APPENDIX B

U. S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-298/86-36 License: DPR-46

Docket: 50-298

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

Facility Name: Cooper Nuclear Station (CNS)

Inspection At: Cooper Nuclear Station, Nemaha County, Nebraska

Inspection Conducted: December 1-31, 1986

Inspectors: E. A. Plettner 1/12/87
E. A. Plettner, Resident Inspector, (RI) Date

D L DuBois 1/12/87
D. L. DuBois, Senior Resident Inspector, (SRI) Date

Approved: J P Jaudon 1/23/87
J. P. Jaudon, Chief, Project Section A, Date
Reactor Project Branch

Inspection Summary

Inspection Conducted December 1-31, 1986 (Report 50-298/86-36)

Areas Inspected: Routine, unannounced inspection of complex surveillance, reactor shutdown margin, core thermal power evaluation, core power distribution limits surveillance, calibration of nuclear instruments, refueling, cold weather preparation, operational safety verification, and monthly surveillance and maintenance activities.

Results: Within the areas inspected, two violations were identified (failure to have documented procedure evaluation, paragraph 6; and failure to follow procedure, paragraph 7).

DETAILS

1. Persons Contacted

Principal Licensee Employees

*G. R. Horn, Division Manager of Nuclear Operations
*W. E. Crawford, Supervisor, Maintenance
*J. Sayer, Manager, Radiation Protection
*C. R. Goings, Regulatory Compliance Specialist
*D. C. Shrader, Assistant Supervisor, Operations
*H. A. Jantzen, Supervisor, Instrumentation and Control
*C. R. Moeller, Supervisor, Technical Staff
*G. Smith, Acting Manager, Quality Assurance

The NRC inspectors also interviewed other licensee employees during the course of the inspection.

*Denotes those present during exit interview January 12, 1987.

2. Complex Surveillance

The purpose of this inspection was to verify that complex surveillance tests, applicable to safety-related systems and subsystems, were performed in accordance with the requirements established in the CNS Operating License and Technical Specifications.

The RI observed the performance of the following procedures on the indicated dates:

- December 30, 1986: Surveillance Procedure (SP) 6.3.4.3 "CS, RHR, and Diesel Auto Start and Loading," Revision 21, dated August 22, 1985;

- December 31, 1986 and January 1, 1987: General Operating Procedure (GOP) 2.1.14 "Reactor Vessel In-Service Leak Test," Revision 16, dated April 3, 1986.

Reviews and observations verified that:

- Testing was performed using approved procedures, which were consistent with regulatory requirements, industry standards, and the Technical Specification.

- Permanent or temporary procedure revisions were accomplished according to administrative requirements and controls.

- Qualified personnel conducted the tests and performed the final reviews and approvals of completed test data.

- The official test copy was available and used by test personnel.

- Procedures contained the purpose, objectives, references, prerequisites, test equipment, precautions, limitations, and acceptance criteria.

- Test equipment required by the procedures was calibrated and in service.

- Procedures provided sufficient direction to accomplish necessary evolutions.

- Systems were returned to normal lineup following completion of testing.

The requirements for the diesel generator sequential loading are stated in the Updated Safety Analysis Report (USAR) in Section VIII Part 5.0. "Standby-A-C Power Source." The NRC inspectors performed a comparative review of Procedure SP 6.3.4.3 with the requirements in the USAR.

From the review it was determined that the procedure may not meet all the requirements as stated in the USAR in Section VIII, Part 5.0. The NRC inspectors are waiting on information from the licensee to the following questions:

- Are Motor Control Centers (MCC) K and S part of the load shed and load sequencing on the 1F for K or 1G for S emergency switch gear buses?

- Are there time delays associated with the starting of the Standby Gas Treatment fans which receive their power from MCC K for train A or MCC S for train B?

- When is the less than or equal to 10 second timing requirement for the diesel to start and attain rated voltage and frequency applicable; I.E. monthly or once per cycle?

It was noted that the current SP 6.3.4.3 does not time the diesel from start until it has attained rated voltage and frequency. Instead, the procedure times the period from diesel start to output breaker closing. USAR allows a three-second delay in the typical sequential loading of diesel generator (DG) from the time the diesel attains rated voltage and frequency until the output breaker closes. SP 6.3.12.1, performed monthly, requires that the start time be recorded with a stop watch but has no acceptance criteria stated in the procedure.

Pending receipt of additional information from the licensee, these questions will be considered to be an unresolved item (298/8626-01).

On December 29, 1986, SP 6.3.4.3 was conducted twice. The first test was conducted by tripping breaker 1FA from Board C instead of deenergizing undervoltage relays 27/1F2 and 27/1FA2 as specified by SP 6.3.4.3, Step VIII.A.19. However, the remainder of the test was conducted as required, and DG No. 1 started and loaded properly. The second test was run as specified in the procedure and demonstrated the proper response to a simulated undervoltage condition (i.e., diesel generator automatic start).

This is an apparent violation of failure to adhere to procedure. A notice of violation will not be issued, since the violation was self-identified by the licensee, and it meets the remaining four criteria stated in Appendix C, Section V.A of 10 CFR Part 2.

One unresolved item was identified in this area.

3. Reactor Shutdown Margin

On December 21, 1986, the RI observed the performance of Nuclear Performance Procedure (NPP) 10.16, "Shutdown Margin Evaluation," Revision 14, dated October 16, 1986. Included were checks to ensure that test prerequisites were completed, testing was performed according to procedures, appropriate precautions and limitations were observed, and test results were adequately reviewed for accuracy and completeness. The RI also reviewed GE supplied nuclear engineering data, which was used to determine the sequence of rod withdrawals, expected reactivity addition rates, and predicted shutdown margin values. The RI verified that test results conformed with Technical Specification requirements.

No violations or deviations were found in this area.

4. Core Thermal Power Evaluation

The RI reviewed licensee records, data sheets, and procedures applicable to reactor core thermal power evaluation and performance which were conducted during the period June 1 through October 3, 1986. The RI verified the following:

- Procedure prerequisites were met prior to performing the core thermal power evaluations.
- Figures and curves corresponding to specific reactor conditions were interpreted properly and recorded on data forms.
- Calculations were correct.
- The evaluation frequency met CNS Technical Specification requirements.

The following CNS procedures were reviewed:

- Nuclear Performance Procedure (NPP) 10.1, "APRM Calibrations," Revision 16, dated October 30, 1986
- NPP 10.2, "IRM Power Calibration," Revision 11, dated May 11, 1984
- NPP 10.3, "Core Thermal Power Evaluation," Revision 7, dated October 31, 1986
- NPP 10.5, "LPRM Calibration," Revision 17, dated January 25, 1985

The reviews were conducted to verify that CNS is operated within the licensed core thermal power limits.

No violations or deviations were identified in this area.

5. Core Power Distribution Limits Surveillance

The RI conducted discussions with CNS reactor engineers and operations department personnel, reviewed records and data applicable to core thermal limits, and verified that appropriate corrective actions were taken when core thermal data indicated an approach to limiting conditions. The review and discussions included the following:

- Verification that the linear heat generation rate (LHGR), core maximum peaking factors (CMPF), minimum critical power ratio (MCPR), and average planar linear heat generation rate (APLHGR), were within prescribed Technical Specification limits.
- Examinations of local power range monitor (LPRM) and BASE distribution calculations as typed out by the OD-1, "LPRM Calibration and Base Data," on-demand typewriter. Typed alarms, errors, and other inprocess messages were also reviewed.
- Verification that traversing incore probe (TIP) machine normalization factors were properly obtained.
- Examination of licensee procedures for ascertaining operation within licensed limits, should the process computer become unavailable.
- Verification that average power range monitor (APRM) channel gains were adjusted as necessary following an LPRM calibration.
- Verification that following an APRM gain adjustment, a subsequent P-1 was run to assure that APRM gain adjustment factor (GAF) reflected such gain adjustments.
- Examination of licensee procedures which are used to correct abnormal core thermal conditions.

The following computer printouts edited from June 1 to October 3, 1986, were reviewed:

- OD-1, "LPRM Calibration and BASE Data"
- OD-3, "Core Thermal Power and APRM Calibration"
- OD-6, "Thermal Data for Specified Fuel Bundles"
- OD-7, "Control Rod Notch Positions"
- OD-8, "LPRM Console Readings"
- OD-10, "Edit Specified Data Array"
- OD-15, "Computer Shutdown and Outage Recovery Monitor"
- OD-16, "Target Exposure and Power Data"
- OD-17, "Edit Periodic Core Performance Logs"
- P-1, "Periodic Core Performance"

The following CNS procedures were reviewed:

- NPP 10.4, "Core Thermal Hydraulic Evaluation," Revision 8, dated April 24, 1986
- NPP 10.7, "Maximum Average Planar and Peak Linear Heat Generation Rates and Minimum Critical Power Ratio," Revision 8, dated October 30, 1986
- NPP 10.8, "Reactivity Follow Check," Revision 9, dated October 18, 1984
- NPP 10.9, "Control Rod Scram Time Evaluation," Revision 13, dated November 6, 1986
- NPP 10.10, "Limiting Control Rod Pattern Determination," Revision 7, dated January 21, 1985
- NPP 10.11, "Control Rod Sequence Exchange," Revision 8, dated October 30, 1986
- NPP 10.13, "Control Rod Sequence and Movement Control," Revision 13, dated October 30, 1986

The reviews and discussions were conducted to verify that the plant is being operated within licensed power distribution limits.

No violations or deviations were identified in this area.

6. <u>Calibration of Nuclear Instruments</u>

The purpose of the inspection was to determine that Source Range Monitors, (SRM), Intermiate Range Monitors (IRM), Local Power Range Monitors (LPRM), and Average Power Range Monitors (APRM) had been properly calibrated in accordance with approved procedures and met the calibration frequency required by Technical Specification.

The RI reviewed the results of SRM, IRM, LPRM, and APRM calibrations to verify the following:

- All precautions and prerequisites were met.
- Power supply voltages were all within tolerance.
- The instruments and calibration equipment used were traceable to the National Bureau of Standards.
- Test equipments used and their serial numbers were recorded on the procedure.
- Licensee's procedures ensure that the setpoint levels for alarms, permissive and prohibitive interlocks are in compliance with the appropriate Technical Specifications.
- Calibration results were reviewed, approved, and documented in accordance with the licensee's administrative control procedures.

The RI reviewed the following procedures:

- SP 6.1.3, "APRM System Excluding 15% Trip Functional Test," Revision 15, dated September 18, 1986.
- SP 6.1.17, "IRM Calibration and Functional Test, (Mode Switch not In Run)," Revision 10, dated August 8, 1985.
- SP 6.1.17A, "IRM Calibration and Functional Test (Mode Switch In Run)," Revision 3, dated May 22, 1982
- SP 6.1.19, "LPRM Calibration Test," Revision 7, dated March 21, 1986
- SP 6.1.21, "SRM Calibration and Functional Test (Reactor Not In Run)," Revision 14, dated October 8, 1984
- SP 6,1,21A, "SRM Calibration and Functional test (Reactor In Run)," Revision 6, dated October 10, 1984
- SP 6.1.22, "APRM System 15% High Flux and Inop Trip Functional Test," Revision 11, dated September 11, 1986

- SP 6.1.29, "APRM System (Flow Bias and Startup) Calibration and/or Functional Test," Revision 14, dated January 6, 1986

The RI performed an audit of calibration data, Attachments "A," of the following procedures for the time frame of September 1985 to December 1986:

- Instrument and Control Procedure (I&C) 7.5.2.1, "SRM Quarterly Calibration Procedure," Revision 13, dated October 30, 1986.

- I&C 7.5.2.2, "IRM Calibration Procedure," Revision 11, dated October 30, 1986

- I&C 7.5.2.3, "LPRM Calibration Procedure," Revision 11, dated April 10, 1986

- I&C 7.5.2.4, "APRM Calibration Procedure," Revision 12, dated June 19, 1986

The results of the audit revealed that the "reviewed by" signature blank and associated "date" blank had not been completed on the following:

- 4 times in I&C 7.5.2.1 Procedure, Attachment "A"

- 2 times in I&C 7.5.2.4 Procedure, Attachment "A"

Appendix B Criterion XI to of 10 CFR Part 50 and the Licensee's Quality Assurance Plan require that test results shall be documented and evaluated to assure that test requirements have been satisfied. The failure to document the review of Attachments "A" in the above procedures is an apparent violation (298/8636-02).

One violation was identified in this area.

7. Refueling

The RI held discussions with fuel handling personnel, observed fuel movement from the reactor to the spent fuel pool, verified spent fuel pool fuel assemblies locations, accountability records, and status board updates. The RI also reviewed the licensee's procedures and records concerning the movement of fuel and storage of fuel assemblies.
On December 16, 1986, the RI observed the following fuel movement activities:

STEP	FUEL BUNDLE NO.	FROM	TO
434	LY2089	11-C-11	41-28
435	LY2034	11-B-10	43-26
438	LY2120	11-H-10	41-26

439	LY2079	11-G-11	43-28
440	LYD808	12-M-11	41-30
441	LYD802	12-I-12	43-32

On December 12, 1986, at 2:20 p.m. refueling activities started.

On December 16, 1986, the RI performed a review of procedures being used on the refueling floor. The results of the review revealed that Nuclear Performance Procedure (NPP) 10.25 "Refueling," was Revision 6, dated August 7, 1986. The latest issued revision of Procedure NPP 10.25 was Revision 7, dated December 10, 1986. Procedure NPP 10.25 in Section VII.A.2 requires the licensee to verify that the procedures in the operations refueling floor procedure book are of the latest revision and update as necessary.

The licensee's failure to verify that the procedure NPP 10.25 in the operations refueling floor procedure book was of the latest revision as required by procedure NPP 10.25 is an apparent violation (298/8636-03).

The RI reviewed Report G-HPO-6-424, dated December 3, 1986, which General Electric (GE) sent to the licensee. The report addressed the impact and lost parts analysis for the double blade quide problem discussed in NRC Inspection Report 50-298/86-26, paragraph 6. The GE report concluded:

- The falling blade guide has caused no structural damage to either the fuel bundles or core top guide assembly.
- The lost pieces of the aluminum blade guide handle in the reactor pressure vessel would not compromise safe reactor operation.

No further action is required.

One violation was identified in this area.

8. Cold Weather Preparation

IE Bulletin 79-24, "Frozen Lines," requested the licensee to verify that adequate protective measures had been taken to prevent safety-related process, instrument, and sampling lines from freezing during extremely cold weather. The RI verified the following:

- Individual plant systems operating procedures identified heating requirements and equipment including power supplies, temperature controls and settings, indication circuits, insulation requirements, heat tracing, and space heaters as required.
- Backup freeze protection was provided during extended plant shutdown in areas that are normally kept warm by heat losses from operational systems.

- Plant procedures used during maintenance or modification of existing systems provided reasonable assurance that cold weather protective measures were reestablished following completion of those activities.

- Plant preventive maintenance requirements associated with cold weather preparation were completed on September 17, 1986.

The following documents were reviewed:

- Preventive Maintenance (PM) routine 04047 and 01271

- General Operating Procedure (GOP) 2.1.11, "Station Operators Tour," Revision 39, dated October 30, 1985.

- Attachment "C" to GOP 2.1.11, "Station Operators Tour - R/W, AOG, ARW Areas and Outside."

- System Operating Procedure (SOP) 2.2.30, "Fire Protection System," Revision 24, dated October 17, 1985.

The discussions, reviews, and walkdowns were performed to verify that the licensee has maintained an effective program of cold weather protective measures for safety-related components and systems.

No violations or deviations were identified in this area.

9. Operational Safety Verification

The NRC inspectors observed control room operations, instrumentation, controls, reviewed plant logs and records, conducted discussions with control room personnel, and performed system walkdowns to verify that:

- Minimum shift manning requirements were met.

- Technical Specification requirements were observed.

- Plant operations were conducted using approved procedures.

- Plant logs and records were complete, accurate, and indicative of actual system conditions and configurations.

- System pumps, valves, control switches, and power supply breakers were properly aligned.

- Licensee systems lineup procedures/checklists, plant drawings, and as-built configurations were in agreement.

- Instrumentation was accurately displaying process variables and protection system status to be within permissible operational limits for operation.

- When plant equipment was found to be inoperable or when equipment was removed from service for maintenance, it was properly identified and redundant equipment was verified to be operable. Also, the NRC inspectors verified that applicable limiting conditions for operation were identified and maintained.
- Equipment safety clearance records were complete and indicated that affected components were removed from and returned to service in a correct and approved manner.
- Maintenance work requests were initiated for equipment discovered to require repair or routine preventive upkeep, appropriate priority was assigned, and work commenced in a timely manner.
- Plant equipment conditions such as cleanliness, leakage, lubrication, and cooling water were controlled and adequately maintained.
- Areas of the plant were clean, unobstructed, and free of fire hazards. Fire suppression systems and emergency equipment were maintained in a condition of readiness.
- Security measures and radiological controls were adequate.

The RI performed a lineup verification of the valves in normally inaccessible areas in the following systems:

- Reactor Core Isolations Cooling (RCIC)
- High Pressure Coolant Injection (HPCI)
- Standby Liquid Control (SLC)
- Main Steam and Turbine Bypass (MS)
- Feedwater

In preparation for performing the system walkdown of the MS and Feedwater systems, the RI conducted a review of and comparison between the following licensee MS and Feedwater systems valve checklist and applicable as-built drawings:

System Operating Procedure (SOP) 2.2.56, "Mainsteam and Turbine Bypass System," Revision 21, dated August 7, 1986, Appendix A, "Valve Checklist"

- As-Built drawing - Burns & Roe 2002; for MS System
- As-Built drawing - Burns & Roe 2041; for MS System
- As-Built drawing - GE 115D6014; for MS System

- As-Built drawing - Burns & Roe M-107; for MS System
- As-Built drawing - Burns & Roe M-109; for MS System
- As-Built drawing - NPPD I.D.-1; for MS System
- As-Built drawing - NPPD I.D.-3; for MS System
- As-Built drawing - NPPD I.D.-18; for MS System

SOP 2.2.28 "Feedwater System," Revision 39, dated July 17, 1986; Appendix A "Valve Checklist"

- As-Built drawing - Burns & Roe 2002; for Feedwater
- As-Built drawing - Burns & Roe 2004; for Feedwater
- As-Built drawing - Burns & Roe 2005; for Feedwater
- As-Built drawing - Burns & Roe 2043; for Feedwater
- As-Built drawing - Burns & Roe 2044; for Feedwater
- As-Built drawing - Burns & Roe M-110; for Feedwater
- As-Built drawing - Burns & Roe M-111; for Feedwater
- As-Built drawing - NPPD I.D.-4; for Feedwater
- As-Built drawing - NPPD I.D.-6; for Feedwater

The review identified that SOP 2.2.56 Appendix A, listed 194 instrument related valves that were not numbered or labeled on applicable as-built drawings 115D6014, M-107, M-109, I.D.-1, I.D.-3, or I.D.-18.

The review identified that SOP 2.2.38 Appendix A, listed 149 instrument related valves that were not numbered or labeled on applicable as-built drawings M-110, M-111, I.D.-4, or I.D.-6.

These deficiencies are similar to the violation that was documented in NRC Inspection Report 50-298/86-14, paragraph 5 and similar to Open Item 298/8626-01 identified in NRC Inspection Report 50-298/86-26, paragraph 5. These two items will be tracked as an open item pending review of licensee's corrective action (298/8636-04).

The RI performed an equipment safety clearance record verification for clearance orders 86-1008 and 86-688. During the performance of the verification the RI noted on Clearance Order 86-1008 that Tag Number 24 had been removed, but the "Return to Normal By" blank had no initials in the blank as required. This item was brought to the licensee's attention. Discussions with the licensee revealed that Tag No. 24 had been removed.

The person who removed the tag had initialed the wrong blank on the clearance order. The responsible individual corrected the discrepancy in a timely manner.

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.

10. Monthly Surveillance Observations

The NRC inspectors observed Technical Specification required surveillance tests. Those observations verified that:

- Tests were accomplished by qualified personnel in accordance with approved procedures.
- Procedures conformed to Technical Specification requirements.
- Tests prerequisites were completed including conformance with applicable limiting conditions for operation, required administrative approval, and availability of calibrated test equipment.
- Test data was reviewed for completeness, accuracy, and conformance with established criteria and Technical Specification requirements.
- Deficiencies were corrected in a timely manner.
- The system was returned to service.

The NRC inspectors observed the licensee's performance of the following surveillance tests on the indicated dates:

- December 10, 1986: Surveillance Procedure (SP) 6.2.2.5.12A, "RHR Loops A and B Pump and Valve Control Logic Reactor Vessel Pressure Less Than or Equal to 75 psig Functional Test," Revision 9, dated November 13, 1986.
- December 12, 1986: SP 6.1.1, "SRM Functional Test (Reactor Not In Run)," Revision 14, dated October 30, 1986.
- December 12, 1986: SP 6.1.26, "Refueling Platform Interlock and System Functional Tests," Revision 20, dated December 4, 1986, Attachments B and E.
- December 12, 1986: SP 6.1.27, "Refueling Platform Interlock and System Functional Tests," Revision 20, dated December 4, 1986, Attachments E and D.

- December 16, 1986: SP 6.2.27, "Refueling Platform Interlock and System Functional Tests," Revision 20, dated December 4, 1986, Attachment B.

- December 17, 1986: SP 6.4.3.1, "HPCI Turbine Overspeed Functional Test," Revision 10, dated May 15, 1986, with temporary procedure change added.

- December 21, 1986: General Operating Procedure (GOP), 2.1.18, "Control Rod Drive Friction Test," Revision 10, dated March 21, 1986, with temporary change procedure added.

- December 21, 1986: SP 6.4.1.2,"Withdrawn Control Rod Operability," Revision 16, dated December 18, 1986.

- December 21, 1986: SP 6.4.1.3, "CRD Coupling Integrity Check," Revision 7, dated October 30, 1986.

The 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.

11. Monthly Maintenance Observation

The NRC inspectors observed preventive and corrective maintenance activities. These observations verified that:

- Limiting conditions for operation were met.

- Redundant equipment was operable.

- Equipment was adequately isolated and safety tagged.

- Appropriate administrative approvals were obtained prior to commencement of work activities.

- Work was performed by qualified personnel in accordance with approved procedures.

- Radiological controls, cleanliness practices, and appropriate fire prevention precautions were implemented and maintained.

- Quality control checks and postmaintenance surveillance testing were performed as required.

- Equipment was properly returned to service.

The NRC inspectors observed the licensee's performance of the following maintenance activities on the indicated date.

- December 18, 1986: "A" Service Water Pump
- December 21, 1986: Design Change (DC) 86-134, "Replacement of Reactor Building Personnel Airlock Doors"
- December 22, 1986: Maintainance on Reactor Equipment Cooling (REC) Heat Exchanger A and B Outlet Valves
- December 22, 1986: Diesel Generator No. 1 Head Torque and Water Jacket Gasket Repair

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 Specification.

CNS issued a justification for Interim Operation in response to a self-identified problem in the Standby Gas Treatment (SGT) System. Reference Licensee Event Report 86-28, "SGT System Seismic Design Deficiencies," dated November 20, 1986. A review of the original design documentation revealed that no weld records existed to identify the type of welding rod used to weld the dissimilar metals (stainless steel to carbon steel.) To ensure that the welding rod used to make the dissimilar metal welds was correct the RI performed a visual inspection of 12 different welds selected at random. The following list of hanger welds were visually inspected for stress cracking (SC) and weld degradation (WD) with results noted:

Hanger Designation	Result	Hanger Designation	Result
F01	No SC or WD	F02	No SC or WD
301	No SC or WD	302	No SC or WD
303 (2)	No SC or WD	304	No SC or WD
306	No SC or WD	307	No SC or WD
309	No SC or WD	313	No SC or WD
317	No SC or WD		

Since no SCs or WDs were found, it was concluded that the appropriate welding rod material had been used to weld the dissimilar metals. No further action was required.

No violations or deviations were identified in this area.

12. Unresolved Items

An unresolved item is one about which additional information is required in order to determine if the item is acceptable, a violation, or a deviation. The following unresolved items were identified during this inspection.

Item	Paragraph	Subject
298/8636-01	2	Diesel Sequential Loading

13. Exit Interviews

Exit interviews were conducted at the conclusion of each portion of the inspection. The NRC inspectors summarized the scope and findings of each inspection segment at those meetings.