oversight Program Performance Indicators	Revision 1
PMP 7110.PIP.001, Data sheet 14	Regulatory Oversight Program Performance Indicators, "Occupational Exposure Control Effectiveness" Documentation Packets, CY 2000, 4th Quarter, CY 2001, 1st , 2nd , and 3 rd Quarter(s),	Revision 0
PMP 7110.PIP.001, Data sheet 15	Regulatory Oversight Program Performance Indicators, "RETS/ODCM Radiological Effluent Occurrences Exposure Control Effectiveness" Documentation packets, CY 2000, 4th Quarter	Revision 0

	Performance Indicator Verification Summary Sheets, "Occupational Exposure Control Effectiveness, Effluent Water dose-Mixed Fission Products, and Effluent Airborne Dose-Total Body"	November 11, 2001

4OA3 Event Follow-up

50-315/2000-007; 50-315/2000-007-01	Licensee event reports: SF Ventilation System Inoperable Due To Technical Specification Surveillance Test Methodology.	October 19, 2000; August 2, 2001
P-00-11175	OE11256 - Control Room emergency Filtration Inoperable due to Testing Methodology	August 10, 2000

4OA7 Licensee Identified Violations

CR 01323024	Technical Specification 3.9.7 violation due to movement of rod control cluster assembly handling tool	November 19, 2001
CR 01343015	Discovered emergency diesel generator Unit 1 "B" Train voltage potentiometer settings incorrect	December 9, 2001