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

May 16, 2000

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

SUBJECT: NRC SUPPLEMENTAL INSPECTION REPORT NO. 50-298/00-05

Dear Mr. Swailes:

This refers to the inspection conducted on April 17 to 20, 2000, at the Cooper Nuclear Station facility. The enclosed report presents the results of this inspection. The results of this inspection were discussed on April 20, 2000, with Mr. B. Rash and other members of your staff and telephonically on May 9, 2000, with Ms. S. Mahler of your staff.

Based on the results of this inspection, the NRC has determined that one Severity Level IV violation of NRC requirements occurred. The violation is being treated as a noncited violation (NCV), consistent with the Enforcement Policy. The NCV is described in the subject inspection report. If you contest the violation or severity level of the NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response, if requested, will be placed in the NRC Public Document Room.

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.
50-298/00-05

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-298
License No.:	DPR 46
Report No.:	50-298/00-05
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska 68321
Dates:	April 17-20, 2000
Inspectors:	J. Russell, Resident Inspector, San Onofre Nuclear Generating Station M. Hay, Resident Inspector, Cooper Nuclear Station
Approved By:	C. Marschall, Chief, Branch C Division of Reactor Projects

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	NRC's Revised Reactor Oversight Program

SUMMARY OF FINDINGS

Cooper Nuclear Station
NRC Inspection Report No. 50-298/00-05

This report covers a one week period of supplemental inspection.

The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

- Green. The inspectors determined that licensee management generally understood the causes of poor engineering performance; however, the causes were not yet corrected.

  This focused inspection was performed by the NRC to assess a licensee engineering self-assessment performed during September and October 1999. The inspection is being documented as a supplemental inspection; however, no “white” issue characterization caused the inspection. In 1995, the licensee relocated and reorganized the engineering staff. The licensee completed an engineering self-assessment and a follow up self-assessment in 1996 in order to evaluate the effectiveness of the engineering organization, staff, and processes. In 1998 the licensee implemented a strategy for achieving engineering excellence. Included in this strategy was an action to perform a self-assessment in 1999, again reviewing the effectiveness of the engineering organization, staff, and processes, and measuring progress made. In letters from the licensee to the NRC dated October 7, 1998, and May 19, 1999, the licensee outlined commitments in order to improve engineering performance and documented the licensee’s understanding of the NRC’s plans for monitoring licensee engineering performance. One of the NRC’s plans was to evaluate the licensee’s 1999 assessment. That evaluation is contained in this report. This inspection was performed using portions of Inspection Procedures 95001 and 71152. During this inspection, the inspectors determined that licensee engineering management generally understood the causes of poor engineering performance. The 1999 licensee self-assessment failed, however, to emphasize the effects of the engineering backlog and failed to emphasize design issues associated with the 250/125 volt dc system. The inspectors also determined that the causes of poor engineering performance were not fully corrected; however, planned corrective actions were reasonable and improvements had been made (Section 02.03).

- Green. Licensee engineers failed to incorporate the correct design of a valve actuator into a calculation and failed to evaluate the scope of the issue.

  During review for replacement of a high pressure core injection steam isolation valve completed on December 23, 1997, licensee engineers found that an inaccurate, nonconservative valve actuator weight had been used in the existing pipe stress calculation. This was a violation of 10 CFR Part 50, Appendix B, Criterion XVI, for failing to implement measures to promptly identify and correct other possible examples of this error. We are treating this violation as noncited in accordance with the NRC Enforcement Policy. This violation was included in the licensee’s corrective action program as Problem Identification Report 4-08665. This issue was determined to be of low safety significance by the Safety Determination Process because allowable pipe stresses were not exceeded and the pipe remained fully operable. Specifically, licensee engineers failed to scope this issue to determine if this nonconservative weight had been used in other pipe stress calculations for other actuators of the same type (Section 02.02.b).

- Green. Engineers failed to incorporate the correct design of service water piping into calculations for evaluating pipe stresses.

  During calculations to evaluate the effects of wall thinning on service water piping, the engineers failed to include seismic considerations as required by design requirements. This was a violation of 10 CFR Part 50, Appendix B, Criterion III. We are treating this as noncited in accordance with the NRC Enforcement Policy. The inspectors noted that the probability of a seismic event was very low. As a result, the lack of evaluating seismic stresses imposed very low risk significance. The licensee replaced the affected piping during the refueling outage (Section 02.02c).

Report Details

01 Inspection Scope

This focused inspection was performed by the NRC to assess a licensee-performed self-assessment of engineering. The licensee had committed to perform the self-assessment as a result of poor performance in engineering. Engineering performance is related to the initiating events, mitigating systems, and barrier integrity cornerstones in the reactor safety strategic performance area.

02 Evaluation of Inspection Requirements

02.01 Was the Self-Assessment Performed by Appropriately Trained Personnel (95001, 71152)

The licensee used 22 evaluators and a team leader to perform the 1999 self-assessment. The 22 evaluators consisted of 15 in-house engineers, one operations liaison, one maintenance liaison, one quality assurance liaison, and four contractors. Fourteen of the evaluators had some root cause or assessment training. The quality assurance organization had no direct involvement in the assessment.

The inspectors determined that the self-assessment was performed by appropriately trained personnel because: (1) the majority of the personnel had some training in root cause or assessment, and (2) all of the personnel were technically competent in the areas they assessed. The inspectors did note, however, that in letters from the licensee to the NRC dated May 19, 1999, and October 7, 1998, the licensee stated that it was their intent that the assessment be performed by a majority of offsite experts. The four contractors did not represent a majority of the 22 evaluators. The licensee generated Problem Identification Report 4-08667 to document this discrepancy and to initiate training for licensing engineers in order to ensure sensitivity to written statements made to the NRC. The majority of the assessment team were not offsite experts because the licensee's internal commitment tracking system did not reflect this intention and because the number of evaluators was increased after the offsite experts were contracted for. The inspectors noted that the assessment was self-critical, but were unable to determine if the assessment would have been different if the assessment team had been composed of a majority of offsite experts.

02.02 Evaluate the Thoroughness of Licensee Self-Assessment Efforts (95001, 71152)

.a Licensee Self-Assessment Scope

The objectives of the 1999 assessment were to determine if engineering strategy performance objectives were being met, to validate whether engineering strategy performance issues still persisted, to compare assessment results to a 1996 assessment, and to verify that the design basis of the 125 volt and 250 volt dc systems had been correctly implemented. The assessment included an evaluation of engineering support of operations and of maintenance and an evaluation of engineering programs, system engineering, and design changes. The assessment also included a review of the 125 volt and 250 volt dc systems.

The inspectors determined that the overall scope of the 1999 assessment was sufficient to evaluate the engineering organization, staff, and processes. However, the assessment did not explicitly consider items assigned to engineering that were not completed and resulted in a backlog. Based on interviews and a review of data, the inspectors found that this backlog, both in items not overdue and overdue for completion, did have an effect on the individual engineering personnel as well as on engineering programs. The backlog included approximately 650 items to be dispositioned as a result of assignments from the corrective action program, approximately 1500 maintenance work requests to be reviewed for unauthorized modifications, and approximately 230 assignments from design changes requiring disposition by engineering. Separate from the 1999 self-assessment, the senior engineering manager showed the inspectors external engineering assessments that did comment on the backlog as having a high number of items. In addition, the licensee trended the backlog as an engineering performance indicator. The inspectors also reviewed the documents reviewed by the licensee evaluators in the area of engineering support of operations. Three of these documents were not relevant to this area. Unreviewed Safety Question Evaluation 99-0069 was an evaluation of an installation of an external bearing on Valve CW-MOV-103MV, a nonsafety-related main condenser water box inlet valve. Unreviewed Safety Question Evaluation 99-0021 was an evaluation of an organization change to the nuclear power group within engineering. Plant Temporary Modification 95-14 was an evaluation of moving the Westinghouse office to the south end of the turbine building. While these documents may have had some effect on operations, the inspectors found that these documents were not directly relevant to assessing engineering support of operations, because in each case the change had a minimal impact on operations. Since these were only three of the 18 documents that the licensee evaluators reviewed, the inspectors concluded that the evaluators reviewed a sufficient number of relevant documents.

.b Assessment of Data Used by Licensee

The licensee evaluators reviewed documents, conducted interviews, and observed engineering work in progress in order to assess engineering support of operations. The evaluators concluded that communications were ineffective, that engineering was more reactive than proactive, and that engineering and operations assigned activities different priorities or did not effectively communicate the priorities. The evaluators noted some improvement since the 1996 licensee audit. The inspectors found that the data used for the assessment supported the licensee’s conclusions.

Although it did not affect the inspectors’ conclusions as stated above, the inspectors found that engineering personnel did not properly identify the extent of an error found during a design change. The failure to properly evaluate the scope of the error was not identified by the licensee evaluators. Modification Package 95-103 was used to replace Valve HPCI-MOV-M014, the high pressure core injection pump turbine steam isolation valve, because the valve was leaking steam past its seat. During engineering review, the engineers noted that Pipe Stress Calculation 89-149, Revision 6, used an incorrect weight for the Limitorque SB-1 actuator for this motor-operated valve. The new valve to be installed was to use the actuator from the old valve. The weight used in the calculation was 550 pounds, while the actual actuator weight was 630 pounds. For this specific modification, licensee engineers used engineering judgment and determined that, since there was margin from the calculated pipe stress to the allowable pipe stress, the additional 80 pounds would not cause this particular pipe to exceed allowable stress. The inspectors agreed with this engineering judgment because margin was approximately 50 percent of the allowable stress. However, since these actuators were used in other locations throughout the plant, the inspectors questioned whether this incorrect weight had been used in other pipe stress calculations. In response, the licensee generated Problem Identification Report 4-08665 to further evaluate the extent of this error. Because pipe stress calculations generally contained a margin between stresses caused by internal pressure, deadweight, a safe shutdown earthquake, a design basis earthquake, and allowable pipe stresses, the inspectors concluded that the operability of the high pressure core spray pump was not impacted. As a result, the use of incorrect valve weight had very low risk significance.

10 CFR Part 50, Appendix B, Criterion XVI, requires measures to promptly identify and correct conditions adverse to quality. Failure to determine whether incorrect actuator weight was incorrectly used in other pipe stress calculations is a violation. This violation is being treated as a noncited violation consistent with the Enforcement Policy. This violation is in the licensee's corrective action program as Problem Identification Report 4-08665 (NCV 298/200005-01).

.c Inspectors' Independent Review of Various Engineering Data

To further evaluate the thoroughness of the licensee self-assessment, the inspectors reviewed various engineering products and documents.

The inspectors reviewed Problem Identification Report 4-06248 and the associated operability determination. During January 2000, the licensee had identified wall thinning of an approximately 6-inch long pipe designated SW-Z8-2852-8. This pipe is a capped horizontal branch connection from the service water Heat Exchanger A discharge pipe. The thinning was below the erosion/corrosion program minimum wall thickness of 0.192 inches. Wall thinning down to 0.041 inches was measured near the welded connection to the service water pipe. A licensee system engineer performed an operability determination by calculating the minimum wall thickness required to sustain internal pipe system pressure, which was 0.017 inches. The system engineer consequently determined that the pipe was operable. The nominal pipe wall thickness was 0.28 inches. The inspectors noted that stresses due to dead weight, the design basis earthquake, and the safe shutdown earthquake were not considered in the operability determination; only "hoop stress," stress due to internal pressure, was considered. Since the service water system was seismically qualified, the inspectors questioned whether the pipe would remain intact, given the wall thinning, if a seismic event occurred. In response, the licensee generated Problem Identification Report 4-08689 to perform a seismic stress analysis of the pipe. The pipe was replaced during Refueling Outage 19 before the end of this inspection period; consequently, no operability concern existed at the end of this inspection period.

10 CFR Part 50, Appendix B, Criterion III, states, in part, that design changes shall be subject to design control measures commensurate with those applied to the original

design and be approved by the organization that performed the original design unless the applicant designates another responsible organization. The Cooper Nuclear Station Updated Safety Analysis Report, Section 8.1.5, states that the service water system other than that portion supplying the turbine generator building and other nonessential areas is designed in accordance with Class I seismic criteria. Contrary to this requirement, on January 9, 2000, the licensee identified wall thinning on Pipe SW-Z8-2852-8, a branched line off of the Division I service water piping, but failed to apply design control measures commensurate with those applied in the original design, by failing to analyze this thinned-wall configuration to ensure that the design was in accordance with Class I seismic criteria. Since the probability of a seismic event is small, the contribution of a pipe failure due to a seismic event contributes very little impact to risk of core damage. As a result, inspectors conclude this issue had very low risk significance. We are treating this violation of 10 CFR Part 50, Appendix B, Criterion III, as noncited in accordance with the NRC Enforcement Policy (NCV 298/200005-02).

Overall, the inspectors found that issues identified during the independent data review conducted by the inspectors were of a type found by the licensee during the 1999 self-assessment. However, the licensee concluded that, for the vertical review of 250 volt and 125 volt DC systems, no significant issues were identified. The inspectors noted, however, that, for this system, there were relatively significant and unresolved issues previously identified by members of the engineering organization, including the validity of calculations for battery load profile. These issues were noted on the data section of the 1999 self-assessment, but were not emphasized in the body of the assessment.

02.03 Consider the Adequacy of the Licensee’s Corrective Actions (95001, 71152)

The corrective actions for the 1999 self-assessment were documented in Problem Identification Report 4-06088. A formal root cause determination was not performed; however, an apparent root cause determination, which was less rigorous and relied more on the evaluators’ judgement than a formal root cause determination, was used to develop the corrective actions. The evaluators determined that five underlying causes that were present during a 1996 self-assessment were still present. These were that: (1) engineering management of open items was inadequate, (2) processes were cumbersome, (3) responsibilities of engineers were not clearly defined, (4) the staff was not being held accountable, and (5) work was not effectively prioritized. Corrective actions were to revise a conduct of engineering procedure, to revise an engineering strategy for excellence plan, to establish interface agreements with other organizations on site (i.e., operations and maintenance), and to establish effective engineering work management. These corrective actions were not completed as of the end of this inspection period. Separate from the corrective actions documented in Problem Identification Report 4-06088, during 1998 and 1999 the licensee had increased engineering staffing levels, certified almost 75 percent of system engineers in plant technology, decreased the number of systems that system engineers were responsible for, and was updating the Updated Safety Analysis Report.

The inspectors determined that, overall, licensee engineering management understood the causes of poor engineering performance. In terms of the 1999 self-assessment, omitting an explicit evaluation of backlogged engineering items and not emphasizing 250 volt and 125 volt dc system design issues were exceptions to this determination.

The inspectors also determined that the engineering department's planned corrective actions were reasonable. These corrective actions, as outlined in Problem Identification Report 4-06088, and other actions outlined in an engineering strategy for success, had not fully corrected the causes of poor engineering performance as of the end of this inspection period. This was despite many of these causes having been noted in a licensee self-assessment performed in 1996. The licensee evaluators did observe, during the 1999 self-assessment, that there had been improvement in many engineering areas. Based on interviews and corrective actions that had been completed as outlined above, the inspectors agreed with this conclusion that improvements had been made.

03 Exit Meeting Summary

The inspectors presented the inspection results to Mr. B. Rash, Senior Engineering Manager, and other members of licensee management at the conclusion of the inspection on April 20, 2000. The inspectors also conducted a telephonic exit meeting with Ms. S. Mahler, Assistant Nuclear Licensing and Safety Manager, on May 9, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

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T. Ackerman, Erosion/Corrosion Program Supervisor
P. Adelung, Senior Civil Structural Engineer, Design Engineering Department
C. Blair, Licensing Engineer
S. Domikaitis, Engineering Program Supervisor
P. Donahue, Plant Engineering Design Manager
B. Houston, Quality Assurance Operations Manager
D. Madsen, Licensing Engineer
S. Mahler, Assistant Nuclear Licensing and Safety Manager
J. Peters, Licensing Engineer
M. Pinto, Assistant to Senior Manager of Engineering
B. Rash, Senior Engineering Manager
D. Vorpaul, Service Water System Engineer

NRC

J. Clark, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

298/200005-01	NCV	inadequate corrective action for a design calculation error (Section 2.02.a)
298/200005-02	NCV	failure to consider seismic effects on service water pipe operability (Section 2.02.c)

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
NCV	noncited violation
NRC	Nuclear Regulatory Commission

PARTIAL LIST OF DOCUMENTS REVIEWED

Maintenance Work Requests:

00-0158, “Replacement of the B 125 VDC Battery”
00-0278, “Replacement of the B 250 VDC Battery”

99-3302, “Replacement of the A 125 VDC Battery”
00-0277, “Replacement of the A 250 VDC Battery”

Change Engineering Documents:

1999-0122, “250 VDC Battery Cell Replacement,” July 7, 1999
1999-0055, “250 VDC Battery Cell Replacement,” July 20, 1999
1999-0055, Revision 1, “250 VDC Battery Cell Replacement,” August 6, 1999

Surveillance Procedures:

6.EE.605, “250V Battery Service Test,” Revision 7C2
6.EE.609, “125/250 Station Battery Intercell Connection Testing,” Revision 7
6.HV.103, “Control Room Emergency Fan Filter Train Differential Pressure Test,” Revision 3
6.SW.102, “Service Water System Post-LOCA Flow Verification,” Revision 8
6.2SW.101, “Service Water Surveillance Operation (Division 2),” Revision 8

Other Documents:

Design Calculation 87-131B, Rev. 7, “250 VDC Division II Load and Voltage Study”

Condition Report 97-1425, Battery Degradation Factor

Battery Load Test Report BCT-2000

“Strategy for Achieving Engineering Excellence,” Revisions 0, 1, 2, and 3

“Cooper Nuclear Station Engineering Self Assessment Follow-Up,” October, 1996

“1996 Engineering Self Assessment,” February, 1996

“1999 Engineering Self Assessment,” Revision 0

“Plans for Monitoring Progress of the CNS Strategy for Achieving Engineering Excellence,” NLS980165

Significant Condition Report 98-0712, “Incorrect Bussman Fuse Interrupt Rating,” Revision D

Procedure Change Request 4-04460

Resolve Condition Report 99-0306, “Discharge Testing of Replacement Cells,”

Engineering Backlog Performance Indicator Information, March 2000

“Student Qualification Histories” for 1999 Self-Assessment Team dated April 10, 2000

Plant Temporary Modification 96-33, “Disable Diesel Fire Pump (FP-P-D) Remote Control Room Stopping Capability”

Vendor Bulletin PT21 Rosemount 0911, “Notification Under 10 CFR Part 21 for Rosemount Model 1153B Alphaline Nuclear Pressure Transmitter,” dated January 11, 1999

Significant Condition Report 98-0167, “Missing Supports in the Off-Gas Fan and Filter Building”

Significant Condition Report 97-0742, “RHR-MX-B RE17 As-Found Sitting Condition”

Event Report OI PS 6649, “Large Bore Primary Containment Vent and Purge Valve Operation,” dated February 21, 1997

NRC Information Notice 88-76, “Recent Discovery of a Phenomenon Not Previously Considered in the Design of Secondary Containment Pressure Control”

Operator Work-Around 98-21, “Control Room Should Have Indication of Fuel Pool Cooling Skimmer Level”

Nebraska Public Power District “Project Plan for Unauthorized Modifications Follow-Up Project,” Revision O

Project Plan, “Significant Conditions Adverse to Quality Corrective Action Effectiveness Review,” Revision 1

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (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.