APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-298/89-19 Operating License: DPR-46

Docket: 50-298

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

Facility Name: Cooper Nuclear Station (CNS)

Inspection At: CNS, Brownsville, Nebraska

Inspection Conducted: May 1-5 and 15-19, 1989

Inspectors: P.C. Wagner 6-2-89
P. C. Wagner, Reactor Inspector, Plant Systems Section, Division of Reactor Safety Date

T.F. Stetka for 6/2/89
A. Singh, Reactor Inspector, Plant Systems Section, Division of Reactor Safety Date

Approved: T.F. Stetka 6/2/89
T. F. Stetka, Chief, Plant Systems Section Division of Reactor Safety Date

Inspection Summary

Inspection Conducted May 1-5 and 15-19, 1989 (Report 50-298/89-19)

Areas Inspected: Routine, unannounced inspection of the licensee's actions in response to NRC requirements for motor operated valve (MOV) testing and the licensee's programs for instrument calibration and piping supports. The inspection included gathering survey information related to drywell temperatures and diesel fuel oil storage and handling, and discussions of the implementation of Regulatory Guide 1.97 instrumentation. The NRC inspectors also reviewed the corrective actions related to the Type SJO electrical cable problems and the actions completed in response to previous NRC inspection findings.

Results: Within the scope of the inspection, one violation of NRC requirements was identified (paragraph 3.b). The violation involved two examples where component testing was performed utilizing instrumentation which had not been calibrated. The NRC inspectors found the MOV testing program to be good but limited in scope in that only those valves covered by IE Bulletin 85-03 were being tested. The NRC inspectors also considered the basis for the repair of the SJO cable to be less substantial than would have been desirable.

The NRC inspectors determined that the calibration program for instruments listed in the Technical Specifications to be a good program.

DETAILS

1. Persons Contacted

NPPD

+*L. Bray, Regulatory Compliance Specialist
+ R. Brungardt, Operations Manager
+ B. Fehrman, Project Manager
*J. Flaherty, Plant Engineering Supervisor
+*S. Freborg, Assistant Plant Engineering Supervisor
+*R. Gardner, Maintenance Manager
B. Jansky, Outage and Modification Manager
*J. Hall, Mechanical Supervisor
+*G. Horn, Plant Manager, Nuclear Operations
+ H. Jantzen, Instrument and Controls (I&C) Supervisor
+ L. Kohles, Nuclear Project and Construction Manager
+*E. Mace, Engineering Manager
+*J. Meacham, Senior Manager of Operations
*D. Robinson, Quality Assurance (QA) Supervisor
+ G. Smith, QA Manager, CNS

NPPD - Columbus Office (via telephone)

+ J. Branch, Supervisor, Engineering
+ A. Heymer, Manager, Configuration Management
+ S. McClure, Manager, Nuclear Engineering
+ H. Parris, Vice President, Production
+ G. Smith, Licensing Supervisor
+ G. Trevors, Division Manager, Nuclear Support
+ K. Walden, Licensing Manager
+ R. Wilber, Division Manager, Nuclear Engineering and Construction
+ V. Wolstenholm, Division Manager, QA

NRC, Region IV

+ J. Jaudon, Deputy Division Director, Division of Reactor Safety
*L. Ellershaw, Reactor Inspector
*W. McNeill, Reactor Inspector
+ L. Gilbert, Reactor Inspector
*G. Pick, Resident Inspector
+ W. Bennett, Senior Resident Inspector

*Denotes personnel present at the May 5, 1989, exit meeting.
+ Denotes personnel present at the May 19, 1989, exit meeting.

The NRC inspectors also contacted and interviewed other NPPD personnel during the performance of this inspection.

2. Motor Operated Valves (25573)

Because of problems identified with the torque switch and limit switch settings of important motor operated valve (MOV) actuators, the NRC issued IE Bulletin (IEB) 85-03. The IEB was provided to ensure that switches were set and maintained so as to accommodate the most severe loadings expected during a design basis event. NRC inspection guidance was provided in Temporary Instruction (TI) 2515/73.

NRC inspection of the licensee's response to IEB 85-03 was initiated in November 1987 and documented in NRC Inspection Report 50-298/87-30. The IEB remained open pending completion of the MOV testing. The MOV testing at CNS was performed utilizing the "MOVATS" system and equipment. Subsequent to the initial NRC inspection, NPPD provided additional information in response to IEB 85-03 in letters dated June 2 and September 20, 1988. The NRC requested some clarification to the NPPD submittals by letter dated November 9, 1988, which NPPD provided in their January 9, 1989, response.

The NRC technical review of the licensee's response is contained in subparagraph a. below. The NRC inspection of the remaining requirements of TI 2515/73 are contained in subparagraphs b. through d. below. A discussion of NPPD review results related to NRC Information Notices is contained in subparagraph e. below.

a. Technical Review of IEB 85-03 Submittals

Review of the licensee's January 9, 1989, response and their letter of June 2, 1988, in response to Supplement 1 of IEB 85-03, indicates that their selection of the applicable safety-related valves to be addressed and the valves' maximum differential pressures meet the requirements of the IEB and that the program to assure valve operability requested by action item e. of the IEB and its supplement is now acceptable.

b. Procedure Review

The NRC inspector reviewed the Maintenance Procedures (MP) related to MOVs listed in Attachment 1 to this report. The NRC inspector found all of the procedures to contain easily understood, detailed, step-by-step instructions. The procedures also included frequent requirements for independent verification of actions. There were a total of 96 MOVs listed in MP-7.3.36, but only 19 of these MOVs were included in the IEB 85-03 testing program. The 19 MOVs included in the IEB 85-03 program consisted of 10 Reactor Core Isolation Cooling (RCIC) system and 9 High Pressure Coolant Injection (HPCI) system valves; all of the valves except 1 RCIC and 1 HPCI were DC motor operated. During discussions with licensee personnel about the MOVATS program, the NRC inspector was informed that the remaining DC MOVs will be MOVATS tested in the fall of 1989 (see paragraph e. below).

c. Training

Since paragraph 6.5 of MP-7.3.35-1 required qualified personnel to perform or supervise the MOVATS testing, the NRC inspector reviewed the training program lesson plans listed in Attachment 1. The three MOV lessons, along with their related laboratory exercises, were prerequisites for the MOVATS training course. The NRC inspector found all of the lesson plans to contain sufficient detail for the functions to be performed.

The NRC inspector also noted that EQP-011-02-03 included maintenance update information from Limitorque Corporation (the MOV vendor) and information on DC motor cable sizing (see paragraph e. below).

d. MOV Testing

NRC Inspection Report 50-298/87-30 stated that 6 of the 19 IEB 85-03 listed MOVs would be required to be tested under pressure. This statement was based on an August 3, 1987, letter to NPPD from MOVATS Incorporated; subsequently, NPPD provided MOVATS additional information, and by letter dated March 22, 1988, MOVATS rescinded their earlier recommendation on the need to test the valves at pressure.

The NRC inspector reviewed the records for the MOV testing completed in April and May 1988, on 10 of the 19 MOVs listed in NPPD's January 9, 1989, submittal to ensure that proper testing had been performed. While the NRC inspector found the testing records to be properly completed with acceptable results, he noted that two valves had new actuators installed prior to the tests and that five other valves had the actuator spring packs replaced. In addition, all of the MOV actuators were cleaned and lubricated prior to testing. Therefore, the NRC inspector questioned how the operability prior to testing designated in the NPPD letter had been determined. NPPD personnel stated that the initial operability determination was based on the testing performed in 1986; the questions arising from any abnormal indications during those tests and the results of evaluations performed subsequent to those tests led to the replacements noted in the 1988 test records. The replacement activities were performed to enhance the MOV operations not to provide operability.

The NRC inspector also questioned how the MOVs had been relubricated, specifically concerning the possibility of mixing the old and new lubricants. The licensee personnel stated that the lubrication had properly consisted of cleaning all of the old grease from the mechanism and then applying fresh, approved grease.

e. DC Motor MOV Activities

The NRC issued Information Notices (INs) 88-72, "Inadequacies in the Design of DC Motor Operated Valves;" and 89-11, "Failure of DC Motor Operated Valves to Develop Rated Torque Because of Improper Cable Sizing," to alert licensees of potential problems. The NRC inspector discussed the status of NPPD's efforts to evaluate the conditions presented in the INs with the responsible corporate engineer. The NRC inspector was informed that NPPD had evaluated the active valves included in the IEB 85-03 listing (i.e., those valves required to reposition during an accident) and had found them to be acceptable. The 125VDC valves were evaluated in calculations NED 89-131C and D; the 250VDC valves in NED 89-131A and B. The NRC inspector briefly reviewed those preliminary calculations and found them to contain conservative assumptions. The calculations concluded that adequate current would be provided to the actuator motors to produce the required torque for valve operation. The torque values were stated to have been calculated in accordance with vendor methodology and the current values were calculated using the presently installed motor feeder cables. NPPD engineering personnel stated that some DC motor starting resistors had been previously removed based on a Limitorque maintenance recommendation. The NPPD engineering personnel also stated that the remaining DC motor MOVs would be evaluated after the in progress refueling outage, and that those evaluations were expected to be completed in the fall of 1989.

The NRC inspector found the MOV testing program that had been implemented at the CNS in response to IEB 85-03 to be a good program. The NRC inspector, however, recommended that NPPD consider expanding the scope of that program to include additional safety-related valves.

No violations or deviations were identified.

3. Calibrations (56700)

In order to ascertain if the licensee had implemented a program for the calibration of installed plant instrumentation that was in accordance with regulatory requirements and accepted industry guidance, the NRC inspector reviewed selected plant documents. The documents reviewed included calibration, surveillance and testing procedures, and the records of the completion of those procedures. A partial listing of the documents reviewed is contained in Attachment 1 to this report.

a. Technical Specification Instruments

The NRC inspector reviewed a number of Calibration and Function Test procedures for Technical Specification (TS) listed instruments and found them to be acceptable. The majority of the procedures had been revised within the preceding year into a new, more easily followed, format. The NRC inspector noted attributes such as: readability;

easily understood, step-by-step instructions; clearly stated caution statements, limits and tolerances; and requirements for independent verification for return to service and for documenting any noted out of tolerance conditions.

The NRC inspector also noted that the procedures required a management review of the post-test/calibration data. The NRC inspector questioned how trending of data was accomplished and was informed that the systems engineers informally tracked and trended test results.

The NRC inspector also noted that most of the TS listed instruments were required to be functionally tested once per month and calibrated once per 3 months. Further, the functional test procedures usually performed a calibration check (e.g., a pressure source was connected to the pressure transmitter and varied to ensure proper response). The NRC inspector found both the calibration frequency and the functional test complexity to be more than usually experienced.

The NRC inspector reviewed the 1988 and 1989 test and calibration records for those instrument procedures that had been reviewed. The records were all complete, containing such attributes as proper sign-offs for completed evolutions and postimplementation verifications and reviews, documentation of the test equipment utilized, and the initiation of nonconformance reports for out of tolerance conditions when discovered. The NRC inspector questioned how the three times normal setpoint for the Main Steam Radiation Monitor (Surveillance Procedure (SP) 6.1.4) was determined and was directed to SP 9.4.3. Review of SP 9.4.3 indicated that the detector was calibrated by exposure to a traceable radiation source and that the normal, 100 percent power, background radiation level, in the area of the main steam line monitor, was determined prior to return to service from each refueling outage by calculating an average from the previous cycle data.

The NRC inspector also questioned how the HPCI pump low discharge flow setpoint of 2.4 inches water gauge (wg) contained in SP 6.2.2.3.13 was determined. A review of the scaling factors showed that the TS setpoint of 400 gpm was calculated to represent a 1.92-inch wg differential pressure at the flow transmitter based on the flow element curves. The procedure setpoint of 2.4-inch wg provided an acceptable margin to ensure flows greater than the TS required minimum.

Based on the sample reviewed, the NRC inspector found the calibration program for those instruments listed in the TS to be a good program. The NRC inspector had no further questions on the program and no violations or deviations were identified.

b. Nontechnical Specification Instruments

In order to ensure that the licensee was implementing an acceptable calibration program for those instruments which were utilized in safety-related functions but were not specifically identified in the TS, the NRC inspector reviewed operating and surveillance procedures and selected a sample of instruments utilized therein to determine proper equipment operability.

(1) Emergency Diesel Generator (EDG) Operability

The NRC inspector reviewed the EDG Operability Test Procedure (SP 6.3.12.1) and selected five parameters listed on the data sheets for review of the calibration status of the instruments utilized during the testing of No. 2 EDG. The review of the calibration records disclosed the following:

- Jacket Water Temperature - Temperature Indicator DG-JW-3145 was calibrated on a 4-year interval in accordance with Preventative Maintenance Procedure (PM) 01470.

- Lube Oil Filter Inlet and Outlet - Pressure Gauges PI-3143 and -3145 were not on a scheduled calibration program, but had been calibrated in November 1986 as part of another work item.

- Voltage, Frequency, and Current - Local electrical meters were not on a scheduled calibration program and had not had their calibration checked since initial startup in 1973.

The NRC inspector reviewed applicable procedures and records and verified that the protective functions (e.g., low lube oil pressure, overcurrent, differential overcurrent, etc.) were calibrated on an acceptable frequency. However, the NRC inspector also reviewed the procedure for shutdown outside the control room (Emergency Procedure 5.2.1) and noted that the local meters and gauges discussed above were utilized to operate and protect the EDGs during the performance of that procedure.

Violation (298/8919-01): The failure to have a procedure which ensured the proper, routine calibration of the local instrumentation utilized to verify the proper operation of the EDGs is an apparent violation of TS 6.3.3.D.

(2) Core Spray (CS) System Operability

The NRC inspector selected another system to evaluate if the instruments utilized during the surveillance test were being calibrated. The CS Operability Test (6.3.4.1) contained data sheets for the inservice test of CS pumps A and B which specified the instruments to be utilized during the test. The NRC inspector selected pump discharge flow, inlet pressure,

outlet pressure, and motor current as the parameters important to operation. Review of calibration records disclosed the following:

- Discharge Flow - FI-50B was calibrated on a quarterly frequency.
- Pump Pressures - PI-36B and -48B were calibrated on an annual frequency.
- Motor Current - CSP1B Ammeter was not included in a scheduled calibration program.

The NRC inspector reviewed the calibration records for the pump discharge flow and pressures and found them to be acceptable; however, the motor ammeters had not had calibration verified since the initial startup testing in 1973.

The failure to have a procedure to calibrate routinely instrumentation used during the performance of surveillance testing of the CS system is an additional example of the apparent violation described in paragraph 3.b(1) above.

4. Electrical Cable Repairs (92701)

The NRC inspector questioned if the preliminary notification (PNO-ADSP-89-01) issued on March 23, 1989, for a problem identified at the Browns Ferry Nuclear Facility was applicable to the CNS. The problem involved the deterioration of the Buna-S rubber insulation on the Type SJO electrical cable provided by General Electric Company (GE). The conductor insulation had become hard and brittle and had developed cracks. NPPD personnel stated that they had been informed of the problem by both GE and the Institute of Nuclear Power Operations (INPO) and had identified approximately 89 installations of the SJO cable at the CNS.

The SJO cables at the CNS were either 2 or 3 conductor, 18 AWG wire size, and usually provided as interconnecting wire. The cable had the appearance of normal appliance cord, similar to what would be used on an electric drill. The problems with the cable at the CNS included instances in which the insulation was completely missing from both conductors. The exposed portions of the conductor insulation had been brittle and in most cases had developed radial cracks; however, the portions of the insulation that had remained covered by the cable's PVC outer jacket material had not become brittle and remained flexible. NPPD had samples of the removed cable analyzed by GE and were verbally informed that the unexposed portions of the Buna-S insulation remained acceptable for continued used.

NPPD evaluated the cable problem and determined that continued use of the SJO cable pending final corrective action recommendations by GE was acceptable provided the cables were reterminated. The retermination

process included the removal of previously exposed portions of Buna-S insulation, stripping the cable jacket back to allow relugging of the freshly exposed conductors, then reterminating. The NRC inspector was informed that these interim actions were in accordance with GE recommendations. NPPD also referenced the 3-year expected shelf-storage life for SJO cable listed in Military Standardization Handbook for Rubber Products (MIL-HDBK-695C, 27 March 1985) as additional bases for the continued use of reterminated cables.

Inspector Followup Item (298/8919-02): While the NRC inspector did not determine that continued interim use of SJO cable would be detrimental to safe plant operations, followup inspection of the long-term corrective actions will be performed to ensure that the SJO cables are replaced prior to exceeding their expected service lifetime.

5. Postaccident Monitoring Instrumentation (25587)

By letter dated December 17, 1982, (Generic Letter 82-33) the NRC provided all reactor licensees and applicants with the "Requirements for Emergency Response Capability." Included in these requirements was the application of RG 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." The provisions for the instrumentation described in RG 1.97 were endorsed by the NRC to ensure that nuclear power plant operators would have sufficient and reliable information available for preventing and/or mitigating the consequences of a reactor accident. The NRC inspector initiated the inspection of this instrumentation in accordance with TI 2515/87.

The NRC inspector reviewed the NRC Safety Evaluation dated October 27, 1986, and noted that some of the instrumentation being utilized at the CNS differed from that described in NPPD's December 4, 1985, submittal. NPPD engineering personnel discussed the changes and indicated that the installed instrumentation was an enhancement to that originally proposed. NPPD personnel had committed to have the RG 1.97 instrumentation operable prior to restart from the in-progress refueling outage and stated that the commitment would be met.

The NRC inspector made a brief tour of the control room to verify that all of the RG 1.97, Category I, instruments were installed, including required redundancy and recorders. The NRC inspector found that the licensee had installed appropriate instrumentation; however, additional inspection activity will be required to ensure that the implementation of RG 1.97 at the CNS meets the NRC acceptance criteria. NPPD personnel agreed to update their December 4, 1985, response to indicate clearly which instruments were being utilized to fulfill the CNS commitments. The updated response will be submitted by August 21, 1989.

No violations or deviations were identified.

6. Information on Drywell Temperatures (25598)

Because of occurrences of higher than anticipated temperatures inside containments (PWRs) and drywells (BWRs), the NRC issued TI 2515/98 to obtain plant specific data. The average temperature profiles requested in Exhibit 1 to TI 2515/98 were to be used to determine if the high temperatures previously identified were plant specific problems or indicative of a more generic problem.

The NRC inspector compiled the data requested by the TI and attached a copy of the completed Exhibit 1 as Attachment 2 to this report.

No violations or deviations were identified.

7. Information on Diesel Fuel Oil (255100)

In order to verify that the licensee had implemented a program to maintain adequate quality of the fuel oil for the EDGs, the NRC inspector implemented TI 2515/100. This TI also requested information in the form of appended questions. The survey was completed and is included as Attachment 3.

The NRC inspector found the CNS programs to be acceptable but questioned the normal alignment of running both portions of the duplex fuel filter in parallel instead of one side or the other. Licensee personnel stated that no fuel oil problems had ever been encountered at the CNS.

No violations or deviations were identified.

8. Testing of Piping Support and Restraint Systems (70370)

The purpose of this inspection was to ascertain whether the licensee had established an adequate program and procedures pertaining to the examination and testing of piping and restraint systems.

a. Procedure Review

The NRC inspector reviewed the CNS second 10-year Inservice Inspection (ISI) Plan submitted to, and approved by, the NRC on January 27, 1986. The NRC inspector also reviewed the implementing procedures for the ISI plan listed below:

Procedure No.	Title	Date
6.3.10.9.1 Revision 15	Surveillance Procedure, Snubber Operability	02/28/89
7.2.34.1 Revision 1	Snubber Inspection	02/09/89

7.2.34.2 Revision 1	Pipe Snubbers Removal and Installation	01/17/89
IV3-W812 Revision 3	Visual Examination VT-3	02/19/88
IV4-W812 Revision 2	Visual Examination VT-4	02/18/87
ED-88-C4 Revision 1	Inservice Inspection (ISI) Activities	03/28/89
7.2.57 Revision 1	Maintenance Procedure for ASME Category F-A, F-B, and F-C Components Supports, Inspection, and Adjustment	08/30/88
7.2.34.7 Revision 0	Grinnell Figure 200/201 Hydraulic Snubber Functional Test	03/15/88
7.2.34.8 Revision 0	Pacific Scientific Snubber Functional Test	03/15/88
60-1 Revision 0	Receipt, Review, and Recording Calculations	01/20/89
60-2 Revision 0	Preconstruction Review of Proposed Pipe Support Modifications	01/20/89
60-3 Revision 0	Large Bore Piping and Pipe Support Design Criteria	01/20/89
60-4 Revision 0	Piping Analysis Procedure	01/20/89

Procedures reviewed by the NRC inspector appeared to be adequate.

b. Field Observations of Pipe Supports

The NRC inspector examined approximately 50 snubbers and pipe supports of various types and on various systems. Attributes selected for visual examination were:

- deterioration, corrosion, physical damage, or deformations were not evident;
- all required bolts, locking devices, nuts, and washers were installed;
- extension rods, support plates, and connecting joints were not deformed, or loose;

- snubber settings;
- pipes, supports, or other associated equipment or components were not restricted or in contact with other surfaces as a result of thermal expansion;
- springs in hangers were not obstructed by foreign material;
- indicators or spring hangers show either "cold" or "hot" position, consistent with plant condition; and
- threaded connections were secured by locknuts, fasteners, and cotter pins.

Snubbers and supports examined are listed below:

Snubbers

MS-S1B
MS-S16A
RH-S9
RH-S10
SW-H23A through H23H
RH-S52
RH-S36
RH-S37
MS-S13A
MS-S13B
MS-S17

Supports

RH-H47A	RF-HS5	MS-H101A
RCC-S86	SW-S29A	RCC-S67
SW-H149	RH-H56B	RH-H41A
BS-H92	RCC-S61	RH-H74
RF-H42A	RH-H17A	RH-H75
HP-H34	RH-H61A	RCC-S69
SW-H137	MS-H104	RCC-H65
CRD-BB5	RH-H96A	RCC-H69
MS-H113		
CS-H19A	RH-H50	
RCC-H137	SW-H184	
RCC-H138	SW-H181	
RCC-H139	RH-H71	

Overall, the pipe supports examined appeared acceptable. The NRC inspector also noted, during the walkdown, that several modifications were in progress.

c. Inservice Tests for Snubbers

The NRC inspector witnessed functional testing on the snubbers. Observations by the NRC inspector indicated the following:

- Personnel performing the testing were qualified.
- Proper instructions and procedures were followed.
- The functional test machine and accessories were calibrated as required.
- As-found drag force, activation/acceleration and as-left drag force were within acceptable limits.

The licensee's testing program was being conducted by NPPD personnel and was found to be adequate.

The licensee has established an adequate program and implementing procedures for the examination and testing of pipe supports.

d. Operability Evaluation of Essential Piping Systems for CNS

During this inspection, the NRC inspector also reviewed the licensee's long-term plan for achieving full code qualification of all essential large bore pipe supports. The licensee committed to this plan in a letter to the NRC dated August 12, 1988. The licensee submitted a contractor prepared report (CYGNA Report 88037A) which contained a detailed evaluation of 122 supports. This report showed that those Class IN pipe supports met the design criteria for full code qualification. However, on May 12, 1989, the licensee reported that they had been informed by their contractor (CYGNA) that they had identified two pipe supports included in the above report which required modifications to satisfy the design criteria. Based upon this report, the NRC inspector stated his concern about the code qualification of the remaining supports. As the result of this concern, the licensee committed to rereview all the remaining calculations to ensure that they meet the design criteria and, if required, to implement modifications prior to plant startup.

No violations or deviations were identified.

9. Action on Previous Inspection Findings

(Closed) Open Item (298/8627-01): Completion of design change package (DCP) reviews for final closeout. An NRC inspector reviewed six DCPs and noted that they had not been closed even though the physical modification effort had been completed. The NRC inspector reviewed the completion records of the previously reviewed DCPs and found them to be

acceptable. However, the completion of these DCPs appeared to be protracted (e.g., the work on DCP 83-023 was finished in April 1986, but the DCP was not closed until April 1989). Since the DCPs referenced in the earlier NRC inspection report had been properly closed, this open item is closed.

(Closed) Violation (298/8817-01): The use of "green tags" during the performance of an integrated leak rate test (ILRT) was not properly implemented (green tags were used to designate proper alignments for testing conditions). In response to this NRC finding, NPPD committed to provide training on the use of the green equipment tags. The NRC inspector reviewed the General Employee Industry Safety Training Lesson Plan (Revision 2) and noted the inclusion of a discussion on the reasons for, and proper use of, the green tags. The NRC inspector also verified, through a review of training plans, that the required training had been conducted.

10. Exit Meetings (30703)

The NRC inspectors summarized the scope and findings of the inspection during exit meetings conducted on May 5 and 19, 1989, with the personnel identified in paragraph 1 above. The licensee acknowledged the NRC inspectors' findings and agreed to provide the submittals discussed in paragraphs 5 and 8, of this report. The licensee did not identify as proprietary any of the material provided to, or reviewed by, the NRC inspectors during this inspection.

ATTACHMENT 1

LIST OF DOCUMENTS REVIEWED

Motor Operated Valves Procedures

- 7.3.35.1, "Testing of Motor Operated Valves Using Motor Operated Valve Analysis and Testing System (MOVATS)," Revision 1
- 7.3.35.2, "Periodic Monitoring of Motor Operated Valves Using MOVATS Motor Load Unit," Revision 0
- 7.3.35.3, "Periodic Monitoring of Motor Operated Valves Using MOVATS Motor Torque Unit," Revision 0
- 7.3.36, "Limit and Torque Switch Checkout and Adjustment for Rising Stem Limitorque MOVs," Revision 3

MOVATS Training Lesson Plans

Troubleshooting and Repair of MOVs:

- EQP 011-02-01 and related laboratory - 051
- EQP 011-02-02 and related laboratory - 052
- EQP 011-02-03 and related laboratory - 053

MOVATS Data Acquisition:

- EQP 011-05-01

Calibration and Function Test Procedures

- 6.1.4, "Main Steam Line Radiation Monitor," Revision 40
- 6.1.5, "High Reactor Pressure Transmitter," Revision 16
- 1.9, "L[illegible] Reactor Vessel Transmitter," Revision 23
- 6.1.4.1, "Main Steam Line Process Radiation Monitor," Revision 2
- 6.2.2.3.4, "Emergency CST Level Transmitter," Revision 19
- 6.2.2.3.13, "HPCI Pump Low Flow Transmitter," Revision 17
- 7.3.6.2, "Diesel Generator Annual Electrical Inspection," Revision 1
- 7.3.1.2, "Timed and Instantaneous Overcurrent Relay Testing and Calibration," Revision 3
- 7.3.1.3, "Differential Current Relay Testing and Calibration," Revision 2
- 9.4.3, "Main Steam Line Process Radiation Monitor," Revision 6

System/Component Surveillance Procedures

- 6.3.4.1, "Core Spray System Operability Test," Revision 25
- 6.3.4.3, "Sequential Loading of Emergency Diesel Generators," Revision 26
- 6.3.12.1, "Diesel Generator Operability Test," Revision 27
- 6.3.12.3, "Diesel Fuel Oil Quality Test," Revision 10
- 7.2.53.1, "Diesel Generator Engine Mechanical Inspection," Revision 0
- 14.17.1, "DG-1 Annual Inspection," Revision 0

Operating/Emergency Procedures

- 5.2.1, "Shutdown from Outside the Control Room," Revision 18

ATTACHMENT 2

TI 2515/98 - Exhibit 1

1. Plant Name: Cooper Nuclear Station (CNS)

2. Unit and Docket Number: 50-298

3. What are the average containment/drywell (C/D) temperatures during power operation as recorded by the licensee? Note: We are interested in the peak operating temperatures during the hottest summer months.

   Average weighted temperatures from July 18 through September 27, 1988, ranged from approximated 130°F to 148°F. These weighted averages were calculated in accordance with the Daily Surveillance Log (Procedure 6.2.4.1). A discussion of drywell temperatures is included in paragraph 3.1.1 of NRC Inspection Report 50-298/88-200.

4. Containment temperature at which the plant is licensed to operate (i.e., operating temperature specified in the FSAR).

   The CNS USAR Section V.2.3.2 states that the drywell is designed for an internal pressure of 56 psig coincident with a temperature of 281°F.

5. Review the temperature readings and provide your assessment as to whether or not you believe the average temperature readings accurately reflect containment/drywell conditions, or if there is a significant difference, due to temperature sensor location or stratification of containment atmosphere which could produce hot spots.

   The temperature readings are a weighted average of the temperatures at five points within the drywell; in addition, seven other drywell temperature points are recorded. Temperature sensing locations are denoted on Figure 1 attached.

6. What temperature(s) is used by the licensee in its equipment environmental qualification program when calculating the remaining qualified lifetime for all equipment inside C/D, and are these temperatures consistent with temperatures being experienced?

   A temperature of 150°F was used in the equipment qualification program. A review of temperature data showed that some local areas exceed the 150°F by up to 25°F during the period discussed in 3 above.

7. Administrative temperature limit for the containment/drywell, if no technical specification limit exists.

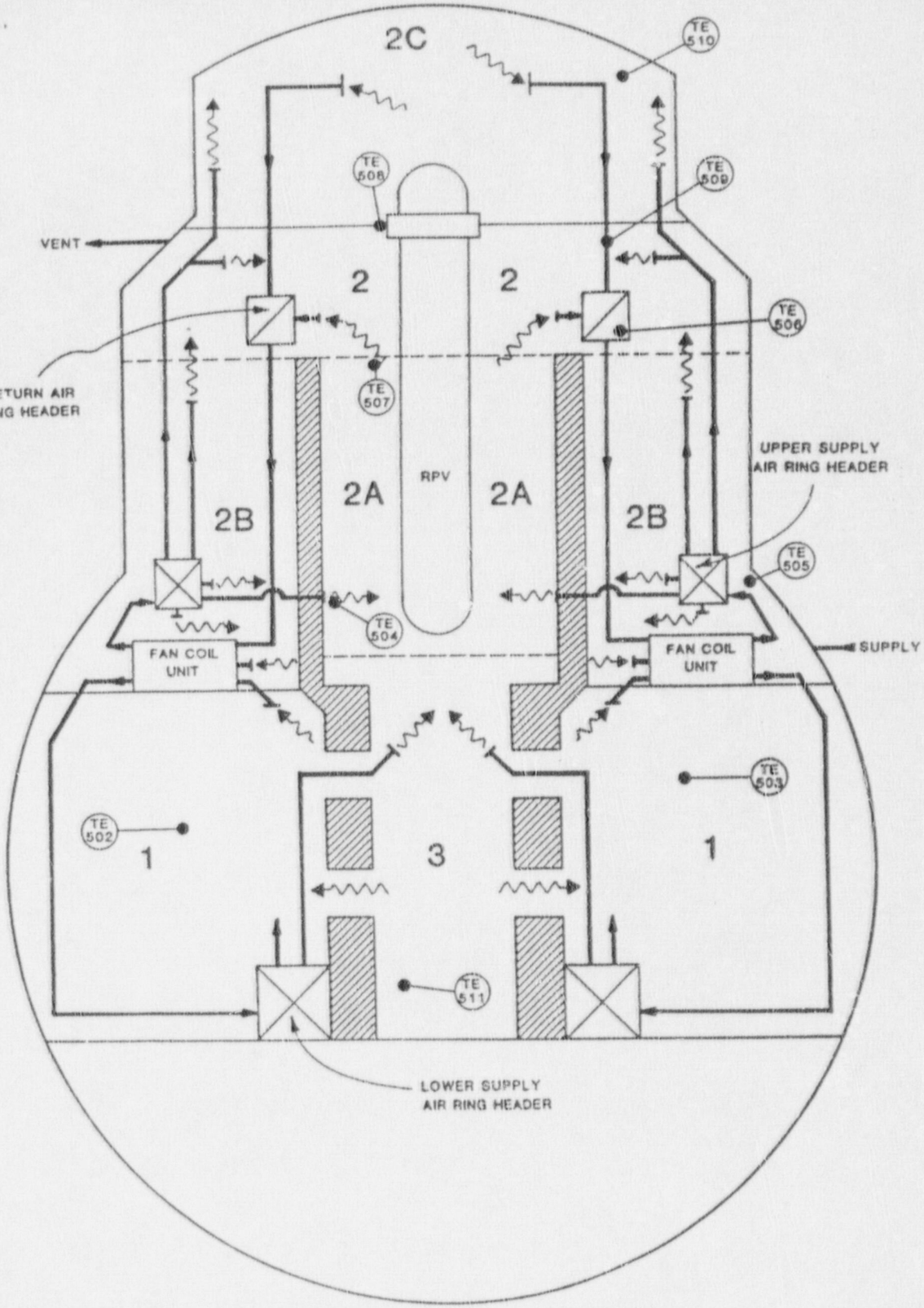
   Although the temperature instruments are listed in the Technical Specifications, no temperature restrictions are included. An administrative limit on average temperature is 150°F.

8. Recent history of temperatures inside containment. Provide C/D average air temperature in addition to the containment air temperatures used to

compute the average C/D temperatures for the months of April, May, June, July, August, and September 1987, if the plant has not operated during those months, use an operating period close to these months.

Temperature Element

1987 :	505A	505B	505C	505D	505E
Apr :	132	129	137	130	143
May :	142	140	147	141	153
Jun :	141	139	148	142	161
Jul :	148	146	155	148	156
Aug :	154	154	163	155	164
Sep :	137	136	146	140	145



DRYWELL VENTILATION RECIRCULATION

FIGURE 1

ATTACHMENT 3

SURVEY OF LICENSEE'S RESULTS TO
SELECTED EDG FO ISSUES

PLANT NAME AND UNIT: Cooper Nuclear Station

1. Has the licensee adequately reviewed and evaluated IE Information Notice 87-04 issued on January 16, 1987, as a result of the ANO Unit 2 EDG FO starvation event which occurred on June 27, 1986?

   Review of the NPPD engineering response dated March 6, 1987, and the Operating Experience Review Transmittal which was closed on March 31, 1987, indicated on adequate review and evaluation.

2. Does the licensee have a permanent FO storage tank recirculation system which allows for complete FO inventory cleaning by filtering each refueling outage to remove accumulated particulates?

   No.

3. Are all FO storage tanks being cleaned and inspected at a minimum of 10-year intervals in accordance with of Regulatory Guide 1.137?

   Yes, in accordance with Section VIII.C.2 of Reference 2.

4. Does the licensee's FO program include a regular analysis of FO samples and bottom testing for accumulated water, at the lowest point in the FO day tanks and FO storage tanks?

   Yes, monthly and after use in accordance with Reference 2.

5. Is a fuel additive being used as a fuel stabilizer which will function to prevent oxidation and bacterial growth?

   Yes, a stabilizer (Power Service Diesel Fuel Supplement) is added in accordance with Reference 1.

6. Does the licensee effectively ensure that periodic FO bottom sampling and analysis are being performed to detect high particulate concentrations in the FO supply which occurs over long-term storage due to the effects of oxidation, and biological contamination in accordance with ASTM D270-1975?

   Yes, every 6 months in accordance with Section VIII.C.1 of Reference 2.

7. Are day tanks and integral tanks being checked for water monthly, as a minimum, and after each operation of the diesel where the period of operation was 1 hour or longer?

   Yes, in accordance with Section V.B.5 of Reference 2.

8. Is accumulated water removed immediately if it is determined that water is present in the storage, integral or day tanks?

   Yes, in accordance with Section VIII.A.3 of Reference 2.

9. Is the licensee replacing FO in a short period of time (about a week) if it is determined that the FO does not meet the applicable specifications?

   Yes, in accordance with Section V.B.2 of Reference 2.

10. Are FO components which may be prone to fouling being routinely monitored for indications of fouling?

    Fuel injector nozzles are tested and the fuel filters are replaced during the performance of Reference 3.

11. Are FO filters and strainers being cleaned and inspected on a periodic basis per the vendor recommendations?

    Fuel filters are replaced annually (see No. 10 above) and fuel strainers are inspected each cycle in accordance with PM 04779.

12. Does the FO system utilize dual element filters and strainers which permits on line cleaning of the elements, in the event of fouling, to allow continuous operation of the EDG?

    There is a single strainer in the suction to each pump with a duplex filter in the common discharge line (mounted on engine).

13. Is there a differential pressure indicator for each duplex filter/strainer for indication of fouling in accordance with ANSI N195-1976?

    No.

14. Are FO alarms annunciated in the main control room or incorporated into a general control room trouble alarm with local individual alarms, in accordance with ANSI N195-1976?

    Fuel oil alarms are annunciated in the control room and locally.

15. Are any of the instruments that perform a control function and provide an alarm seismically qualified in accordance with the IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Station, IEEE-344-1975?

    The original design did not include seismic qualification; however, replacement parts are being qualified.

References:

1. OP 2.1.16, Revision 11, "Heating Boiler and Diesel Fuel Oil Unloading"
2. SP 6.3.12.3, Revision 10, "Diesel Fuel Oil Quality Test"
3. SP 7.2.53.1, Revision 0, "Diesel Generator Engine Mechanical Inspection"
4. Drawing 2011, Sheet 1, "Diesel Oil System"
5. Drawing 2077, "Flow Diagram - Diesel Generator Auxiliary Systems"