UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

February 14, 2005

Randall K. Edington, Vice
President-Nuclear and CNO
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 05000298/2004005

Dear Mr. Edington:

On December 31, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Cooper Nuclear Station. The enclosed integrated inspection report documents the inspection findings which were discussed on January 6, 2005, with Mr. S. Minahan, General Manager of Plant Operations, 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 NRC identified four findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC also determined that there were four violations associated with these findings. However, because these violations were of very low safety significance and the issues were entered into the licensee's corrective action program, the NRC is treating these findings as noncited violations (NCVs), consistent with Section VI.A.1 of the NRC's Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the subject or significance of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper Nuclear Station facility.

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

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Michael C. Hay, Chief
Project Branch C
Division of Reactor Projects

Docket: 50-298
License: DPR-46

Enclosure:
NRC Inspection Report 05000298/2004005
w/attachment: Supplemental Information

cc w/enclosure:
Michael T. Boyce, Nuclear Asset Manager
Nebraska Public Power District
1414 15th Street
Columbus, NE 68601

John C. McClure, Vice President
and General Counsel
Nebraska Public Power District
P.O. Box 499
Columbus, NE 68602-0499

P. V. Fleming, Licensing Manager
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

Michael J. Linder, Director
Nebraska Department of
Environmental Quality
P.O. Box 98922
Lincoln, NE 68509-8922

Chairman
Nemaha County Board of Commissioners
Nemaha County Courthouse
1824 N Street
Auburn, NE 68305

Sue Semerena, Section Administrator
Nebraska Health and Human Services System
Division of Public Health Assurance
Consumer Services Section
301 Centennial Mall, South
P.O. Box 95007
Lincoln, NE 68509-5007

Ronald A. Kucera, Deputy Director
for Public Policy
Department of Natural Resources
P.O. Box 176
Jefferson City, MO 65101

Jerry Uhlmann, Director
State Emergency Management Agency
P.O. Box 116
Jefferson City, MO 65102-0116

Chief, Radiation and Asbestos
Control Section
Kansas Department of Health
and Environment
Bureau of Air and Radiation
1000 SW Jackson, Suite 310
Topeka, KS 66612-1366

Daniel K. McGhee
Bureau of Radiological Health
Iowa Department of Public Health
401 SW 7th Street, Suite D
Des Moines, IA 50309

William J. Fehrman, President
and Chief Executive Officer
Nebraska Public Power District
1414 15th Street
Columbus, NE 68601

Jerry C. Roberts, Director of
Nuclear Safety Assurance
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

Chief Technological Services Branch
National Preparedness Division
Department of Homeland Security
Emergency Preparedness & Response Directorate
FEMA Region VII
2323 Grand Boulevard, Suite 900
Kansas City, MO 64108-2670

Electronic distribution by RIV:
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DRP Director **(ATH)**
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket.:	50-298
License:	DPR-46
Report:	05000298/2004005
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	September 24 through December 31, 2004
Inspectors:	S. Schwind, Senior Resident Inspector S. Cochrum, Senior Resident Inspector (temporary) D. Carter, Health Physicist P. Elkmann, Emergency Preparedness Inspector G. Replogle, Senior Reactor Inspector
Approved By:	M. Hay, Chief, Branch C, Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000298/2004005; 09/24/04 - 12/31/04; Cooper Nuclear Station, Fire Protection, ALARA Planning and Controls and other activities.

The report covered a 3-month period of inspection by resident inspectors and region-based inspectors. Four Green noncited violations and one Green finding were identified. The significance of the issues is indicated by their color (Green, White, Yellow, or Red) and was determined by the significance determination process in Inspection Manual Chapter 0609. Findings for which the significance determination process does not apply are indicated by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of Technical Specification 5.4.1.d for failure to implement the station’s fire watch procedure. Specifically, on October 22, 2004, the inspectors identified that a compensatory fire watch, responsible for protecting equipment important to safety from fire damage, was not alert and therefore was inattentive to the areas assigned as directed by procedural requirements.

  This finding was considered more than minor since the finding would become a more significant safety concern if left uncorrected. The finding was determined to be of very low safety significance, since the finding was assigned a moderate fire protection barrier degradation rating and did not degrade the automatic water-based fire suppression system in the fire area. This finding had crosscutting aspects associated with problem identification and resolution due to the licensee’s failure to enter this condition into the corrective action program until prompted by the inspectors approximately 10 days following its identification (Section 1R05).

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, involving the failure to promptly identify and correct conditions adverse to quality. Specifically, on numerous occasions the licensee failed to promptly identify that environmental temperatures outside design specifications could potentially affect the function of equipment important to safety. As a result, the licensee failed to promptly evaluate this adverse condition in a timely manner. The failure to promptly identify and correct this condition adverse to quality involved crosscutting aspects associated with problem identification and resolution.

  The inspectors determined that the issue had more than minor safety significance because it impacted the mitigating systems cornerstone objective and could have affected the ability of safety-related systems to perform their design basis functions. The finding was of very low risk significance because it was a design/qualification deficiency that did not result in a loss of function per Generic Letter 91-18, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," Revision 1 (Section 4OA5).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a noncited violation of Technical Specification 5.7.1, since the licensee failed to barricade and conspicuously post a high radiation area. On November 30, 2004, the inspector identified piping located in the Residual Heat Removal B heat exchanger room that had dose rates elevated to greater than 100 millirem per hour. The licensee performed a survey and confirmed dose rates were 600 millirem per hour on contact with the pipe and 160 millirem per hour at 12 inches from the pipe. The area was immediately barricaded and posted. The licensee entered this issue into its corrective action program.

  This finding is greater than minor because it was associated with the cornerstone attribute (exposure control) and affected the cornerstone objective because failure to post a high radiation area with dose rates greater than 100 millirem per hour could increase the risk of personnel dosage. The finding was of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose (Section 2OS2).

- Green. The inspector reviewed a self-revealing, noncited violation of Technical Specification 5.7.1 because the licensee failed to provide an individual a radiation monitoring device that could be detected when a preset integrated dose alarm was received. On December 15, 2003, an individual unknowingly exceeded the alarm setpoint of a required electronic dosimeter while working in an area with radiation levels as high as 200 millirem per hour. The electronic dosimeter was set to alarm at 20 millirem, but upon exiting the area, the electronic dosimeter read 31 millirem and was alarming. The individual did not hear the alarm until the area was exited. The licensee entered this issue into its corrective action program.

  This finding is greater than minor because it was associated with the cornerstone attribute (exposure control) and affected the cornerstone objective because the inability to detect an alarming device in a high radiation area could increase personnel dose. The finding was of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. This finding also had crosscutting aspects associated with human performance (Section 2OS2).

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

The plant was operating at full power at the beginning of this inspection period. On October 19, 2004, the reactor was shut down due to elevated main turbine rotor vibrations. Following repair of the main turbine on November 10, 2004, full power operations resumed for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors completed an inspection sample of licensee activities involving preparations for cold weather conditions on two risk significant systems. These activities included:

- A review of maintenance work orders completed to prepare the systems for cold weather conditions

- A review of deficiency tags and condition reports associated with cold weather protection measures to determine their impact on the systems

- A walkdown of Emergency Diesel Generator (EDG) 2 to verify proper ventilation alignments were implemented

- A walkdown of the ventilation screens in the intake structure to verify that the licensee had completed the required actions identified in the work orders

The two risk significant systems evaluated during this inspection included:

- Portions of the EDG 2 system

- The intake structure and environmental controls located in the service water pump room

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Partial Equipment Alignment Inspections

a. Inspection Scope

The inspectors performed two partial equipment alignment inspections. The inspections verified that critical portions of the selected systems were correctly aligned in accordance with system operating procedures. The following two equipment alignment inspections were performed:

- EDG 1, while EDG 2 was inoperable during cleaning and coating of Diesel Fuel Storage Tank 2 on October 25, 2004. The walkdown included accessible portions of the system in the diesel generator room as well as temporary diesel fuel tanks, hoses, and other equipment staged to support operability of EDG 1 during this work. The inspectors also performed an as-found inspection of Diesel Fuel Storage Tank 2 to assess its condition and an as-left inspection prior to refilling the tank.

- EDG 2, while EDG 1 was inoperable during cleaning and coating of the Diesel Fuel Storage Tank 1 on November 1, 2004. The walkdown included accessible portions of the system in the diesel generator room as well as temporary diesel fuel tanks, hoses, and other equipment staged to support operability of EDG 2 during this work. The inspectors also performed an as-found inspection of Diesel Fuel Storage Tank 1.

a. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Walkdowns

a. Inspection Scope

The inspectors performed six fire zone inspections to verify the licensee was maintaining those areas in accordance with the fire hazards analysis. The fire zones were chosen based on their risk significance as described in the individual plant examination of external events. The walkdowns focused on control of combustible materials and ignition sources, operability and material condition of fire detection and suppression systems, and the material condition of passive fire protection features. The following fire zones were inspected:

- Fire Zone 1F/G, Control and computer rooms
- Fire Zone 2A, Control rod mechanism North

- Fire Zone 3A/B, Critical switchgear room
- Fire Zone 8D, Control Building Elevation 903
- Fire Zone 8F/8G, Division 1 battery room and DC switchgear room
- Fire Zone 20A/B, SW pump room

b. Findings

Introduction. The inspectors identified a noncited violation (NCV) of Technical Specification (TS) 5.4.1.d for failure to implement the station's fire watch procedure. Specifically, on October 22, 2004, the inspectors identified that a compensatory fire watch, responsible for protecting equipment important to safety from fire damage, was not alert and therefore was inattentive to the areas assigned as directed by procedural requirements.

Description. On October 22, 2004, the inspectors conducted an inspection of the fire protection features in the northeast section of the reactor building 903 level. The inspectors identified that a compensatory fire watch, assigned to watch for fires in this area, was not alert nor attentive to the area assigned. Following questioning by the inspectors, the fire watch stated he was tired and therefore was not attentive to assigned fire watch duties. The inspectors discussed the requirements of Administrative Procedure 0.39, "Fire Watches," Section 6.3.1, with the fire watch. The section states, in part, that the fire watch shall observe the affected area and be alert for signs of fire, smoke, and changing conditions. The inspectors then informed the shift manager and outage manager who confirmed the fire watch appeared very tired and not alert and had the watch relieved.

Analysis. The failure to implement the procedural requirements of Administrative Procedure 0.39, "Fire Watches," Revision 31, was considered a performance deficiency which affected the mitigating systems cornerstone since compensatory fire watches are used throughout the plant to protect safety-related equipment when fire protection systems are degraded. This finding was considered more than minor since the finding would become a more significant safety concern if left uncorrected. Inspection Manual Chapter 0609, "Significance Determination Process," Appendix F, was used to assess the safety significance of this finding. Based on the results of a significance determination process Phase 1 evaluation, the finding was determined to have very low safety significance (Green) since the finding was assigned a moderate fire protection barrier degradation rating and did not degrade the automatic water-based fire suppression system in the fire area.

This finding had crosscutting aspects associated with problem identification and resolution. This assessment was based on the fact that the licensee failed to enter this condition into the corrective action program until prompted by the inspectors approximately 10 days later.

Enforcement. TS 5.4.1.d states, "Written procedures shall be established, implemented, and maintained covering the fire protection program." Administrative Procedure 0.23, "CNS Fire Protection Plan," Revision 41, Section 3.3, states, "Fire Watches are controlled by Procedure 0.39." Administrative Procedure 0.39, "Fire Watches," Revision 31, Section 6.3.1, states, in part, that the fire watch shall observe the affected area and be alert for signs of fire, smoke, and changing conditions. Contrary to this requirement, the fire watch failed to observe the affected area and remain alert to fire, smoke, and changing conditions. Because this violation was of very low safety significance and was entered into the corrective action program as Condition Report CR-CNS-2004-07109, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000298/2004005-01, Failure to Implement the Station Fire Watch Procedure.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On December 7, 2004, the inspectors observed one session of licensed operator requalification training in the plant simulator. The training evaluated operator ability to recognize, diagnose, and respond to a loss of dc power and a reactor scram. Observations were focused on the following key attributes of operator performance:

- Crew performance in terms of clarity and formality of communications
- Ability to take timely and appropriate actions
- Prioritizing, interpreting, and verifying alarms
- Correct implementation of procedures, including the alarm response procedures
- Timely control board operation and manipulation, including high-risk operator actions
- Oversight and direction provided by the shift supervisor, including the ability to identify and implement appropriate TS requirements, reporting, emergency plan actions, and notifications
- Group dynamics involved in crew performance

The inspectors also verified that the simulator response during the training scenario closely modeled expected plant response during an actual event.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed one equipment performance issue to assess the licensee's implementation of their maintenance rule program. The inspectors verified that components which experienced performance problems were properly included in the scope of the licensee's maintenance rule program and that the appropriate performance criteria were established. Maintenance rule implementation was determined to be adequate if it met the requirements outlined in 10 CFR 50.65 and Administrative Procedure 0.27, "Maintenance Rule Program," Revision 15. The inspectors reviewed the following equipment performance problem:

- Failure of breakers related to Lighting Panel EE-PNL- LPIS1 (CR 2004-07124)

b. Findings

No findings of significance were identified

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

a. Inspection Scope

The inspectors reviewed two risk assessments for planned or emergent maintenance activities to determine if the licensee met the requirements of 10 CFR 50.65(a)(4) for assessing and managing any increase in risk from these activities. Evaluations for the following maintenance activities were included in the scope of this inspection:

- Corrective maintenance on the reactor core isolation cooling system to replace a fuse and fuse holder on September 30, 2004 (CR 2004-06582)
- Surveillance procedure on Residual Heat Removal (RHR) Loop B requiring shutdown cooling to be secured on November 4, 2004 (Work Order 4360509)

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Evolutions (71111.14)

a. Inspection Scope

For the nonroutine event described below, the inspectors reviewed operator logs, plant computer data, and strip charts to determine what occurred, how the operators responded, and whether the response was in accordance with plant procedures.

- On October 18, 2004, the inspectors responded to the control room following identification of elevated main turbine bearing vibrations. Based on the elevated vibrations, operators performed a normal shutdown to inspect the main turbine blades. The inspectors observed and evaluated the reactor shutdown, followup actions by the operators, actions required by procedures, and monitoring of plant conditions.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed three operability determinations associated with mitigating system capabilities to ensure that the licensee properly justified operability and that the component or system remained available so that no unrecognized increase in risk occurred. These reviews considered the technical adequacy of the licensee's evaluation and verified that the licensee considered other degraded conditions and their impact on compensatory measures for the condition being evaluated. The inspectors referenced the Updated Safety Analysis Report, TSs, and the associated system design criteria documents to determine if operability was justified. The inspectors reviewed the following equipment conditions and associated operability evaluations:

- Reactor equipment cooling system leakage (CR 2004-06015)
- Rod block monitor setpoint error (CR 2004-06893)
- RHR Valve MO-13D failure to close (CR 2004-06776)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors performed a review of all open operator workaround items to evaluate their cumulative effect on mitigating systems and the operator's ability to implement abnormal or emergency procedures. In addition, open operability determinations and selected condition reports were reviewed, and operators were interviewed to determine if there were additional degraded or nonconforming conditions that could complicate the operation of plant equipment.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed plant modification Change Evaluation Document (CED) 6008140 that replaced the reactor feedwater startup valves.

b. Findings

Introduction. A Green self-revealing finding was identified involving the failure to perform an adequate design change for the reactor feed system startup flow control valves. The inadequate design change failed to ensure component temperature ratings were not exceeded resulting in adversely affecting valve operation. Specifically, the licensee's evaluation failed to recognize and address acceptable O-ring types for the temperatures of the reactor feed system.

Description. In September and October of 2004, several reactor feed pump (RFP) trouble alarms, related to startup flow control valve abnormal operation, were received in the control room. These valves are used to control feed water flow to the reactor vessel during startup, cooldown, and depressurization. Based on initial testing and operation following the alarms, the licensee determined that failed valve positioners were causing the alarms. Both valve positioners were replaced September 25, 2004, however, during testing the valves did not cycle as expected. During the investigation into the cause of valve positioner failures, the licensee discovered the valve piston o-ring seal was degraded, failing to provide the required seal. As a result, the valves would operate erratically and deviate from normal demand.

The licensee's apparent cause investigation discovered that the actuator O-rings were square in shape indicating they had taken a permanent set due to exceeding the O-ring temperature rating. The vendor manual states the temperature limit for this type of actuator with nitrile O-rings is 175EF. During normal operation, the reactor feedwater temperatures flowing thought the startup flow control valves reach as high as 360EF exceeding the nitrile O-ring 175EF limit. In October of 2002, the licensee implemented Modification CED 6008140, installing the inappropriate O-rings for the reactor feed system startup flow control Valves RF-AOV-FCV11AA and RF-AOV-FCV11BB.

Analysis. The inadequate design review of the RFP startup flow control valve modification (CED 6008140) was considered a performance deficiency, which affected the Mitigating Systems cornerstone. This finding is greater than minor because it affected both the Initiating Events and Mitigating Systems cornerstone attribute of design control, reducing the reliability and capability of the RFP startup flow control valves to mitigate events or potentially result in an initiating event based on loss of feedwater control to the reactor vessel. This issue is unresolved for significance determination and the appropriate regulatory characterization (URI 05000298/2004005-02, Review Safety Significance or Degraded Startup Flow Control Valves).

Enforcement. No violation of regulatory requirements occurred because the RFP startup flow control valves are not classified as safety-related. The licensee entered this finding into their corrective action program as Condition Report CR-CNS-2004-06997. This finding is identified as FIN 05000298/2004005-02, Inadequate Design Review of System Modification.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed or observed four selected postmaintenance tests (four inspection samples) to verify that the procedures adequately tested the safety function(s) that were affected by maintenance activities on the associated systems. The inspectors also verified that the acceptance criteria were consistent with information in the applicable licensing basis and design basis documents and that the procedures were properly reviewed and approved. Postmaintenance tests for the following maintenance activities were included in the scope of this inspection:

- Emergency Diesel Generator 2 fuel oil pump replacement (Work Order 4376678)
- Emergency Diesel Generator 2 fuel oil strainer cleaning and inspection (Work Order 4384984)
- High pressure coolant injection exhaust drip leg drain extension (Work Order 4327601)
- RHR system Valve RHR-MO-13D control switch replacement (Work Order 4406741)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors observed outage-related activities during a forced outage following a main turbine blade failure on October 18, 2004. Activities included reactor shutdown, plant cooldown, placing the RHR system in the shutdown cooling mode of operation, and startup activities.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed or reviewed the following four surveillance tests (four inspection samples) to ensure that the systems were capable of performing their safety function and to assess their operational readiness. Specifically, the inspectors verified that the following surveillance tests met TS requirements, the Updated Safety Analysis Report, and licensee procedural requirements:

- 6.HPCI.103, “HPCI IST and 92 Day Test Mode Surveillance Operation,” Revision 26, performed on Oct 15, 2004

- 6.1DG.105, “Diesel Generator Starting Air Compressor Operability (IST) (DIV 1),” Revision 13C1, performed on October 4, 2004

- 6.2RHR.706, “RHR Loop B Injection Valve Time Delay Channel Functional Test (DIV 2),” Revision 2, performed on November 4, 2004

- 6.DWLD.302, “Drywell Floor Drain Sump 1F Flow Loop Channel Calibration,” Revision 6, performed on December 14, 2004

b. Findings

No findings of significance were identified.

Cornerstones: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspector performed an in-office review of Revision 48 to the Cooper Nuclear Station Emergency Plan, submitted October 20, 2004, and Revision 49 to the Cooper Nuclear Station Emergency Plan, submitted October 25, 2004. This review included the following changes that had been made to the plan:

- Revised the physical description of installed meteorological instruments because of replacement of the instruments, along with their associated ranges, tolerances, and accuracy

- Revised the description of the emergency notification system from the Emergency Broadcast System to the Emergency Alerting System, with associated details related to system activation

- Revised the location from which the emergency notification system is activated in Atchinson County, Missouri, from the Sheriff's Department to the 911 Center

- Revised the site emergency preparedness training guide from a guide document to a procedure

- Removed emergency preparedness from the site Quality Assurance for Operation Policy Document

- Revised the titles of several procedures.

The revision was compared to: its previous revision; the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1; the criteria of ANSI/ANS-2.5-1984, "American National Standard for Determining Meteorological Information at Nuclear Power Sites;" the criteria of Federal Emergency Management Agency (FEMA) Report REP-10, "Guide for the Evaluation of Alert and Notification Systems for Nuclear Power Plants;" and the requirements of 10 CFR 50.47(b) and 50.54(q) to determine if the revision decreased the effectiveness of the plan.

b. <u>Findings</u>

No findings of significance were identified.

1EP6 <u>Drill Evaluation (71114.06)</u>

a. <u>Inspection Scope</u>

The inspectors observed the licensee perform one emergency preparedness drill on December 9, 2004. Observations were conducted in the control room, technical support center, and emergency operations facility. During the drill, the inspectors assessed the licensee's performance related to classification, notification, and protective action recommendations. Following the drill, the inspectors reviewed the licensee's critique to determine if issues were appropriately identified and documented. The following documents were reviewed during this inspection:

- Emergency Plan for Cooper Nuclear Station

- Emergency Plan Implementing Procedures for Cooper Nuclear Station

- Cooper Nuclear Station Emergency Preparedness Drill Scenario for December 9, 2004

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by TSs as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Current 3-year rolling average collective exposure
- Three on-line maintenance work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures.
- Site-specific trends in collective exposures, plant historical data, and source-term measurements
- Site-specific ALARA procedures
- Intended versus actual work activity doses and the reasons for any inconsistencies
- Integration of ALARA requirements into work procedure and radiation work permit (or radiation exposure permit) documents
- Person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements
- Shielding requests and dose/benefit analyses
- Method for adjusting exposure estimates, or replanning work, when unexpected changes in scope or emergent work were encountered
- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding
- Workers use of the low dose waiting areas

- Exposures of individuals from selected work groups
- Declared pregnant workers during the current assessment period, monitoring controls, and the exposure results

The inspector completed 8 of the required 15 samples and 5 of the optional samples.

b. Findings

(1) Introduction. The inspector identified an NCV of TS 5.7.1 for failure to barricade and conspicuously post a high radiation area.

Description. On November 30, 2004, during walkdowns of the Reactor Building 931-foot elevation, the inspector performed independent radiation measurements and identified dose rates greater than 100 millirem per hour coming from RHR B heat exchanger piping. This area was not posted and barricaded as a high radiation area, and it was accessible from a scaffold platform. The radiation survey tag attached to the scaffold ladder indicated general area dose rates of approximately 20 to 35 millirem per hour. The licensee performed a survey of the area and confirmed dose rates of 600 millirem per hour on contact with the pipe and 160 millrem per hour at one foot from the pipe. The area was immediately barricaded and posted as a high radiation area.

Analysis. The failure to barricade and post a high radiation area is a performance deficiency. This NRC-identified finding is greater than minor because it was associated with a cornerstone attribute (exposure control), and affected the associated cornerstone objective, to ensure the adequate protection of worker's health and safety from exposure to radiation, because not properly controlling high radiation areas could increase personnel dose. The finding involved the potential for an individual's unplanned or unintended dose, which could have been significantly greater as a result of a single minor reasonable alteration of the circumstances. When processed through the Occupational Radiation Safety Significant Determination Process, the finding was of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose.

Enforcement. TS 5.7.1 requires that each area in which radiation levels are in excess of 100 millirem per hour but less than 1000 millirem per hour shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a specific work request. Contrary to this, the licensee failed to barricade and post a high radiation area. Because the failure to barricade and post a high radiation area was determined to be of very low safety significance and has been entered into the licensee's corrective action program as Condition Report CR-CNS-2004-07496, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000298/2004005-03, Failure to Barricade and Post a High Radiation Area.

(2) Introduction. The inspector reviewed a self-revealing NCV of TS 5.7.1 for failure to provide an individual a radiation monitoring device that could be detected when a preset integrated dose alarm was received in a high radiation area.

Description. On December 15, 2003, an individual unknowingly exceeded the alarm setpoint of a required radiation monitoring device (electronic dosimeter) while working in an area with radiation levels as high as 200 millirem per hour. A worker exited the high radiation area (the condenser area) with the electronic dosimeter in alarm. The worker's electronic dosimeter alarm was set at 20 millirem; upon exiting the area the electronic dosimeter was reading 31 millirem. The worker did not hear the electronic dosimeter alarm until the area was exited, and the alarm became self-revealing to the worker.

In addition to being unable to hear the electronic dosimeter alarm, the licensee's apparent cause determination identified that: (1) the worker failed to properly monitor his dose during work, (2) the worker and the radiation protection technician failed to communicate a specific stay time for a job, and (3) the radiation protection technician failed to ensure the proper electronic dosimeter was used for the job, as required by the radiation work permit.

Analysis. The failure to provide a radiation monitoring device that could be detected when it alarms in a high radiation area is a performance deficiency. This self-revealing finding is greater than minor because it was associated with a cornerstone attribute (exposure monitoring), and affected the associated cornerstone objective, to ensure the adequate protection of the worker's health and safety from exposure to radiation, because being unable to detect a radiation alarming device in a high radiation area could increase personnel dose. The finding involved an individual's unplanned or unintended dose. When processed through the Occupational Radiation Safety Significant Determination Process, the finding was of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose. The finding also had crosscutting aspects associated with human performance.

Enforcement. TS 5.7.1 requires that any individual entering an area in which radiation levels are in excess of 100 millirem per hour but less than 1000 millirem per hour shall be provided with a monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. Contrary to this, the licensee failed to provide a monitoring device that can be detected when it alarms. Because the failure to provide a radiation monitoring device that could be detected when it alarms was determined to be of very low safety significance and has been entered into the licensee's corrective action program as Condition Report CR-CNS-2003-02009, this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000298/2004005-04, Failure to Provide a Monitoring Device that could Detect High Radiation in a Work Area.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

The inspectors sampled two licensee PIs listed below for the period October 2003 through September 2004 (two inspection samples). The definitions and guidance of Nuclear Energy Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 2, were used to verify that the licensee accurately reported PI data during the assessment period. Licensee PI data was reviewed against the requirements of Procedure 0-PI-01, "Performance Indicator Program," Revision 16.

Reactor Safety Strategic Area

- Reactor Coolant System Specific Activity
- Reactor Coolant System Leak Rate

The inspectors reviewed a selection of licensee event reports, portions of operator log entries, monthly reports, and PI data sheets to determine whether the licensee adequately collected, evaluated, and distributed PI data for the period reviewed. The inspectors also interviewed licensee personnel responsible for collecting and evaluating PI data.

b. Findings

No findings of significance were identified

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed a selection of condition reports written during the inspection period to verify the licensee was entering conditions adverse to quality into the corrective action program at an appropriate threshold. Additionally, the inspectors verified that condition reports were appropriately categorized and dispositioned in accordance with the licensee's procedures, and in the case of significant conditions adverse to quality, to review the adequacy of licensee root cause determinations, extent of condition reviews, and implemented corrective actions.

b. Findings

No findings of significance were identified.

.2 Occupational Radiation Safety Sample Review

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's problem identification and resolution processes regarding exposure tracking, higher than planned exposure levels, and radiation worker practices. The inspector reviewed the corrective action documents listed in the attachment against the licensee's problem identification and resolution program requirements.

b. Findings

No findings of significance were identified.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors performed a semiannual assessment of trends in the licensee's corrective action program to determine if any more significant safety issues exist. Specifically, the inspectors reviewed the licensee's corrective action program database to determine if the licensee had identified trends in any of the following areas:

- RFP controller alarms
- Reactor equipment cooling leakage results
- Intake structure silting
- Contractor control
- Average power range monitor alarms
- Ronan computer multiplexer failures
- 345kv transformer alarms
- 120v ac lighting breaker tripping
- Service air compressor failures
- Procedure adherence
- Operability determinations

These areas were chosen based on information gathered by the inspectors during the previous 6 months. For those areas where trends were documented in the corrective action program, the inspectors verified that the licensee had corrective actions planned or in place to address the trend. For the remainder of the issues in the scope of this inspection, the inspectors reviewed control room logs, system health reports, Quality Assurance Audits, and department self-assessments and interviewed selected licensee staff to determine if any adverse trends existed.

b. Findings

The inspectors concluded that, in general, the licensee had adequately identified trends in areas within the scope of this inspection; however, these trends were not always explicitly documented in the corrective action program. This was a result of the licensee's practice of closing new condition reports regarding similar equipment issues to an existing condition report which was already open to evaluate the condition. In all cases, the licensee was taking adequate corrective actions to address the trends.

There were six condition reports written during this 6-month period regarding lighting panel breaker tripping related to Panel EE-PNL-LPIS1, which did not cross any statistical thresholds in the licensee's trending program. The inspectors identified several additional lighting breaker trips that were not documented in the correction action program. During a discussion of these events, engineering personnel concluded that a trend may exist. Condition Report CR-CNS-2004-07124 was written to document a potential trend regarding breaker trips related to lighting Panel EE-PNL-LPIS1.

.4 Identification and Resolution of Problems Crosscutting Aspects of Findings

Sections 1R05 and 4OA5 describe findings with crosscutting aspects associated with problem identification and resolution.

4OA4 Human Performance Crosscutting Aspects of Findings

Sections 1R17 and 2OS2 describe findings with crosscutting aspects associated with human performance.

4OA5 Other Activities

(Closed) Unresolved Item 05000298/2004004-05: Plant temperatures outside Updated Safety Analysis Report Limits

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion XVI, involving the failure to promptly identify and correct conditions adverse to quality. Specifically, on numerous occasions the licensee failed to promptly identify that environmental temperatures outside design specifications could potentially affect the function of equipment important to safety. As a result, the licensee failed to promptly evaluate this adverse condition in a timely manner. The failure to promptly identify and correct this condition adverse to quality involved crosscutting aspects associated with problem identification and resolution.

Description. The Cooper Updated Safety Analysis Report states, in part, that the design of the station heating, ventilating, and air conditioning systems are based on a minimum outdoor temperature of -5EF and a maximum outdoor temperature of 97EF. The inspectors reviewed historical plant temperature data and noted that in the last 2 years site temperatures had exceeded the 97 degree limit on 12 occasions and had dropped below the -5EF threshold five times. During this same period plant temperatures were as high as 104EF and as low as -10EF. Plant areas that directly rely on outside air for temperature control during design basis accidents include the emergency diesel generator rooms, portions of the reactor building, the control room, and the control building.

The inspectors determined that engineering personnel failed to follow corrective action program requirements resulting in the failure to promptly identify and correct a condition adverse to quality. Specifically, although engineering was aware of the deviations from the design temperature specifications, this nonconforming condition was not entered into the corrective action program for resolution, resulting in the licensee’s failure to evaluate the potential adverse affect to equipment being subjected to temperatures outside design values.

Analysis. The inspectors determined that the issue had more than minor safety significance because it impacted the mitigating systems cornerstone objective and could have affected the ability of safety-related plant systems to perform their design basis functions. The finding was of very low risk significance because it was a design/qualification deficiency that did not result in a loss of function per Generic Letter 91-18, “Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions,” Revision 1.

This finding had crosscutting aspects associated with problem identification and resolution. This assessment was based on the fact that the licensee failed to enter this known nonconforming condition into the corrective action program.

Enforcement. The inspectors identified a violation of 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Actions), because the licensee had failed to properly identify conditions adverse to quality. The noted regulation requires licensees, in part, to promptly identify and correct conditions adverse to quality. Contrary to this requirement, the inspectors identified 17 instances, in the last 2 years, where outside ambient temperatures had exceeded design specifications for safety-related heating and ventilation systems (conditions adverse to quality) and the licensee had failed to address or evaluate the occurrences. Because the violation was of very low safety significance, and was entered into the licensee's corrective action program (Condition Report 2004-6820), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000298/2004005-05).

4OA6 Meetings, Including Exit

On December 2, 2004, the inspectors presented the ALARA Planning and Controls inspection results to Mr. S. Minahan, General Manager of Plant Operations, and other members of his staff who acknowledged the findings.

On December 13, 2004, the inspectors conducted a telephonic exit meeting of the emergency preparedness inspection results to Mr. J. Bednar, Emergency Preparedness Manager, and other members of his staff who acknowledged the findings.

On January 6, 2005, the inspectors presented the results of the resident inspector activities to Mr. S. Minahan, and other members of his staff, who acknowledged the findings.

The inspectors confirmed that proprietary information was not provided or examined during the inspection.

40A7 Licensee-Identified Violations

The following violation of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

Cornerstone: Barrier Integrity

- TS 5.6.5(a)3 required the licensee to determine the rod block monitor upscale allowable values and document the values in the core operating limits report prior to each reload cycle. Contrary to this requirement, this determination was not completed prior to Reload Cycle 22. The licensee used generic values provided by the vendor vice determining cycle-specific values. This resulted in nonconservative rod block monitor upscale setpoints. This finding affected the Barrier Integrity cornerstone and was of very low safety significance since it did not represent an actual degradation of a fission product barrier. This was identified in the licensee's corrective action program as Condition Report CR-CNS-2004-06893.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Bednar, Emergency Preparedness Manager
C. Blair, Engineer, Licensing
D. Cook, Technical Assistant to General Manager
J. Christensen, Co-Director of Nuclear Safety Assurance
S. Minahan, General Manager of Plant Operations
T. Chard, Radiological Manager
K. Chambliss, Operations Manager
K. Dalhberg, General Manager of Support
J. Edom, Risk Management
R. Estrada, Corrective Actions Manager
J. Flaherty, Site Regulatory Liaison
P. Fleming, Licensing Manager
D. Knox, Maintenance Manager
W. Macecevic, Work Control Manager
J. Roberts, Director, Nuclear Safety Assurance
R. Shaw, Shift Manager
J. Sumpter, Senior Staff Engineer, Licensing
K. Tanner, Shift Supervisor, Radiation Protection
R. Hayden, Emergency Preparedness Staff

NRC Personnel
L. Ricketson, Senior Health Physicist

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000298/2004005-02	URI	Review Safety Significance of Degraded Startup Flow Control Valves (Section 1R17)

Opened and Closed

05000298/2004005-01	NCV	Failure to Implement the Station Fire Watch Procedure (Section 1R05)
05000298/2004005-03	NCV	Failure to barricade and conspicuously post a high radiation area (Section 2OS2)

05000298/2004005-04	NCV	Failure to provide a radiation monitoring device that could detect high radiation in a work area (Section 2OS2)
05000298/2004005-05	NCV	Plant Temperatures Outside Design Specifications (Section 4A05)

Closed

05000298/2004004-05	URI	Plant temperatures outside Updated Safety Analysis Report limits (Section 4A05)

LIST OF DOCUMENTS REVIEWED

Condition Reports

2004-06045	2004-07124	2004-06582	2004-06835
2004-06015	2004-06567	2004-06757	2004-07120

Section 2OS2: ALARA Planning and Controls (71121.02)

Corrective Action Documents

2003-1787, 2003-61136, 2003-7706, 2004-1699, 2004-5224, 2004-5969, 2004-5970, 2004-5973, 2004-6327, and 2004-6360 and 10299462 and 10314221

Audits and Self-Assessments

Snap Shot Assessment dated July 19-23, 2004
Snap Shot Assessment dated May 24, 2004
Snap Shot Assessment dated June 2, 2004

Radiation Work Permits

2003-1111	Condenser tube leak and repair
2004-1047	Condenser water box cleaning
2004-1050	Fan B cooling unit bearing replacement

Procedures

0.ALARA.1	CNS ALARA Program, Revision 3
0.ALARA.2	ALARA Organization and Management, Revision 7
9.ALARA.4	Radiation Work Permits, Revision 4
9.RADOP.3	Area Posting and Access Control, Revision 16
9.ALARA.1	Radiation Protection at CNS, Revision 4
9.ALARA.5	ALARA Planning and Controls, Revision 12
9ALARA.12	Hot Spot Reduction Program, Revision 0
9.EP 3.14	Temporary Shielding, Revision 15

ALARA Committee Meeting Minutes

2003

February 19, 2003
June 6, 2003, ALARA Special Committee Meeting Minutes
1st Quarter 2003
3rd Quarter 2003
September 11, 2003, Water Box Cleaning
4th Quarter 2003

2004
2004-02, 2004-03, 2004-04, 2004-05, 2004-06, 2004-07, 2004-08, 2004-09, 2004-10, 2004-11, 2004-12, and 2004-13

Miscellaneous

2003 ALARA Program and RE21 Review

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
CED	change evaluation document
CFR	*Code of Federal Regulations*
EDG	emergency diesel generator
FIN	finding
NCV	noncited violation
NRC	U.S. Nuclear Regulatory Commission
PI	performance indicator
RFP	reactor feed pump
RHR	residual heat removal
TS	Technical Specification