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

March 17, 2000

J. H. Swailes, Vice President of
Nuclear Energy
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

SUBJECT: NRC INSPECTION REPORT NO. 50-298/00-01

Dear Mr. Swailes:

This refers to the inspection conducted on January 2 through February 19, 2000, at the Cooper Nuclear Station facility. The enclosed report presents the results of this inspection. The results of this inspection were discussed on February 12, 2000, with Mr. J. McDonald and other members of your staff.

The inspectors examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspectors examined a selection of procedures and representative records, observed activities, and conducted interviews with personnel.

Based on the results of this inspection, the NRC has determined that three violations of NRC requirements occurred. These violations are being treated as noncited violations (NCV's), consistent with the Interim Enforcement Policy for pilot plants. These NCV's are described in the subject inspection report. If you contest any of the violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the 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 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, if requested, will be placed in the NRC Public Document Room (PDR).

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

Sincerely,

/RA/

Charles S. Marschall, Chief
Project Branch C
Division of Reactor Projects

Docket No.: 50-298
License No.: DPR-46

Enclosure:
NRC Inspection Report No.
50-298/00-01

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-298
License No.:	DPR 46
Report No.:	50-298/00-01
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	January 2 through February 19, 2000
Inspectors:	J. Clark, Senior Resident Inspector M. Hay, Resident Inspector A. Earnest, Security Specialist
Approved By:	Charles S. Marschall, Chief, Project Branch C Division of Reactor Projects

ATTACHMENTS:

1. Supplemental Information
2. NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

Cooper Nuclear Station
NRC Inspection Report 50-298/00-01 (DRP)

This report covers a 7-week period of baseline resident inspection.

The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process in draft Inspection Manual Chapter 0609. The body of the report is organized under the broad categories of Reactor Safety and Other Activities as reflected in the summary below.

Cornerstone: Mitigating Systems

Green. Plant personnel failed to perform required evaluations because licensed operators inappropriately declared the reactor equipment cooling system operable following the discovery of a system leak greater than design allowances.

On December 30, 1999, the NRC granted a Notification of Enforcement Discretion indicating our intention to exercise discretion not to enforce compliance with Technical Specification 3.7.3, "Reactor Equipment Cooling System." This discretion only related to the noncompliance with Technical Specification 3.7.3 resulting from continued operation of the plant with excessive reactor equipment cooling system leakage. Technical Specification 5.5.11 requires that plant personnel perform an evaluation under the safety function determination program whenever safety-related support systems are inoperable and cross-divisional equipment is out of service. Because operators erroneously declared the reactor equipment cooling system operable, the required assessment was not conducted prior to removing Residual Heat Removal System B from service for testing. However, because the duration of testing was short, and no actual loss of safety function was experienced during this time, this issue had very low safety significance based upon the significance determination process. Licensing personnel documented this issue in their corrective action process as Problem Information Report 4-05856. (Section 1R15.)

Green. Licensee engineers determined that the turbine bypass valve fast open feature was not enabled until 33 percent of rated thermal power.

Technical Specification 3.7.7 requires that the fast open feature be enabled prior to exceeding 25 percent of rated thermal power. However, the fast open feature was inoperable whenever the reactor was operated between 25 and 33 percent. The cause of the inoperable fast open feature for the turbine bypass valves was a design error made during original construction of the facility that was not identified prior to the implementation of Improved Technical Specifications in August of 1998. Engineers inappropriately designed the turbine bypass valve controller resulting in the blocking of the fast open feature of the valves until approximately 33 percent rated thermal power. The turbine bypass valve fast open feature is required to prevent exceeding minimum critical power ratio limits for certain transients while the reactor is operating between 25 and 30 percent of rated thermal power. However, the plant is infrequently operated in this region, resulting in a low probability of occurrence for these transients. Reactor engineering personnel also provided corollary data and vendor information to demonstrate that there was still considerable margin to safety limits. As a result, this issue was characterized as having very low safety significance based upon the significance determination process. Licensing personnel documented that personnel had failed to identify this deficiency in the corrective action process as Significant Condition Report 2000-0024 (Section 1R22).

Green. Plant personnel failed to adequately test the permissive for the fast opening feature of the turbine bypass valves.

Although technicians tested that the fast open feature would function when the reactor was near rated thermal power, the functional testing did not include a verification that the permissive enabled the fast open feature at 25 percent rated thermal power. This was in noncompliance with Technical Specification Surveillance Requirement 3.7.7.2. The turbine bypass valve fast opening feature is required to prevent exceeding minimum critical power ratio limits for certain transients while the reactor is operating between 25 and 30 percent of rated thermal power. However, the plant is infrequently operated in this region, resulting in a low probability of occurrence for these transients. Reactor engineering personnel also provided corollary data and vendor information to demonstrate that there was still considerable margin to safety limits. As a result, this issue was characterized as having very low safety significance based upon the significance determination process. Licensing personnel documented the procedure inadequacy in their corrective action process as Significant Condition Report 2000-0024 (Section 1R22).

Report Details

During most of this inspection period, the plant operated at 100 percent power, with the exception of minor power reductions for control valve testing and control rod pattern adjustments. On January 7, 2000, operations personnel reduced reactor power to approximately 15 percent to permit an entry into the drywell to assess leakage from the reactor equipment cooling system. On January 8, 2000, operators conducted a plant shutdown to repair the identified leak. The plant was restarted on January 14, 2000, and achieved full power on January 17, 2000. The plant continued to operate at 100 percent power for the remainder of the period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R03 Emergent Work

a. Inspection Scope

On February 4, 2000, the inspectors reviewed the activities associated with a leak in Service Water Subsystem A. Technicians identified a pin-hole leak in the reactor equipment cooling heat exchanger backwash line. The inspectors discussed the problem with operations and engineering personnel. The inspectors also reviewed the limiting conditions for operation log entries and the established boundaries for the emergent work.

On February 10, 2000, the inspectors evaluated the reported failure of the drywell atmospheric monitor. The monitor provides particulate, noble gas, and iodine analysis capability for the drywell. The inspectors reviewed the limiting conditions for operation with operations personnel. The inspectors also discussed the compensatory actions and associated equipment operability with the operators.

b. Observations and Findings

The inspectors did not identify any findings.

1R04 Equipment Alignments

.1 Routine Inspection

a. Inspection Scope

The inspectors performed a partial walkdown of the standby liquid control system. The inspection included a review of the component alignments designated in System Operating Procedure 2.2.74A, “Standby Liquid Control System Component Checklist,” Revision 6.

While Diesel Generator 2 was out of service for routine maintenance, the inspectors performed a partial walkdown of Diesel Generator 1. The inspection included a review of the component alignments designated in System Operating Procedure 2.2.20.2A, “Standby AC Power System (Diesel Generator) Component Checklist (Div 1),” Revision 0.

b. Observations and Findings

The inspectors did not identify any findings.

.2 Semiannual Inspection

a. Inspection Scope

The inspectors conducted a complete walkdown of the 125/250 Volt direct current (dc) system. The inspectors verified correct configuration of the breakers and charging equipment. The inspectors observed the physical characteristics of the battery cells and interconnections. The inspectors also verified that support equipment, including ventilation and monitoring, were properly aligned and operating. The inspection included a review of System Operating Procedures 2.2.24 / 2.2.24A / 2.2.25 / 2.2.25A, “250 VDC Electrical System / Power Checklist” and “125 VDC Electrical System / Power Checklist.”

b. Observations and Findings

The inspectors did not identify any findings.

1R05 Fire Protection

.1 Monthly Routine Inspection

a. Inspection Scope

The inspectors performed fire protection walkdowns to assess the material condition of plant fire protection equipment and proper control of transient combustibles. Specific risk-significant areas included Battery Rooms 1A and 1B and the auxiliary relay room.

b. Observations and Findings

The inspectors did not identify any findings.

1R09 Inservice Testing

a. Inspection Scope

The inspectors observed significant portions, or reviewed the performance of, the following in-service test procedures:

- Surveillance Procedure 6.1RHR.201, ”RHR Power Operated Valve Operability Test (IST)(Div 1),” Revision 8C3

- Surveillance Procedure 6.1RHR.101, "RHR Test Mode Surveillance Operation (IST)(Div 1)," Revision 10
- Surveillance Procedure 6.2CS.101, "Core Spray Test Mode Surveillance Operation (IST)(Div 2)," Revision 12

b. Observations and Findings

The inspectors did not identify any findings.

1R10 Large Containment Isolation Valves

a. Inspection Scope

On January 14, 2000, the inspectors reviewed the pressure testing of the drywell personnel airlock seals. Technicians performed this test, as required, prior to plant mode change after the forced outage. The inspectors also reviewed the adequacy of the test equipment used to conduct the test.

b. Observations and Findings

The inspectors did not identify any findings.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the maintenance rule data for the service water, reactor equipment cooling, and residual heat removal booster pump systems. The inspectors reviewed maintenance history of these systems, and conducted interviews with system engineers, to assess the accuracy of availability, functional failure, and goal-setting data. The inspectors also discussed reporting relationships between maintenance rule data and baseline inspection performance indicators with the applicable system engineers.

b. Observations and Findings

The inspectors did not identify any findings.

1R13 Maintenance Work Prioritization

a. Inspection Scope

The inspectors reviewed the work prioritization and control activities associated with the forced outage period of January 8-14, 2000. The inspectors discussed selected activities with operations and work control personnel regarding risk evaluations and overall plant configuration control.

The inspectors reviewed weekly and daily work schedules to identify scheduled risk significant activities. Operations and work control personnel discussed selected activities with the inspectors regarding risk evaluations and overall plant configuration control. The inspectors discussed emergent work issues with work control center personnel and reviewed the prioritization of scheduled activities when scheduling conflicts occurred. Specifically, the inspectors reviewed the repair of a weld at the interface of the service water and reactor equipment cooling system.

b. Observations and Findings

The inspectors did not identify any findings.

1R14 Nonroutine Plant Evolutions

a. Inspection Scope

On January 8, 2000, the inspectors monitored the control room activities associated with the forced shutdown to repair a reactor equipment cooling system leak. The inspectors observed the operators conduct the plant shutdown and assessed their use of required procedures.

On January 14, 2000, the inspectors monitored the control room activities associated with the plant startup after the forced outage. The inspectors observed the operators conduct the plant startup and assessed their use of required procedures.

b. Observations and Findings

The inspectors did not identify any findings.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following operability evaluations for technical adequacy, applicable compensatory measures, and impact on continued plant operation:

- Service water piping operability with pin-hole leaks
- Average and local power range monitor operability after half-scram on February 2, 2000
- Operability determination for reactor equipment cooling operation and associated equipment realignments

b. Observations and Findings

The inspectors did not identify any findings associated with the service water piping or power range monitor problems.

On December 30, 1999, the NRC granted a notification of enforcement discretion indicating our intention to exercise discretion not to enforce compliance with Technical Specification 3.7.3, “Reactor Equipment Cooling System.” This discretion only related to the noncompliance with Technical Specification 3.7.3 resulting from continued operation of the plant with excessive reactor equipment cooling system leakage. On December 31, 1999, the inspectors noted that operations personnel had declared the system operable after receiving the notification. Operators had misinterpreted the discretion granted and considered the reactor equipment cooling system to be operable.

Technical Specification 5.5.11 requires that plant personnel perform an evaluation under the safety function determination program whenever safety-related support systems are inoperable and cross-divisional equipment is out of service. The inspectors determined that on December 31, 1999, the residual heat removal system was tested, resulting in the low pressure coolant injection mode being declared inoperable. With reactor equipment cooling inoperable, the support functions to both divisions is lost; therefore, either train of low pressure coolant injection is cross-divisional equipment. Because operators erroneously declared the reactor equipment cooling system operable, the required assessment was not conducted prior to removing Residual Heat Removal System B from service for testing.

The failure to evaluate a potential loss of safety function in accordance with Technical Specification 5.5.11 is a violation. However, the duration of testing was short, and no actual loss of safety function was experienced during this time. Additionally, the reactor equipment cooling system was still capable of providing more than 7 days of cooling at that time. As a result, this issue had very low safety significance based upon the significance determination process. Licensing personnel documented this issue in their corrective action process as Problem Information Report 4-05856. We are treating the violation as a noncited violation consistent with the Interim Enforcement Policy for pilot plants (50-298/2001-01).

1R16 Operator Workarounds

.1 Routine Inspection

a. Inspection Scope

The inspectors reviewed the licensee’s list of operator workarounds. The inspectors selected several items for specific review. These items were reviewed for their overall significance and the ability of operators to implement the workaround during various plant conditions, including abnormal and emergency conditions. The specific workarounds reviewed were:

- 98-020: Plant heating and ventilation for battery rooms inadequate to maintain temperature in cold conditions

- 96-007: Manual / auto controllers for reactor feed pumps can be lowered to point causing lockout

- 96-026: Fire protection panel alarm problems

- 98-018: Nuisance rod withdrawal block and APRM upscale alarms during operation

b. Observations and Findings

The inspectors did not identify any findings.

.2 Semiannual Inspection

a. Inspection Scope

The inspectors performed an assessment of the cumulative effects of operator workarounds. The inspectors reviewed the potential for misoperation of systems, the increased risk of initiating events caused by the workarounds, and whether operators would be able to respond to transients in a correct and timely manner.

b. Observations and Findings

The inspectors did not identify any findings.

1R19 Postmaintenance Testing

a. Inspection Scope

The inspectors observed or evaluated the following postmaintenance tests to determine whether the tests adequately confirmed equipment operability:

- Reactor equipment cooling system restoration after leak repair on Drywell Fan Cooling Unit C

- Service water to reactor equipment cooling heat exchanger pipe replacement testing

- Testing and documentation to return the Division 2 emergency diesel generator to service on December 31, 1999

b. Observations and Findings

The inspectors did not identify any findings.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed or reviewed the following tests:

- 6.CRD.301, “Control Rod Withdrawal/Operability Test,” Revision 6C1
- 6.2DG.101, “Diesel Generator 31 Day Operability Test (Div 2),” Revision 17
- 6.2SW.101, “Service Water Surveillance Operation (Div 2),” Revision 8
- 6.TG.301, “Main Turbine Steam Bypass System Response Time Testing,” Revision 3
- 6.RPS.302,”Main Turbine Stop Valve Closure and Steam Valve Functional Test,” Revision 10C1

b. Observations and Findings

The inspectors did not identify any findings associated with Surveillances 6.CRD.301, 6.2DG.101, and 6.2SW.101.

Technical Specification 3.7.7, “The Main Turbine Bypass System,” requires the turbine bypass system to be operable at greater than or equal to 25 percent rated thermal power. On January 8, 2000, the licensee identified that the turbine bypass system fast open feature was not operable until 33 percent power. The cause of the inoperable fast open feature for the turbine bypass valves was a design error made during original construction of the facility that was not identified prior to the implementation of Improved Technical Specifications in August of 1998. Engineers inappropriately designed the turbine bypass valve controller, resulting in the blocking of the fast open feature of the valves until approximately 33 percent rated thermal power. The turbine bypass valve fast open feature is required to prevent exceeding minimum critical power ratio limits for certain accident transients while the reactor is operating between 25 and 30 percent of rated thermal power.

The inspectors noted that the plant is infrequently operated between 25 and 33 percent of rated thermal power, resulting in a low probability of occurrence for these accidents. Reactor engineering personnel also provided corollary data and vendor information to demonstrate that there was still considerable margin to safety limits. As a result, this issue was characterized as having very low safety significance based upon the significance determination process. Licensing personnel documented that personnel had failed to identify this deficiency in the corrective action process as Significant Condition Report 2000-0024.

10 CFR Part 50, Appendix B, Criterion 16, “Corrective Action,” requires, in part, that measures shall be established to assure that conditions adverse to quality, such as

deficiencies, are promptly identified and corrected. The failure of licensee engineers to identify that the main turbine bypass system contained a design deficiency during the review of plant design related to the implementation of Improved Technical Specifications is a violation of this requirement. We are treating this violation as a noncited violation, consistent with the Interim Enforcement Policy for pilot plants (50-298/0001-02).

Surveillance Requirement 3.7.7.2 requires the performance of a system functional test of the main turbine bypass system every 18 months. Surveillance Procedures 6.TG.301 and 6.RPS.302 were performed and tracked by the licensee to satisfy Surveillance Requirement 3.7.7.2. Upon review of these surveillance procedures, the inspectors noted that the procedures failed to perform a complete functional test of the main turbine bypass valves. Although technicians tested that the fast open feature would function when the reactor was near rated thermal power, the functional testing did not include a verification that the permissive enabled the fast open feature at 25 percent rated thermal power.

The licensee agreed that the surveillance procedures being implemented to ensure compliance with Surveillance Requirement 3.7.7.2 were inadequate. Licensee personnel informed the inspectors that an instrumentation and controls procedure had been written to test the functionality of the permissive circuit used to enable the fast open feature at greater than or equal to 25 percent power. However, this procedure was not conducted to ensure compliance with the requirements. The failure to perform a functional test of the fast opening feature of the turbine bypass valves while continuing to operate the plant is a violation of Technical Specification 3.7.7. We are treating this violation as a noncited violation consistent with the Interim Enforcement Policy for pilot plants (50-298/2001-03). Licensing personnel documented the procedure inadequacy in their corrective action process as Significant Condition Report 2000-0024.

OTHER ACTIVITIES

4OA1 Closing of Unresolved Item

(Closed) Unresolved Item 9810-01: Nuclear Energy Institute Vendor Audit AUD-SPS-97-01, dated March 17, 1997, identified that, prior to transferring an employee to Cooper Nuclear Station, another utility had completed an “update” investigation without performing character reference checks as part of its background investigation. NRC review of 10 CFR 73.56 and NUMARC 89-01 (appended to Regulatory Guide 5.66, “Access Authorization Program for Nuclear Power Plants”) determined that a reference check is not required as part of an updated background investigation.

4OA2 Meetings

.1 Exit Meeting Summary

On February 16, 2000, the inspectors conducted a meeting with J. McDonald, Plant Manager, and other members of plant management and presented the inspection results. The plant management acknowledged the findings presented. Plant management also informed the inspectors that no proprietary material was examined during the inspection.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Caudill, Senior Manager
T. Chard, Radiological Manager
J. Dillich, Fuels and Reactor Engineering Manager
J. Flaherty, Acting Design Engineering Manager
M. Hale, Site Support Senior Manager
B. Houston, Quality Assurance Operations Manager
J. Lewis, Fuels and Reactor Engineering Supervisor
W. Macecevic, Operations Manager
S. Mahler, Assistant Licensing Manager
E. McCutchen, Licensing Engineer
J. McDonald, Plant Manager
R. Sessoms, Quality Assurance Senior Manager
J. Swailes, Vice-President - Nuclear
R. Thorson, Work Control Manager
R. Wachowiak, Risk Management Supervisor

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-298/2000-01	NCV	Failure to perform safety function determination program in accordance with Technical Specification 5.5.11
50-298/2000-02	NCV	Inoperability of fast open feature of turbine bypasses
50-298/2000-03	NCV	Inadequate surveillance testing for turbine bypasses

Closed

50-298/9810-01	URI	Nuclear Energy Institute Vendor Audit Number AUD-SPS-97-01, dated March 17, 1997
50-298/2000-01	NCV	Failure to perform safety function determination program in accordance with Technical Specification 5.5.11
50-298/2000-02	NCV	Inoperability of fast open feature of turbine bypasses
50-298/2000-03	NCV	Inadequate surveillance testing for turbine bypasses

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The NRC revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety:	Radiation Safety:	Safeguards:
•Initiating Events	•Occupational	•Physical Protection
•Mitigating Systems	•Public	
•Barrier Integrity		
•Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process and assigned colors of GREEN, WHITE, YELLOW, OR RED. GREEN findings are indicative of issues that, while they may not be desirable, represent little effect on safety. WHITE findings indicate issues with some increased importance to safety, which may require additional NRC inspections. YELLOW findings are more serious issues with an even higher potential to effect safety and would require the NRC to take additional actions. RED findings represent an unacceptable loss of safety margin and would result in the NRC taking significant actions that could include ordering the plant shut down.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing incremental degradation in safety: GREEN, WHITE, YELLOW, AND RED. The color for an indicator corresponds to levels of performance that may result in increased NRC oversight (WHITE); performance that results in definitive, required action by the NRC (YELLOW); and performance that is unacceptable but still provides adequate protection to public health and safety (RED). GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an action matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. As a licensee's safety performance degrades, the

NRC will take more and increasingly significant action as described in the matrix. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings.

More information can be found at: http:\\www.nrc.gov\NRR\OVERSIGHT\index.html.