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

July 26, 2002

David L. Wilson, Vice President of
Nuclear Energy
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

SUBJECT: NRC INSPECTION REPORT 50-298/02-02 AND NOTICE OF VIOLATION

Dear Mr. Wilson:

On July 6, 2002, the NRC completed an inspection at your Cooper Nuclear Station. The enclosed report documents the inspection findings which were discussed on July 12, 2002, 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

The NRC has increased security requirements at Cooper Nuclear Station in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

Based on the results of this inspection, the NRC has identified three issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. 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; 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/

Jeffrey A. Clark, Chief
Project Branch F
Division of Reactor Projects

Docket: 50-298
License: DPR-46

Enclosure:
NRC Inspection Report
50-298/02-02

cc w/enclosure:
Michael T. Coyle
Site Vice President
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

John R. McPhail, General Counsel
Nebraska Public Power District
P.O. Box 499
Columbus, Nebraska 68602-0499

D. F. Kunsemiller, Risk and
Regulatory Affairs Manager
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

Dr. William D. Leech
Manager - Nuclear
MidAmerican Energy
907 Walnut Street
P.O. Box 657
Des Moines, Iowa 50303-0657

Ron Stoddard
Lincoln Electric System
1040 O Street
P.O. Box 80869
Lincoln, Nebraska 68501-0869

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

Chairman
Nemaha County Board of Commissioners
Nemaha County Courthouse
1824 N Street
Auburn, Nebraska 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, Nebraska 68509-5007

Ronald A. Kucera, Deputy Director
for Public Policy
Department of Natural Resources
205 Jefferson Street
Jefferson City, Missouri 65101

Jerry Uhlmann, Director
State Emergency Management Agency
P.O. Box 116
Jefferson City, Missouri 65101

Vick L. Cooper, Chief
Radiation Control Program, RCP
Kansas Department of Health
and Environment
Bureau of Air and Radiation
1000 SW Jackson, Suite 310
Topeka, Kansas 66612-1366

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

William R. Mayben, President
and Chief Executive Officer
Nebraska Public Power District
1414 15th Street
Columbus, Nebraska 68601

Electronic distribution by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(EEC)**
Senior Resident Inspector **(SCS)**
Branch Chief, DRP/F **(KMK)**
Senior Project Engineer, DRP/F **(JAC)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Jim Isom, Pilot Plant Program **(JAI)**
RidsNrrDipmLipb
Scott Morris **(SAM1)**
CNS Site Secretary **(SLN)**

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MCHay	SCSchwind	CSMarschall	JAClark	
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-298
License:	DPR 46
Report:	50-298/02-02
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	April 7 through July 6, 2002
Inspectors:	S. Schwind, Senior Resident Inspector M. Hay, Resident Inspector M. Runyan, Senior Reactor Inspector P. Goldberg, Reactor Inspector G. Miller, Reactor Inspector
Approved By:	J. Clark, Chief, Branch F, Division of Reactor Projects

SUMMARY OF FINDINGS

Cooper Nuclear Station
NRC Inspection Report 50-298/02-02

IR 05000298-02-02; 04/07/2002-07/06/2002; Nebraska Public Power District; Cooper Nuclear Station. Integrated Resident/Regional Report; Permanent Plant Modifications, Access Authorization, Identification & Resolution of Problems.

The inspection was conducted by resident inspectors and regional specialists. During the inspection the NRC identified three Green findings, of which three were noncited violations. The significance of each issue is indicated by its color (Green, White, Yellow, or Red) and was determined by the Significance Determination Process (SDP) in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

Green. The licensee failed to conduct required design control measures prior to implementing a design change in the service water system, in which a coating previously not evaluated was applied to the internal surface of several pipe riser columns. This was identified as a violation of Criterion III of Appendix B to 10 CFR Part 50, “Design Control.”

This finding is characterized under the significance determination process as having very low safety significance because there was no loss of function in the service water system. Because of the very low safety significance and because the licensee included the item in their corrective action program as Notification 10156239, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (Section 1R17.b).

Cornerstone: Mitigating Systems

Green. The licensee failed to identify and correct deficient documentation supporting environmental qualification of safety-related equipment in the steam tunnel and acceptable voltage applications for Buchanan 0241 terminal blocks. These findings were determined to be two examples of a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee documented this issue in their corrective action process as Notifications 10163954 and 10167990. This finding also had crosscutting aspects associated with problem identification and resolution.

This finding was determined to have a credible impact on safety because there was no assurance that the equipment would perform its design function during accident conditions since it was not operating in a previously tested or analyzed configuration. This noncited violation was characterized under the significance determination process as having very low safety significance (Green) based on the performance of an acceptable analysis that demonstrated the affected equipment was environmentally qualified (Section 4OA2).

Cornerstone: Physical Protection

Green. The inspectors identified a finding regarding inconsistencies in the licensee's implementation of the testing for cause requirements of 10 CFR 26.24. This finding was identified during a followup inspection of an unresolved item discussed in NRC Inspection Report 50-298/0108 (URI 50-298/0108-08).

No violation of NRC requirements was identified; however, this finding had a credible impact on safety since inconsistent implementation of the fitness-for-duty requirements could reduce the effectiveness of the program in deterring and detecting potential substance abuse. Manual Chapter 0609 has no significance determination process to address fitness for duty without affects on radiological sabotage. Therefore, in accordance with Appendix B of NRC Manual Chapter 0612, this issue is considered a Green non-SDP finding (Section 3PP1).

Report Details

The plant was operating at 100 percent power at the beginning of the inspection period. On May 10, 2002, reactor power was reduced to approximately 10 percent for a planned drywell entry to complete corrective maintenance on a fan cooling unit. The plant returned to full power on May 12. On May 13 reactor power was reduced to approximately 10 percent again in order to take the main generator offline to complete emergent repairs to a potential transformer. On May 14, while attempting to synchronize the generator with the grid prior to power ascension, a digital electrohydraulic system pump failed and operators manually scrammed the reactor from approximately 20 percent power. After completion of repairs, the plant was restarted and returned to full power on May 20. On May 27, reactor power was reduced to approximately 65 percent in response to high vibration readings on both reactor feed pump turbines. The reactor returned to full power on May 28.

1. **REACTOR SAFETY**
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

.1 Complete Equipment Alignment Inspections

a. Inspection Scope

On May 6, 2002, the inspectors performed one complete system alignment inspection of the emergency standby power system, which included the Divisions I and II emergency diesel generators. The inspectors verified that the system was in the appropriate configuration per the system operating procedures and that it was installed and capable of performing its design functions as described in the Updated Final Safety Analysis Report. A review of maintenance work orders and corrective action documents for the past 12 months was also performed. A walkdown of the system was performed in each diesel generator room to assess material condition, such as system leaks and housekeeping issues, that could adversely affect system operability.

b. Findings

No findings of significance were identified.

.2 Partial Equipment Alignment Inspections

a. Inspection Scope

The inspectors performed three partial equipment alignment inspections, which included the high pressure coolant injection system, reactor equipment cooling system, and steam tunnel ventilation system. The inspectors verified that the systems were in the appropriate configuration per the system operating procedures and that they were installed and capable of performing their design functions as described in the Updated Final Safety Analysis Report. A visual inspection of all accessible portions of these systems located in the reactor building was completed. This inspection accessed the material condition of the systems and verified that supports and hangers were visually showing no signs of degradation.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Routine Fire Area Walkdowns

a. Inspection Scope

The inspectors performed nine 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 Zone 14A - Division I emergency diesel generator room
- Fire Zone 14B - Division II emergency diesel generator room
- Fire Zone 13C - Electric shop
- Fire Zone 13D - Instrument and controls shop
- Fire Zone 20B - Intake Structure
- Fire Zone 1C - Division I residual heat removal pump room
- Fire Zone 1D - Division II residual heat removal pump room
- Fire Zone 6 - Refueling floor
- Fire Zone 5A - Standby liquid control system room

b. Findings

No findings of significance were identified.

.2 Annual Fire Drill Inspection

a. Inspection Scope

The inspectors observed the plant fire brigade during a fire drill on May 31, 2002, to assess its ability to fight fires. The fire was simulated in the technical support center in 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

a. Inspection Scope

The inspectors performed a semiannually inspection of internal flood protection features. The high pressure coolant injection pump room was chosen for this inspection based on its location in the plant and its risk significance. The inspection included a review of the Updated Final Safety Analysis Report, selected Design Criteria Documents (DCD) and design calculations, and a walkdown of flood protection features in the pump room. Specific documented reviewed included:

- Cooper Nuclear Station DCD 36, “High Energy Line Break (HEL)/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

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed and reviewed the cleaning, inspecting, and testing of the Division 1 reactor equipment cooling water system heat exchanger that was performed on April 22, 2002. A review of the heat exchanger performance evaluation was conducted to identify potential deficiencies that could mask degraded performance. The inspectors reviewed the type, location, and calibration of instrumentation used to acquire the data to verify its acceptability for the evaluation. The evaluation review was conducted and documented in accordance with Performance Evaluation Procedure 13.15.1, “Reactor Equipment Cooling Heat Exchanger Performance Analysis,” Revision 14.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalifications

a. Inspection Scope

On June 5, 2002, the inspectors observed licensed operator simulator training. The simulator training evaluated the operators’ ability to recognize, diagnose, and respond to a security threat, loss of the startup transformer, and loss of the intake structure. The inspectors observed and evaluated the following areas:

- Formality of communication
- Prioritizing, interpreting, and verification of alarms
- Procedure implementation
- Control board operation and manipulation of controls
- Oversight and direction provided by the shift supervisor
- The crew’s and evaluator’s critiques

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed several equipment performance issues to assess the licensee implementation of their maintenance rule program. The inspectors verified that the systems, structures, and components (SSC's) that experienced these problems were properly included in the scope of the licensee's maintenance rule program, the appropriate performance criteria were established, and in the case of SSC's monitored under paragraph a(1) of the rule, that the established goals and corrective actions were appropriate. 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 11. The inspectors reviewed the following seven equipment performance problems:

- Degraded gland water flow supporting Service Water Booster Pump C
- Degraded stroking characteristics for reactor equipment cooling system valve (REC-MOV-711MV)
- Noncomforming material used for the main steam isolation valve limit switch rollers
- Packing leakage on the steam supply valve (RHR-MO-920) to the augmented offgas air ejectors
- Incorrect background settings used for the turbine building high range radiation monitoring assembly (RMV-RM-20B)
- Failed source check on the radwaste building high range radiation monitoring assembly (RMV-RM-30B)
- Incorrect background settings used for the radwaste building high range radiation monitoring assembly (RMV-RM-30B)

b. <u>Findings</u>

No findings of significance were identified.

1R13 <u>Maintenance Risk Assessments and Emergent Work Evaluation</u>

a. <u>Inspection Scope</u>

The inspectors reviewed six 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. The following risk evaluations were reviewed:

- On April 6, 2002, service water booster Pump C was declared inoperable and required emergent repairs
- On May 29, 2002, the high pressure coolant injection system was declared inoperable for scheduled maintenance

- On June 3, 2002, service water system Valve SW-MO-MO89B was declared inoperable and required emergent repairs
- On June 1, 2002, the 345 kV-161 kV auto-transformer was removed from service for scheduled maintenance
- On June 10, 2002, the Division II emergency diesel generator was declared inoperable for scheduled maintenance
- On June 12, 2002, the Booneville 345 kV distribution line was disabled during a storm concurrent with a scheduled outage on the Division II emergency diesel generator

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

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 such 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 DCDs to determine if operability was justified. The inspectors reviewed the following equipment conditions and associated operability evaluations:

- A fuel oil leak from the Division I emergency diesel generator engine-driven fuel pump (Notification 10145175)
- A failed switch contact in the reactor manual control system that prevented continuous insertion of control rods (Notification 10162815)
- Control Rod 30-19 could not be withdrawn past position 46 (Notification 10169478)
- Foreign material was found in high pressure coolant injection turbine drain Valve HPCI -SOV-SSV64 (Notification 10173856)
- Reliability requirements of the anticipated transient without scram recirculation pump trip circuit (Notification 10159692)

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed 12 permanent plant modification packages and associated documentation, such as review screens and safety evaluations, to verify that they were performed in accordance with regulatory requirements and plant procedures. The inspectors also reviewed procedures governing plant modifications to evaluate the effectiveness of the programs for implementing modifications to risk-significant SSCs such that these changes did not adversely affect the design and licensing basis of the facility. Permanent plant modifications and procedures reviewed are listed in the attachment to this report.

The inspectors interviewed the cognizant design and system engineers for the identified modifications as to their understanding of the modification packages.

The inspectors evaluated the effectiveness of the licensee's corrective action process to identify and correct problems associated with the performance of permanent plant modifications. In this effort, the inspectors reviewed the corrective action documents listed in the attachment to this report.

b. Findings

Inadequate Design Control Measures for Change to Service Water System

The inspectors identified a violation of very low safety significance (Green) associated with the use of an epoxy coating (Belzona) in the repair of service water pump riser columns by a third-party vendor. The inspectors noted that, in the repair of the service water pump riser columns, the licensee had actually implemented a design change without performance of the required design control measures.

Under Purchase Order 991103, the licensee sent four service water pump riser columns to a vendor for repairs to corroded areas around the pipe flanges. In addition to performing a weld buildup repair, the vendor applied Belzona coating to the flange faces as a corrosion/erosion inhibitor. The coating extended beyond the flange approximately 2 inches along the inside surface of the piping (in the flow path). Since the vendor certified to the licensee that the repaired piping was equivalent in form, fit, and function to the original piping, the licensee did not classify the repair activity as a design change. The repaired piping was accepted by the licensee under Component/Part Equivalency Change Evaluation Document 2000-0015.

When requested by the inspectors, the licensee staff was unable to produce a safety evaluation or documentary evidence of the specification, codes, or standards for the Belzona coating to verify the adequacy of the coating for use in the service water system. The licensee staff stated that, because the repaired piping had been treated as a component/part equivalency (i.e., “like-for-like” replacement), a safety evaluation under 10 CFR 50.59 was not required in accordance with Engineering Procedure 3.4 “Configuration Change Control.”

The inspectors concluded that, since the Belzona coating had not been used previously in the service water system (or any other safety-related system) at the facility, the application of the coating in the flow path represented a design change to the system. The licensee staff agreed that they had improperly characterized the change to the service water system as a component/part equivalency replacement.

Requirements in 10 CFR Part 50, Appendix B, Criterion III, state that design changes to a facility shall be subject to design control measures sufficient to verify the adequacy of the design through reviews, calculational methods, or a suitable testing program.

The inspectors found that the failure to verify the adequacy of the Belzona coating for application in the service water system represented a violation of 10 CFR Part 50, Appendix B, Criterion III. This finding had a credible impact on safety because a postulated failure of the coating could result in a failure of the service water system through clogging of the pump discharge strainer or individual component piping tolerances in the system. This could credibly impact the operability and availability of the service water system (mitigating systems cornerstone), which provides cooling water to systems necessary to mitigate the effects of an accident. Using Phase 1 of the Significance Determination Process, the inspectors determined that the finding was a design deficiency confirmed not to result in loss of function. Therefore, this finding was determined to have a very low safety significance (Green).

Because of the very low safety significance, and because the licensee included the item in their corrective action program as Notification 10156239, this violation is being treated as a noncited violation (50-298/0202-01) in accordance with Section VI.A.1 of the Enforcement Manual.

1R19 <u>Postmaintenance Testing</u>

a. <u>Inspection Scope</u>

The inspectors verified that postmaintenance tests were adequate to verify system operability and functional capabilities. The inspectors verified that testing met design and licensing bases, Technical Specifications, the Updated Final Safety Analysis Report, the inservice test program, and licensee administrative procedures requirements. The inspectors reviewed the testing results for the following seven components:

- Primary containment isolation Valve PC-MOV-230MV following planned maintenance on May 7, 2002 (Work Order 4172372)

- Service water system Pump C following planned maintenance on May 22, 2002 (Work Order 4226732)

- Service water system Valve SW-MO-MO89B following emergent repairs on June 4, 2002 (Work Order 4246716)
- Reactor manual control system normal rod movement switch following emergent repairs on May 13, 2002 (Work Order 4242216)
- Division II emergency diesel generator fuel oil transfer pump following emergent repairs on May 19, 2002 (Work Order 4242711)
- Division I emergency diesel generator fuel pump following planned maintenance on June 17, 2002 (Work Order 4229256)
- Division II emergency diesel generator Relay 14RV1 following planned maintenance on June 10, 2002 (Work Order 4169211)

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed or reviewed the following four 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:

- Surveillance Procedure 6.2RHR.501, “ASME Section XI Periodic Pressure Test of the Class 2 Residual Heat Removal System Loop B,” Revision 6, performed on April 16, 2002
- Surveillance Procedure 6.CRD.201, “North and South SDV Vent and Drain Valve Cycle, Open Verification, and Timing Test,” Revision 6C2, performed on May 6, 2002
- Surveillance Procedure 6.SC.603, “Technical Specification Verification of Secondary Containment Manual Valves and Blind Flanges,” Revision 4, performed on May 23, 2002
- Surveillance Procedure 6.1DG.101, “Diesel Generator 31 Day Operability Test (IST) (DIV I),” Revision 23, performed on May 24, 2002

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed two temporary plant modifications to ensure that the modifications did not adversely affect system operability or design requirements specified in the Updated Final Safety Analysis Report and Technical Specifications. This review also included the testing requirements after installation and removal of the temporary modifications as well as how configuration control on the affected system was maintained. The following two modifications were reviewed:

- On June 28, 2002, the licensee installed a temporary modification to the ventilation systems in the reactor water cleanup pump rooms to determine airflow patterns in the rooms. This information was needed to develop a permanent plant modification to the ventilation systems in order to prevent thermal stratification in the rooms.

- On May 29, 2002, the licensee installed a temporary diesel generator to support the 12.5 kV distribution system during planned maintenance on the 345Kv-161 kV auto-transformer. This diesel was used to supply power to the electric fire pump, the emergency operations facility, technical support center, and other house loads.

b. Findings

No findings of significance were identified.

Emergency Preparedness

1EP6 Drill Evaluation

a. Inspection Scope

On July 2, 2002, the inspectors observed the licensee perform an emergency preparedness drill. Observations were conducted in the control room simulator and technical support center. 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 July 2, 2002

b. Findings

No findings of significance were identified.

2 RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs during routine operations. The inspector also conducted plant walkdowns within the radiological controlled area and conducted independent radiation surveys of selected work areas.

The inspector attended and compared to regulatory requirements a prejob briefing for Radiation Work Permit 2002-1035, “RadWaste Clean Up (RWCU) ‘A’ Pump Rebuild/Replace,” and observed the movement of a high integrity container (HIC) fill head from a full to an empty container under Radiation Work Permit 2002-1019, “Fill and Prepare Condensate HIC for Shipment.”

The following items were reviewed and compared with regulatory requirements:

- Quality Assurance Audit Report #01-11, “RE20 Outage,” Quality Assurance Field Observation FO-0202, and Radiological Protection Self-Assessment SA02005, “Internal Dose Assessment”

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

- Radiological surveys involving airborne radioactivity areas and high radiation areas

- Locked high radiation area key control program

- Access controls, surveys, and radiation work permits for significant high dose work areas during the inspection

- Dosimetry evaluation and placement when work involved a significant dose gradient

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

- A summary of operational radiation protection corrective action documents written since November 1, 2001, and selected examples (4-13725, 4-14738, 10120838, 10122910, 10123681, 10125472, 10125503, 1012811, 10128483, 10132294, 10142945, 10150952, and 10163219)

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP1 Access Authorization

a. Inspection Scope

The inspectors reviewed the licensee's revised for-cause drug and alcohol testing program performed following an accident that resulted in serious injury to a person that fell off a flatbed trailer. This inspection was performed in order to resolve an inspection item opened in NRC Inspection Report 50-298/0108 (URI 50-298/0108-08).

b. Findings

The inspectors identified a Green finding regarding inconsistencies in the implementation of the licensee's fitness-for-duty program. As discussed in NRC Inspection Report 50-298/0108, on October 2, 2001, a radiation protection worker was performing radiological surveys on new fuel that was arriving on site for an upcoming refueling outage. While performing these surveys on top of a flatbed truck trailer, the individual apparently lost track of his position and stepped off the edge of the trailer. The individual sustained serious injuries which required medical attention at the local hospital.

Following the incident, the inspectors questioned whether the licensee had administered a for-cause drug and alcohol test on the individual. The licensee stated that they had not tested the individual based on the lack of any observed unusual behavior or the smell of alcohol. The individual's supervisor had made this determination; however, there was no documentation of this decision. Following the licensee's review of this issue, they acknowledged that their fitness-for-duty procedure and their implementation of the procedure were not adequate to ensure that the requirements of 10 CFR 26.24 were consistently met and, subsequently, they revised their fitness-for-duty procedure.

No violation of NRC requirements was identified; however, this finding had a credible impact on safety since inconsistent implementation of the fitness-for-duty requirements of 10 CFR 26.24 could reduce the effectiveness of the program in deterring and detecting potential substance abuse. Manual Chapter 0609 has no significance determination process to address fitness for duty without affects on radiological sabotage. Therefore, in accordance with Appendix B of NRC Manual Chapter 0612, this issue is considered a Green non-SDP finding.

The inspectors reviewed the revised fitness-for-duty procedure and determined that additional controls were put in place that appeared adequate to consistently implement the testing-for-cause program. This revision precluded the immediate supervisor from being able to make the decision to not test for cause individuals without concurrence from the licensee's security manager or nuclear security services supervisor.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Initiating Events and Barrier Integrity Performance Indicators

a. Inspection Scope

The inspectors reviewed logs, notifications, and plant records for the first quarter of 2002 to verify the accuracy of reported data for the following three performance indicators:

- Unplanned Scrams
- Scrams with Loss of Normal Heat Removal
- Reactor Coolant System Leakage

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed corrective action program records involving locked high radiation areas (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 work 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 inspector 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 inspector 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

a. Inspection Scope

The inspectors reviewed selected notifications placed into the licensee's corrective action process to verify that equipment, human performance, and program issues were being identified at an appropriate threshold and the associated immediate and long-term corrective actions taken or planned were commensurate with the significance of the issues.

b. Findings

The inspectors identified two examples of the failure to identify and correct deficient conditions related to the environmental qualification of equipment important to safety. The first example involved the licensee's failure to analyze the affects of accelerated aging on equipment located in the steam tunnel due to exceeding the expected service life temperatures of the equipment. The second example resulted from inadequate corrective actions for terminal blocks located in the drywell that were not tested for the proper accident conditions. In this example, the licensee's root cause failed to identify that the same equipment was not successfully tested for the appropriate voltage applications. These were determined to be two examples of a violation of 10 CFR Part 50, Appendix B, Criterion XVI. These findings were determined to have crosscutting aspects associated with problem identification and resolution.

Failure to Analyze for Accelerated Aging

The inspectors reviewed two notifications regarding heat related degradation of components located in the steam tunnel. Notification 1013545, dated December 16, 2001, documented Buchanan 0241 nylon terminal blocks that were found to be brittle, discolored, and crumbled when handled. Notification 10125579, dated November 23, 2001, documented heat damaged to electrical cables. During interviews, the licensee stated that the degraded conditions found in the steam tunnel were due to elevated temperatures that resulted in accelerated aging of affected equipment. According to the environmental qualification data packages for these components, the service life aging conditions were based on a maximum temperature of 150°F.

The inspectors reviewed historical notifications and identified that on multiple occasions the licensee had documented concerns associated with exceeding the 150°F limit tied to the environmental qualifications of equipment in the steam tunnel. When questioned by the inspectors, the licensee stated that no formal evaluations had been performed to address accelerated aging of equipment due to the elevated temperatures. Furthermore, the licensee had previously identified this concern with Buchanan 0241 terminal blocks installed in the drywell and subsequently replaced them. However, they failed to recognize the same concern for these components located in the steam tunnel.

Appendix B, Criterion XVI, of 10 CFR Part 50, states that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. The failure to identify and correct the deficient documentation supporting environmental qualification of safety-related equipment in the steam tunnel, as required for 10 CFR 50.49, was determined to be the first example of a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (50-298/0202-02). The licensee documented this issue in their corrective action process as Notification 10163954.

This finding affected the mitigating systems cornerstone since it dealt with the reliability and availability of mitigation system equipment. Furthermore, since the finding was associated with the operability, availability, reliability, or function of a system or train in a mitigating system, its safety significance was evaluated using Manual Chapter 0609, “Significance Determination Process.” This finding was characterized under the significance determination process as having very low safety significance (Green) based on the performance of an acceptable analysis that demonstrated the affected equipment was environmentally qualified and system functionality was not lost.

Failure to Test for the Appropriate Voltage Applications

The inspectors reviewed the environmental qualification test report for the Buchanan 0241 nylon terminal blocks and noted that, during testing under accident conditions, the terminal blocks had excessive leakage current for configurations using 250 volts direct current and 600 volts alternating current. The equipment qualification data package for the Buchanan 0241 terminal blocks indicated that they met the functional requirements for electrical circuits of 600 volts and below under accident conditions. When questioned by the inspectors, the licensee acknowledged that the qualification data package was in error and that the Buchanan 0241 terminal blocks were only qualified at a tested configuration of 125 volts.

The inspector questioned the licensee as to whether Buchanan 0241 terminal blocks were installed in any applications greater than 125 volts that were required to remain functional during accident conditions. The licensee reviewed their records and discovered that Service Water Valve SW-MO-MO89B utilized a nylon Buchanan 0241 terminal block in a 480 volt application that was required to remain functional during accident conditions. The licensee immediately declared the valve inoperable on June 3, 2002, and replaced the terminal block with an environmentally-qualified component.

In April of 2000, the licensee identified that the environmental qualification data package for the Buchanan terminal blocks indicated that the blocks were not environmentally qualified for use in the drywell because they were not tested to the appropriate accident conditions. This concern was documented in Problem Identification Report 4-07970. In response to this deficient condition the licensee performed an engineering evaluation to replace the Buchanan terminal blocks with Weidmuller terminal blocks. Change Evaluation Document (CED) 2000-0059 documented this evaluation. CED 2000-0059 stated that "This CED will evaluate the applicable critical characteristics of the existing Buchanan Terminal Blocks against the Weidmuller Terminal Blocks." This evaluation also stated that both terminal blocks were rated for 600 volt applications. During this review, the licensee failed to identify that the Buchanan terminal blocks were not rated for 600 volt applications under harsh environment conditions.

As previously discussed, on December 16, 2001, the licensee identified that Buchanan terminal boards located in the steam tunnel were found in a degraded condition. The licensee’s corrective actions included a review of the Buchanan terminal board qualification documentation. This review also failed to identify that the environmental qualification documentation for these terminal boards was inadequate for voltage applications of greater than 125 volts.

Appendix B, Criterion XVI, of 10 CFR Part 50, states that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. The failure to identify and correct the deficient documentation supporting environmental qualification of safety-related equipment, as required by 10 CFR 50.49, was determined to be the second example of a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (50-298/0202-02). The licensee documented this issue in their corrective action process as Notification 10167990.

This finding affected the mitigating systems cornerstone since it dealt with the reliability and availability of mitigation system equipment. Furthermore, since the finding was associated with the operability, availability, reliability or function of a system or train in a mitigating system, its safety significance was evaluated using Manual Chapter 0609, “Significance Determination Process.” This finding was characterized under the significance determination process as having very low safety significance (Green) based on the performance of an acceptable analysis that demonstrated the affected equipment was environmentally qualified and system functionality was not lost.

4OA6 Meetings, including Exit

On April 18, 2002, the inspectors presented the results of the permanent plant modification inspection to Mr. D. Wilson, Vice President-Nuclear, and other members of licensee management. A supplemental exit meeting was conducted by telephone on April, 29, 2002, with Mr. D. Kunsemiller, Manager, Risk and Regulatory Affairs, and other licensee staff members.

On May 16, 2002, the inspectors presented the results of the radiological protection inspection to Mr. J. Hutton, Plant Manager, and other members of licensee management. On May 22, 2002, the inspectors discussed details of an additional issue with Mr. P. Fleming, Acting Risk and Regulatory Affairs Manager, and other licensee personnel.

On July 12, 2002, the results of the resident inspector activities were discussed with Mr. D. Wilson, Vice President-Nuclear, and other staff personnel.

During all meetings, licensee management acknowledged the inspection findings presented. Additionally, the inspectors were informed that none of the material examined during the inspection should be considered proprietary.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee:

A. Bacha, Senior Civil Engineer
D. Madsen, Senior Licensing Engineer
B. Champlin, Senior I&C Engineer
J. Lechner, Engineering Support Department Project Manager
S. Freborg, Assistant Engineering Support Manager
J. Hutton, Plant Manager
M. Coyle, Site Vice President
D. Wilson, Vice President-Nuclear
C. Blair, Licensing Engineer
T. Chard, Radiation Protection Chemistry Manager
P. Fleming, Acting Risk and Regulatory Affairs Manager
D. Meyers, Senior Manager Site Support
K. Tanner, Radiation Protection Supervisor

NRC:

M. Hay, Resident Inspector
B. Baca, Health Physicist
S. Schwind, Senior Resident Inspector

ITEMS OPENED, CLOSED AND ADDRESSED

Opened and Closed

50-298/0202-01	NCV	Failure to apply required design control measures for a change to the service water system (Section 1R17.b)
50-298/0202-02	NCV	Failure to adequately document environmental qualification of safety-related equipment (Section 4OA2)

Closed

50-298/0108-08	URI	Inconsistent implementation of fitness-for-duty requirements (Section 3PP1)

Addressed

50-298/0108-10	NCV	IR 50-298/0108 referenced an incorrect notification number regarding this NCV. The correct number is Notification 10139333.

DOCUMENTS REVIEWED

Design Modifications

1999-0043, “HPCI Suction Vent Valve,” April 4, 2002

2000-0068, “Re-classification of DG 1 &2 Turbo Inter-cooler Temperature Controllers and Installation of Essential Pressure Regulating Valves,” October 4, 2000

6006863, “IA-CV-18CV Check Valve Model Addition Change,” February 25, 2001

91-144, “RHR Heat Exchanger Tube Plugging,” October 29, 2001

93-257, “Control Room Emergency Bypass Fan Upgrade,” January 27, 1995

CED 6005601, “ERP to Z Sump Drain Line Isolation Valve Addition,” November 21, 2001

CED 6005426, “Diesel Generator Cooling Requirements Upgrade,” November 20, 2001

CED 1999-0229, “REC Heat Exchanger Tube Plugs,” January 4, 2000

CED-2000-0015, “Service Water Pump Column Repair, Belzona Coating of Eroded Areas,” January 29, 2001

CED-2001-0020, "CNS-2-HPCI-CV-15CV Replacement," Revision 0

CED-4163326, "Replacement of ASCO Series 8342-4-Way Solenoid Operated Valve (SOV) with Automatic Valve Company Model U0403AABR-AAS 4-Way SOV," Revision 1

MP 96-103, “DG Exhaust Gas Bypass Modification,” November 4, 1998

Problem Identification Reports

10092523	RCR 1999-0628
10095255	RCR 2000-0806
10122256	RCR 2000-1128
10155287	RCR 2001-0409
10156239	RCR 2002-0087
SCAQ 98-0390	

Procedures

0.8 “Safety Assessments and Unreviewed Safety Question Determinations,” Revision 6
3.4, “Configuration Change Control,” Revision 32
5.3AC12.5, “Loss of 12.5 KV,” Revision 0
6.ADS.302, “ADS Accumulator Functional Test,” Revision 4

Work Orders

98-1385	4179411
98-1797	4179683
00-2466	4204305
4173365	4207645

Miscellaneous

Purchase Order 991103 and supporting documentation
QA Audit 17354
Source Surveillance 99-063