UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-8064

May 3, 2001

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

SUBJECT: COOPER NUCLEAR STATION - NRC INSPECTION REPORT 50-298/01-03

Dear Mr. Swailes:

On April 6, 2001, the NRC completed a 1-week onsite team triennial fire protection baseline inspection of your Cooper Nuclear Station. Additional in-office inspection was performed during the week of April 9-13, 2001. The enclosed report presents the results of this inspection. We discussed the results of the onsite inspection with you (via telecom), and other members of your staff on April 6, 2001. On April 26, 2001, we conducted a telephonic exit meeting with you, and members of your staff to inform you of the results of the additional onsite inspection.

The inspection involved an examination of the effectiveness of activities conducted under your license as they related to the implementation of your NRC-approved Fire Protection Program and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified issues that were evaluated under the risk significance determination process as having very low safety significance (green). The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations, consistent with Section VI.A of the Enforcement Policy. These noncited violations are described in the subject inspection report. If you deny these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper Nuclear Station facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be 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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/**RA**/

Jeffrey L. Shackelford, Chief
Engineering and Maintenance Branch
Division of Reactor Safety

Docket: 50-298
License: DPR-46

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NRC Inspection Report
50-298/01-03

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-298
License:	DPR 46
Report No.:	50-298/01-03
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	April 6, 2001
Inspectors:	C. E. Johnson, Senior Reactor Inspector Engineering and Maintenance Branch C. A. Clark, Reactor Inspector Engineering and Maintenance Branch R. P. Mullikin, Senior Reactor Inspector Engineering and Maintenance Branch W. M. McNeill, Reactor Inspector Engineering and Maintenance Branch R. Nease, Senior Reactor Inspector Engineering and Maintenance Branch
Accompanying Personnel:	F. J. Wyant, Contractor Sandia National Laboratories
Approved By:	Jeffrey L. Shackelford, Chief Engineering and Maintenance Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000298/01-03; on 04/02-06/2001, onsite and 04/09-13/01, in-office; Cooper Nuclear Station; Triennial Fire Protection Inspection

This report covers a 1-week onsite inspection by a team of five Regional inspectors and one contractor from Sandia National Laboratories during April 2-6, 2001. Additional in-office inspection was performed by the team members during the week of April 9-13, 2001. The inspectors used NRC Inspection Procedure 71111.05 to evaluate the licensee's implementation of their NRC-approved fire protection program. However, certain associated circuit issues, which are the subject of an ongoing, voluntary industry initiative, were not reviewed in this inspection. This portion of the inspection procedure was not performed in order to permit the industry to develop an approach and methodology to resolving the associated circuits issues that the NRC can endorse and to provide for licensees to implement the resolution methodology once approved.

Four issues identified during the inspection are discussed in the report. The significance of the issues is indicated by their color (green, white, yellow, red) and was determined through the use of the Significance Determination Process as described in NRC Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the Significance Determination Process does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Cornerstone: Mitigating Systems

- Green. The team identified a noncited violation in Fire Zone 20A (service water pump room) in which equipment required for safe shutdown of the plant following a fire was not separated by 20 feet horizontal distance, and there were intervening combustibles (Rubatex insulation) that were not part of an exemption, nor included in the licensee’s engineering evaluation. This was not in accordance with Section III.G.2(b) of Appendix R. This violation was entered into the licensee’s corrective action program as Notification 10075408 and 10076323. (Section 1R05.2b)

  This finding was of very low safety significance because the area-wide fire suppression and detection systems were not degraded, and the increase in combustible loading of the Rubatex insulation did not substantially increase the severity of a postulated fire in the fire area.

- Green. The team identified a noncited violation in Fire Zone 7A (control room basement) in that emergency lighting was not aligned properly to adequately perform safe shutdown operator actions in accordance with Section III.J of Appendix R to 10 CFR Part 50. This violation was entered into the licensee’s corrective action program as Notification 10076810. (Section 1R05.7b).

This finding was of very low safety significance because the operators would have available dedicated hand held lights that would assist them in performing required actions.

- Green. The team identified a noncited violation in three areas (control room, diesel generator room, and the 1001-foot elevation of the reactor building) in which the licensee failed to install detectors as documented in the safety evaluation report which was not in accordance with 10 CFR 50.48(b). This violation was entered into the licensee's corrective action program as Notification 10078580, 10078607, and 10078606. (Section 4A05.2.1)

  This finding was determined to be of very low safety significance due to the number of mitigating systems remaining.

- Green. The team identified that on October 20, 1985, the licensee implemented modification design change MDC 85-48 in which they replaced 1/2-inch diameter sprinkler heads with 1/4-inch diameter sprinkler heads in the reactor recirculation pump motor generator set lube oil pump area (958-foot elevation of the reactor building) and in the reactor recirculation pump motor generator lube oil pump area (976 foot elevation of the reactor building). The licensee failed to perform calculations to ensure that the reduction in the diameter of the sprinkler heads did not adversely affect the suppression requirements in these fire areas, as required by the National Fire Protection Association Code 13. This was not in accordance with 10 CFR 50.48(b). This violation was entered into the licensee's corrective action program as Notification 10073757. (Section 4A05.3.2).

  This finding was determined to be of very low safety significance, because there were no safe shutdown systems in the areas that could be affected by a postulated fire.

Report Details

1. REACTOR SAFETY

1R05 Fire Protection

The purpose of this inspection was to review the Cooper Nuclear Station fire protection program for selected risk significant fire areas with emphasis on verification that the post-fire safe shutdown capability and the fire protection features provided for ensuring that at least one post-fire safe shutdown success path is maintained free of fire damage. The inspection was performed in accordance with the new NRC regulatory oversight process using a risk-informed approach for selecting the fire areas and attributes to be inspected. The team as a group used the Cooper Nuclear Station individual plant examination of external events to choose several risk-significant areas for detailed inspection and review. The fire areas chosen for review during this inspection were:

- Fire Zone/Area 9A/VII (cable spread room)
- Fire Zone/Area 9B/VII (cable expansion room)
- Fire Zone/Area 10B/VII (control room)
- Fire Zone/Area 20A/XI (service water pump room)

For each of these fire areas, the team focused their inspection on the fire protection features and on the systems and equipment necessary for the licensee to achieve and maintain safe shutdown conditions.

.1 Systems Required to Achieve and Maintain Post-Fire Safe Shutdown

a. Inspection Scope

The team reviewed the licensee's piping and instrumentation diagrams and the list of safe shutdown equipment documented in the Appendix R, Post Fire Shutdown Topical Design Criteria Document, to verify whether the licensee's shutdown methodology had properly identified the components and systems necessary to achieve and maintain safe shutdown conditions for equipment in the fire areas selected for review. The team focused on the following functions that must be ensured to achieve and maintain post-fire safe shutdown conditions: (1) reactor coolant makeup capable of maintaining the reactor coolant level inventory, (2) reactor heat removal capable of achieving and maintaining decay heat removal, and (3) supporting system capable of providing all other services necessary to permit extended operation of equipment necessary to achieving and maintaining hot shutdown conditions.

A review was also conducted to ensure that all required electrical components in the selected systems were included in the licensee's safe shutdown analysis. The team identified the systems required for each of the primary safety functions necessary to shut down the reactor. These systems were then evaluated to identify the systems that interfaced with the fire areas inspected and were the most risk significant for reaching both hot and cold shutdown. The following systems were selected for review:

- Reactor core isolation cooling system
- High pressure coolant injection system
- Automatic depressurization system
- Core spray system
- Residual heat removal system (low pressure coolant injection mode)

b. Findings

No findings of significance were identified.

.2 Fire Protection of Safe Shutdown Capability; Fire Protection Systems, Features, and Equipment

a. Inspection Scope

For the selected fire areas, the team evaluated the adequacy of fire suppression and detection systems, fire area barriers, penetration seals, and fire doors to ensure that at least one train of safe shutdown equipment was free of fire damage. To do this, the team observed the material condition and configuration of the installed fire detection and suppression systems, fire barriers, and construction details and supporting fire tests for the installed fire barriers. In addition, the team reviewed the license documentation, such as exemptions and National Fire Protection Association code deviations to verify that the fire barrier installations met license commitments.

b. Findings

The findings are discussed below by fire area.

Fire Area XI, Fire Zone 20A, Service Water Intake Structure, Elevation + 903' 6"

Fire Zone 20A contains all four service water pumps. The zone is protected by an automatic Halon 1301 extinguishing system and flame, smoke, and heat detectors. The service water system provides a heat sink for the reactor equipment cooling, residual heat removal, and diesel generator cooling systems under transient and accident conditions.

In a letter dated June 28, 1982, the licensee requested an exemption for the service water intake structure from the provisions of 10 CFR Part 50, Appendix R, Section III.G.2. Specifically, the exemption request stated that 20 feet of horizontal separation free of intervening combustibles, along with automatic detection and suppression, did not exist. However, automatic suppression and detection did exist within the fire zone, but not between the redundant service water pumps. The licensee proposed to extend the automatic suppression and detection to all areas separating the pumps if the NRC granted the exemption from the 20-feet separation requirement of Appendix R. The NRC, in a letter dated September 21, 1983, granted the exemption for Fire Zone 20A (described as Fire Zone A in the exemption request) based on the following:

- The diesel driven fire pump would be removed and all cables are in conduits,

- The only significant in-situ combustible is the lubricating oil from the service water pump motors, and

- The probability of ignition of the lubricating oil is low because the oil has a high flashpoint (approximately 450°F) and that sufficiently hot surfaces do not exist to cause the ignition of the lubricating oil.

The licensee performed Engineering Evaluation 87-1, dated May 11, 1987, in order to justify discrepancies found between the original exemption requests, as approved by the NRC in September 1983, and Revision 1 to the Fire Hazards Analysis. One of the discrepancies noted was that the fire severity in Fire Zone 20A increased from 1.1 to 3 minutes in duration. The licensee considered that this increase could still be considered negligible given the large floor area (1270 square feet) and the fact that automatic suppression was provided. This was based on an estimated total in-situ and transient combustible loading of 1,568,000 Btu used in the 1983 exemption, and a revised total combustible loading of 5,694,000 Btu used in Engineering Evaluation 87-1. The licensee concluded that the deviation regarding combustible loading did not adversely affect the basis of the 1983 exemption for Fire Zone 20A.

During the inspection, the team observed large amounts of Rubatex insulation that had been installed on the service water piping since 1993. The licensee was questioned as to whether this was a combustible material since it was not listed on the Fire Hazards Analysis. The licensee subsequently determined that the Rubatex insulation material is a combustible material and that the licensee no longer conformed with 10 CFR Part 50, Appendix R, exemption for Fire Zone 20A. The licensee performed Calculation NEDC 01-011, "SW Pump Room Combustible Loading Due To Rubatex Insulation," and determined that the Rubatex insulation increased the combustible loading in Fire Zone 20A by 4,430,800 Btu and the fire severity by 2.62 minutes.
10 CFR 50.48(b) states, that "Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of Appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate before January 1, 1979."
Section III.G.2 of Appendix R to 10 CFR Part 50 describes three methods acceptable for ensuring that at least one train of redundant safe shutdown equipment is free of fire damage: (a) redundant trains be located in different fire areas separated by 3-hour rated fire barriers; (b) redundant trains in the same fire area be separated by 20 feet of horizontal distance with no intervening combustible or fire hazards, and the fire area be equipped with area-wide detection and suppression; or (c) one redundant train be separated from the other redundant trains by enclosing it in a 1-hour fire-rated barrier, and the fire area be equipped with area-wide detection and suppression.
Section III.G.2.(b) is the method chosen by the licensee to ensure that one train of service water pumps is free of fire damage. The team determined that both trains of service water pumps could be damaged by a fire in Fire Zone 20A since redundant service water pumps are separated by less than 20 feet (16.5 feet) and there are intervening combustibles that were not part of the exemption, nor included in the licensee's engineering evaluation. This violation is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy (50-298/0103-01). In

response to this finding, the licensee initiated Notifications 10075408 and 10076323, and initiated a continuous fire watch in Fire Zone 20A. The significance of the finding is evaluated below.

<u>Risk Significance of the Lack of 20-feet Horizontal Separation in Fire Zone 20A</u>: The team reviewed Appendix B to NRC Manual Chapter 0610*, "Power Reactor Inspection Reports." The team determined that the finding met the Group 1, Question (1), in that, the loss of all service water pumps would have a credible impact on safety. The team also determined that the Group 2 fire protection question was satisfied in that the finding involved the degradation of a fire protection feature, which was the 20-feet separation between redundant trains of safe shutdown equipment. Thus, since the finding satisfied both Group 1 and 2 questions, the team evaluated the finding using the significance determination process as described in Appendix F to Manual Chapter 0609.

The following was considered in evaluating the risk for Fire Zone 20A:

- A fire ignition frequency (IF) of 6.55 x 10^{-3} per year was determined from the licensee's individual plant examination of external events document (log IF=-2.18).

- Degradation of the 20-feet separation was determined to be high (FB = 0).

- Although a fire brigade drill was not witnessed by the team, one adverse observation was noted by the NRC within the last 2 years. However, the licensee had taken appropriate corrective action; therefore, manual suppression (MS) was considered to be in its normal operating state (MS = -1.0).

- Automatic suppression was determined to be within its normal operating state (AS = -1.25).

- A common cause term (CC) was not used since the automatic suppression system utilized Halon and the manual suppression extinguishing agent was water (CC = 0).

- A fire mitigation frequency (FMF) was calculated to be -4.45 using the formula, FMF = log IF + FB + AS + MS + CC.

- Based on the length of time the condition existed (greater than 30 days), the likelihood for the initiating event occurrence during the degraded period was rated E (Table 5.5 of Appendix F to Manual Chapter 0609).

- Entering the transient worksheet from the risk-informed inspection notebook for Cooper Nuclear Station, resulted in a remaining mitigating capacity of -2. The team input the remaining mitigating capacity of -2 into Table 5.6 (Appendix F to Manual Chapter 0609), and concluded that this finding was of very low safety significance (Green).

In addition, the team provided the technical details associated with the Rubatex insulation to the fire protection branch in the Office of Nuclear Reactor Regulation. An evaluation was performed, which determined that the additional combustible loading would not have significantly increased the size or magnitude of a postulated fire such that an appreciable increase in risk would have resulted.

The team concluded that the finding for Fire Zone 20A was determined to be within the licensee response band (Green).

.3 Post-fire Safe Shutdown Circuit Analysis

a. Inspection Scope

The team, on a sample basis, verified that safety-related and nonsafety-related cables for equipment in the four selected fire areas had been analyzed to show that they would not prevent safe shutdown because of hot shorts, open circuits, or shorts to ground. Additionally, the team verified, on a sample basis, that circuit breaker coordination and fuse protection were acceptable as a means of protecting the power sources of the designated alternate safe shutdown equipment.

b. Findings

No findings of significance were identified.

.4 Alternative Safe Shutdown Capability

a. Inspection Scope

The team performed a review to determine if the licensee had appropriate procedures in place and had identified the plant components and systems required to achieve and maintain safe shutdown conditions. The team reviewed the capability of the identified systems and components and the adequacy of the procedures that were identified as required to achieve alternative safe shutdown. The team then reviewed procedures and system operating capabilities to verify they were adequate to perform plant cooldown to hot and cold shutdown conditions from outside of the control room. The team's methodology was to focus on the overall adequacy of the identified systems, components, and use of procedures to perform actions necessary to increase core shutdown margin, control reactor pressure, provide reactor coolant makeup, and remove core decay heat. The team also reviewed the adequacy of process monitoring and needed support system functions.

The team reviewed, on a sample basis, the transfer of control from the control room to the alternative location to determine if it could be affected by fire-induced circuit faults (e.g., by the provision of separate fuses and power supplies for alternative shutdown control circuits).

b. Findings

No findings of significance were identified.

.5 Operational Implementation of Alternative Shutdown Capability

a. Inspection Scope

The team performed a walkdown of the actions defined in Operating Procedure 5.4.3.2, "Post-Fire Shutdown to Mode 4 Outside Control Room," Revision 24. This procedure documented the method for performing an alternative shutdown of the plant from the remote shutdown panel and by manipulating certain equipment locally in the plant. The team reviewed the ability of operators to perform the procedural actions within applicable plant shutdown time requirements and that equipment labeling was consistent with the procedure. The team reviewed the training program for licensed and nonlicensed personnel to verify it included training on the alternative safe shutdown capability.

b. Findings

No findings of significance were identified.

.6 Communications

a. Inspection Scope

The team reviewed the adequacy of the communication system to support plant personnel in the performance of alternative safe shutdown functions and fire department duties. The licensee credited the plant radio system for post-fire safe shutdown actions that require prompt control room operator response.

The team observed the radios in the alternate shutdown cabinet and reviewed records to assure that the radios were being maintained in an operable condition. The team reviewed a sample of preventative maintenance activities to verify radios were available and operational for emergency use by operators and fire department members.

b. Findings

No findings of significance were identified.

.7 Emergency Lighting

a. Inspection Scope

The team reviewed the emergency lighting system required for safe shutdown activities in the selected fire areas to verify it would provide for adequate access to perform manual actions required to achieve and maintain hot shutdown conditions. The team also reviewed the adequacy of emergency lighting for performing actions required in Procedure 5.4.3.2, "Post-Fire Shutdown to Mode 4 Outside Control Room," Revision 24, which included access and egress routes. The team reviewed test procedures and test data to verify that the individual battery operated units were able to supply light for the required 8-hour period. The team reviewed vendor and licensee data, which determined the maximum temperatures at which the battery-powered lighting units would operate for 8 hours, in order to verify operability under maximum ambient temperatures. The team reviewed vendor documentation to verify that the battery power supplies were rated with at least an 8-hour capacity. The team also verified whether routine preventive maintenance was being performed to assure that the 8-hour battery powered lights were being maintained in an operable manner.

b. Findings

The team performed a walkdown of Emergency Procedure 5.4.3.2, "Post-Fire Shutdown to Mode 4 Outside Control Room," Revision 24, to determine whether the installed emergency lighting was adequate to perform the required functions, assuming a loss of normal lighting. Attachment 3 to Procedure 5.4.3.2, "Control Building Actions," requires that an operator perform the following actions in the control building basement at Elevation 882'-6":

- Step 1.3.9 Remove pipe plug from end of IA-26, Reliable Air Header Drain Valve (C-882-SE corner of room).
- Step 1.3.10 Open IA-26.

During the walkdown, the team observed that in order to perform these two steps, an operator would have to climb on piping, which is approximately 5 feet off the floor and reach above his/her head. Emergency Lighting Unit EE-LTG-C30 has two lamps, which illuminate the access and egress route, and illuminates the work area to perform these actions. The team observed that the lamp, which is required to illuminate the work area, had been positioned so that illumination was directed away from the work area. The team determined that the emergency lighting was inadequate to perform Steps 1.3.9 and 1.3.10. 10 CFR 50.48(b) states, "Appendix R to this part establishes fire protection features required to satisfy Criterion 3 of Appendix A to this part with respect to certain generic issues for nuclear power plants licensed to operate before January 1, 1979." Section III.J of Appendix R to 10 CFR Part 50 states that emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto. The licensee informed the team that the reason that Steps 1.3.9 and 1.3.10 are performed is to mitigate the potential consequences of a hot short circuit caused by a fire in either the control room, cable spreading room, or the cable expansion room. A hot short circuit could result in the closure of air-operated Valve SW-TCV-451B, which is the service water discharge valve to Reactor Equipment Cooling Heat Exchanger 1B.

This event would eliminate cooling to the high pressure coolant injection (HPCI) pump room, and result in the subsequent loss of HPCI, since HPCI is a single train system. In a transient that would require alternative shutdown, HPCI would provide core cooling during a plant shutdown assuming a loss of offsite power. However, a loss-of-cooling accident would not be postulated. This violation is being treated as a noncited Volation in accordance with Section VI.A of the NRC Enforcement Policy (50-298/0103-02). In response to this finding, the licensee initiated Notification 10076810 and realigned the lamp. Additionally, the team noted that operators would carry handheld lights that are dedicated for use during alternate shutdown, and these lights are under a periodic maintenance activity. The significance of the finding is evaluated below.

Risk Significance of the Inadequate Emergency Lighting: The team reviewed Appendix B to NRC Manual Chapter 0610*, "Power Reactor Inspection Reports." The team determined that the finding met the Group 1 Question (1) in that the loss of HPCI would have a credible impact on safety. The team also determined that the Group 2 fire protection question was satisfied in that the finding involved the degradation of a fire protection feature, which was Appendix R emergency lighting. The team then evaluated the risk of this finding in accordance with the significance determination process described in Appendix F to Manual Chapter 0609. In evaluating the risk, the team chose to evaluate the control room (Fire Zone 10 B), which was determined to have the worst-case ignition frequency of the 3 areas (9A, 9B, or 10B) involved. The following was considered in evaluating the risk of this finding in the control room:

- A fire ignition frequency (IF) of 1.09×10^{-2} per year was determined from the licensee's individual plant examination of external events document. This was the ignition frequency for Fire Zone 10B, which was the most conservative (Log IF=-1.96).

- The control room does not have fire barriers or separation between redundant trains of safe shutdown equipment; therefore, no credit was given for barriers. (FB = 0).

- Although a fire brigade drill was not witnessed by the team, one adverse observation was noted by the NRC within the last 2 years. However, the licensee had taken appropriate corrective action; therefore, manual suppression (MS) was considered to be in its normal operating state (MS = -1.0).

- There is no automatic suppression in the control room (AS = 0).

- A common cause term (CC) was not used since there is no automatic suppression (CC = 0).

- A fire mitigation frequency (FMF) was calculated to be $10^{-2.96}$ using the formula, FMF = log IF + FB + AS + MS + CC.

- Based on the length of time the condition existed (greater than 30 days), the likelihood for the initiating event occurrence during the degraded period was rated C (Table 5.5 of Appendix F of Manual Chapter 0609).

- Entering the transient worksheet from the risk-informed inspection notebook for Cooper Nuclear Station, resulted in a remaining mitigating capacity of -7. The team input the remaining mitigating capacity of -7 into Table 5.6 (Appendix F to Manual Chapter 0609), and concluded that this finding was of very low safety significance (Green).

The team concluded that the finding for a lack of adequate emergency lighting was determined to be within the licensee response band (Green).

.8 Cold Shutdown Repairs

a. Inspection Scope

The team reviewed licensee procedures to determine which repairs were required to achieve cold shutdown and whether repair material was available onsite.

b. Findings

No findings of significance were identified.

.9 Compensatory Measures

a. Inspection Scope

The team verified that adequate compensatory measures were in place by the licensee for out-of-service, degraded or inoperable fire protection and post-fire safe shutdown equipment, systems or features (e.g., detection and suppression systems, or passive fire barrier features).

b. Findings

No findings of significance were identified.

4. **OTHER ACTIVITIES (OA)**

4OA2 Identification and Resolution of Problems

a. Inspection Scope

The team reviewed a sample of condition reports to verify that the licensee was identifying fire protection-related issues at an appropriate threshold and entering those issues into the corrective action program.

b. Findings

No findings of significance were identified.

40A5 Other

.1 (Closed) Unresolved Item 50-298/9625-04, Use of Automatic Depressurization and Core Spray Systems for Alternate Safe Shutdown

NRC Inspection Report 50-298/96025, dated March 4, 1997, noted that although the re-analysis of their alternate safe shutdown methodology using automatic depressurization plus core spray systems was considered acceptable as stated in NRC Inspection Report 50-298/87-04, the licensee should have requested an exemption from 10 CFR Part 50, Appendix R.

The team reviewed the licensee's methodology for performing alternate shutdown in accordance with 10 CFR Part 50 Appendix R, Section III.L.2 and found that the licensee does not use automatic depressurization and core spray systems for achieving and maintaining hot shutdown under alternate shutdown conditions. The licensee does use, however, this method as redundant means for achieving and maintaining safe shutdown under 10 CFR Part 50, Appendix R, Section III.G. In a letter dated December 12, 2000, to James M. Kenny, Chairman of the BWR Owners Group, the NRC concluded that the use of automatic depressurization plus low pressure systems meets the requirements of a redundant means of post-fire safe shutdown under Section III.G.2 of 10 CFR Part 50, Appendix R. This unresolved item is closed.

.2 (Closed) Unresolved Item 50-298/9625-05: Fire Detection System Capability

In NRC Inspection Report 50-298/96025, dated March 4, 1997, the NRC identified that fire detection did not appear to be installed in accordance with industry standards, manufacturers' recommendations, or NFPA codes. In addition, the inspection report noted that the licensee had not performed a design/code evaluation of the fire detection system to demonstrate the system could rapidly detect an incipient fire.

.2.1 Compliance with National Fire Protection Association (NFPA) Codes 72D and E:

Subsequent to the NRC's inspection conducted in November and December of 1996, the licensee performed a NFPA code conformance review of the fire suppression and detection systems at Cooper Nuclear Station. The results of this evaluation are documented in Engineering Evaluation EE01-006, dated March 19, 2001. In this report, the licensee identified numerous plant areas where NFPA codes were not met. Each of these deviations was evaluated for its effect on the ability of the fire protection system to perform its original function. In addition, the licensee either provided justification for accepting the deviation or provided recommendations made for bringing the system into an acceptable condition. The licensee initiated Problem Identification Report (PIR) 4-03052 to document and track resolution of these deviations from NFPA codes. This PIR is still open.

In January of 2001, the licensee performed a fire protection self-assessment in which they identified that fire detector location and spacing did not appear to conform to the requirements of NFPA 72E, "Automatic Fire Detectors 1974." These non-conformances were evaluated in Engineering Evaluation EE01-007, dated March 30, 2001. The licensee initiated PIR 4-14696 on March 9, 2001, PIRs 4-14697 and 4-14698 on March 8, 2001, and Notification 10075862 on March 30, 2001, to document and track resolution of these non-conformances. At the time of this inspection these PIRs were still open.

In Engineering Evaluation EE01-007, the licensee stated that the plant alarm system was evaluated for conformance to NFPA 72D, "Proprietary Protective Signaling Systems 1975"; however, no specific commitment had been made to meet the detector location and spacing requirements of NFPA 72E. Section 3330 of NFPA 72D-1975 states that fire detecting equipment shall be installed in accordance with NFPA 72E, "Automatic Fire Detectors." NFPA 72D-1975 does not provide guidance concerning fire detector location and spacing, merely invokes the requirements of NFPA 72E. Further review by the team revealed that the licensee included detailed drawings which showed fire detector locations in their December 17, 1976, Appendix A to Branch Technical Position, BTP 9.5-1 submittal for NRC review. The resulting NRC "Fire Protection Safety Evaluation Report by the Office of Nuclear Reactor Regulation in the Matter of Nebraska Public Power District, Cooper Nuclear Station - Unit 1, Docket No. 50-298" (SER), dated May 23, 1979, stated, "The system meets all the major provisions for NFPA 72D-1975 except for supervision of the trip circuit for the carbon dioxide system in the diesel generator room." It is, therefore, the team's position that for those areas for which the licensee did not submit drawings showing fire detector locations in their December 17, 1976, submittal, the licensee is committed to meeting the requirements of NFPA 72E for installation of fire detectors. Furthermore, for those areas for which the licensee submitted drawings showing detector locations, any changes in the location of detectors or in the configuration of the areas that would affect fire detection, would also require the licensee to meet NFPA 72E. The team discussed the NRC's position in detail with the licensee. The licensee acknowledged the teams discussion.

.2.2 Ventilation Air Flow Analysis

The team reviewed the fire protection licensing basis for Cooper Nuclear Station, and found that Section 4.2 of the NRC's SER of May 23, 1979, stated that the licensee would conduct a ventilation air flow analysis and install additional detectors to correct any deficiencies identified. When questioned by the team, the licensee stated that not all recommendations in the ventilation air flow analysis had been implemented. In particular, the licensee did not install detectors in the control room, diesel generator rooms, and on the 1001-foot elevation of the reactor building, as discussed below.

1. Control Room (Fire Zone 10B): The ventilation air flow analysis recommended three additional detectors be installed in the back corridor of the control room, behind the back panels. As of this inspection, these detectors had not been installed. On April 12, 2001, the licensee entered this issue into their corrective action program as Notification 10078580. The licensee did not post compensatory measures, because the control room is continuously occupied.

2. Diesel Generator Rooms 1A (Fire Zone 14A) and 1B (Fire Zone 14B): The ventilation air flow analysis recommended that 8 additional detectors be installed in diesel generator rooms 1A and 1B. As of this inspection, these detectors had not been installed. On April 12, 2001, the licensee entered this issue into their corrective action program as Notification 10078607, and posted compensatory measures.

3. 1001' Elevation of the Reactor Building (Fire Zone 6): The ventilation air flow analysis recommended that 2 additional heat detectors be installed. As of this inspection, these detectors had not been installed. On April 12, 2001, the licensee entered this issue into their corrective action program as Notification 10078606, and posted compensatory measures.

Title 10 of the Code of Federal Regulations, Part 50.48(b) states that, except for Sections III.G, III.J, and III.O, the provisions of Appendix R to this part shall not be applicable to plants licensed to operate prior to January 1, 1979, to the extent that fire protection features proposed or implemented have been accepted by the NRC staff as satisfying the provisions of Appendix A to Branch Technical Position BTP APCSB 9.5-1 as reflected in NRC fire protection safety evaluation reports issued prior to the effective date of this rule (November 19, 1980). Section 4.1 of the NRC's SER of May 23, 1979, states, "The licensee will conduct a detailed ventilation air flow analysis to determine the effectiveness of the existing detector locations. Additional detectors will be provided to correct any deficiencies noted by the study. We find that, subject to implementation of the above described modifications, the fire detection system satisfies the objectives identified in Section 2.2. of this report and is, therefore, acceptable." The team found that the failure to install detectors in the control room, diesel generator rooms 1A and 1B, and on the 1001-foot elevation of the reactor building as recommended by the ventilation air flow analysis was a violation of 10 CFR 50.48(b) with three examples. This violation is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy (50-298/0103-03).

<u>Risk Significance of the Failure to Install Detectors Recommended in the Ventilation Air Flow Analysis</u>:

This violation was evaluated in accordance with Appendix B to NRC Manual Chapter 0610*, "Power Reactor Inspection Report." The team determined that it met Group 1 Question (1), in that the failure to install fire detectors where necessary would have a credible impact on safety. The Group 2 fire protection question was satisfied, because the finding involved a degradation of fire detection, a fire protection feature. The team then evaluated the risk of this finding in accordance with the significance determination process described in Appendix F to Manual Chapter 0609. In evaluating the risk, the team chose to evaluate the control room (Fire Zone 10 B), which was determined to have the worst-case ignition frequency of the 3 areas involved. The following was considered in evaluating the risk of this finding in the control room:

- A fire ignition frequency (IF) of 1.09×10^{-2} per year was determined from the licensee's individual plant examination of external events document. (Log IF = -1.96)

- The control room does not have fire barriers or separation between redundant trains of safe shutdown equipment; therefore, no credit was given for barriers. (FB = 0)

- Detectors recommended by the ventilation air flow analysis were not installed, therefore, manual suppression was considered to be moderately degraded. (MS = -1.0)

- The control room does not have automatic suppression; therefore, no credit was given for automatic suppression. (AS = 0).

- There were no dependencies between the defense-in-depth elements, and no common cause contributions identified. (CC=0)

- A fire mitigation frequency (FMF) was calculated to be $10^{-2.96}$ per year using the formula, FMF = log IF + FB + AS + MS + CC.

- Based on the length of time the condition existed (greater than 30 days), the likelihood for the initiating event occurrence during the degraded period was rated C (Table 5.5 of Appendix F to Manual Chapter 0609).

- The team determined that a worst-case fire would occur in Control Panel 9-3 and would cause a reactor trip due to inadvertent closure of main steam isolation valves or inadvertent opening of automatic depressurization valves. Entering the transient worksheet from the risk-informed inspection notebook for Cooper Nuclear Station, resulted in a remaining mitigating capacity of -5. The team input the remaining mitigating capacity of -5 into Table 5.6 (Appendix F to Manual Chapter 0609), and concluded that this finding was of very low safety significance (Green).

The team concluded that the finding was determined to be within the licensee response band (Green).

This unresolved item is closed, however, a violation was identified.

.3 (Closed) Unresolved Item 50-298/9625-06: Fire Suppression System Capability

In NRC inspection report 50-298/96025, dated March 4, 1997, the NRC identified that the fire protection sprinkler system in the reactor recirculation pump motor generator (RRMG) set area on the 976-foot elevation of the reactor building did not appear to be installed in accordance with NFPA codes, and the curbed area around the RRMG sets did not appear to be adequate to contain spilled lube oil plus sprinkler discharge. In addition, the NRC questioned the licensee's smoke removal capability for a fire in the RRMG set area.

.3.1 NFPA Code Non-conformances

In response to the observation documented in NRC Inspection Report 50-298/96025, the licensee performed a NFPA code conformance review of the fire suppression and detection systems at Cooper Nuclear Station. The results of this evaluation were documented in engineering evaluation EE01-006, dated March 19, 2001. In this report, the licensee identified numerous plant areas where NFPA codes were not met. Each of these deviations was evaluated for its effect on the ability of the fire protection system to perform its original function. In addition, the licensee either provided justification for accepting the deviation or provided recommendations made for bringing the system into an acceptable condition. On July 17, 1999, the licensee initiated PIR 4-03052 and on April 23, 2001 initiated Notification 10073746 to document and track resolution of these deviations from NFPA codes. Both PIR 4-03052 and Notification 10073746 were still open at the time of this inspection.

On March 10, 1999, a licensee contractor completed a hydraulic analysis of sprinkler systems in the plant that identified numerous fire systems where the hydraulic demand may not be met by the plant's water supply. The systems in question were:

- System 4: Reactor Building, 958-foot elevation, sprinklers for RRMG set lube oil pumps;

- System 5: Control Building, 918-foot elevation, cable spreading room pre-action sprinklers;

- System 6: Reactor Building, 976-foot elevation, RRMG set pre-action sprinklers;

- System 7: Turbine Building, 882-foot elevation, lube oil storage tank deluge system;

- System 9: Turbine Building, 882-foot elevation, reactor feed pump 1A deluge system;

- System 13: Turbine Building, 903 and 932-foot elevation, main turbine generator oil piping area and bearings deluge system;

- System 27: Reactor Building, 931-foot elevation, sprinklers for RRMG oil coolers; and

- System 38: Multi-purpose Building, 903-foot elevation, decontamination room pre-action sprinklers.

On March 27, 2001, the licensee completed Engineering Evaluation EE01-008 to address the potential deficiencies in the hydraulic demand and supply for the above-listed systems. In EE01-008, the licensee stated that they could not verify the inputs the contractor used in their hydraulic analysis. This issue was included in PIR 4-03052, dated July 15, 1999, which was still open at the time of this inspection. In EE01-008, the licensee concluded that all the sprinkler systems were adequate. However, the licensee noted that in 1985, they had replaced 1/2" diameter sprinkler heads with 1/4" diameter sprinkler heads in sprinkler Systems 4, 6, and 27 (listed above). In reviewing Calculation NEDC 85-053, performed in support of this modification, the licensee found errors, and on March 12, 2001, initiated PIR 4-14779 to address these errors. This portion of the unresolved item is closed.

.3.2 <u>Under-designed Curbing</u>

In NRC Inspection Report 50-298/96025, dated March 4, 1997, the NRC observed that the curbed area surrounding the RRMG sets did not appear to be able to contain discharge from the automatic sprinkler system plus spilled RRMG set lube oil. On October 20, 1985, the licensee completed modification design change MDC 85-48, in which sprinklers with 1/2-inch orifices were replaced with sprinklers with 1/4" diameter orifices in the RRMG set area on the 976-foot elevation and the RRMG lube oil pump area on the 958-foot elevation of the reactor building. This modification was performed due to a concern that the curbed areas surrounding these areas were thought to be undersized. In their December 17, 1976, response to Appendix A to Branch Technical Position APCSB 9.5-1,” Guidelines for Fire Protection for Nuclear Power Plants,” the licensee stated that the automatic sprinkler systems were installed in compliance with NFPA 13, “Installation of Sprinkler Systems 1975.” The team found that although NFPA

13-1975 permits smaller orifices for “locations and conditions not requiring as much water as is discharged by nominal 1/2-inch orifice sprinkler,” the licensee failed to perform calculations to ensure that this reduction in the diameter of the sprinkler heads did not adversely affect the suppression requirements in these areas. Therefore, if a fire in this area is not suppressed sufficiently, there would be the potential for damage to the 125 V DC Starter Rack which is equipment required for safe shutdown. Also a fire in this area damages the RRMG sets which controls power to the reactor recirculation pumps. Thus, such a fire could increase the likelihood of a reactor transient.

Title 10 of the Code of Federal Regulations, Part 50.48 states, “Except for the requirements of [S]ections III.G, III.J, and III.O, the provisions of [A]ppendix R to this part shall not be applicable to nuclear power plants licensed to operate prior to January 1, 1979, to the extent that fire protection features proposed or implemented by the licensee have been accepted by the NRC staff as satisfying [A]ppendix A of Branch Technical Position BTP APCSB 9.5-1 reflected in staff fire protection safety evaluation reports issued prior to the effective date of this rule...” Appendix A to BTP APCSB 9.5-1, “Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1 1976,” states, “Automatic sprinkler systems as a minimum should conform to the requirements of appropriate standards, such as NFPA 13, Standard for the Installation of Sprinkler Systems.” Section 3-15.2.5 of NFPA 13-1975 permits smaller orifices for “locations and conditions not requiring as much water as is discharged by nominal 1/2-inch orifice sprinkler.” In their December 17, 1976, response to Appendix A to Branch Technical Position APCSB 9.5-1,” Guidelines for Fire Protection for Nuclear Power Plants,” the licensee stated that the automatic sprinkler systems were installed in compliance with NFPA 13, which included the Reactor Building 958 and 976-foot elevations as listed in Section .3.1 above. Section 4.3.1.5 of the NRC’s “Safety Evaluation Report by the Office of Nuclear Reactor Regulation in the Matter of Nebraska Public Power District Cooper Nuclear Station - Unit 1 Docket No. 50-298,” (SER) dated May 23, 1979, states, “We find that, subject to implementation of the above described modifications, the water suppression systems satisf[y] the objectives identified in Section 2.2 of the report and are, therefore, acceptable.” Section 2.2 of the SER of May 23, 1979, states, “Guidance on the implementation of GDC-3 for existing nuclear power plants has been provided by the NRC staff in “Appendix A” of Branch Technical Position 9.5-1, “Guidelines for Fire Protection for Nuclear Power Plants.” On October 20, 1985, the licensee completed modification design change MDC 85-48, in which sprinklers with 1/2-inch orifices were replaced with sprinklers with 1/4" diameter orifices in the RRMG lube oil pump area on the 958-foot elevation of the reactor building and in the RRMG set area on the 976-foot elevation of the reactor building. However, the licensee failed to perform calculations to ensure that the locations and conditions of this fire area did not require as much water as is discharged by nominal 1/2-inch orifice sprinkler, as required by NFPA 13. This is a violation of 10 CFR 50.48, with two examples. This violation is being treated as a noncited violation in accordance with Section VI.A of the NRC Enforcement Policy (50-298/0103-04). On March 23, 2001, the licensee entered this issue into their corrective action program as Notification 10073757 and posted compensatory measures for these areas.

<u>Risk Significance of the Failure to meet NFPA 13</u>: The violation, described above, was evaluated in accordance with Appendix B to NRC Manual Chapter 0610*, “Power Reactor Inspection Report.” the team determined that it met Group 1 Question (1), in

that the failure to ensure that the conditions of this fire area did not require as much water as is discharged by nominal 1/2-inch orifice sprinkler would have a credible impact on safety. Specifically, a fire in this area without adequate automatic suppression capability, could result in failure of the reactor recirculation pumps. The Group 2 fire protection question was satisfied, because the finding involved a degradation of fire suppression, a fire protection feature. The team then evaluated the risk of this finding in accordance with the significance determination process described in Appendix F to Manual Chapter 0609. In evaluating the risk, the team chose to use the ignition frequency of the RRMG lube oil pump area on the 958-foot elevation of the reactor building (Fire Zone 4D), which was determined to have the worst-case ignition frequency of the two areas in which this non-conformance was identified.

- A fire ignition frequency (IF) of 1.22×10^{-2} per year was determined from the licensee's individual plant examination of external events document. (Log IF = -1.91)

- The area does not contain redundant safe shutdown systems; therefore barriers were considered to be in a normal operating state. (FB = -2)

- Although a fire brigade drill was not witnessed by the team, the NRC resident inspector noted a weakness when observing a fire brigade drill performed within the last 2 years. However, the licensee took appropriate corrective action; therefore, manual suppression was considered to be in a normal operating state. (MS = -1.0)

- The licensee identified potential deficiencies in the hydraulic demand and supply of certain sprinkler systems in the plant, among them are the areas which are the subject of this finding. In addition, the licensee identified errors in the hydraulic analysis performed as a result of changing out 1/2" sprinkler heads for 1/4" in the areas which are the subject of this finding. The team identified that the licensee failed to perform calculations to verify that the discharge from 1/4" sprinkler heads can meet the fire demand in these areas which are the subject of this finding. Therefore, the automatic suppression was considered to be moderately degraded. (AS = -0.75)

- There were no dependencies between the defense-in-depth elements, and no common cause contributions identified. (CC= +0.25)

- A fire mitigation frequency (FMF) was calculated to be $10^{-5.41}$ per year using the formula, FMF = log IF + FB + AS + MS + CC.

- Based on the length of time the condition existed (greater than 30 days), the likelihood for the initiating event occurrence during the degraded period was rated F in Table 5.5, Appendix F to Manual Chapter 0609.

- The team determined that a fire in the RRMG lube oil pump area of the 958-foot elevation of the reactor building would cause a loss of recirculation pumps resulting in a reactor trip. Entering the transient worksheet from the Risk-informed Inspection Notebook for Cooper Nuclear Station, resulted in a remaining mitigating capacity of -8. The team input the remaining mitigating capacity of -8 into Table 5.6 (Appendix F to Manual Chapter 0609), and concluded that this finding was of very low safety significance. (Green)

The team concluded that the finding was determined to be within the licensee response band (Green).

.3.3 Inadequate Smoke Removal

In NRC Inspection Report 50-298/96025, dated March 4, 1997, the NRC observed that the fire protection design did not appear to consider smoke propagation in the event of a fire in the RRMG set area on the 976-foot elevation of the reactor building. In particular, the NRC was concerned about smoke migrating into the refueling floor area.

The team reviewed the licensee's pre-fire plan, Procedure CNS-FP-221, Revision N02, and found that for a fire in the RRMG set area, smoke would vent through the ceiling hatch to the refueling floor. The licensee stated that the smoke would not cause actuation of sprinklers on the refueling floor, and the fire brigade would not need to enter the refueling floor to respond to a fire in the RRMG set area (976-foot elevation of the reactor building). In addition, the licensee has smoke removal equipment available to the fire brigade for use in smoke removal. This part of the unresolved item is closed.

.4OA6 Meetings, including Exit

On April 6, 2001, at the conclusion of the team's onsite inspection, the team debriefed Mr. J. Swailes (via telecom), and other licensee staff members on the preliminary inspection results.

On April 26, 2001, a tele-conference exit meeting was held with Mr. J. Swailes, Vice President, Nuclear, and other licensee staff members, during which the team leader characterized the results of the inspection. The licensee's management acknowledged the findings presented.

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

ATTACHMENT

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Boyce, Manager, Regulator Affairs
R. Dyer, Fire Protection Program Engineer
F. Diya, Manager, Plant Engineering
A. Dostal, Manager, Performance and Strategy (via telecom)
D. Ferraro, Counsel, Watson and Strawm
C. Fidler, Manager, Maintenance
R. Gardner, Senior Manager, Quality Assurance
J. Gillan, Manager, Work Control
B. Hannaford, Mechanical Engineer, Engineering Support Department
K. Jones, Manager, Design Engineering
K. Kirkland, Manager, Nuclear Information Systems
B. Leech, Vice President, Mid American Energy (via telecom)
D. Madsen, Licensing Engineer, Regulatory Affairs
W. Macecevic, Manager, Operations
C. Markert, Manager, Engineering Support Department
J. McDonald, Plant Manager, (via telcom)
J. Shrader, Nuclear Instructor, Training
J. Swailes, Vice President, Nuclear (via telcom)
J. Salisbury Assistant Manager, Engineering Support
K. Sutton, Senior Staff Risk Management Engineer
L. Schilling, Manager, Administrative Services Department
R. Stoddard, Consultant Representative, Lincoln Electrical System (via telecom)
N. Wetherell, Assistant Senior Engineering Manager
J. Ranalli, Senior Manager, Engineering
K. Newcomb, Fire Marshall, Engineering Support Department

ITEMS OPENED AND CLOSED

Opened and Closed

50-298/0103-01	NCV	Failure to provide 20 feet seperation between redundant service water equipment.
50-298/0103-02	NCV	Failure to provide adequate emergency lighting to perform operator actions for safe shutdown.
50-298/0103-03	NCV	Failure to install detectors as documented in the safety evaluation report which was not in accordance with 10 CFR 50.48(b).

50-298/0103-04	NCV	Failure to to perform calculations to ensure that the locations and conditions of fire areas did not require as much water as is discharged by nominal 1/2-inch orifice sprinkler, as required by NFPA 13 Code requirements.

Closed

50-298/9625-04	URI	Adequacy Of Alternate Safe Shutdown Method For Fire
50-298/9625-05	URI	Adequacy Of Fire Detection
50-298/9625-06	URI	Adequacy Of Fire Suppression System

LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of this inspection:

DRAWINGS

Drawing Number	Title	Revision
2006, Sh 1	Circulating, Screen Wash & Service Water Systems	N44
2006, Sh 3	Circulating, Screen Wash & Service Water Systems	N41
2006, Sh 4	Control Building Service Water System	N36
2028	Flow Diagram Reactor Building & Drywell Equipment Drain System	N42
2031, Sh 1	Reactor Building Closed Cooling Water System	N20
2031, Sh 2	Reactor Building Closed Cooling Water System	N56
2031, Sh 3	Reactor Building Closed Cooling Water System	N23
2036, Sh 1	Reactor Building Service Water System	N70
2040, Sh 1	Residual Heat Removal System	N72
2040, Sh 2	Residual Heat Removal System Loop "B"	N09
2041	Flow Diagram Reactor Building Main Steam System Cooper Nuclear Station	N69
2043	Reactor Core Isolation Coolant and Reactor Feed Systems	N45
2044	High Pressure Coolant Injection and Reactor Feed Systems	N65
2045, Sh 1	Core Spray System	N54

Drawing Number	Title	Revision
2082	General Arrangement Turbine Building Operating Floor Plan	N15
3002, Sh 1	Cooper Nuclear Station Auxiliary One Line Diagram MCC Z, SWGR Bus 1A, 1B, 1E, & Critical SWGR Bus 1F 1G	N32
3004, Sh 3	Cooper Nuclear Station Auxiliary One Line Diagram MCC C, D, H, J, DG1 & DG2	N20
3005, Sh 4	Cooper Nuclear Station Auxiliary One Line Diagram Motor Control Centers M, N, P, U, V, & W	N39
3006, Sh 5	Cooper Nuclear Station Auxiliary One Line Diagram Starter Racks LZ and TZ MCC's K, L, LX, RA, RX, S, T, TX, X	N66
3007, Sh 6	Cooper Nuclear Station Auxiliary One Line Diagram Motor Control Centers E, Q, R, RB, & Y	N75
3022, Sh 6	Cooper Nuclear Station 4160V Switchgear Elementary Diagrams	N23
3023, Sh 7	Cooper Nuclear Station 4160V Switchgear Elementary Diagrams	N15
3037, Sh 6	Control Elementary Diagrams	N26
3037, Sh 6A	Control Elementary Diagrams Sheet	N12
3040, Sh 9	Cooper Nuclear Station Control Elementary Diagrams	N23
3045, Sh 14	Control Elementary Diagrams	N38
3047, Sh 16	Control Elementary Diagrams	N11
3051	Main One Line Diagram	10
3058	Cooper Nuclear Station DC One Line Diagram	N41
3059, Sh 1	D.C. Panel Schedules	N31
3071	Control Elementary Diagram	N21
3076, Sh 19	Control Elementary Diagram	N31
3214, Sh 1	Cooper Nuclear Station Control Building Cable Tray Loading Schedule	N25
3255, Sh 4	Connection Wiring Diagram Control Room - Control Panels	N38
3256, Sh 3	Connection Wiring Diagram Relay Panels	N15
3256, Sh 6	Connection Wiring Diagram Relay Panels	N16
3256, Sh 7	Connection Wiring Diagram Relay Panels	N15

Drawing Number	Title	Revision
3256, Sh 8	Cooper Nuclear Station Connection Wiring Diagram Relay Panels	N23
3257, Sh 67	Cooper Nuclear Station Alternate Shutdown HPCI Panel External Connections	N16
3257, Sh 73	Cooper Nuclear Station HPCI Panel Layout Alternate Shutdown	N05
3257, Sh 74	Alternate Shutdown RHR Panel Layout	N03
3257, Sh 75	Alternate Shutdown ADS Panel Layout	N01
3401	Cooper Nuclear Station Auxiliary One Line Diagram MCC CA, CB, MR, OG1 & OG2	N27
791E253, Sh 2	Elem. Diag. Automatic Blowdown Sys.	N19
791E253, Sh 3	Cooper Nuclear Station Elementary Diagram Automatic Blowdown System	N09
791E261, Sh 1	Elem. Diag. Residual Heat Removal Sys.	N14
791E261, Sh 15	Elem. Diag. Residual Heat Removal Sys.	N10
791E261, Sh 16	Elem. Diag. Residual Heat Removal Sys.	N07
791E261, Sh 17	Elem. Diag. Residual Heat Removal Sys.	N13
791E261, Sh 18	Elem. Diag. Residual Heat Removal Sys.	N10
791E261, Sh 19	Residual Heat Removal System	N21
791E261, Sh 20	Elem. Diag. Residual Heat Removal Sys.	N12
791E261, Sh 21	Elem. Diag. Residual Heat Removal Sys.	N09
791E261, Sh 22	Elem. Diag. Residual Heat Removal Sys.	N10
791E261, Sh 23	Elem. Diag. Residual Heat Removal Sys.	N06
791E261, Sh 24	Cooper Nuclear Station Residual Heat Removal System Elementary Diagrams	N01
791E264, Sh 5	Elementary Diagram RCIC Sys	N15
791E264, Sh 6	Elementary Diagram RCIC Sys	N10
791E264, Sh 7	Elementary Diagram RCIC Sys	N13
791E265, Sh 2	Elem. Diag. Core Spray System	N19
791E265, Sh 3	Elem. Diag. Core Spray System	N19

Drawing Number	Title	Revision
791E265, Sh 4	Elem. Diag. Core Spray System	N12
791E266, Sh 12	Cooper Nuclear Station Elementary Diagram Primary Containment Isolation System (16-23)	N12
791E271, Sh 1	Elem. Diag. HPCI System	N39
791E271, Sh 3	Elem. Diag. HPCI System	N17
791E271, Sh 4	Elem. Diag. HPCI System	N21
791E271, Sh 4A	Elementary Diagram HPCI System	N02
791E271, Sh 5	Elem. Diag. HPCI System	N19
791E271, Sh 6	Elem. Diag. HPCI System	N15
791E271, Sh 6A	Cooper Nuclear Station Elementary Diagram HPCI System	N03
791E271, Sh 7	Elem. Diag. HPCI System	N17
791E271, Sh 8	Cooper Nuclear Station HPCI System	N18
944E689, Sh 1	Elem. Diag. Low-Low Set	N10
E501, Sh 33	Motor Operated Valves Connection Diagrams	N07
E501, Sh 44	Cooper Nuclear Station Motor Operated Valves Connection Diagram	N02
E507, Sh 18A	Cooper Nuclear Station Connection Wiring Diagram for Terminal Boxes 554, 556, 560 & 562	N05
E507, Sh 19	Cooper Nuclear Station Connection Diagram for Wireway (Terminal Box 372) at 125 V. D.C. HPCI Starter Rack	N12
E507, Sh 42A	Cooper Nuclear Station Connection Diagram for EE-STR-125 HPCI (MO17) HP-558MV	N08
EE-181	Safe Shut Down Component Locations & Emergency Route Lighting 932'-6" Critical Switchgear Rooms	4
EE-184	Safe Shut Down Component Locations & Emergency Route Lighting 932'-6" Turbine Generator Building	4
EE-189	Safe Shut Down Component Locations & Emergency Route Lighting 877'-6" & 903'-6" Control Building	7
FP-2	Cable Expansion Room Fire Protection Wet Sprinkler System #29	3

ENGINEERING DOCUMENTS

Number	Description	Revision
Calc. No. 86-105B	Critical AC Bus Coordination Study	8
Calc. No. 86-105D	CNS Critical DC Bus Coordination Study	7
Calc. No. NEDC 01-011	SW Pump Room Combustible Loading Due to Rubatex Insulation	1
Change Evaluation Document 1998-0134	Installation of Emergency Lighting BBESI-EBL Battery Assembly	0
Change Evaluation Document 1998-0191	Installation of Emergency Lighting BBESI-EBL Battery Assembly	0
Change Package 96-02	Safe Shutdown Component Block Diagrams and Safe Shutdown Cable Data Sheets	0
Change Package 96-03	Safe Shutdown Component Block Diagrams and Safe Shutdown Cable Data Sheets	0
Change Package 96-04	Safe Shutdown Component Block Diagrams and Safe Shutdown Cable Data Sheets	0
Change Package 96-06	Safe Shutdown Component Block Diagrams and Safe Shutdown Cable Data Sheets	0
Change Package 98-01	Safe Shutdown Component Block Diagrams and Safe Shutdown Cable Data Sheets	0
Change Package 98-07	Safe Shutdown Component Block Diagrams and Safe Shutdown Cable Data Sheets	0
Change Package 00-02	Safe Shutdown Component Block Diagrams and Safe Shutdown Cable Data Sheets	0
Design Control Document 02	High Pressure Coolant Injection (HPCI) System	5
Design Control Document 03	Service Water (SW) and Residual Heat Removal Service Water (RHRSW) Booster System	2
Design Control Document 12	Core Spray (CS) System	2

Number	Description	Revision
Design Control Document 13	Residual Heat Removal (RHR) System	2
Design Control Document 18	Reactor Core Isolation Cooling (RCIC)	1
Minor Design Change 84-7	Appendix R - Fire Protection for the Cable Expansion Room	5/11/84
	Cooper Nuclear Station Inservice Testing Program Basis Document	3.2
	Appendix R Post Fire Shutdown Topical Design Criteria Document	March 1998
	CNS Fire Hazard Analysis Report	06/99

ENGINEERING EVALUATIONS

NUMBER	TITLE	DATE
EE01-006	Disposition of NFPA Code Compliance Deviations (Systems in Fire Zones Requiring an IPEEE Phase 2 Screening)	March 19, 2001
EE01-007	Fire Detector Location and Spacing	March 30, 2001
EE01-008	Assessment of RJA Hydraulic Calculation for CNS Sprinkler Systems	March 27, 2001

NOTIFICATIONS

10073746	10075862	10076509	10077199	10078512	10078606
10073757	10076422	10076810	10078497	10078580	10078607
10074895	10076488	10076904	10078511		

PREVENTATIVE MAINTENANCE ACTIVITIES

Preventative Maintenance	Procedure	Completion Date
4730	5.4.3.2	3/21/01
4730	5.4.3.2	2/01/01
4730	5.4.3.2	1/10/01

Preventative Maintenance	Procedure	Completion Date
4730	5.4.3.2	12/02/00
4730	5.4.3.2	11/05/00
4730	5.4.3.2	10/06/00
4730	5.4.3.2	9/01/00
4730	5.4.3.2	8/05/00
4730	5.4.3.2	7/05/00
4730	5.4.3.2	6/02/00
4730	5.4.3.2	5/02/00
4730	5.4.3.2	4/04/00
4730	5.4.3.2	3/03/00
4730	5.4.3.2	2/05/00
4730	5.4.3.2	1/03/00

PROBLEMS IDENTIFICATION REPORTS

0-00317	4-03469	4-08923	4-09803	4-11451	4-13778
B-03936	4-06430	4-09152	4-09810	4-11651	4-14128
B-06471	4-06486	4-09161	4-09830	4-11774	4-14171
0-10251	4-06555	4-09162	4-09832	4-12189	4-14211
1-01249	4-06700	4-09163	4-10125	4-12190	4-14214
1-03416	4-08159	4-09164	4-10668	4-12241	4-14215
1-22441	4-08182	4-09475	4-10670	4-12694	4-14342
2-01390	4-08295	4-09532	4-10671	4-12695	4-14465
2-01703	4-08416	4-09566	4-10689	4-12700	4-14696
2-20354	4-08532	4-09708	4-10782	4-13148	4-14697
2-26441	4-08533	4-09719	4-11136	4-13174	4-14698
4-01395	4-08537	4-09741	4-11137	4-13643	4-14754
4-01663	4-08539	4-09801	4-11138	4-13657	4-14779
4-03052	4-08540	4-09802	4-11225	4-13678	4-15021

PROCEDURES

Number	Title	Revision
0.7.1	Control of Combustibles	11
0.23	Cooper Nuclear Station Fire Protection Plan	29

Number	Title	Revision
0.31	Equipment Status Control	9
0.7.1	Control of Combustibles	11
2.2.4	Communication Systems	28
3.19.1	Fuse Control	7
5.2.1	Shutdown From Outside the Control Room	26
5.4.3.1	Post-Fire Operational Information	18
5.4.3.2	Post-Fire Shutdown to Mode 4 Outside Control Room	23
5.7.22	Communications	19
15.EE.302	90 Second Emergency Lighting Functional Test	10C1
15.EE.303	1.5 Hour Emergency Lighting Functional Test	6
EDP-06	Design Inputs	5

SURVEILLANCE TESTS

Procedure	Revision	Test Date
15.EE.302	10C1	3/20/01
15.EE.302	10C1	2/13/01
15.EE.302	9	4/16/00
15.EE.303	5	4/07/00
STP 94-075	o	1/13/95

MISCELLANEOUS DOCUMENTS

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Facility Operating License, dated January 18, 1974

Fire Protection Safety Evaluation Report by the Office of Nuclear Reactor Regulation in the Matter of Nebraska Public Power District Cooper Nuclear Station - Unit 1, Docket No. 50-298, dated May 23, 1979

Supplement No. 1 to the Fire Protection Safety Evaluation Report by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission in the Matter of Nebraska Public Power District, Cooper Nuclear Station, Docket No. 50-298, dated November 21, 1980

Safety Evaluation by the Office of Nuclear Reactor Regulation Outstanding Fire Protection Modifications, Cooper Nuclear Station, Docket No. 50-298, dated August 21, 1985

Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 98 to Facility Operating License No. DPR-46, Nebraska Public Power District, Cooper Nuclear Station, Docket No. 50-298

Attachment A, "Halon Discharge Test Results, " to Design Change 85-01

NRC Memorandum to R. Mattson from L. Rubenstein, "Use of the Automatic Depressurization System (ADS) and Low Pressure Coolant Injection (LPCI) to meet Appendix R, Alternate Shutdown Goal," December 3, 1982

Vendor Manual No. VM-1730, Big Beam Emergency Systems Inc. Emergency Lighting, Revision 1

CNS Memorandum: R. Wachowaik to CNS Licensing, "Sensitivity Analysis for RAI Response to NLS2000078," PRA00007, September 20, 2000.

"CNS Plant Low Voltage Short Circuit Study: Rev 11," Jan 12, 1993.

Fax: D. Madsen, CNS, to USNRC Region-IV, "Training Information Regarding ADS Portion of Main Steam - 5 Pages," 3/28/01.

Fax: R. Dyer, CNS, to F. Wyant, SNL, "Tracking Item #115," 4/11/01.

Letter: J. Swailes, CNS, to USNRC, "Response to Supplemental Request for Additional Information - Individual Plant Examination for External Events (IPEEE)," September 22, 2000.

Notification 10076509, "Fire Protection App. A SER Air Flow Test," 04/03/2001.

Notification 10078497, "Verification of Power Cables in CSR," 04/11/2001.

"Safe Shutdown Cable(s)," 12/18/00.

"Tracking Item #115," Rev. 1, 4/12/01.

Letter to V. Stello, NRC from J. Pilant, Nebraska Public Power District, dated June 3, 1976

Letter to D. L. Ziemann, NRC from J. Pilant, Nebraska Public Power District, dated December 17 1976, transmitting the Response to Appendix A to Branch Technical Position APCSB 9.5-1

Letter to D.L. Ziemann, NRC from J. Pilant, Nebraska Public Power District, dated February 4, 1977

Letter to D.L. Ziemann, NRC from J. Pilant, Nebraska Public Power District, dated march 31, 1977 transmitting the Fire Hazards Analysis, Response to Appendix A to Branch Technical Position APCSB 9.5-1

Letter to D.L. Ziemann, NRC from J. Pilant, Nebraska Public Power District, dated April 6, 1977, transmitting Revisions and Additional Information, Fire Protection Review

Letter to D. K. Davis, NRC from J. Pilant, Nebraska Public Power District, dated July 20, 1977

Letter to D. K. Davis, NRC from J. Pilant, Nebraska Public Power District, dated December 19, 1977

Letter to G. E. Lear, NRC from J. Pilant, Nebraska Public Power District, dated May 11, 1978

Letter to V. Stello, NRC from J. Pilant, Nebraska Public Power District, dated June 21, 1978

Letter to T. Ippolito, NRC from J. Pilant, Nebraska Public Power District, dated August 16, 1978

Letter to T. Ippolito, NRC from J. Pilant, Nebraska Public Power District, dated December 11, 1978

Letter to T. Ippolito, NRC from J. Pilant, Nebraska Public Power District, dated April 12, 1979

Fire Protection Safety Evaluation Report by the Office of Nuclear Reactor Regulation in the matter of Nebraska Public Power District Cooper Nuclear Station - Unit 1, dated May 23, 1979

Safety Evaluation Report by the Office of Nuclear Reactor Regulation Outstanding Fire Protection Modifications Cooper Nuclear Station, dated August 21, 1985

Memorandum for R. N. Vollmer, Director, Division of Engineering from R. J. Mattson, Director, Division of Systems Integration, dated July 2, 1982

Letter to J. M. Kenny, Chairman BWR Owners Group from S. A. Richards, Director, Project Directorate IV and Decommissioning, NRC dated December 12, 2000

Letter to G. R. Horn, Senior Vice President, Nebraska Public Power District, from K. Brockman, Acting Director, Division of Reactor Safety, Region IV, NRC, dated March 4, 1997, transmitting NRC Inspection Report 50-298/96-25

Letter to G. R. Horn, Senior Vice President, Nebraska Public Power District, from J. N. Donohew, Senior Project Manager, NRC, dated July 31, 1998, transmitting Conversion to Improved Technical Specifications for the Cooper Nuclear Station - Amendment No. 178 to Facility Operating License No. DPR-46

Letter to D. N. Madsen and R. J. Dyer, Nebraska Public Power District from T. C. Poindexter, Winston & Strawn, dated April 12, 2001

NATIONAL FIRE PROTECTION ASSOCIATION CODES

NUMBER	TITLE	DATE
13	Installation of Sprinkler Systems	1975
13	Installation of Sprinkler Systems	1976
72D	Proprietary Protective Signaling Systems	1975
72E	Automatic Fire Detectors	1974
72E	Automatic Fire Detectors	1982