UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

February 4, 2002

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

SUBJECT: COOPER NUCLEAR STATION - NRC INSPECTION REPORT 50-298/01-07

Dear Mr. Wilson:

On December 29, 2001, the NRC completed an inspection at your Cooper Nuclear Station. The enclosed report documents the inspection findings which were discussed with you and other members of your staff on January 3, 2002.

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 covered selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Since September 11, 2001, Cooper Nuclear Station has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to Nebraska Public Power District. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

Based on the results of this inspection, the NRC has identified four findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. Because the violations were of very low safety significance, and because they were entered into your corrective action program, the NRC is treating the findings as noncited violations, in accordance with Section VI.A of the NRC's Enforcement Policy. If you contest these violations, you should provide a response with the basis for your denial within 30 days of the date of this inspection report, 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/

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

Docket: 50-298
License: DPR-46

Enclosure:
NRC Inspection Report
50-298/01-07

cc w/enclosure:
G. R. Horn, Senior Vice President
of Energy Supply
Nebraska Public Power District
1414 15th Street
Columbus, Nebraska 68601

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

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(MCH2)**
Branch Chief, DRP/C **(KMK)**
Senior Project Engineer, DRP/C **(vacant)**
Staff Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Jim Isom, Pilot Plant Program **(JAI)**
RidsNrrDipmLipb
Scott Morris **(SAM1)**
CNS Site Secretary **(SLN)**
Dale Thatcher **(DFT)**

R:_CNS\2001\CN2001-07RP-MCH.wpd

RIV:SRI:DRP/C	RI:DRP/C	SPE:DRP/C	C:DRS/EMB	C:DRS/PSB	C:DRP/C
JAClark	MCHay	DPLoveless	CSMarschall	GMGood	KMKennedy
/RA/	***E- KMKennedy***	***KMKennedy for***	***/RA/***	***LTRicketson for***	***/RA/***
1/28/02	2/4/02	2/4/02	1/31/02	1/31/02	2/4/02

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-298
License:	DPR-46
Report:	50-298/01-07
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	October 7 through December 29, 2001
Inspectors:	J. Clark, Senior Resident Inspector M. Hay, Resident Inspector P. Elkmann, Emergency Preparedness Inspector D. Carter, Health Physicist C. Clark, Reactor Inspector
Approved by:	K. Kennedy, Chief, Project Branch C Division of Reactor Projects

SUMMARY OF FINDINGS

Cooper Nuclear Station
NRC Inspection Report 50-298/01-07

IR 05000298-01-07; 10/07/2001-12/29/2001; Nebraska Public Power District; Cooper Nuclear Station. Integrated Resident/Regional Report; Fire Protection, Operability Evaluations, Surveillance Testing, Occupational Radiation Safety.

The inspection was conducted by resident inspectors and regional specialists. The inspection identified four Green findings, all of which are noncited violations. The significance of the issues are indicated by their color (Green, White, Yellow, Red) and were determined by the Significance Determination Process in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

Green. The licensee failed to ensure that combustible material was removed or protected from hot work resulting in a fire on November 26, 2001, located in the reactor building on the torus area floor. This was determined to be a violation of Technical Specification 5.4.1.d. This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This issue has been entered into the licensee's corrective action process as Notification 10126869.

This issue was determined to have a credible impact on safety because an actual fire inside the reactor building occurred. This noncited violation was characterized under the significance determination process as having very low safety significance because the fire was quickly identified and extinguished and the fire did not, and could not, affect any equipment necessary for maintaining safe shutdown conditions. Specifically, the reactor cavity was flooded to greater than 23 feet, the spent fuel pool gates were open, a division of shutdown cooling was operable, and emergency core cooling system instrumentation was not affected (Section 1RO5.1).

Cornerstone: Mitigating Systems

Green. The licensee failed to identify and correct a condition adverse to quality. Power cables to the safety-relief valve solenoid valves were not maintained in conformance with 10 CFR 50.49 requirements from 1995 through October of 2001. The licensee had several opportunities to identify and correct this condition from April 2000 to October 2001. This was determined to be 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. This issue has been entered into the licensee's corrective action process as Notification 10092693.

This finding was more than minor because, if left uncorrected, it would have posed a more significant issue. This noncited violation was characterized under the significance determination process as having very low safety significance because the safety-relief valves were later determined to have been qualified (Section 1R15.1).

Cornerstone: Barrier Integrity

Green. The licensee failed to implement effective corrective actions resulting in repetitive failures of reactor feedwater check valves to pass local leak rate testing requirements from 1983 through November of 2001. This was determined to be 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. This issue has been entered into the licensee's corrective action process as Significant Condition Report 2001-1161.

This issue was considered to have a credible impact on safety, in that the failure of these valves caused a higher than normal containment leakage. This noncited violation was characterized under the significance determination process as having very low safety significance. The finding was a Type B finding in accordance with the significance determination process because these valve failures did not affect core damage frequency. Type B findings related to containment isolation valves in plants with Mark I containments and are considered to be Green, based on Table 3 of Inspection Manual Chapter 0609-H, "Containment Integrity Significance Determination Process" (Section 1R22.1).

Cornerstone: Barrier Integrity

Green. The licensee failed to implement effective corrective actions, resulting in repetitive scaffold construction nonconformances potentially affecting the operation of equipment important to safety. Examples included scaffolding built in the proximity of and over safety-related equipment, as well as scaffold components that could have interfered with the safety function of plant components. This violation of 10 CFR Part 50, Appendix B, Criterion XVI, is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This issue has been entered into the licensee's corrective action process as Notification 10127237.

This issue was considered to have a credible impact on safety, in that the failure to properly construct scaffolds could affect the operation of equipment important to safety. This noncited violation was characterized under the significance determination process as having very low safety significance because the failure to construct scaffolds in accordance with the procedural requirements did not result in any equipment failure or loss of safety function (Section 4OA2).

Report Details

The plant operated at 100 percent power from October 7 through November 3, 2001. On November 3, 2001, the plant was shut down for a refueling outage and remained in that condition for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R01 Adverse Weather

a. Inspection Scope

The inspectors reviewed or observed implementation of General Operating Procedure 2.1.11, Attachment 7, "Station Operator Ice and Snow Inspection," Revision 101, pertaining to the licensee's response to adverse weather. The inspectors assessed the adequacy of the procedure for ensuring that safety-related equipment would be appropriately monitored for operability during ice and snow storms.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments

.1 Partial Equipment Alignment Inspections

a. Inspection Scope

The inspectors performed partial equipment alignment inspections of the Division 1, 250 Vdc system and the high pressure coolant injection system. The inspectors verified that the systems were installed and capable of performing their design functions as described in the Updated Final Safety Analysis Report. They reviewed system operating procedures, surveillance procedures, and design documents to assess that these systems were properly operated and maintained.

b. Findings

No findings of significance were identified.

.2 Complete Equipment Alignment Inspections

a. Inspection Scope

The inspectors performed a complete equipment alignment inspection of the Division 1 4160-volt essential distribution system. The ability of the Division 1 diesel generator to supply power to the essential busses during accident conditions was also assessed. The ability of the Division 1 diesel generator to supply power to the essential buses was assessed by reviewing the design requirements contained in the Updated Safety Analysis Report. The following additional documents were reviewed during this inspection:

- Special Test Procedure 87-10, "Electrical Load Study"
- NEDC 00-111, "CNS AC Load Study," Revision 1
- NEDC 91-094, "125 VDC/250 VDC Battery Charger Analysis"
- System Operating Procedure 2.2.20.1, "Diesel Generator Operations," Revision 15
- NEDC 99-046, "LOCA Analysis," Revision 16C1

The inspectors also performed a complete equipment alignment inspection of the spent fuel pool cooling system. The inspectors assessed the ability to provide spent fuel pool cooling using the residual heat removal system during refueling operations with the fuel pool gates open. The inspectors reviewed the design requirements contained in the Updated Safety Analysis Report, and the following documents:

- NEDC 00-0105, "Fuel Pool Decay Heat Loads/Alternate Decay Heat Removal Time Limitations," Revision 3
- NEDC 98-049, "Reactor Vessel Natural Circulation," Revision 0

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Failure to Implement a Fire Protection Program Procedure

a. Inspection Scope

The inspectors responded to a fire located in the reactor building torus room on November 26, 2001. Following the fire, the inspectors interviewed licensee responders to determine the cause of the fire and actions that were taken following its discovery by the assigned fire watch for the area.

b. Findings

The inspectors determined that the assigned fire watch failed to implement the procedural requirements of Administrative Procedure 0.39, "Fire Watches," Revision 24. The failure to follow this procedure is a violation of Technical Specification 5.4.1.d which states, "Written procedures shall be established, implemented, and maintained covering the fire protection program implementation."

On November 26, 2001, at approximately 2 p.m., the inspectors responded to a fire located in the reactor building on the torus area floor. Once the fire was extinguished, the inspectors interviewed the fire watch assigned to monitor the affected area. The fire watch stated that rags had ignited below the area where hot work was being performed and that he was cognizant of the rags and had been watching them closely since assuming the watch because sparks were landing on them. After the rags caught on fire, the fire watch immediately reported the fire to the control room.

Additionally, the licensee discovered that the fire watch, who was observing the area on a video monitor, did not know the physical location of the fire and, therefore, could not put it out. The fire watch was monitoring the hot work areas using a video camera located on the 903-foot elevation of the reactor building. The fire was located on the 859-foot elevation of the reactor building on the torus area floor. Although the fire watch was not cognizant of the location of the fire, the watch immediately reported the fire to the control room, and plant operators quickly responded to extinguish it.

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 24, 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.” The inspectors determined that the failure of the fire watch to ensure that the combustible material below the hot work was either removed from the affected area 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 (NCV) (50-298/0107-01) consistent with Section VI.A of the NRC Enforcement Policy. This issue has been entered into the licensee’s corrective action process as Notification 10126869.

This issue was determined to have a credible impact on safety because an actual fire inside the reactor building occurred. This NCV was characterized under the risk significance determination process as having very low safety significance because the fire was quickly identified and extinguished and the fire did not affect any equipment necessary for maintaining safe shutdown conditions. Specifically, the reactor cavity was flooded to greater than 23 feet, the spent fuel pool gates were open, a division of shutdown cooling was operable, and emergency core cooling system instrumentation was not affected.

.2 Routine Fire Protection Inspections

a. Inspection Scope

The inspectors reviewed the following areas throughout the inspection period to determine if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capabilities, and maintained passive fire protection features in good material condition.

The following areas were inspected:

- High pressure coolant injection room
- Reactor building 903 level
- Reactor building northeast quadrant
- Cable expansion room
- Control building basement

b. Findings

No findings of significance were identified.

.3 Routine Fire Drill Inspection

a. Inspection Scope

On December 10, 2001, the inspectors observed and evaluated a fire brigade drill. The drill consisted of a simulated fire located in the service water pump room. The inspectors assessed the fire brigade's performance in the following areas:

- Appropriate protective clothing donned in a timely manner
- Self-contained breathing apparatus properly worn and used
- Fire hoses utilized were capable of reaching the fire
- Effective command and control provided by the fire brigade leader

The inspectors also reviewed the licensee's fire drill critique to verify that areas for improvement were properly identified and all the scenario objectives were met.

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 2 residual heat removal system heat exchanger. 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 13.17, "Residual Heat Removal Heat Exchanger Performance Testing," Revision 10.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination (NDE) Activities

The Cooper Nuclear Station "Third Ten-Year Interval Inservice Inspection (ISI) Program and Schedule," Revision 1.1, was committed to the 1989 Edition, with no Addenda, of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Code. The Refueling Outage RE-20 inspections were the last inspections of the second 40-month period of the third 10-year inspection interval of the program.

c. Inspection Scope

The inspector reviewed the licensee's NDE records for work that was performed for Refueling Outage RE-20. This review was performed to verify that NDE activities were accomplished in accordance with ASME Boiler and Pressure Vessel Code requirements. Nondestructive examination records reviewed are listed in the attachment.

The inspector observed licensee's contractor personnel (Sonic Systems International, Inc.) perform the ASME Code Section XI specified examinations listed below:

System	Component/Weld Identification	Examination Method
Residual Heat Removal	RHR-IR-IA	Ultrasonic Examination
Reactor Water Cleanup	CU-H50	Visual Examination

During the performance of each examination, the inspector verified that the correct NDE procedure was used, procedural requirements or conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period.

The inspector reviewed the NDE certification packages of the contractor personnel who performed the above examinations and verified that they had been properly certified in accordance with ASME Code requirements. The inspector also verified that the correct NDE procedure was used and had been properly qualified.

b. Findings

No findings of significance were identified.

.2 ASME Code Section XI Repair and Replacement Activities

a. Inspection Scope

The inspector reviewed three ASME Section XI Code repair and replacement packages for repair work scheduled to be performed to reactor closed cooling pipe Weld W51, replacement of service water piping, and replacement of High Pressure Core Injection Valve HPCI-CV-15CV.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

Periodic Evaluation of Maintenance Rule Implementation

a. Inspection Scope

During the inspection period, the inspectors reviewed licensee implementation of the maintenance rule program. The inspectors verified that systems, structures, and components were properly scoped and characterized; the appropriate safety significance and performance criteria were established; and the goals established and corrective actions were appropriate. The inspectors assessed the licensee's implementation of the Maintenance Rule to the requirements outlined in 10 CFR 50.65, Administrative Procedure 0.27, "Maintenance Rule Program," Revision 11, and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The inspectors reviewed the following components and/or systems that displayed performance problems:

- High pressure coolant injection system overspeed trip device
- Division I diesel generator
- Service water system piping
- Reactor Core Isolation Cooling System Limit Switch 74
- Service Water Booster Pump D
- Demineralized water system

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

The inspectors reviewed the licensee's risk assessment for equipment outages as a result of planned and emergent activities. The inspectors compared the licensee's risk assessment and risk management activities to the requirements of 10 CFR 50.65(a)(4) and the recommendations of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2. The inspectors also discussed the planned and emergent work activities with planning and maintenance personnel. The inspectors reviewed the following risk evaluations:

- October 14, 2001, high pressure coolant injection system declared inoperable upon failure of the overspeed trip mechanism to reset
- November 11, 2001, maintenance activities associated with pipe replacements of the reactor equipment cooling system during the refueling outage
- November 29, 2001, maintenance activities on Division II, 4160-volt essential bus during the refueling outage
- December 12, 2001, maintenance activities associated with Division II residual heat removal system during the refueling outage

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

.1 Safety-Relief Valve (SRV) Operability Evaluation

a. Inspection Scope

The inspectors reviewed corrective action documents and engineering evaluations associated with the environmental qualification of the safety-relief valve power cables.

b. Findings

The licensee failed to identify and correct a condition adverse to quality. The inspectors determined that the localized temperature and aging affects of the SRV cables, and the brand of the cable used, were different than that specified in the environmental qualification packages. The licensee had several opportunities to identify and correct this condition from April 2000 to October 2001. This was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This issue has been entered into the licensee's corrective action process as Notification 10092693.

On June 21, 2001, Cooper engineers identified concerns regarding the environmental qualifications of power cables going to the SRV solenoids. The engineers determined that the environmental qualification packages did not properly account for temperature effects and the associated acceleration in aging of the cables. The engineers entered the problem into the corrective action program as Notification 10092693. The engineers stated in the notification that the cables were designed for a 40-year qualified life. However, actual ambient temperature conditions could greatly reduce the life expectancy. If the cables exceeded their expected life, there would be no reasonable assurance that the SRV's would operate in a harsh drywell environment during a design basis event.

On July 16, NRC inspectors identified that no operability determination had been performed for this condition and that the engineer's written information provided only a qualitative assessment of SRV operability. Based on the inspectors' concern, operations personnel requested an operability evaluation from the engineering department. The inspectors also presented to the engineers several erroneous assumptions in the original evaluation. An example of this was the use of average drywell ambient temperature instead of actual temperature encountered in the SRV enclosures. Based upon these comments, and further engineering analyses, it was determined that the cable qualified life was reduced from 40 years to approximately 9 years. The cables had been installed in 1995. Therefore, engineers wrote a work order to replace the cables during the next refueling outage.

On October 9, 2001, the inspectors reviewed the proximity of terminal boxes to the solenoid enclosures. In an effort to show the inspectors this configuration, engineers retrieved photographs of the terminal boxes taken in April of 2000. When the engineers examined the photograph of SRV C, they noted that a cable had manufacturer's markings of "Brand-Rex." The environmental qualification package, and all design documentation for the SRV's, stated that Rockbestos cable was used in this application. The engineers informed the inspectors that the use of Brand-Rex cable, in this application, was inconsistent with the brand of cable specified in the environmental qualification packages.

The engineers searched numerous files and databases, from October 9-12, to find the traceability of the Brand-Rex cable. The licensee determined that this cable was installed in 1995. The engineers then developed an operability evaluation to justify the qualification of this cable in this application. The SRV wiring and cables were worked on in several outages since 1995. The cables and configurations were examined several times for qualification requirements during the Environmental Qualification outage in April to May 2000. The cables were also spliced together, after removing terminal blocks, in the 2000 midcycle outage. Finally, the engineers stated that they had conducted a thorough review of SRV cable qualification for the inspector concerns raised from July to October 2001. These unreliable qualification records were examples of the licensee's continuing lack of auditable environmental qualification records for essential equipment.

Section (j) of 10 CFR 50.49 requires that a record of the environmental qualification must be maintained in an auditable form to permit verification that each item of electric equipment important to safety is qualified for its application and meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function. 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. In the case of significant conditions adverse to quality, the measures shall ensure that the cause of the condition is determined and corrective action is taken to preclude repetition."

Contrary to the above, from 1995 to October 2001, the localized temperature and aging affects of the SRV cables, and the brand of cable used to power the SRV solenoids, were different than that specified in the environmental qualification packages. The inspectors determined that this configuration was less conservative than the tested configuration of the SRV cables. The licensee had several opportunities to identify and correct this configuration from April 2000 to October 2001. The failure to identify and correct this condition is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as an NCV (50-298/0107-02) consistent with Section VI.A of the NRC Enforcement Policy.

This finding was more than minor because, if left uncorrected, the cables would have exceeded their qualified life and, therefore, would pose a more significant issue. Based on the subsequent engineering evaluations, it was determined that the cable life was significantly reduced and that the SRV cables required replacement in the upcoming refueling outage. Without such determination, operation in future cycles would present an elevated risk because SRV operation could not be ensured. The inspectors determined that the current risk is very low, because the SRV cables are presently qualified.

This issue also had crosscutting aspects associated with problem identification and resolution. The failure to recognize the problem, with numerous human reviews, represents an example of a problem identification and resolution issue.

.2 Periodic Review of Operability Evaluations

a. Inspection Scope

The inspectors reviewed the technical adequacy of multiple operability evaluations to verify that they were sufficient to justify continued operation of a system or component. The inspectors verified that, although equipment was degraded, the operability evaluation provided adequate justification that the equipment could still meet its Technical Specification, Updated Final Safety Analysis Report, and design bases requirements and that any potential risk increase contributed by the degraded equipment was thoroughly evaluated. The following evaluations were reviewed:

- Operability evaluation for lower than expected cylinder pressure on Division I diesel generator Cylinder 4L (Notification 10131185)
- Operability evaluation for the failure of reactor core isolation cooling system Limit Switch 74 to open Valve RCIC-AO-32 (Notification 001011877)
- Operability evaluation for higher than normal differential pressure across the Division II residual heat removal heat exchanger (Notification 10115221)

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors performed a review of operator workarounds in the control room focusing on annunciators that were either disabled or not functioning properly. The review of a lower-tiered list of operator concerns was also performed to determine if any of these issues should be considered operator workarounds. This review also evaluated the cumulative affects of current operator workarounds.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing

a. Inspection Scope

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 components:

- Main steam isolation valves following maintenance on limit switches
- Reactor coolant system leak test following outage activities
- Drywell personnel airlock local leak rate test following outage activities
- Division II diesel generator balancing run following overhaul
- Repairs on the high pressure coolant injection system overspeed trip device
- Division I service water system following piping replacements

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities

a. Inspection Scope

The inspectors observed control of outage activities to determine whether the licensee appropriately considered the impact on risk. In particular, inspectors observed or reviewed the following outage-related activities:

- Licensee's outage risk control plan
- Portions of the reactor plant cooldown following reactor shutdown
- Selected risk significant outage activities such as configuration control
- Selected clearance and restoration of risk significant equipment and systems
- Control of containment, decay heat removal, inventory, and reactivity
- Selected refueling activities associated with fuel handling and control rod drive refurbishment
- Portions of the reactor plant heatup and startup

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Inappropriate Corrective Actions for Repairs on Reactor Feed Check Valves

a. Inspection Scope

The inspectors reviewed surveillance testing associated with local leak rate testing of the reactor feed primary containment isolation check valves. The inspectors assessed the performance of these valves to satisfy Technical Specification and design basis requirements. A review of corrective action documents, design documents, and discussions with engineering personnel was performed.

b. Findings

The licensee failed to implement effective corrective actions, resulting in repetitive failures of reactor feedwater check valves to pass local leak rate testing requirements from 1983 through November 2001. The licensee had not effectively identified and corrected this significant condition adverse to quality. This was determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This issue has been entered into the licensee's corrective action process as Significant Condition Report 2001-1161.

The inspectors reviewed the results of surveillance tests that obtained the as-found local leak rates associated with Reactor Feed Check Valves RF-CV-13CV, -14CV, -15CV, and -16CV that were performed at the onset of the refueling outage that started November 3, 2001. The test results indicated that RF-CV-13CV, -14CV, and -15CV all exceeded acceptable leakage requirements. RF-CV-15CV and -16CV are the “A” feed line inboard and outboard primary containment isolation check valves. RF-CV-13CV and -14CV are the “B” feed line inboard and outboard primary containment isolation check valves. Both “B” reactor feed line inboard and outboard check valves failed their as-found local leak rate test. Therefore, the licensee appropriately reported this condition to the NRC on November 9, 2001, in accordance with 10 CFR 50.72(b)(3)(ii) as an event or condition that results in the nuclear power plant, including its principal safety barriers, being seriously degraded.

The inspectors reviewed past local leak rate test results for these check valves and noted that there was a high failure rate. Specifically, since 1983 these valves had collectively experienced approximately a 61 percent failure rate. The inspectors discussed this high failure rate with the responsible system and design engineers. The engineers stated that previous maintenance activities did not ensure that appropriate clearances were implemented, necessary for proper valve seat alignment. The engineers stated that they came to this conclusion with the assistance of outside technical experts. The failure to incorporate the appropriate design controls resulted in the seating surfaces of the valves failing to be properly aligned, which caused the valves to rapidly degrade over an operating cycle.

Title 10 of the Code of Federal Regulations, Part 20, Appendix B, Criterion XVI, states in part, that “Measures shall be established to assure that conditions adverse to quality, such as failures, are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.” The failure to determine the appropriate cause for the reactor feed check valve failures from 1983 through November 2001, that resulted in inappropriate corrective actions and subsequent failures, is a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (50-298/0107-03). The licensee documented this issue in their corrective action process as Significant Condition Report 2001-1161.

This issue was considered to have a credible impact on safety, in that the failure of these valves caused a higher than normal containment leakage. This NCV was characterized under the significance determination process as having very low safety significance. Using Inspection Manual Chapter 0609, Appendix H, “Containment Integrity SDP,” the finding was determined to be a Type B finding because these valve failures did not affect core damage frequency. Type B findings related to containment isolation valves in plants with Mark I containments are considered to be Green, based on Table 3 of Appendix H.

This issue was also considered to have crosscutting aspects associated with problem identification and resolution. The failure to correct the problem, with numerous human reviews, represents the second example of a problem identification and resolution issue.

.2 Routine Inspection of Surveillance Tests

a. Inspection Scope

The inspectors observed or reviewed the following surveillance tests to ensure 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.HPCI.103, "HPCI IST and 92 Day Test Mode Surveillance Operation," Revision 16C1
- Surveillance Procedure 6.EE.608, "250V Station Battery Performance Discharge Test," Revision 9
- Surveillance Procedure 6.SLC.102, "SLC Test Mode Surveillance Operation," Revision 9C1

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspector performed an in-office review of Revision 36 to the Cooper Nuclear Station Emergency Plan, submitted October 16, 2001, against 10 CFR 50.54(q) to determine if the revision decreased the effectiveness of the plan.

b. Findings

No findings of significance were identified.

2. RADIATION PROTECTION

Cornerstone: Occupational Radiation Safety and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs during routine and refueling outage operations. The inspector also conducted plant walkdowns within the controlled access area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Area posting and other controls for airborne radioactivity areas, radiation areas, high radiation areas, and very high radiation areas
- Radiation exposure permits and radiological surveys involving airborne radioactivity areas, high radiation areas, and electronic dosimeter alarm setpoints
- Access controls, surveys, and radiation exposure permits for the following significant high dose work areas from Refueling Outage 20: RWP 1102, "RWCU high point vent," RWP 2001-1085, "Under vessel activities in the Drywell"
- Radiation protection program procedures
- Dosimetry placement when work involved a significant dose gradient
- Locked high radiation area key control program
- Controls involved when handling highly radioactive items
- A summary of corrective action documents written since March 1, 2000, that involved high radiation area and work practice incidents (eight Notification reports were reviewed in detail: 4-06989, 4-07021, 4-07130, 4-07715, 4-08306, 4-08244, 00100826, and 0100864)
- Radiation protection self-assessments and field observations
- Radiation Protection Quality Assurance Audit Reports: 01-03, "Radiological Controls and Chemistry," and Surveillance Report S412-0102, "Public Radiation Safety"

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspector interviewed members of the licensee's staff responsible for implementing the radiological environmental, meteorological monitoring, and radioactive material control programs. The inspector observed the following activities and equipment to verify that the above programs were implemented consistent with the licensee's Technical Specifications and/or Offsite Dose Assessment Manual.

- Preparation of airborne particulate and charcoal sample holders for sample collection
- Meteorological instrument condition and data displays at the meteorological tower and control room
- Eight environmental air sampling stations (1, 2, 3, 5, 6, 7, 9, and 10), two surface water sampling stations (1 and 2), and eight thermoluminescent dosimetry stations (1, 2, 3, 5, 6, 7, 9, and 10)

The following items were reviewed and compared with regulatory requirements to determine whether the licensee had an adequate program to verify the impact of radioactive effluent releases to the environment and to ensure that the licensee's surveys and controls were adequate to prevent the inadvertent release of licensed materials into the public domain:

- Implementing procedures for the radiological environmental monitoring program
- Number and location descriptions of the environmental sampling stations as specified in the Offsite Dose Assessment Manual
- Environmental sample analytical results
- Calibration and maintenance records for environmental air sampling equipment and radiation measurement instrumentation
- 1999, 2000, and 2001 environmental assessments
- 1999 and 2000 Annual Radiological Environmental Operating Reports
- The environmental laboratory's performance in the interlaboratory comparison program
- Implementing procedures for the meteorological monitoring program
- Meteorological instrument operability, reliability, and annual meteorological data recovery
- Procedures, methods, and instruments used to survey, control, and release materials from the radiologically controlled area
- Calibration procedures and records for instruments used to perform radiological surveys prior to material release
- Detection sensitivities of radiation survey instruments used for the release of potentially contaminated materials from the radiologically controlled area

- Criteria used for the unrestricted release of potentially contaminated material from the radiologically controlled area

- Environmental sections of Quality Audit Report 00-01, Surveillance Reports S412-0101 and S412-0102, and vendor audits (TLD Processing Supplier Audit SA2000-050 and NUPIC Audit Report A1269823)

- A summary of meteorological, environmental, and release of licensed-radioactive-material-related corrective action reports written since July 1, 1999 (13 of these reports were reviewed in detail: 4-08542, 4-06306, 4-06105, 4-06039, 4-06052, 4-04635, 4-03932, 4-10808, 4-12770, 4-12975, 4-13675, 10090617, and 10102934)

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

.1 Reactor Safety Performance Indicators

a. Inspection Scope

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

- Emergency AC power system unavailability

- High pressure injection system unavailability

- Heat removal system unavailability

- Safety system functional failures

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed corrective action program records for Technical Specification required locked high radiation areas, very high radiation areas, and unplanned exposure occurrences since March 2000 to confirm that these occurrences were properly recorded as performance indicators. Radiologically controlled area entries with exposures greater than 100 millirem were reviewed and selected examples were examined to determine whether they were within the dose projections of the governing radiation exposure permits. Internal dose estimates were reviewed if the radiation worker received a committed effective dose equivalent of more than 100 millirem.

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 since March 2000 to determine if any events exceeded the performance indicator thresholds.

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 are 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 licensee failed to implement effective corrective actions, resulting in repetitive scaffold construction nonconformances potentially affecting the operation of equipment important to safety. Examples included scaffolding built in the proximity of and over safety-related equipment, as well as scaffold components that could have interfered with the safety function of plant components. This issue has been entered into the licensee’s corrective action process as Notification 10127237.

The inspector reviewed Notification 10127237 pertaining to scaffolding that was constructed which potentially blocked the stroke paths of Main Steam Isolation Valves 80B, 80C, 86B, 86C, and 86D. The licensee identified this condition on November 27, 2001, prior to performance of Surveillance Procedure 6.MS.303, “Main Steam Isolation Valve Limit Switch as Found Values,” that required stroking of the valves.

NRC Inspection Report 50-298/00-04 documented an example where scaffold construction was not performed in accordance with Maintenance Procedure 7.0.7, “Scaffolding Construction and Control.” This violation, identified on February 26, 2000, prevented operation of secondary containment isolation Valve HV-AOV-257AV. This occurrence was documented as a violation of Technical Specification 5.4.1(a) for the failure to follow Maintenance Procedure 7.0.7 (50-298/0004-03). In September 2001, during an NRC team inspection of the licensee’s problem identification and resolution program, the inspectors identified numerous examples where scaffolding was not constructed in accordance with Maintenance Procedure 7.0.7. These occurrences were documented in NRC Inspection Report 50-298/01-10 and determined to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI, based on ineffective corrective actions.

The inspectors reviewed the corrective actions associated with the scaffold issues identified in September 2001 during the problem identification and resolution inspection. These corrective actions were documented in Significant Condition Report 2001-0895. The inspector noted that all immediate and long-term corrective actions assigned to prevent recurrence of the issue had been completed prior to the problem being identified on November 27, 2001. This issue indicated that the licensee’s corrective actions associated with Significant Condition Report 2001-895 were not effective in preventing a recurrence.

Title 10 of the Code of Federal Regulations, Part 20, Appendix B, Criterion XVI, states in part that “Measures shall be established to assure that conditions adverse to quality, such as nonconformances, are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.” The failure to establish corrective actions to preclude repetition of scaffold nonconformances is being considered a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (50-298/0107-04). The licensee documented this issue in their corrective action process as Notification 10127237.

This issue was considered to have a credible impact on safety, in that the failure to properly construct scaffolds could affect the operation of equipment important to safety. This NCV was characterized under the significance determination process as having very low safety significance because the failure to construct scaffolds in accordance with the procedural requirements did not result in any equipment failure or loss of safety function.

4OA3 Event Followup

.1 <u>(Closed) Licensee Event Report (LER) 05000298/2001-004</u>

On September 7, 2001, a combination of a lightning strikes and an equipment malfunction resulted in the loss of both qualified offsite power sources. An NRC inspection was performed on this event and determined that a violation of Technical Specification 5.4.1(a) resulted. NRC Inspection Report 50-988/2001010, Section 4OA2(a), documented the details associated with this finding.

.2 (Closed) LER 05000298/2001-05

On October 17, 2001, the high pressure coolant injection system was declared inoperable following failure of the overspeed trip mechanism to reset following testing. The licensee identified that a spring in the trip mechanism needed to be replaced. Following replacement of the spring, the overspeed trip mechanism was tested and the high pressure coolant injection system was declared operable. The licensee determined that the spring failed at normal end of life. This event did not constitute a violation of NRC requirements.

.3 (Closed) LER 05000298/2001-006

On November 2, 2001, the licensee identified that personnel inadvertently placed the operating switches to close for Suppression Chamber Vacuum Relief Valves PC-AO-243 and PC-AO-244 while performing a tagout of another system for maintenance. This was determined to be a violation of Technical Specification 5.4.1(a). This violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy (50-298/0107-05). This issue has been entered into the licensee's corrective action process as Notification 10120889.

These valves provide a primary containment isolation function and also provide a suppression chamber vacuum relief function. The primary containment isolation function was not impacted since the valves were closed. This issue was considered to have a credible impact on safety, in that the suppression chamber vacuum relief function was disabled. This event was characterized as having very low safety significance because licensed operators identified that the switches were in the incorrect position and corrected the condition within approximately 3 hours.

.4 (Closed) LER 05000298/2001-07

On November 9, 2001, the licensee identified during testing that Reactor Feedwater Check Valves RF-CV-13CV, -15CV, and -16CV had leakage that exceeded requirements. This event was inspected and documented in Section 1R22.1, "Inappropriate Corrective Actions for Repairs on Reactor Feed Check Valves," of this inspection report.

4OA6 Meetings

.1 Exit Meeting Summary

On January 3, 2001, the results of the inspection were discussed by the resident inspectors with Mr David Wilson, Vice President-Nuclear.

The public radiation safety inspector presented the inspection results to Mr. M. Coyle, Assistant Vice President, Nuclear, and other members of licensee management at the conclusion of the inspection on October 18, 2001.

The occupational radiation safety inspector and the inservice inspection inspector presented their inspection results to Mr. M. Coyle, Assistant Vice President, Nuclear, and other members of licensee management at the conclusion of the inspection on November 16, 2001.

The emergency preparedness inspector presented the inspection results to Mr. P. Hays, Interim Emergency Preparedness Manager, and other members of licensee management during a telephonic exit interview conducted on November 29, 2001.

During these exit meetings, Cooper management acknowledged all the findings presented. Additionally, the inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information is discussed in this inspection report.

4OA7 Licensee Identified Violations

The following findings of very low safety significance 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 an NCV.

If you deny any of the NCVs, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper Nuclear Station.

NCV Tracking Number	Requirement Licensee Failed to Meet
50-298/0107-06	Technical Specification 5.4.1 requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Section 2.2 of Procedure 0.24, “Working Over or in Reactor Vessel or Fuel Pool Requirements,” Revision 14, requires that items hanging in the spent fuel pool that would create a high radiation area if removed from the water will be locked and controlled by radiation protection personnel. On October 26, 2001, the licensee identified that two buckets suspended in the spent fuel pool had underwater contact radiation levels of approximately 40 and 120 Rem per hour. These buckets were not locked or controlled and would have created a high radiation area if removed from the water, as described in the licensee’s corrective action program, reference Notification 010120317. Prior to implementing corrective actions associated with Notification 010120317, the licensee identified an additional example of a failure to control and lock a bucket of radioactive material in the spent fuel pool on November 29, 2001. The underwater contact radiation levels on the bucket were

approximately 200 Rem per hour. The licensee entered this issue into the corrective action process as Notification 10127300. This violation is being treated as a noncited violation.

The safety significance of this violation was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no actual overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.

50-298/0107-07

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 Equipment Controls. On November 9, 2001, the licensee identified that, during performance of a tagout, personnel inappropriately racked out the Residual Heat Removal Pump B breaker. This is being treated as an NCV. The licensee entered this issue into their corrective action process as Notification 10122626.

The safety significance of this violation was determined to be very low. Residual heat removal Pump B was not in use when the breaker was removed and did not affect the ability to maintain the plant in a safe shutdown condition. Also, the removal of the wrong breaker was immediately identified by the licensee and it was returned to service within 1 hour.

50-298/0107-08

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 Equipment Controls. On November 15, 2001, the licensee identified that personnel had inappropriately removed a seismic restraint/pipe support (SW-H138) from an operable and running service water piping system. The licensee entered this issue into their corrective action process as Notification 10123800.

The safety significance of this violation was determined to be very low. Although the operators declared the service water system inoperable, the removal of the support hanger did not affect the service water system’s ability to perform its safety function to maintain the plant in a safe shutdown condition. The section of piping affected was immediately isolated following discovery of the missing hanger until repairs were performed.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Allen, Quality Assurance Specialist
J. Bebb, Security Fitness-for-Duty Supervisor
C. Blair, Licensing Engineer
T. Chard, Radiation Protection and Chemistry Manager
M. Coyle, Assistant Vice President-Nuclear
F. Diya, Plant Engineering Department Manager
K. Dorwick, Outage/Work Control Department
P. Fleming, Licensing Manager
J. Flaherty, Site Regulatory Liaison
R. Gardner, Quality Assurance Senior Manager
P. Hays, Interim Emergency Preparedness Manager
B. Houston, Quality Assurance Operations Manager
J. Hutton, Plant Manager
A. Jacobs, Performance Analysis Department/Corrective Actions Program
K. Kirkland, Acting Site Support Senior Manager
D. Kunsemiller, Risk and Regulatory Affairs Manager
D. Linnen, Senior Manager, Training
T. McClure, ISI Engineer
W. Macecevic, Operations Manager
D. Madsen, Licensing Engineer, Regulatory Affairs
C. Markert, Engineering Support Department Manager
D. Myers, Senior Manager, Site Support
J. Ranalli, Senior Manager, Engineering
M. Schaible, Operations Training Supervisor
R. Steele, Design Engineering Department, Electrical Supervisor
C. Stipp, Environmental Coordinator, Radiation Protection
J. Sumpter, Project Manager, Licensing
N. Wetherell, Maintenance Manager
D. Wilson, Vice President-Nuclear

ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

05000298/2001-004	LER	A combination of a lightning strike and an equipment malfunction resulted in the loss of both qualified offsite power sources (Section 4OA3.1)
05000298/2001-05	LER	The high pressure coolant injection system was declared inoperable following failure of the overspeed trip mechanism to reset (Section 4OA3.2)

05000298/2001-006	LER	Personnel inadvertently placed the operating switches to closed for Suppression Chamber Vacuum Relief Valves PC-AO-243 and PC-AO-244 operating switches to close (Section 4OA3.3)
05000298/2001-07	LER	Reactor Feedwater Check Valves RF-CV-13CV, -15CV, and -16CV had leakage that exceeded requirements (Section 4OA3.4)

<u>Opened and Closed During this Inspection</u>

50-298/07-01	NCV	Failure to follow procedure resulting in a fire (Section1R05)
50-298/07-02	NCV	Failure to identify and correct 10 CFR 50.49 requirement nonconformances associated with SRV cables (Section 1R15.1)
50-298/07-03	NCV	Failure to identify and correct design control deficiencies associated with the reactor feed check valves (Section 1R22.1)
50-298/07-04	NCV	Ineffective corrective actions resulting in repetitive scaffold construction nonconformances (Section 4OA2)
50-298/07-05	NCV	Failure to follow procedure resulting in inadvertently closing suppression chamber vacuum relief valves (Section 4OA3)
50-298/07-06	NCV	Failure to follow procedure for controlling radioactive material in the spent fuel pool (Section 4OA7)
50-298/07-07	NCV	Failure to follow procedure resulting in inappropriately racking out a residual heat removal pump breaker (Section 4OA7)
50-298/07-08	NCV	Failure to follow procedure resulting in inappropriately removing a service water support hanger (Section 4OA7)

LIST OF ACRONYMS USED

ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
ISI	inservice inspection
LER	licensee event report
NCV	noncited violation
NDE	nondestructive examination
SRV	safety relief valve

LIST OF DOCUMENTS REVIEWED

Procedures

Administrative Procedure 0.31, “Equipment Status Control,” Revision 10

Administrative Procedure 0.9, “Tagout,” Revision 28

Administrative Procedure 0.50, “Outage Management Program,” Revision 12

Alarm Procedure 2.3_9-5-1, “Panel 9-5-Annunciator 9-5-1,” Revision 2

Surveillance Procedure 6.MS.301, “Main Steam Isolation Valve Limit Switch Channel Calibration,” Revision 4

Surveillance Procedure 6.PC.511, “High Pressure Coolant Injection Local Leak Rate Tests,” Revision 6

Surveillance Procedure 6.PC.519, Reactor Core Isolation Coolant Local Leak Rate Tests,” Revision 6

Surveillance Procedure 6.PC.524, “Primary Containment Airlock Local Leak Rate Tests,” Revision 10

General Operating Procedure 2.1.1, “Startup Procedure,” Revision 98

General Operating Procedure 2.1.1.2, “Technical Specifications Pre-Startup Checks,” Revision 26

Corrective Action Documents

CR 94-0796, “Excessive leakage of RF-CV-14CV”

RCR 98-0696, “RF-CV-14CV and 16CV failed LLRT”

RCR 2000-0203, “RF-CV-15CV failed LLRT”

SCR 2001-1036, “HPCI Automatic Trip Reset Failure”

Work Order 4203468, “Disassemble HOV-PCV2770"

Notification 001011877, “Limit Switch 74 failed to open RCIC-AO-32”

Other

Cooper Nuclear Station IST Basis Document, Revision 4

Design Drawing 920-3, “Tilting Disc Check Valve,” Revision G, 1971

NUMBER	DESCRIPTION	REVISION/ DATE
N/A	Interval 3, Inspection Period 2, Fall 2001 RFO ASME Code Examination Plan	
N/A	Third 10-Year Interval Inservice Inspection Program	1.1
CSP-ISI-70	Magnetic Particle Examination for Cooper Nuclear Station	2
CSP-ISI-211	Manual Ultrasonic Examination of Nozzle Inner Radii for Cooper Station	CC1
PDI-UT-1	PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds	C
PDI-UT-2	PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds	C
PDI-UT-3	PDI Generic Procedure for the Ultrasonic through Wall Sizing in Pipe Welds	C
PDI-UT-5	PDI Generic Procedure for Straight Beam UT Examination of Bolts and Studs	C
SA-01-004	Engineering Support Department Self-Assessment, ASME Section XI Inservice Inspection (ISI) Program	11/08/01
SSI-A-005	Qualification and Certification of Nondestructive Examination Personnel	18
SSI-A-011	Qualification and Certification of Visual Testing Personnel	8
SSI-A-020	Personnel Visual Acuity Examination Procedure	0
N/A	Ultrasonic Testing Calibration Standards Report	12/17/1997

Nondestructive Examination Data Reports

R-010	R-122	R-150	R-154	R-164
R-084	R-139	R-151	R-157	R-168
R-085	R-146	R-152	R-158	R-169
R-120	R-149	R-153		

<u>Work Packages</u>

4159603
4159899
4160040