APPENDIX B

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-298/84-20 License: DPR-46

Docket: 50-298

Licensee: Nebraska Public Power District (NPPD)
P. O. Box 499
Columbus, Nebraska 68601

Facility Name: Cooper Nuclear Station (CNS)

Inspection At: Cooper Nuclear Station, Nemaha County, Nebraska

Inspection Conducted: October 1-November 30, 1984

Inspector: D L DuBois 12/20/84
D. L. DuBois, Senior Resident Inspector (SRI) Date

Approved: J P Jaudon 12/28/84
J. P. Jaudon, Chief, Project Section A, Date
Reactor Project Branch (RPB) 1

Inspection Summary

Inspection Conducted October 1-November 30, 1984 (Report 50-298/84-20)

Areas Inspected: Routine, unannounced inspection of operational safety verification, monthly surveillance and maintenance observations, licensee event followup, surveillance, and followup of IE bulletins. The inspection involved 147 inspector-hours onsite by one NRC inspector.

Results: Within the six areas inspected, one violation was identified (unreviewed safety question - standby gas treatment system, paragraph 6).

DETAILS

1. Persons Contacted

Principal Licensee Personnel

*P. Thomason, Division Manager of Nuclear Operations
*K. Wire, Operations Manager
*D. Whitman, Technical Staff Manager
*C. Goings, Regulatory Compliance Specialist

The NRC inspector also interviewed other licensee and contractor personnel.

NRC

*D. Garrison, Reactor Inspector, RIV

*Indicates presence at exit meeting.

2. Operational Safety Verification

The SRI observed control room operations, instrumentation, controls, reviewed applicable logs, and conducted discussions with control room operators. The SRI verified operability of:

- Number 2 Diesel Generator
- 125 VDC Distribution System
- Standby Liquid Control System

The SRI reviewed safety clearance records, including verification that affected components were removed from and returned to service in a correct and approved manner, that redundant equipment was verified operable, and that limiting conditions for operation were adequately identified and maintained. The SRI also verified that maintenance requests had been initiated for equipment discovered to require repair or routine preventive upkeep, appropriate priority was assigned, and maintenance commenced in a timely manner commensurate with assigned priorities.

Tours of accessible areas of the facility were conducted to verify that minimum shift crew requirements were met and to observe normal security practices, plant and equipment conditions including cleanliness, radiological controls, fire suppression systems, emergency equipment, potential fire hazards, fluid leaks, excessive vibration, and instrumentation adequacy.

The tours, reviews, and observations were conducted to verify that facility operations were performed in accordance with the requirements established in the CNS Operating License and Technical Specification.

No violations or deviations were identified in this area.

3. Monthly Surveillance Observations

The SRI observed Technical Specification required surveillance tests. These observations verified that:

- Test prerequisites were completed
- Testing was performed in accordance with approved procedures
- Test instrumentation was in calibration
- Limiting conditions for operation were met
- Return to service was accomplished
- Test results were reviewed
- Deficiencies were corrected in a timely manner

These reviews and observations were conducted to verify that facility surveillance operations were performed in accordance with the requirements established in the CNS Operating License and Technical Specification.

No violations or deviations were identified in this area.

4. Monthly Maintenance Observations

The SRI observed preventive and corrective maintenance activities during this inspection period. Observations included checks for the availability of redundant equipment and for adequate isolation and clearance. The SRI also found that the work was accomplished by qualified personnel in accordance with approved procedures and the Technical Specification requirements. Additionally, the performance of quality control checks and the adequacy of health physics coverage were verified, as were appropriate cleanliness controls. The SRI monitored postmaintenance surveillance testing which was performed to demonstrate operability of affected systems and components.

These reviews and observations were conducted to verify that facility maintenance operations were performed in accordance with the requirements established in the CNS Operating License and Technical Specifications.

No violations or deviations were identified in this area.

5. <u>Licensee Event Report Followup (LER)</u>

The following LER is closed on the basis of the SRI's inoffice review, review of licensee documentation, and discussions with licensee personnel:

- LER 84-011, HPCI Overspeed Trip Control Valve failure

6. Surveillance

The SRI completed a review of numerous CNS surveillance tests performed on the standby gas treatment (SGT) system. The following is a list of procedures that were reviewed including descriptions of identified inadequacies:

- 6.3.19.2, Revision 7, "SGT Filter Differential Pressure and Heater Output Test," performed on April 20, 1984.

  Procedure 6.3.19.2, is performed in order for the licensee to verify that the SGT system high efficiency and charcoal filters are not plugged to the point of negating their intended functions. Also, SGT system heaters are tested to verify that they will reduce the relative humidity of gas flow through the charcoal filters thus assisting the maintenance of efficient charcoal filter operation. SGT system filter differential pressure testing and heater performance evaluations are required to be measured at the system design flow rate.

  CNS Technical Specification, Section 4.7.B.1.a, states, "Pressure drop across the combined HEPA filters and charcoal absorber banks is less than six inches of water at the system design flow rate."

  The SRI noted that Procedure 6.3.19.2, performed on April 20, 1984, was conducted at a SGT system flow rate of 1350 cfm and not at the system design flow rate. The failure to perform Procedure 6.3.19.2 at design flow conditions is an apparent violation. This item is discussed further in the summary section of paragraph 6.

- 6.3.19.3, Revision 7, "SGT HEPA Filters Leak and Housing Door Seal Leak Test," performed on August 13, 1984.

  Procedure 6.3.19.3, is conducted by the licensee in order to determine HEPA filter DOP removal capability. CNS Technical Sepcification, Section 3.7.B.2.a, requires cold DOP and halogenated tests to be performed at design flow. Also, Procedure 6.3.19.3 is performed in order to determine if filter housing door seal leakage can also meet Technical Specification requirements while at design flow conditions.

  The SRI noted that Procedure 6.3.19.3, performed on August 13, 1984, did not indicate the system flow that was established and maintained during the test performance. However, a review of data associated with the performance of Procedure 6.3.19.4, conducted earlier on August 13, 1984, indicates that system flow was established and maintained at 1250 cfm. The actual system flow of 1250 cfm does not meet the USAR and Technical Specification required design flow rate. The failure to perform Procedure 6.3.19.3 at design flow conditions is an apparent violation. This item is discussed further in the summary section of paragraph 6.

- Procedure 6.3.19.4, Revision 10, "SGT Charcoal Filter Leak and Fan Capacity Test," performed on August 13, 1984.

6.3.19.4, is performed in order for the licensee to determine inplace leakage of the carbon filters and also to verify SGT fan capacities. During his review of Procedure 6.3.19.4, the SRI noted that the following were in disagreement with Technical Specification requirements:

1) SGT fans flow data obtained during the performance of the test was 1250 cfm.

   CNS USAR, Volume II, Section V, subsection 3.3.4, paragraph 2, states that each SGT fan has a design flow of 1780 cfm.

   CNS Technical Specifications section 3.7.B.2.c, states, "Fans shall be shown to operate with $\pm$ 10% of design flow."

   The failure to perform procedure 6.3.19.4 at design flow conditions is an apparent violation. This item is discussed further in the summary section of paragraph 6.

2) Procedure 6.3.19.4, Section VI, Precautions, subsection D, states in Part, ". . . Preop flow rate was measured at 1750 cfm and accepted. This is now considered <u>design flow</u>." Section VI, subsection E, states, "Fans must be shown to operate with $\pm$ 10% of design flow (1575 cfm to 1925 cfm)."

   The licensee arbitrarily revised Procedure 6.3.19.4 subsections D and E design flow requirements to equal 1750 cfm. This failure to perform a safety review of test data prior to accepting that data as the new design flow, constitutes an unreviewed safety question and an apparent violation. This item is discussed further in the following summary paragraph.

In summary, the licensee failed to perform surveillance test procedures 6.3.19.2, 6.3.19.3, and 6.3.19.4 at the system design flow rate. Collectively, performance of the preceeding tests, at other than design flow, is contrary to USAR and Technical Specification requirements. Presently, testing at the design flow rate establishes the basis for demonstrating system/subsystem operability. The failure to demonstrate

SGT system/subsystem operability as stated above, constitutes an apparent violation. However, discussions with NRR technical branches has determined that the function of the SGT system is to reduce and maintain the secondary containment atmospheric pressure at a minimum of 0.25 inches of water vacuum while directing all flow through the SGT filtration assemblies. The amount of SGT system flow required to reduce and hold vacuum is dependent upon the amount of in-leakage to the secondary containment. The licensee's surveillance procedures have met the intent of the Technical Specification to verify that the SGT system functions as designed. However, the USAR and the Technical Specification exhibit the following descrepancies:

- The design function of the SGT system is not explicitly defined.
- The absolute value of system design flow is not stated nor is it readily apparent.
- Procedures 6.3.19.2 and 6.3.19.3 do not specify a system flow rate that should be established and maintained during the tests performance that would provide a consistent and sound technical basis for determining system/subsystem operability at that given flow condition.

Thus, the Technical Specification requirements for determining operability of the SGT system/subsystems are ambiguous. Pending necessary revision to and clarification of the USAR and Technical Specification by the licensee, and a review and acceptance of those proposed revisions by NRR, this will remain an unresolved item. (298/8420-01)

Revising the value of SGT system fan design flow from 1780 cfm to 1750 cfm in Procedure 6.3.19.4, Section VI, subsections D and E, is a change to the facility as described in the USAR. The licensee did not make hardware changes to the SGT system as a result of the preoperational test. Therefore, the revision made to Procedure 6.3.19.4, as stated above, has minor safety significance. However, failure to revise Procedure 6.3.19.4 as outlined in 10 CFR Part 50.59 constitutes an apparent violation. (298/8420-02)

7. <u>Followup of IE Bulletins (IEB)</u>

a. IEB 84-01 (Closed), "Cracks In Boiling Water Reactor Mark I Containment Vent Headers."

IEB 84-01 was issued February 3, 1984, as a result of the identification of through wall cracks around the primary containment vent header at the Hatch Unit 2 nuclear plant. Specifically, the cracks appeared in that portion of vent header piping located inside the torus. The cause of the crack at the Hatch Unit 2, appeared to result from embrittlement of the vent header piping due to impingement of cold nitrogen directly upon the failed area. Nitrogen is used in the containment inerting process and is supplied from the nitrogen inerting system at a temperature of between 50° and 100°F.

IEB 84-01 requested operating boiling water reactor (BWR) plants to review their plant data on differential pressure between the wet-well and drywell for any anomalies that could be indicative of cracks. CNS was operating on the date IEB 84-01 was issued. The licensee took the following immediate actions on February 3, 1984:

- Reviewed approximately 30 days of daily technical specification logs which note N_2 useage.

- Reviewed approximately 60 days of control room drywell/torus differential pressure recorder chart data.

- Reviewed drywell/torus pump around system compressor accumulative run time meters.

The above reviews did not identify any anomolous conditions. The licensee notified the SRI of the results of their review on February 3, 1984.

On February 14, 1984, General Electric (GE) issued Service Information Letter (SIL) number 402, "Wetwell/Drywell Inerting." In SIL 402, GE recommended that all BWR owners having Mark I or Mark II containment systems, perform the following actions with respect to the nitrogen inerting system:

- Evaluate system design.

- Evaluate system operating characteristics.

- Review maintenance and operating procedures for adequacy.

- Test for drywell/torus bypass leakage.

- Perform nondestructive examinations (NDE) of accessible piping, welds, and containment penetrations that are located downstream of the containment isolation valves.

- Perform visual examination of primary containment vent headers and downcomers located in the vicinity of nitrogen injection lines.

- Inspect the containment shell or liner within six inches of the nitrogen injection penetrations.

The licensee reported in a letter from Mr. J. M. Pilant (NPPD) to Mr. D. B. Vassallo (NRC), dated September 12, 1984, that all SIL 402 recommendations had been completed. Visual and nondestructive examination results were documented on CNS maintenance work request (MWR) number 84-0319.

The SRI verified that the licensee completed all of the reviews, evaluations, tests, and inspections described above. The SRI independently reviewed plant logs and recorder data, system layout prints, maintenance and operating procedures, and nondestructive examination (NDE) test results. The licensee appears to have satisfactorily completed all the requirements of IEB 84-01.

b. IEB 84-02 (Closed), "Failures of General Electric Type HFA Relays in Class IE Safety Systems."

IEB 84-02 was issued March 12, 1984, as a result of an increase failure rate of General Electric (GE) type HFA relays. GE has attributed recent failures to end-of-life situations. CNS uses normally energized and normally de-energized HFA relays in safety-related applications. IEB 84-02 requires the following actions from all holders of operating licenses:

- Plans and schedules for replacing HFA relay coils or entire relays must be developed. The replacement program should be completed within two years from the date of this bulletin. HFA relays used in normally energized or de-energized safety-related applications are to be included for replacement.

- During the period prior to relay replacement, develop and implement monthly functional tests that will verify that relay contacts change state when the normally energized relays are operated. Also, visually inspect all safety-related normally energized relays for evidence of relay contacts and relay coils deteriation.

- Justify the basis for continued reactor operation for the period of time preceeding relay replacement.

- If the subject HFA relays are to continue to be used in non-safety-related applications, administrative controls must be developed and implemented that will prevent inadvertent installation of these relays in safety-related systems during subsequent maintenance efforts.

- If different types and/or models of relays are in use in safety-related systems, review past operating history and manufacturer's recommendations to determine if general concerns apply. If concerns are identified, submit short and long-term corrective active plans and implementation schedules.

- Provide a written report of the above required actions to the NRC within 120 days of the receipt of this bulletin.

In a letter from Mr. L. G. Kuncl (NPPD) to Mr. J. T. Collins (NRC - RIV), dated July 16, 1984, the licensee submitted the following responses:

- The normally energized and normally de-energized HFA relays used in safety-related applications are being replaced during the present plant outage. The replacement relays are the GE "Century Series" model.

- CNS presently performs monthly functional tests of all reactor trip system normally energized relays. Visual inspection of all safety-related normally energized relays is an on-going practice at CNS.

- Justification for continued operation was based upon the plan to replace all safety-related HFA relays during the 1984 outage, monthly visual and functional tests are an on-going practice, and the licensee completed shorted turns testing of all safety-related HFA 120V AC relay coils during the 1983 spring refueling outage.

- The licensee does use the subject HFA relays in nonsafety-related applications at CNS. Administrative procedures do not presently exist that provide controls of these relays to prevent inadvertant installation into safety-related systems. This is considered an open item. (298/8420-03)

- In a letter from Mr. L. G. Kuncl (NPPD) to Mr. R. D. Martin (NRC - RIV), dated October 9, 1984, the licensee stated that their review of non-HFA safety-related relays was completed. Further, the licensee determined that due to the low failure rate of non-HFA relays at CNS, and because the relays are periodically checked by the existing preventive maintenance and surveillance programs, continued satisfactory relay performance will continue.

The SRI has observed the implementation of the licensee's reviews, plans, and schedules as outlined above. The SRI has verified that the replacement of all safety-related GE type HFA relays is presently in progress. The SRI will verify satisfactory testing of all newly installed relays prior to plant startup.

c. IEB 84-03 (Closed), "Refueling Cavity Water Seal."

IEB 84-03 was issued August 24, 1984, in response to the failure of a refueling cavity water seal while the cavity was flooded in preparation for refueling operations. The affected seal assembly consisted of an annular plate with two pneumatic seals.

CNS was at power operation when IEB 84-03 was issued. Therefore, IEB 84-03 required the licensee to evaluate the potential for and consequences of a refueling cavity water seal failure and provide a summary report of these actions to the NRC Regional Administrator, prior to the commencement of the 1984 refueling activities.
Such evaluations were to include consideration of:

- Gross Seal failure.

- Maximum leak rate due to failure of active components such as inflated seals.

- Makeup capacity.

- Time to cladding damage without operator action.

- Potential effect on stored fuel and fuel in transfer.

- Emergency operating procedures.

In a letter from Mr. L. G. Kuncl (NPPD) to Mr. J. T. Collins (NRC - RIV), dated September 30, 1984, the licensee reported the results of their evaluations as follows:

- Gross seal failure - the refueling cavity water seal at CNS is not a pneumatic type. The seal consists of a stainless steel bellows, backing plate, self-energizing spring seal, and removable guard ring. The self-energizing spring seal is designed to form a seal against the backing plate by yielding to that plate if a bellows rupture should occur. Also, a seal leak in excess of five gallons per minute causes an alarm to annunciate in the control room.

- Maximum leak rate due to failure of active components - the licensee determined that the seal was of a passive rather than active design. Therefore no leak rate calculations were necessary.

- Makeup capacity - the core spray and residual heat removal systems combined can supply approximately 40,000 gallons per minute of makeup water to the refueling cavity.

- Time to cladding damage - due to plant design, the reactor vessel and spent fuel pool can not be drained to a level below the top of stored fuel if a seal failure would occur. Also, the makeup water capability described above, would provide ample time to place an elevated spent fuel bundle back into the reactor core or a spent fuel pool storage position.

- Potential effect on stored fuel and fuel in transfer - see above response. In addition, the licensee would implement Emergency Operating Procedure (EOP) 5.3.5, "Refueling Floor High Radiation."

  Emergency Operating Procedures - EOP 5.3.5 provides required actions to be taken by operations personnel in the event of a refueling accident. Included are directions for returning an elevated fuel bundle to "wet" storage and providing makeup water to the refueling cavity.

The SRI held discussions with licensee personnel and conducted an independent verification of licensee actions applicable to IEB 84-03. Also, the SRI conducted a review of refueling cavity water seal blue-prints, water supply systems makeup capabilities, refueling cavity and spent fuel pool as built configurations, and plant procedures.

8. Exit Meetings

Exit meetings were conducted at the conclusion of each portion of the inspection. The division manager of nuclear operations was informed of the above findings.