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

April 28, 2003

Clay C. Warren, Vice President of
Nuclear Energy
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

SUBJECT: COOPER NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 50-298/03-04

Dear Mr. Warren:

On December 29, 2002, through March 29, 2003, 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 April 3, 2003, with Mr. Mike Coyle, Site Vice President, 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 six findings that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC also determined that there were violations associated with five of these findings. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violation or significance of these 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 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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

Sincerely,

/RA/

Kriss M. Kennedy, Chief
Project Branch F
Division of Reactor Projects

Docket: 50-298
License: DPR-46

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NRC Inspection Report
50-298/03-04

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket.:	50-298
License:	DPR 46
Report No.:	50-298/03-04
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	December 29, 2002, through March 29, 2003
Inspectors:	S. Schwind, Senior Resident Inspector S. Cochrum, Resident Inspector J. B. Nicholas, Ph.D., Senior Health Physicist M. P. Shannon, Senior Health Physicist P. Elkmann, Emergency Preparedness Inspector G. Pick, Senior Physical Security Inspector W. Sifre, Reactor Inspector
Approved By:	K. Kennedy, Chief, Branch F, Division of Reactor Projects

ATTACHMENT: Supplemental Information

SUMMARY OF FINDINGS

Cooper Nuclear Station
NRC Inspection Report 50-298/03-04

IR 05000298/03-04; 12/29/02-03/29/03, Cooper Nuclear Station. Integrated Resident/Regional Rpt; Adverse Weather, Fire Prot, Personnel Perf During Nonroutine Evolutions, Operator Workarounds, Ident & Resolution of Problems.

The inspection was conducted by resident inspectors and Region IV specialists. During the inspection the NRC identified five noncited violations and one finding. The significance of issues is indicated by their color (Green, White, Yellow, Red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. Frazil ice conditions were observed on the Missouri River on February 25 as well as a patch of ice on the service water intake trash rack. The licensee was not able to support the claim that the intake structure was not susceptible to ice accumulation during shutdown conditions nor did they have a procedure to address ice accumulation or loss of service water due to blockage of the trash racks. The failure to develop and implement a procedure to cope with an act of nature, such as the accumulation of ice in the intake structure, was determined to be a violation of Technical Specification 5.4.1. This finding was considered more than minor since the formation of ice at the intake structure could reasonably be viewed as a precursor to a significant event.

  This noncited violation was characterized as a “green” finding using the significance determination process. The failure to develop and implement a procedure for ice accumulation had very low safety significance since there was no loss of safety function for the service water system (Section 1R01).

- Green. The failure to implement the requirements of the station’s fire watch procedure was considered to be a Green, noncited violation of Technical Specification 5.4.1.d. The inspectors observed a fire watch who had allowed hot work to commence prior to removing all combustible materials from the area as required by station procedures. Furthermore, the fire watch procedure requires annual requalification training for fire watches. The fire watch in question had not completed this training. This finding was more than minor since failure to implement the fire watch procedure could become more safety significant if left uncorrected.

  This noncited violation was characterized as a “green” finding using the significance determination process. The failure to implement the station’s fire watch procedure had very low safety significance since the fire ignition frequency for the area in question was low and fire mitigation capability (operator action) remained (Section 1R05).

- Green. The failure to evaluate the impact of long-standing equipment problems on operator actions required by an alarm response procedure was determined to be a Green finding. The fire protection backup supply of gland water to the service water pumps had been isolated for more than a year due to various equipment failures. Operator actions during an actual loss of service water pump gland water were complicated as a result of this equipment condition. The licensee had not evaluated this condition as an operator work-around. This finding was considered more than minor since it affected the availability of the service water system.

This was characterized as a “green” finding based on the Mitigating Systems significance determination process. The failure to evaluate the impact of long-standing equipment problems had very low safety significance since mitigation capability remained from the non-safety related source of gland water (Section 1R16).

Cornerstone: Barrier Integrity

- Green. On February 20, the drain valve for Feedwater Heater 4A failed closed, causing a partial loss of feedwater heating. According to station procedures, reactor power should have been reduced below 25 percent within 2 hours following this valve failure. However, power was not reduced until approximately 15 hours after the partial loss of feedwater heating, and then, only after repeated questioning by the inspectors regarding procedural adherence. This was considered to be a violation of Technical Specification 5.4.1 for failure to implement a procedure. This finding was considered more than minor since, if left uncorrected, could have become a more safety significant event. This finding had cross-cutting aspects of human performance since it dealt with procedure adherence.

  This noncited violation was characterized as a “green” finding using the significance determination process. The failure to reduce reactor power had very low safety significance since it only affected one of the three fission product barriers (Section 1R14).

Cornerstone: Occupational Radiation Safety

- Green. A self-revealing noncited violation was identified because the licensee failed to follow the requirements of Technical Specification 5.7.1b. Specifically, a worker failed to wear an alarming dosimeter that could be heard while working in the Steam Jet Air Ejector Room, an area with general radiation levels greater than 100 millirem per hour.

  The failure to wear an alarming dosimeter that could be heard is a performance deficiency. The issue was more than minor because it is associated with a cornerstone attribute (program and process) and affected the occupational radiation safety cornerstone objective (to ensure the adequate protection of the worker’s health and safety from radioactive material). The finding involved the failure to control radiological work that was contrary to Technical Specification requirements. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was found to have very low safety significance because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised (Section 2OS2).

- Green. On February 20, a radiation protection technician and a mechanic entered the steam jet air ejector room, which was a locked high radiation area, to perform spot maintenance on a main steam valve. Continuous coverage of the job by the technician was required due to dose rates in the room. The station's conduct of maintenance procedure prohibited the performance of spot maintenance under these conditions. This was considered to be a violation of Technical Specification 5.4.1 for failure to implement the maintenance procedure. This finding had cross-cutting aspects of human performance since it dealt with procedure adherence. The finding was considered more than minor because it affected a cornerstone objective.

  This noncited violation was characterized as a "green" finding using the significance determination process. The failure to follow a station maintenance procedure had very low safety significance since there was no over-exposure or substantial potential for an over-exposure and the ability to assess dose was not compromised (Section 4OA2).

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 their corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

Report Details

The plant began the inspection period operating at 100 percent power. On February 20, reactor power was reduced to approximately 25 percent due to the failure of a feedwater heater drain valve. The reactor was subsequently shut down on February 23 after the licensee determined that the valve could not be repaired while the plant was operating. With the start of a scheduled refueling outage on March 1, the plant remained shut down through the end 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 reviewed the licensee’s response to frazil ice conditions in the Missouri River during the week of February 25. The inspectors reviewed the design basis of the service water system, abnormal and emergency response procedures, and operations standing orders issued to monitor for the accumulation of frazil ice in the intake structure to determine the susceptibility of the intake structure to ice blockage and to determine if the licensee’s procedures were adequate to monitor for and cope with ice accumulation.

b. Findings

The failure to develop and implement a procedure to cope with possible ice accumulation in the service water intake was considered to be a Green, noncited violation.

Frazil ice occurs when ice crystals form on the surface of a lake or river due to super-cooling of the surface water. Water turbulence, either from wind action or the river current, can mix these ice crystals throughout the entire depth of the water. When the crystals come in contact with a substrate material which is at, or below 32°F, such as a trash rack on an intake structure, the crystals begin to adhere to the material and can eventually lead to a complete blockage of flow through the intake. This phenomena can occur on lakes or rivers and is influenced by, among other factors, air temperature and wind speed. The formation of round “pancake ice” or “pad ice” on the surface of the water is indicative of frazil ice conditions.

On February 25, 2 days after the plant had been placed in cold shutdown, control room operators reported that the water temperature in Service Water Intake Bay E was approximately 32.4°F. The reactor was on shutdown cooling with service water providing the ultimate heat sink. The circulating water (CW) system had been secured and was drained in preparation for maintenance during the refueling outage. As a result, the recirculation of warm water from the CW outlet to the intake structure was not available. Also at this time, the inspectors observed “pad ice” formations on the Missouri River and observed a small area of ice formation on the Bay E trash rack approximately 12 to 18 inches above the waterline. This portion of the trash rack had previously been underwater as the river had crested 2 days earlier on February 23.

In response to the control room operators' concerns about the intake water temperature, several portable, forced air heaters were staged in the intake structure with their discharge directed toward the water surface. In addition, the control room initiated inspections of the service water traveling screens every 3 hours to detect any ice formation. Based on industry operating experience, information on frazil ice formation published by the United States Army Corp of Engineers, and the direct observation of ice on the Bay E trash rack, the inspectors questioned whether the actions taken by the licensee were adequate to prevent ice formation in the intake structure and whether the licensee had abnormal or emergency procedures to cope with an ice blockage of the intake structure.

In response to these questions, the licensee stated that previous evaluations had determined that the intake structure was not susceptible to the accumulation of frazil ice, even without recirculating flow from the CW system. The inspectors were referred to Engineering Study NED-87001, "Cooper Nuclear Station Intake Structure Silting and Icing Concerns," August 4, 1987, and Design Calculation NEDC 87-056, "Service Water Bay E Traveling Water Screen Flow Velocities," Revision 2. Calculation NEDC 87-056 showed that, if the traveling screens were blocked at a rate of 10 square inches per hour, regardless of the blockage mechanism, there would be adequate flow to ensure operability of service water for up to 28 days. The licensee stated that it was unlikely that frazil ice conditions would exist for 28 consecutive days and that there would be adequate time to take compensatory actions to prevent a loss of service water in the event of ice accumulation. However, the licensee was unable to provide a basis for the assumption that ice accumulation on the traveling screens or the trash racks would not exceed 10 square inches per hour. On the contrary, Engineering Study NED-87001 quoted experiments conducted at the University of Iowa which demonstrated that "it would take eight to ten minutes for frazil ice to start to cause substantial blockage." Based on this, and the direct observation of ice on the trash racks on February 25, the inspectors concluded that Cooper Nuclear Station did not have an adequate technical basis to support their claim that the intake structure was not susceptible to the accumulation of frazil ice.

As previously stated, the licensee established compensatory actions by placing forced air heaters in Bay E and by inspecting the traveling screens for ice every 3 hours. However, the United States Army Corp of Engineers Cold Regions Technical Digest No. 91-1, March 1991, titled "Frazil Ice Blockage of Intake Trash Racks," states that heating the air around trash racks alone cannot prevent frazil ice accumulation. The heat gain to the trash rack from the warm air will be balanced by the heat loss to the flowing water at a very small depth. Since frazil ice crystals can exist throughout the entire depth of water, this strategy alone cannot be relied upon to prevent ice accumulation on the entire trash rack. Furthermore, the licensee was only inspecting the traveling screens on a routine basis for ice. The traveling screens are automatically rinsed with a high pressure jet of water which might tend to remove any sign of ice accumulation. The licensee was not routinely inspecting the trash racks which were the only component in the intake structure which had shown signs of ice accumulation. Therefore, the inspectors concluded that the licensee's compensatory actions were inadequate to detect or preclude the accumulation of frazil ice.

The inspectors questioned operators as to any contingency plans should there be an accumulation of ice on the Bay E trash racks. The inspectors were referred to Emergency Procedure 5.2SW, "Service Water Casualties," Revision 7. Ice accumulation on the Bay E trash racks could cause an entry condition into this procedure but not until there was indication of lowering service water header pressure or a service water pump trip, either of which would only occur after significant ice accumulation and would indicate a pending loss of service water. In addition, this procedure did not address a loss of service water due to any type of blockage of the trash racks.

As a result of discussions with the inspectors, the licensee established an operations standing order which described the initial conditions required to form frazil ice and more adequately monitored the traveling screens and trash racks for the accumulation of ice. In addition, Emergency Procedure 5.2SW was revised to address the loss of service water due ice accumulation in the intake structure.

This finding affected the mitigating systems cornerstone and was considered more than minor since the loss of service water due to ice accumulation on the intake structure could be reasonably viewed as a precursor to a significant event. It also involved the quality of an operating procedure which is an attribute of the mitigating systems cornerstone objective. This finding was characterized under the significance determination process as having very low safety significance because there was no loss of function of the service water system.

Technical Specification 5.4.1(a) requires that licensees establish, implement, and maintain written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for combating emergencies and other significant events such as acts of nature. Emergency Procedure 5.2SW, "Service Water Casualties," Revision 7, did not meet this requirement in that it did not address the loss of service water due to blockage of the intake structure. The failure to establish a procedure for this condition is a violation of Technical Specification 5.4.1(a). This violation is being treated as a noncited violation (50-298/0304-001) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Notification 10231011.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed two partial equipment alignment inspections. The inspections verified that the critical portions of the selected systems were correctly aligned per the system operating procedures. The following systems were included in the scope of this inspection:

- Division 1 of the service water system while Division 2 was out of service for planned maintenance on January 22. The walkdown included portions of the system in the intake structure and control building.

- Emergency Diesel Generator 2 while Emergency Diesel Generator 1 was out of service for planned maintenance on March 13. The walkdown included portions of the system in the control room and emergency diesel building.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

.1 Routine Fire Area Walkdowns

a. Inspection Scope

The inspectors performed eight fire zone walkdowns to determine if the licensee was maintaining those areas in accordance with their Fire Hazards Analysis Report. 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 material 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 Area IV - Seal water pump area and corridor
- Fire Area V - Reactor Protection System Room 1A
- Fire Area VI - Battery Room 1B
- Fire Area VII - Auxiliary relay room
- Fire Zone 12C - Condensate and heater bay
- Fire Zone 12E - Turbine oil reservoir
- Fire Zone 14B - Diesel Generator Room 1B
- Fire Zone 6 - Refueling floor

b. Findings

A Green, noncited violation was identified regarding the failure to implement the station's fire watch procedure.

On March 14 the inspectors conducted a walkdown of the fire protection features in the northwest section of the turbine building condensate and heater bay (Fire Zone 12C). During the walkdown, the inspectors observed combustible materials directly under and adjacent to welding activities. When inspectors questioned the assigned fire watch on the presence and location of the combustible materials, the fire watch stated that workers had been instructed to move the materials, but they had not yet done so. When the inspectors asked if the job supervisor had been informed, the fire watch replied in the affirmative, but that the materials still had not been removed. The inspectors noted that hot work continued with combustible materials under and around the welding areas. The inspectors discussed the requirements of Administrative Procedure 0.39, "Fire Watches", Section 3.4, with the fire watch. The section states “Immediately prior to the start of hot work, the fire watch shall ensure moveable combustible materials below and within a 35-foot radius of the hot work have been removed or protected by metal guards or fire blanket." Section 5.3.3 also states that "Watch personnel have the authority and responsibility to prevent the commencement, suspend, or terminate any hot work activity that does not meet the hot work permit requirements or is deemed unsafe." When questioned whether the fire watch was aware of these requirements, the fire watch ordered the hot work stopped until the debris and hazards were removed.

The inspectors requested the fire watch's training record and a copy of the lesson plan for fire watch training. Every 12 months, personnel qualified to stand fire watch were required to complete a computer-based training module as well as a practical demonstration on the use of fire extinguishers. The individual fire watch in question had completed the practical demonstration in 2003 but had not completed the computer-based training module since October 2001. The inspectors also noted that the lesson plan material did not address the authorities and responsibilities of a fire watch stated in Administrative Procedure 0.39, Section 5.3.3.

This finding had crosscutting aspects associated with problem identification and resolution. This assessment was based on the fact that the licensee had opportunities to identify and correct the deficiencies with the training and qualification of fire watches. This finding was very similar to a noncited violation (50-298/0107-01) identified in 2001 where a fire watch failed to implement Section 3.4 of Administrative Procedure 0.39 which resulted in a fire in the reactor building.

The failure to implement the procedural requirements of Administrative Procedure 0.39, "Fire Watches," Revision 27, affected the mitigating systems cornerstone since fire watches are used throughout the plant to protect safety-related equipment during hot work. This finding was considered more than minor since the finding would become a more significant safety concern if left uncorrected. Inspection Manual Chapter (MC) 0609, "Significance Determination Process," was used to assess the safety significance of this finding. Phase 1 of the significance determination process concluded that the finding was potentially risk significant since it affected a system that is used to minimize damage to safety-related equipment from an external event (fire). Therefore, a Phase 2 analysis using MC 0609, Appendix F, was required.

The following assumptions were used during the Phase 2 analysis:

- The condition existed for >30 days. This was a conservative assumption used to establish a bounding case.
- Automatic fire detection or suppression was highly degraded due to fire impairments. Manual detection and suppression were relied upon to minimize damage.
- Manual detection and suppression were assumed to be highly degraded. This was a conservative assumption used to establish a bounding case.
- Fire event in Fire Zone 12C (Condensate and heater bay)

- Maximum transient fuel load in Zone 12C
- Ignition source is welding in Zone 12C
- According to Cooper Nuclear Station's Individual Plant Examination Of External Events, the ignition frequency for a fire in Zone 12C was 3.27E-3/year.

This set of assumptions resulted in a fire mitigation frequency of -4 to -5. Based on this frequency and the fact that fire mitigation capability remained unaffected (operator action), this finding was determined to have very low safety significance (Green).

Technical Specification 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 31, Section 3.1, states that "Ignition sources are controlled through Administrative Procedure 0.39." Administrative Procedure 0.39, "Fire Watches," Revision 27, Section 3.4, states that "Immediately prior to the start of the hot work, the fire watch shall ensure that moveable combustible materials below and within a 35-foot radius of the hot work have been removed or protected by metal guards or fire blankets." Section 5.3.3 states "Watch personnel have the authority and responsibility to prevent the commencement, suspend, or terminate any hot work activity that does not meet the hot work permit requirements or is deemed unsafe." The inspectors determined that the failure of the fire watch to stop the work and ensure that the combustible material below the hot work was either removed or protected from the hot work was a violation of Technical Specification 5.4.1.d. This violation is being treated as a noncited violation (50-298/0304-002) consistent with Section VI.A of the NRC Enforcement Policy. This issue has been entered into the licensee's corrective action process as Notification 10233322.

.2 Annual Fire Drill Inspection

a. Inspection Scope

The inspectors observed the plant fire brigade during an unannounced fire drill on February 5 to assess the licensee's ability to fight fires. The fire was simulated to occur on the roof of the administration building. Observations focused on the following aspects of the drill:

- Protective clothing/turnout gear was properly donned.
- Self-contained breathing apparatus equipment was properly worn and used.
- Fire hose lines were capable of reaching all necessary fire hazard locations, the lines were laid out without flow constrictions, and the hose was simulated as being charged with water.
- The fire area of concern was entered in a controlled manner (e.g., fire brigade members stayed low to the floor and felt the door for heat prior to entry into the fire area of concern).

- Sufficient firefighting equipment was brought to the scene by the fire brigade to properly perform their firefighting duties.

- The fire brigade leader's firefighting directions were thorough, clear, and effective.

- Radio communications with the plant operators and between fire brigade members were efficient and effective.

- Members of the fire brigade checked for fire victims and propagation into other plant areas.

- Effective smoke removal operations were simulated.

- The firefighting preplan strategies were utilized.

- The licensee planned drill scenario was followed and the drill objectives acceptance criteria were met.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors performed a semiannual inspection of internal flood protection features. The control building basement was chosen for this inspection based on its location in the plant and risk significance. The inspection included a review of the Updated Final Safety Analysis Report, selected design criteria documents (DCDs) and design calculations. A walkdown was also conducted to verify that flood protection features in this area were installed and maintained per the following documents:

- Cooper Nuclear Station DCD 36, “High Energy Line Break (HELB)/Moderate Energy Line Break (MELB),” Revision 2

- Cooper Nuclear Station DCD 38, “Internal Flooding System,” Revision 2

- Calculation NEDC 91-37, “High Energy Line Break Flooding Evaluation”

- Calculation NEDC 91-069, “Moderate Energy Line Break Flooding Calcs”

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination (NDE) Activities

a. Inspection Scope

The inspectors requested and reviewed the NDE records for work that was performed during the ongoing outage and verified that the licensee performed the required inspections. The inspectors also observed the following visual, magnetic particle, and ultrasonic examinations:

System/Component	Examination Method
Core Spray A Header	Remote Ultrasonic
Core Spray B Header	Remote Ultrasonic
Residual Heat Removal A Elbow	Magnetic Particle
High Head B Connection CC-53	Magnetic Particle
High Head A Elbow-90	Visual (VT-3)
High Head A Elbow-93	Visual (VT-3)

The inspectors reviewed two weld repairs and two indications that were accepted for continued service to determine if they were performed in accordance with ASME Code requirements.

The inspectors reviewed licensee NDE and contractor personnel qualification and certification records to determine if NDE personnel were certified to perform the above examinations.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

The inspectors performed a detailed review of a sample of condition reports initiated within the past 2 years in the area of inservice inspection activities. The review was conducted to ascertain whether plant personnel were identifying performance issues within the inservice inspection program. This review assessed the effectiveness of cause determination, corrective action, and the adequacy of the plant personnel's effort to identify transportability and generic issues. The review also assessed the effectiveness of the plant personnel's efforts to identify and address programmatic issues within the inservice inspection program.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed four 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:

- Service Water Pump D outage for emergent maintenance during the week of December 30, 2002
- Emergent maintenance to clean and inspect Service Water Strainer A on January 17
- Emergent maintenance on 4160V Bus 1F followed by planned maintenance on the reactor core isolation cooling system on January 17
- Service water intake traveling screen maintenance on February 6

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Evolutions (71111.14)

a. Inspection Scope

On February 20, the inspectors responded to the control room after a report of a failed feedwater heater drain valve which caused a reduction in feedwater temperature and a minor reactor power transient. The inspectors verified that the licensee was operating the plant within the limits specified in the Technical Specifications and observed the actions taken by operators as they implemented the applicable abnormal procedures.

b. Findings

A Green, noncited violation was identified regarding the failure to reduce reactor power below 25 percent as required by plant procedures.

On February 20, at approximately 3:30 a.m., control room operators received high level alarms on Feedwater Heaters 4A and 5A and a low level alarm on Feedwater

Heater 3A. Feedwater temperature decreased from approximately 370°F to 355°F, as indicated by Temperature Indicator RF-TI-1 on the main control board, and reactor power was observed to increase to approximately 101.5 percent. Operators executed the immediate actions in the alarm response procedure and reduced reactor power to approximately 97 percent. Subsequent actions determined that the drain valve for Feedwater Heater 4A had failed closed.

In conjunction with the immediate actions of the alarm response procedure, operators entered Abnormal Procedure 2.4EX-STM, "Extraction Steam Abnormal," Revision 1, based on lowering feedwater temperature. Attachment 2 of this procedure provided a graph of feedwater temperature versus reactor thermal power and indicated a NORMAL FEEDWATER HEATING REGION, a LOSS OF FEEDWATER HEATING REGION, and an UNANALYZED REGION. The procedure directed operators to reduce reactor power as necessary to maintain parameters within the NORMAL FEEDWATER HEATING REGION, otherwise to reduce power below 25 percent within 2 hours. If parameters fell inside the UNANALYZED REGION, operators were to scram the reactor. Subsequently, operators reduced reactor power to approximately 90 percent and feedwater temperature stabilized at approximately 354.2°F, as indicated on RF-TI-1 which was the highest reading indication in the control room. According to the graph in Attachment 2, this appeared to be slightly above the minimum temperature allowed within the NORMAL FEEDWATER HEATING REGION.

At 12 p.m. the inspectors were notified by control room personnel that feedwater temperature was within the LOSS OF FEEDWATER HEATING REGION. Operators planned to commence reducing reactor power to less than 25 percent at 1 p.m. This decision was based on the determination that Abnormal Procedure 2.4EX-STM had been developed using Plant Computer Point NSSRP617 for feedwater temperature which was indicating approximately 5 degrees lower (349.46°F) than RF-TI-1. Although the procedure did not specify which temperature indication to use, the decision was made by operators that Computer Point NSSRP617 should be used to enter the graph in Attachment 2. Using this lower value for feedwater temperature, operators determined that the temperature was slightly below the line for the NORMAL FEEDWATER HEATING REGION.

When the inspectors entered the control room to observe the power reduction, they were informed that it would not occur due to new information provided to the operators. Operations department management had provided the operators with two pages from Design Calculation NEDC 97-072, "Nominal Feedwater Heating and 100°F Loss of Feedwater Heating Curve," Revision 0, which contained data tables of thermal power and corresponding feedwater temperatures. These tables were used to develop the graph in Abnormal Procedure 2.4EX-STM, Attachment 2, and operators had been convinced that it was acceptable to use the data in the tables to determine the allowable feedwater temperature for the current power level. Using this method, and interpolation of the data in the tables, operations department management, with input from the engineering department, concluded that feedwater temperature was approximately 0.5°F above the minimum allowable temperature. Based on this information, control room operators determined that it was no longer necessary to reduce reactor power below 25 percent.

The inspectors obtained and reviewed a complete copy of Design Calculation NEDC 97-072 and concluded that the use of the data tables to interpolate allowable feedwater temperature was inappropriate since these tables were developed using empirical data obtained during a previous reactor startup. Because of this, the data in the tables fell within an error band centered around the allowable temperature limit given by the graph. It was therefore impossible to accurately compare a given set of plant parameters to a single point in the data tables. Furthermore, there was no allowance in Abnormal Procedure 2.4EX-STM for the use of this alternate means of determining acceptable feedwater temperatures. Step 4.2.4.2 clearly stated that "If operating outside NORMAL FEEDWATER HEATING REGION, restore feedwater temperature within region within 2 hours or lower reactor power <25% RTP."

Following the licensee's decision that a power reduction was not required, the inspectors obtained current values for reactor power and feedwater temperatures from control room indicators and the plant computer. Using this data, the inspectors evaluated the graph in Attachment 2 and determined that the plant was operating in the LOSS OF FEEDWATER HEATING REGION. As previously stated, plant operation in this region required operators to reduce reactor power to less than 25 percent. When this was brought to the attention of plant management, they did not agree that the plant was being operated in the LOSS OF FEEDWATER HEATING REGION. Further discussions were held between plant management and the NRC Region IV management. At approximately 5 p.m., the licensee determined that the plant was operating in the LOSS OF FEEDWATER HEATING REGION and that a power reduction to less than 25 percent was required. Operators reduced power to less than 25 percent at 8:53 p.m. In their subsequent root cause investigation, the licensee determined that the plant parameters of feedwater temperature and reactor power fell within the LOSS OF FEEDWATER HEATING REGION, and reactor power should have been reduced to less than 25 percent, at 5:20 a.m.

This finding had crosscutting aspects associated with human performance. This assessment was based on the fact that the procedural requirements of Abnormal Procedure 2.4EX-STM were not followed. Operators allowed themselves to be convinced that a power reduction was not required based on incorrect information provided by their management and the engineering department.

This finding affected the barrier integrity cornerstone and was considered more than minor since the feedwater temperature limits were established to ensure that the minimum critical power ratio thermal limit would not be challenged during a loss of feedwater event. The human performance attribute of the barrier integrity cornerstone objective was affected since the plant was not being operated in accordance with a procedure designed to provide reasonable assurance of fuel cladding integrity during an analyzed transient. This finding, if left uncorrected, could also have become a more significant safety concern since operators are expected to implement procedures as written. This finding was characterized under the significance determination process as having very low safety significance because it only affected the fuel cladding barrier.

Technical Specification 5.4.1(a) requires that licensees establish, implement, and maintain written procedures recommended in Regulatory Guide 1.33, Revision 2,

Appendix A, February 1978. Appendix A recommends procedures for abnormal conditions such as feedwater system failures. Abnormal Procedure 2.4EX-STM, "Extraction Steam Abnormal," Revision 1, step 4.2.4.2, required that reactor power be reduced less than 25 percent within 2 hours if feedwater temperature fell within the LOSS OF FEEDWATER HEATING REGION on the graph in Attachment 2. Contrary to this, the plant was operated above 25 percent power for approximately 15 hours with feedwater temperature within this region. The failure to implement Abnormal Procedure 2.4EX-STM is a violation of Technical Specification 5.4.1(a). This violation is being treated as a noncited violation (50-298/0304-003) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Notifications 10227710 and 10228294.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed five operability determinations regarding 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 Final Safety Analysis Report, Technical Specifications, and the associated system design criteria documents to determine if operability was justified. The inspectors reviewed the following equipment conditions and associated operability evaluations:

- Diesel Generator 1 oil leak on January 2 (Notification 10217627)

- Reactor Feed Pump B Speed Mircocontroller Malfunction on December 28, 2002 (Notification 10216820)

- Failure of the enveloping tube on Service Water Pump D on January 5 (Notification 10216796)

- Calibration test failure of Main Steam Differential Pressure Switch MS-DPIS-117D on January 11 (Notification 10215097)

- Potting material observed dripping from 4160V Bus 1F potential transformer cabinet on January 14 (Notification 10219988)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed an equipment performance deficiency regarding the backup supply of gland water to service water pumps to determine if it would pose a challenge to operators while implementing abnormal or emergency procedures.

b. Findings

A Green finding was identified regarding the licensee's failure to evaluate the impact that the isolation of the backup supply of service water pump gland seal water from the fire protection system had on operator actions required to be taken by an alarm response procedure.

On January 5, control room operators received numerous gland water low flow alarms on Service Water Pumps B and D. While carrying out the actions of Alarm Procedure 2.3_SW-GLND-B, "SW Gland Water Supply Panel - Annunciator 1A," Revision 1C1, operators noted that Valve SW-SOV-SSV10 indicated open, which should have aligned the fire protection header to supply water to the service water pump gland water system. However, there was no flow from the fire protection header because Valve FP-V-508 was caution-tagged in the closed position, isolating Valve SW-SOV-SSV10 from the fire protection header. Valve FP-V-508 had been caution-tagged closed in order to isolate Valve SW-SOV-SSV10, which had failed open in August 2002 due to a failed power supply. Operators were eventually able to restore normal gland water flows by starting an additional service water pump and opening throttle valves to the packing glands on Service Water Pumps B and D. However, operator response to and recovery from this condition was complicated by the fact that the fire protection header was isolated from the gland water system.

Alarm Procedure 2.3_SW-GLND-B, "SW Gland Water Supply Panel - Annunciator 1A," Revision 1C1, stated that Valve SW-SOV-SSV10 will automatically open after an 18-second time delay if gland water pressure falls below 28.5 psig. It then directed operators to throttle open the downstream throttle valves or cross-connect the gland water system to establish a 3 to 6 gpm flowrate. The backup supply of gland water from the fire protection header was necessary because the normally closed motor-operated valves that supplied gland water from the essential source (service water pump discharge header) lacked adequate separation to ensure their proper operation under certain fire scenarios (fire on Elevation 903 of the reactor building). Therefore, the backup supply was necessary to demonstrate shutdown capability during a fire scenario.

The inspectors found that there have been multiple maintenance issues with the backup supply to the gland water system beginning as early as September 2001. This supply was isolated between September 2001 and May 2002 due to leakage by Valve SW-SOV-SSV10 between June 2002 and November 2002 due to leaking check valves and between August 2002 and March 2003 due to a failed power supply for Valve SW-SOV-SSV10. These conditions were entered into the licensee's corrective action program as Notifications 10108498, 10168812, and 10108498, respectively. The inspectors found that the licensee had not evaluated the impact of these conditions on operator actions required by Alarm Procedure 2.3_SW-GLND-B, nor were any compensatory actions established to make this supply available during a fire scenario.

This finding affected the mitigating systems cornerstone and was considered more than minor because it concerned the availability of the service water system during a fire. Inspection MC 0609, “Significance Determination Process,” was used to assess the safety significance of this finding. Phase 1 of the significance determination process concluded that the finding was potentially risk significant since it affected a system designed to mitigate an external event (fire) and failure of this system would degrade two or more trains of a multitrain safety system or function. Therefore, a Phase 2 analysis using MC 0609, Appendix F, was required.

The following assumptions were used during the Phase 2 analysis:

- The condition existed for more than 30 days.

- An automatic fire detection and a fixed suppression system was relied upon to minimize damage to redundant divisions of equipment on Elevation 903 of the reactor building. No rated fire barrier or sufficient horizontal separation existed between the redundant divisions. The fire detection and suppression systems were at their normal operating state.

- Manual firefighting effectiveness (fire brigade) was assumed to be at its normal operating state.

- According to Cooper Nuclear Station’s Individual Plant Examination of External Events, the ignition frequency for a fire on Elevation 903 of the reactor building is 2.32E-2/year.

- A fire on Elevation 903 of the reactor building would have disabled the essential supply of gland water to the service water pumps but the normal supply from the riverwell system would have remained available.

This set of assumptions resulted in a fire mitigation frequency of -3 to -4. Based on this frequency, and the fact that the normal supply of gland water was available and would not be affected by fire scenarios that could affect the essential supply, this finding was determined to have very low safety significance (Green).

No violation of NRC requirements was identified regarding the failure to evaluate the impact of this condition on operator actions; however, the failure to issue a fire protection impairment for this condition was considered to be a licensee-identified noncited violation, as discussed in Section 4OA7 of this report. This finding was entered into the licensee’s corrective action program as Notification 10222839.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed or observed selected postmaintenance tests 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 five maintenance activities were included in the scope of this inspection:

- Service Water Pump D overhaul on January 5 (Work Order 4256493)
- Reactor building crane wire rope replacement on January 8 (Work Order 4287267)
- Fastener replacement on reactor building crane gear box on February 26 (Work Order 4292727)
- 250 Vdc Battery B cell replacement on March 3 (Work Order 4235345)
- Diesel Generator 2 intercooler cleaning and inspection on March 13 (Work Order 4223169)

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated the licensee's outage activities associated with Refueling Outage 21 (RE21) to ensure that: risk was considered in developing the outage schedule; administrative risk reduction methodologies were implemented to control plant configuration; mitigation strategies were developed for losses of key safety functions; and the operating license and Technical Specification requirements were satisfied to ensure defense-in-depth. Specifically, the following activities were included in the scope of this inspection:

- A review of the RE21 Outage Schedule, Revision 1, including the outage risk assessment
- Control room observations of the reactor shutdown and initial cooldown
- Daily review of critical parameter associated with reactor vessel level, shutdown cooling operations, and offsite power availability

- Daily review of scheduled work and the outage risk assessment for that work

RE21 continued beyond the conclusion of this inspection period. Further observations will be documented in future inspection reports.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed or reviewed the following six surveillance tests 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 Technical Specifications, the Updated Final Safety Analysis Report, and licensee procedural requirements:

- 6.HPCI.103, “HPCI IST and 92 Day Test Mode Surveillance Operation,” Revision 19, performed on January 6
- 6.HPCI. 306, “HPCI Turbine Steam Inlet Pressure Indication Channel Calibration,” Revision 4, performed on February 6
- 6.RHR.306, “Reactor High Pressure Channel Calibration,” Revision 9, performed on February 10
- 6.PC.511, “High Pressure Coolant Injection (HPCI) Local Leak Rate Test,” Revision 7, performed on March 5
- 6.1DG.401, “Diesel Generator Fuel Oil Transfer Pump IST Flow Test (Div 1),” Revision 12, performed on February 24
- 6.2EE.302, “4160V Bus 1G Undervoltage Relay and Relay Timer Functional Test (Div2),” Revision 8C1, performed on March 11

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Configuration Change 4301609 which was implemented on March 26 to block open the right bank air inlet butterfly valve on Diesel Generator 1. This normally open valve is tripped shut via a cable linkage by the mechanical overspeed trip device. This cable linkage failed to trip the valve due to mechanical binding in the cable sleeve during testing on March 23; therefore, this temporary configuration change was developed to ensure operability of the diesel until a replacement cable could be obtained. The inspectors verified that the change did not represent an unreviewed safety question, that there were adequate controls on the installation and removal of the valve blocking device, and that redundant engine overspeed protection was still available (electronic fuel rack trip, mechanical fuel rack trip, and the mechanical trip of the left bank air intake valve).

a. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors performed an in-office review of Revision 41 to the Cooper Nuclear Station Emergency Plan, submitted November 25, 2002. The inspectors compared Revision 41 to Revision 40 of the plan and 10 CFR 50.54(q) to determine if the revision decreased the effectiveness of the emergency plan.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

To review and assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and locked high radiation areas (LHRAs), the inspectors interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs during RE21. The inspectors discussed changes to the access control program with the radiation protection manager. The inspectors also conducted plant walkdowns within the radiologically controlled area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Area postings and other access controls for airborne radioactivity areas, radiation areas, LHRAs, and very high radiation areas

- Access controls, radiation work permits, and radiological surveys involving airborne radioactivity areas and high radiation areas

- Radiation protection prejob briefings presented prior to maintenance personnel performing work on Motor-Operated Valve 702 in the reactor cleanup heat exchanger room and prior to operations personnel performing valve lineup checks on valves located in the reactor cleanup heat exchanger room

- LHRA key controls

- Internal dose assessment for exposures exceeding 50 mrem Committed Effective Dose Equivalent (No opportunities for review were identified.)

- Setting, use, and response of electronic personal dosimeter alarms

- Conduct of work by radiation protection technicians and radiation workers in areas with the potential for high radiation dose and the associated radiation work permits, radiological surveys, and controls for the work (repair of Motor-Operated Valve 702 in the reactor water cleanup heat exchanger room and work on the feedwater check valve in the drywell/steam tunnel)

- Dosimetry placement when work involved a significant dose gradient

- Controls involved with the storage of highly radioactive items in the spent fuel pool

- Quality Assurance Surveillance Report S103-0201, “Operations Quarterly Assessment”; Radiation Protection Department Self-Assessment SA 02037, “External Radiation Dose Control”; and Radiation Protection Department Self-Assessment SA 02050, “2002 CNS Radiation Dose Reduction”

- A summary of access controls and high radiation area work practice related corrective action documents (Notification Reports) written since May 2002 and selected specific examples

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors conducted independent radiation surveys of selected work areas and interviewed radiation workers and radiation protection personnel to determine their knowledge of as low as is reasonably achievable (ALARA) practices. The inspectors observed high dose work activities involving Main Steam Safety Relief Valve testing (RWP 20031036) and Reactor Recirculation Flow Switch Repairs (RWP 20031057) to determine if personnel work practices were ALARA.

The following items were reviewed and compared with regulatory requirements to assess the licensee's program to maintain occupational exposures ALARA:

- Plant collective exposure history for the past 3 years, current exposure trends, source term measurements, and 3-year rolling average dose information
- ALARA program procedures
- Processes, methodology, and bases used to estimate, justify, adjust, track, and evaluate exposures
- Three ALARA prejob work packages and associated radiation work permits from RE21 activities which were likely to result in some of the highest personnel collective exposures (Refueling Activities-2003AL-01, Inservice Inspection/Erosion Corrosion Activities-2003AL-02, and Undervessel Activities-2003AL-06)
- The use and result of administrative and engineering controls to achieve dose reductions
- Permanent and temporary shielding program and implementation
- Plant source term evaluation and control strategy/program
- Hot spot tracking and reduction program
- Department self-assessments (SA 02037-External Dose Control and SA 02050-2002 CNS Radiation Dose Reduction) reviewing ALARA performance
- ALARA Committee meeting minutes and presentations
- Declared pregnant worker and embryo/fetus dose evaluation, monitoring, and controls
- Twenty-one selected corrective action documents involving exposure tracking, higher than planned exposure levels, and radiation worker practice and repetitive deficiencies since the last inspection in this area

b. <u>Findings</u>

A self-revealing Green noncited violation was identified because the licensee failed to follow the requirements of Technical Specification 5.7.1.b. Specifically, a worker failed to wear an alarming device that could be heard while working in the steam jet air ejector (SJAE) room.

On February 20, 2003, a mechanic and radiation protection technician entered the SJAE room located in the turbine building to stop a packing leak on Main Steam Valve 62. Expected radiological conditions were 5000 millirem contact with the SJAE and 500 millirem general area. However, general work area radiation levels were found to be as high as 3000 millirem per hour by the radiation protection technician prior to the start of work. Electronic dosimeter alarm settings were 5500 millirem dose rate and 40 millirem dose accumulated. The radiation protection technician informed the mechanic of the radiological conditions. However, the radiation protection technician did not stop the job to re-evaluate the radiological controls and electronic dosimeter alarm settings. The mechanic entered the area to perform maintenance; however, due to the background noise level in the area, the mechanic was not able to hear his electronic dosimeter alarm. The mechanic exited the work area and returned to the entrance of the room where he heard the alarm. The mechanic’s electronic dosimetry was in alarm as a result of the accumulated dose of 62 millirem exceeding the alarm setpoint of 40 millirem. Despite the fact that the mechanic’s electronic dosimeter was in alarm, the radiation protection technician allowed the mechanic to reenter the work area to remove equipment and verify that the packing leak had been stopped. The mechanic reentered the work area, completed his task, and exited the work area. The mechanic dose received for the task was 119 millirem.

The failure to wear an alarming device that could be heard is a performance deficiency. The issue was more than minor because it is associated with a cornerstone attribute (program and process) and affected the occupational radiation safety cornerstone objective (to ensure the adequate protection of the worker’s health and safety from radioactive material). The finding involved the failure to control radiological work that was contrary to Technical Specification requirements. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was found to have very low safety significance because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised. This event was identified because of an equipment alarm; therefore, the finding was considered self-revealing.

Technical Specification 5.7.1.b states, in part, that entry into an area with general radiation levels greater than 100 millirem per hour is permitted when a worker has a monitoring device which continuously integrates the radiation dose in the area and alarms when a preset integrated dose is received. The fact that the background noise level was loud, such that the worker could not hear the alarm, made the alarm nonfunctioning. Therefore, the failure to wear an alarming device that could be heard is a violation of Technical Specification 5.7.1.b. Because the finding was determined to be of very low safety significance and was entered into the licensee’s corrective action program as Notification Report 10227893, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (NCV 50-298/0304-04).

3. SAFEGUARDS
Cornerstone: Physical Protection (PP)

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspectors completed an in-office review of the following Physical Security Plan changes to determine if they decreased the effectiveness of the Physical Security Plan and to determine if requirements of 10 CFR 50.54 (p) were met:

- Physical Security Plan, Revision 39, dated February 6, 2002, revised an owner controlled area drawing to reflect the actual configuration of railroad tracks and some administrative title changes.

- Physical Security Plan, Revision 40, dated May 9, 2002, reflected weapons upgrades and some administrative title changes.

- Physical Security Plan, Revision 41, dated February 7, 2003, revised the process to be used for obtaining military work history checks and clarified the number of armed responders.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

.1 Initiating Events Performance Indicators

a. Inspection Scope

The inspectors verified the accuracy of data reported for the fourth quarter of 2002 for the following three NRC performance indicators:

- Unplanned scrams
- Scrams with loss of normal heat removal
- Unplanned power changes

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed corrective action program records involving LHRAs (as defined in Technical Specification 5.7.2), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned exposure occurrences (as defined in NEI 99-02) for the past 12 months to confirm that these occurrences were properly recorded as performance indicators. Radiologically controlled area entries with exposures greater than 100 millirems within the past 12 months were reviewed, and selected examples were examined to determine whether they were within the dose projections of the governing radiation exposure permits. Whole-body counts or dose estimates were reviewed if the radiation worker received a committed effective dose equivalent of more than 100 millirems. Where applicable, the inspectors reviewed the summation of unintended deep dose equivalent and committed effective dose equivalent to verify that the total effective dose equivalent did not surpass the performance indicator threshold without being reported.

b. Findings

No findings of significance were identified.

.3 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors reviewed radiological effluent release program corrective action records, licensee event reports, and annual effluent release reports documented during the past four quarters to determine if any doses resulting from effluent releases exceeded the performance indicator thresholds (as defined in NEI 99-02).

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors reviewed a selection of condition reports written during this period to determine if the licensee was entering conditions adverse to quality into the corrective action program at an appropriate threshold, to determine if the condition reports were appropriately categorized and dispositioned in accordance with the licensee’s procedures, and, in the case of conditions significantly adverse to quality, to determine if the licensee’s root cause determination and extent of condition evaluation was accurate and of sufficient depth to prevent recurrence of the condition. The following condition report was reviewed during this period:

- Significant Condition Report 2003-0350 regarding an unplanned exposure while performing spot maintenance in the steam jet air ejector room

b. Findings

The failure to follow a maintenance procedure concerning the conduct of spot maintenance was determined to be a self-revealing, Green, noncited violation.

On February 20, a radiation protection technician and a mechanic entered the SJAE room to adjust the packing on Main Steam Valve MS-62. As discussed in Section 2OS2 of this report, neither individual was wearing appropriate dosimetry for this area. Furthermore, the SJAE room was an LHRA and the actual dose rates observed upon entry into the room were 1000 mrem per hour to 5000 mrem per hour, which was three to five times higher than what was expected. Rather than re-evaluate the job, the individuals continued with the work as originally planned, which resulted in both individuals exceeding the planned dose for the job.

The mechanic and radiation protection technician were both assigned to the Fix-it-Now Team which is tasked, primarily, with completing minor or spot maintenance. Maintenance Procedure 7.0.4, "Conduct of Maintenance," Revision 22, defines spot maintenance as "skill-of-the-craft" tasks where the use of written work instructions are not required. Manual valve packing adjustment is specifically listed as an approved spot maintenance activity; however, the procedure precludes an activity from consideration as spot maintenance when special radiological controls beyond self-coverage are necessary. The individuals entered this area under Radiation Work Permit 20031005, Revision 2, which required continuous coverage by a radiation protection technician for work in areas above 1000 mrem per hour. Therefore, this activity should not have been considered spot maintenance and should not have been assigned to the Fix-it-Now Team.

This finding had crosscutting aspects associated with human performance. This assessment was based on the fact that there were administrative controls in place to prevent such an occurrence and the individuals involved were trained on those requirements.

This finding affected the occupational radiation safety cornerstone and was considered more than minor because it affected the cornerstone objective to ensure the adequate protection of the worker's health and safety from radioactive material. When processed through the Occupational Radiation Safety Significance Determination Process, the finding was found to have very low safety significance because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised. This event was identified because of an equipment alarm; therefore, the finding was considered self-revealing.

Technical Specification 5.4.1(a) requires that licensees establish, implement, and maintain written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends general procedures for the control of maintenance. Maintenance Procedure 7.0.4, “Conduct of Maintenance,” Revision 22, precluded spot maintenance in areas requiring continuous coverage by a radiation protection technician. Contrary to this, spot maintenance was performed in the SJAE room in a radiation field greater than 1000 mrem per hour, which required continuous radiation protection technician coverage. The failure to implement Maintenance Procedure 7.0.4 is a violation of Technical Specification 5.4.1(a). This violation is being treated as a noncited violation (50-298/0304-005) consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Notification 10227893.

4OA6 Meetings, including Exit

The results of the emergency plan change review were presented to Mr. D. Cook, Senior Manager, Emergency Preparedness, and other members of licensee management during a telephonic exit interview conducted on February 6.

The results of the security plan change review were presented to Ms. Noreen Robinson, Licensing Specialist, telephonically on February 24.

The results of the ALARA planning and controls inspection were presented to Mr. M. Coyle, Site Vice President, and other members of licensee management on March 13.

The results of the inservice inspection review were presented to Mr. M. Coyle, Site Vice President, and other members of licensee management on March 17.

The results of the access control to radiological areas inspection were presented to Mr. M. Coyle, Site Vice President, and other members of licensee management on March 21.

The results of the resident inspector activities were discussed with Mr. M. Coyle, Site Vice President, and other staff personnel on April 3, 2003.

During all meetings, licensee management acknowledged the inspection findings and stated that none of the material examined during the inspection was considered proprietary.

4OA7 Licensee Identified Violations

The following findings of very low safety significance (Green) were identified by the licensee, are violations of NRC requirements, and meet the criteria of Section VI.A of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a noncited violation.

- Technical Specification 5.4.1(a) requires that the licensee establish, implement, and maintain written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A recommends procedures for the plant fire protection program. Administrative Procedure 0.23, “CNS Fire

Protection Plan," Revision 35, required that a permit be issued for all fire protection impairments. Contrary to this, the fire protection backup water supply to the service water pump gland water system was disabled multiple times between September 2001 and January 2003 and was never considered to be a fire impairment even though it was required by the postfire shutdown analysis. This is being treated as a noncited violation. The licensee entered this issue into their corrective action process as Notification 10220280.

The safety significance of this violation was determined to be very low since the nonsafety-related supply of gland water remained available while the credited backup supply was unavailable.

- Technical Specification 5.4.1(a) requires written procedures for access control to radiation areas, including a radiation work permit system. Radiation Protection Shop Guide 15, "Radiation Work Permit-RP Coverage Guidance," Revision 10, states, in part, that under extreme conditions a worker can work with a dose alarm if the following criteria are met: (1) the radiation protection technician provides continuous coverage of the worker and begins the role of timekeeper for the worker's dose, and (2) the dose alarm is not exceeded by more than 50 millirem. However, on February 20, 2003, a mechanic entered the SJAE room located in the Turbine Building to stop a packing leak on Main Steam Valve 62 and was not provided appropriate timekeeping coverage by the assigned radiation protection technician. The worker exited the area with 79 millirem more than the dose alarm setting of 40 millirem. This event is documented in the licensee's corrective action program as Notification Report 10227893.

  This finding is only of very low safety significance because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.

- Technical Specification 5.4.1 requires that written procedures be implemented covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33 Section 9.e states, in part, that general procedures for the control of maintenance should be established and specifies that maintenance that can effect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures. Cooper Nuclear Station Operations Manual, Administrative Procedure 0.40, "Work Control Program," Revision 22C1, Sections 7.3 and 7.3.1 states, in part, shop supervision shall ensure a pre-job briefing is conducted that includes the following: all maintenance personnel involved have qualifications and certifications to perform the activity. Cooper Nuclear Station Operations Manual, Maintenance Procedure 7.3.36, "RHR and CS Motor Maintenance and Inspection," Revision 3, Note 1 states, in part, that Steps 2.13 and 2.14 shall be performed by a CNS machinist qualified for critical measurements. Steps 2.13 and 2.14 address the removal and installation of bearings in the residual heat removal pump motor.

During the November 2001 refueling outage, an electrical maintenance supervisor assigned a CNS electrical technician to perform motor maintenance, including bearing replacement, on a spare residual heat removal pump motor. The assigned electrical technician did not have the required qualifications to perform the specified maintenance. This is being treated as a noncited violation. The licensee entered this issue into their corrective action program as Notification 10127213.

The safety significance of this violation was determined to be very low since the maintenance activity was stopped and re-performed by qualified technicians and the motor was a spare unit that had not been placed in service.

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Bednar, Emergency Preparedness Manager
C. Blair, Engineer, Licensing
M. Boyce, Corrective Action Program Senior Manager
D. Cook, Senior Manager of Emergency Preparedness
M. Coyle, Site Vice President
T. Chard, Radiological Manager
J. Christensen, Operations Manager
J. Edom, Risk Management
R. Estrada, Performance Analysis Department Manager
M. Faulkner, Security Manager
J. Flaherty, Site Regulatory Liaison
P. Fleming, Risk & Regulatory Affairs Manager
M. Gillan, Assistant to Plant Manager
J. Hutton, Plant Manager
D. Kimball, Assistant Radiological Manager
C. Kirkland, Nuclear Information Technology Manager
V. Krueger, Engineer, Engineering Support Division/In-Service Inspection
D. Kunsemiller, Quality Assurance Manager
W. Macecevic, Work Control Manager
D. Pease, Assistant Operations Manager
R. Remmers, Supervisor, Radiation Protection
V. Roppel, Assistant Senior Manager, Engineering
L. Schilling, Administrative Services Department Manager
R. Shaw, Senior Reactor Operator
J. Sumpter, Senior Staff Engineer, Licensing
K. Tanner, Shift Supervisor, Radiation Protection
N. Wetherell, Maintenance Manager
A. Williams, Manager, Engineering Support Division
B. Wulf, Plant Engineering Department Manager

NRC
S. Schwind, Senior Resident Inspector

ITEMS OPENED AND CLOSED

Opened and Closed

50-298/0304-001	NCV	The failure to develop and implement a procedure to cope with an act of nature, such as the accumulation of ice in the intake structure was determined to be a violation of Technical Specification 5.4.1. (Section 1R01)

50-298/0304-002	NCV	The failure to implement the procedural requirements of Administrative Procedure 0.39, “Fire Watches,” Revision 27, affected the mitigating systems cornerstone since fire watches are used throughout the plant to protect safety-related equipment during hot work. (Section 1R05)
50-298/0304-003	NCV	Failure to implement a procedure affecting three fission product barriers as consistent with Section VI.A of the NRC Enforcement. (Section 1R14)
50-298/0304-004	NCV	Failure to wear an alarming device that could be heard in a High Radiation Area (Section 2OS2)
50-298/0304-005	NCV	Failure to follow a maintenance procedure regarding conduct of spot maintenance consistent with Section VI.A of the NRC Enforcement Policy (Section 4OA2)
FINDING	FIN	The failure to evaluate the impact on operator actions required by an alarm response procedure which resulted from long-standing equipment problems was considered to be a Green finding. (Section 1R16)

LIST OF ACRONYMS

ALARA	as low as is reasonably achievable
CFR	Code of Federal Regulations
CW	circulating water
DCD	design criteria document
HPCI	high pressure coolant injection
LHRA	locked high radiation area
MC	manual chapter
NCV	noncited violation
SJAE	steam jet air ejector

DOCUMENTS REVIEWED

Special Work Permits

SWP 20031002, SWP-20031003, SWP-20031015, SWP-20031021, SWP-20031031, SWP-20031040, SWP-20031054, SWP-20031057, SWP-20031059, SWP-20031068, and SWP-20031086

Notification Reports

10144459, 10145278, 10147273, 10147677, 10149769, 10155592, 10157641, 10162863, 10163219, 10165631, 10166771, 10170964, 10188418, 10192076, 10193446, 10194326, 10198857, 10200255, 10207553, 10216482, 10216516, 10217748, 10220648, 10224229, 10227469, 10227893 10228086, 10229019, 10229754, and 10233407

Procedures

54-ISI-029-02	Vision Test Procedure
54-ISI-124-02	Ultrasonic Examination of Ferritic Piping Welds and Vessel Welds Two Inches or Less in Thickness
54-ISI-147-00	UT Exam for Thickness Measurements Using Model 26DL Ultrasonic Thickness Gage
54-ISI-240-40	Visible Solvent Removable Liquid Penetrant Examination Procedure
54-ISI-270-41	Wet or Dry Magnetic Particle Examination Procedure
54-ISI-363-02	Remote Underwater In-Vessel Visual Inspection of Reactor Pressure Vessel Internals, and Associated Repairs in Boiling Water Reactor
54-ISI-366-05	Procedure for VT-1 and VT-3 Visual Examinations
54-ISI-833-01	Procedure for the Ultrasonic Examination of Reactor Equipment Cooling System Piping at Cooper Nuclear Station
54-ISI-835-04	Procedure for Ultrasonic Examination of Ferritic Piping Welds
54-ISI-836-04	Procedure for Ultrasonic Examination of Austinitic Piping Welds

Work Orders

4235683	RHR-A MT
4296827	CS-A Remote UT
4296828	CS-B Remote UT
4235682	RHH-90, RHH-93 VT
4235682	RHB-CC-53 MT

Corrective Action Notifications

10125591	Cracks Identified in the Reactor Vessel Steam Dryer
10135680	One of eight CRD Flange Bolts Had Pitting in Excess of 5%
10224623	REC Weld Has Two Flaw Indications