APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report No. 50-298/92-10

Operating License: DPR-46

Docket: 50-298

Licensee: Nebraska Public Power District
P.O. Box 499
Columbus, Nebraska 68602-0499

Facility Name: Cooper Nuclear Station

Inspection At: Brownville, Nebraska

Inspection Conducted: May 31 through July 11, 1992

Inspectors: R. A. Kopriva, Senior Resident Inspector
W. C. Walker, Resident Inspector
D. L. Kelley, Test Programs Section, Division of Reactor Safety

Approved: P. H. Harrell, Chief, Projects Section C, Division of Reactor Projects — 7/24/92 Date

Inspection Summary

Inspection Conducted May 31 through July 11, 1992 (Report 50-298/92-10)

Areas Inspected: Routine, unannounced inspection of onsite followup of a licensee event report, followup of previously identified inspection findings, operational safety verification, maintenance and surveillance observations, and containment integrated leak rate test review.

Results:

- During a tour with an operations manager, operating personnel were found to be performing their intended functions and housekeeping was good (paragraph 5.b).

- Control room communications (i.e., repeat backs) with operations personnel and other plant personnel was improving (paragraph 7.-).

- Maintenance activities were well planned and executed (paragraph 6.a).

DETAILS

1. Persons Contacted

Principal Licensee Employees

L. E. Bray, Regulatory Compliance Specialist
R. Brungart, Operations Manager
M. A. Dean, Nuclear Licensing and Safety Manager, Onsite
C. M. Estes, Acting Senior Manager, Technical Support Services
R. L. Gardner, Division Manager, Nuclear Operations
H. T. Hitch, Plant Services Manager
S. M. Peterson, Senior Manager Technical Support
J. V. Sayer, Radiological Manager
G. E. Smith, Quality Assurance Manager
G. R. Smith, Nuclear Licensing and Safety Manager, Offsite
R. L. Wenzel, Nebraska Engineering Division Site Manager

The above personnel attended the exit meeting held on July 16, 1992. The inspectors also contacted additional personnel during this inspection period.

2. Plant Status

The reactor operated at essentially full power from May 31, 1992, through the end of this inspection period.

3. Onsite Followup of a Licensee Event Report (92700)

(Closed) Licensee Event Report 298/92-006: Technical Specification Violation Due to Inoperable 250-Volt Battery Chargers Caused By Equipment Deficiencies.

On April 7, 1992, at 7:31 a.m., the 1B 250-volt battery charger input breaker tripped. Following an investigation, the 1B battery and charger were declared inoperable and the Technical Specification limiting condition for operation entered. On April 8 at 1:23 p.m., the input breaker to the 1A 250-volt battery charger tripped, resulting in the 1A charger being inoperable. After an inspection, the charger was reenergized, but tripped again approximately 1 minute later. The float voltage was adjusted and the charger returned to operation at 1:52 p.m. Several hours later, approximately 30 minutes after an additional voltage adjustment, the 1A charger tripped again. The voltage adjustment potentiometer was exercised several times and the charger returned to service. The unit operated satisfactorily. Both chargers being out of service resulted in operation prohibited by the Technical Specifications. The troubleshooting of the 1A charger was subsequently performed.

The cause of the trip of the 1B charger could not be definitively established. Troubleshooting found a failed electronic board, but this discrepancy would not cause the charger to trip without producing any fault indication. The failure of the 1A charger was due to inadequate preventive maintenance combined with setpoint drift of the high voltage shutdown relay. Both chargers were repaired, tested, and returned to service. Preventive maintenance for the chargers will be enhanced and additional investigations into the failure rate of the K-3 relay in the 1A charger will be performed.

4. Followup of Previously Identified Inspection Findings (92701)

a. (Closed) Unresolved Item 298/9127-02. Failure To Declare An Unusual Event.

On July 30, 1991, both diesel generators (DG) were declared inoperable because of inadequate seismic qualifications of the DG heating, ventilation, and air conditioning units. The provisions of the Technical Specification action statements for two inoperable DGs were implemented to reduce reactor power to 25 percent.

Emergency Plan Implementing Procedure 5.7.1, Attachment B, Step 4.1.2, specified that the loss of both onsite DGs, with offsite power available, is a condition requiring the declaration of an unusual event. The licensee did not declare an unusual event.

The licensee's decision to treat the DGs as not capable of performing their function for Technical Specification purposes and, at the same time, as capable of performing their function for emergency plan implementation was inconsistent.

The licensee has completed a procedure change to Emergency Plan Implementing Procedure 5.7.1, "Emergency Classification," which states that the term "loss," which is used in Procedure 5.7.1, is the same as not having "operability" as defined in the Technical Specifications. On the basis of this change, this unresolved item is considered closed.

b. (Closed) Inspection Followup Item 298/9039-01: Removal of References to Quality Control Steps in Solid Radioactive Waste and Radioactive Material Transportation Procedures.

This item involved an NRC inspection of records for solidification of waste performed in accordance with Procedure 2.5.4.1, "Solid Wet Waste Packaging, Storage, and Transfer System." During the inspection, it was noted that a step was included for quality control to verify proper solidification of waste.

As documented by a letter, dated February 15, 1991, the licensee has reviewed all radwaste, chemistry, and auxiliary equipment operating procedures and removed quality control steps and replaced them with verification steps, as necessary. The inspector reviewed documentation for the completion of the corrective actions. Based on the completed corrective actions, this inspection followup item is closed.

c. (Closed) Violation 298/9204-001: 250-Volt DC Battery Cell Voltage Below 2.13 Volts.

- On December 19, 1991, the licensee measured Cell 110 of 250-volt Battery EE-BAT-250(1A) at 2.05 volts, a condition adverse to quality caused by copper contamination, and immediate corrective action was not taken to perform an equalizing charge or to remove the degraded cell from service.

- Actions were not taken to prevent recurrence. Redundant train 250-volt Battery EE-BAT-250(1B) was found, on February 5, 1992, to be degraded due to copper contamination in that Cell 88 was measured at 2.13 volts. Again, no immediate corrective action was taken to perform an equalizing charge or to remove the degraded cell from service. On February 10, 1992, Cell 88 was measured at 2.06 volts.

The following corrective actions were taken:

- Cell 110 of the A 250-volt battery and six other cells in that battery, that were exhibiting indications of advanced copper contamination, were replaced.

- Cell 38 of the B 250-volt battery and two other cells in that battery were also replaced.

- A battery action plan requiring the frequency of individual cell monitoring to be increased and requiring trending of individual cell voltages for those cells, where there was evidence of copper contamination, was formally implemented.

- A temporary design change for jumpering and replacing a cell was developed. Should the need arise, the change would be approved and implemented.

Subsequently, following plant startup, a test discharge of 5 of the cells removed during shutdown, including Cell 110 from the A 250-volt battery, was performed. The test verified that the cells, even though in a degraded condition, would have met their design-basis performance requirement.

Finally, a change to the Technical Specifications was submitted, which delineates specific parameter limits for individual cells and the corresponding effect on battery operability.

The inspectors reviewed the licensee's actions taken to address this violation, as discussed above, and noted that the surveillance test program and operability determination program upgrades had been completed. The inspectors also noted that additional human factors improvements will be completed by the licensee on or before January 1, 1993. Issuance of a change to the Technical Specifications in the near future should further clarify the operability requirements for the station batteries.

d. (Closed) Violation 298/9204-02: Failure to Follow a Procedure.

The battery operability evaluation was performed on January 15, 1992, to confirm the operability of the A 250-volt battery that had the degraded cell. However, the procedural requirements of Cooper Nuclear Station (CNS) Procedure 0.27, "Component Operability," were not followed in that the operability evaluation was not formally submitted for Station Operation Review Committee review and approval.

Surveillance test discrepancies associated with Technical Specification limits now result in an immediate declaration of inoperability. In addition, the operability determination program has been significantly upgraded. Under the revised program, an operability evaluation is performed for all other surveillance test discrepancies that are not associated with Technical Specification limits. The program was developed to be consistent with the guidance provided in Generic Letter 91-18 and requires a Station Operation Review Committee review within 1 working day if the discrepancy is associated with functionality, the discrepancy is existing (has not been corrected), and the affected structure, system, or component is being considered operable. The program was implemented May 1, 1992.

The Station Operability Review Committee review of all operability determinations performed in accordance with CNS Procedure 0.27 is being performed, as specified by the procedure. Full compliance has been achieved.

The human factors improvements to the surveillance procedures will be completed by January 1, 1993.

The inspectors found the licensee's actions acceptable.

5. Operational Safety Verification (71707)

a. Routine Control Room Observations

The inspectors observed operational activities throughout this inspection period. Proper control room staffing was maintained and control room professionalism was observed. Discussions with operators determined they were cognizant of plant status and were aware of plant activities that could affect plant safety. The inspector observed selected shift turnover meetings and noted excellent transfer of information concerning plant status.

b. Plant Tours

The inspectors routinely toured various areas of the plant to verify that proper housekeeping was being maintained. Plant housekeeping was adequate considering the activities ongoing.

On June 9, 1992, the inspectors accompanied a member of licensee management on a tour through the radioactive waste building. Operating personnel were found to be performing their intended functions and housekeeping was good.

c. Radiological Protection Program Observations

The inspectors verified that selected activities of the licensee's radiological protection program were implemented in conformance with facility policies, procedures, and regulatory requirements. Radiation and/or contaminated areas were properly posted and controlled. Health physics personnel were observed to be touring work areas to ensure that proper radiological protection practices and radiological control requirements were properly implemented.

d. Security Program Observations

The inspectors observed various aspects of the licensee's security program. Guards were observed posted on fire watches, when necessary, and were attentive. It was noted that personnel, packages, and vehicles were properly searched before entering the protected area.

e. Potential For Loss of Remote Shutdown Capability During a Control Room Fire

On February 28, 1992, Information Notice 92-18, "Potential For Loss of Remote Shutdown Capability During a Control Room Fire," was issued. The licensee reviewed this information notice and identified a potential unanalyzed condition regarding a hot short in the control cables of the valves controlled from the alternate shutdown panel. The Appendix R motor-operated valves may not maintain mechanical integrity if energized by a hot short and could continue to run in a lock-rotor condition. Operator action could manually position the affected valves outside of the primary containment; however, access to the valves inside primary containment would be restricted. Two valves, in particular, are of concern: the reactor recirculation loop discharge valve and the high pressure core injection steam supply valve. The potential effect of this scenario would be the inability to provide makeup to the reactor vessel.

The licensee has informed operations control room personnel of the scenario and possible consequences of a fire in an alternate shutdown area if control was not transferred to the alternate shutdown panel prior to equipment damage. Procedure 5.4.3.2, "Post-Fire Shutdown To Cold Shutdown Outside Control Room," was changed to ensure prompt direct entry into this procedure on evidence of fire in any of the six alternate shutdown areas. Technical Specifications, Section 3.2.I ? require the alternate shutdown capability to be restored within 30 days or to notify the NRC and provide plans to restore alternate shutdown capability. On July 8, 1992, the licensee committed by letter to NRC to make any necessary additional final corrective actions during the 1993 refueling outage.

f. <u>Thermo-Lag 330 Fire Barrier Systems</u>

On June 24, 1992, NRC issued Bulletin 92-01, "Failure of Thermo-Lag 330 Fire barrier Systems To Maintain Cabling in Wide Cable Trays and Small Conduits." This bulletin informed operating reactor licensees of the inability of Thermo-Lag 330 fire barrier systems to provide the required level of fire safety for the protection of safety-related shutdown cabling. The bulletin required licensees to take appropriate compensatory measures consistent with those taken for inoperable fire barriers and advise the NRC on what actions would be taken to restore the affected fire barriers to an operable status.

The licensee has an area in the ceiling of the service water booster pump room that uses Thermo-Lag 330 to provide a thermal shield for the Division II, 125-volt direct current power leads to the diesel generator control circuitry. Appropriate compensatory measures were established in this area (i.e., continuous fire watch) and the licensee was preparing a written report, as required by NRC Bulletin 92-01, to describe the measures taken to ensure or restore fire barrier operability.

Conclusions

Good control room communications were evident during shift turnover. Operations personnel were found to be performing their intended functions and housekeeping was good. The licensee instituted appropriate compensatory measures to address several safety issues.

6. Maintenance Observation (62703)

The inspector witnessed a reactor water cleanup (RWCU) weld repair on June 25, 1992. This work was performed under Maintenance Work Request 92-1376.

On June 17, the licensee discovered a small steam leak at the joint between a 3-inch flange and 4-inch RWCU piping. An engineering analysis was performed and special instructions were written to isolate the affected portion of the RWCU piping for the weld repair, and to restore the RWCU system to service. The procedures for welding and postmaintenance testing had been reviewed and approved, as designated by appropriate signatures. The results of the acceptance testing for the welding, which consisted of a radiograph, were satisfactory. Also, an inservice leak test was satisfactorily conducted after the new flange was installed per Procedure 7.0.8.1, "Inservice Leak Testing." Procedural compliance was noted throughout this effort.

Conclusions

Maintenance activities observed were well planned and performed in a coordinated controlled manner with adherence to procedures.

7. Surveillance Observation (61726)

On June 10, 1992, the inspectors witnessed operations personnel perform inservice testing, using Procedure 6.3.3.1, "High Pressure Core Injection (HPCI) System Quarterly Inservice Test." This test was performed to verify the operational readiness of the high pressure core injection pump. The inspectors noted attention to detail was apparent throughout the performance of the test. The instrumentation and control technicians coordinated with the control room via telephone and repeat backs were used in recording results to ensure accuracy. In addition, the inspectors independently verified all test results met acceptance criteria. A review of the completed test package showed that all required review and approvals were made.

Conclusions

The surveillance was performed in accordance with the procedure, and good attention to detail was maintained.

8. Containment Integrated Leak Rate Test (70323)

This portion of the inspection deals with the inspector's review of the licensee's final report of the containment integrated leak rate test conducted on December 8 through 10, 1991.

The test was performed using the Surveillance Procedure 6.3.1.3, "Primary Containment Integrated Leak Rate Test." The test was a short duration leak rate test performed in accordance with Bechtel Topical Report BN-TOP-1. The test report consisted of the test summary sheet and integrated leak rate data sheets, instrument lists, verification test data sheets, temperature stabilization data, the edited log of events, and the test results. The inspector examined the test report for any abnormal data points and instrument failures. In addition, several data record points were checked for mathematical accuracy. No instrument failures or abnormal data were identified, nor were any mathematical errors identified.

The test result identified a total as-left leakage rate of 0.30159 percent weight/day which was less than the acceptance criteria of 0.635 percent weight/day.

Conclusions

The inspector's review of the licensee's results of the leak rate testing found the results to be acceptable.

9. Management Meeting (30702)

On July 7, 1992, Mr. Hugh Parris, Vice President of Production for Nebraska Public Power District, Cooper Nuclear Station, and representatives of his staff met in the NRC Region IV office with Mr. J. Milhoan, Region IV Regional Administrator, and members of the NRC staff. The licensee provided the NRC with copies of its response to the Systematic Assessment of Licensee Performance, NRC Inspection Report 50-298/92-99. The licensee also included a presentation on several subjects, including licensed operator training, radiological controls, nuclear procurement program, and operability program/deficiency program improvements.

10. Summary of Open Items

The following is a synopsis of the status of all items closed in this inspection report. No new items were generated.

Inspection Followup Items 9039-001 and 9127-002 were closed.

Violations 9204-001 and 9204-003 were closed.

LER 92-006 was closed.

11. <u>Exit Meeting</u>

An exit meeting was conducted on July 16, 1992, with the licensee representatives identified in paragraph 1. During this meeting, the inspectors reviewed the scope and findings of the inspection. During the exit meeting, the licensee did not identify as proprietary, any information provided to, or reviewed by, the inspectors.